A/P

NUREG–1511

Reactor Pressure Vessel Status Report

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

J. Strosnider, K. Wichman, B. Elliot, C. Fairbanks,
E. Hackett, S. Sheng, J. Tsao, L. Lois, M. Mayfield,
M. Mitchell

AVAILABILITY NOTICE

Availability of Reference Materials Cited in NRC Publications

Most documents cited in NRC publications will be available from one of the following sources:

1. The NRC Public Document Room, 2120 L Street, NW., Lower Level, Washington, DC 20555-0001
2. The Superintendent of Documents, U.S. Government Printing Office, P. O. Box 37082, Washington, DC 20402-9328
3. The National Technical Information Service, Springfield, VA 22161-0002

Although the listing that follows represents the majority of documents cited in NRC publications, it is not intended to be exhaustive.

Referenced documents available for inspection and copying for a fee from the NRC Public Document Room include NRC correspondence and internal NRC memoranda; NRC bulletins, circulars, information notices, inspection and investigation notices; licensee event reports; vendor reports and correspondence; Commission papers; and applicant and licensee documents and correspondence.

The following documents in the NUREG series are available for purchase from the Government Printing Office: formal NRC staff and contractor reports, NRC-sponsored conference proceedings, international agreement reports, grantee reports, and NRC booklets and brochures. Also available are regulatory guides, NRC regulations in the *Code of Federal Regulations*, and *Nuclear Regulatory Commission Issuances.*

Documents available from the National Technical Information Service include NUREG-series reports and technical reports prepared by other Federal agencies and reports prepared by the Atomic Energy Commission, forerunner agency to the Nuclear Regulatory Commission.

Documents available from public and special technical libraries include all open literature items, such as books, journal articles, and transactions. *Federal Register* notices, Federal and State legislation, and congressional reports can usually be obtained from these libraries.

Documents such as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings are available for purchase from the organization sponsoring the publication cited.

Single copies of NRC draft reports are available free, to the extent of supply, upon written request to the Office of Administration, Distribution and Mail Services Section, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at the NRC Library, Two White Flint North, 11545 Rockville Pike, Rockville, MD 20852-2738, for use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from the American National Standards Institute, 1430 Broadway, New York, NY 10018-3308.

NUREG–1511

Reactor Pressure Vessel Status Report

Manuscript Completed: September 1994
Date Published: December 1994

J. Strosnider, K. Wichman, B. Elliot, C. Fairbanks,
E. Hackett, S. Sheng, J. Tsao, L. Lois, M. Mayfield,
M. Mitchell

Division of Engineering
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555–0001

ABSTRACT

The Nuclear Regulatory Commission (NRC) issued Generic Letter 92-01 (GL 92-01) to obtain information needed to assess compliance with requirements and commitments regarding reactor vessel integrity in view of certain concerns raised in the NRC staff's review of reactor vessel integrity for the Yankee Nuclear Power Station.

This report gives a brief description of the reactor pressure vessel (RPV), followed by a discussion of the radiation embrittlement of RPV beltline materials and the two indicators for measuring embrittlement, the end-of-license (EOL) reference temperature and the EOL upper-shelf energy. It also summarizes the GL 92-01 effort and presents, for all 37 boiling water reactor plants and 74 pressurized water reactor plants in the United States, the current status of compliance with regulatory requirements related to ensuring RPV integrity. The staff has evaluated the material data needed to predict neutron embrittlement of the reactor vessel beltline materials. These data will be stored in a computer database entitled the reactor vessel integrity database (RVID). This database will be updated annually to reflect the changes made by the licensees in future submittals and will be used by the NRC staff to assess the issues related to vessel structural integrity.

CONTENTS

Figures

Table

EXECUTIVE SUMMARY

The reactor pressure vessel (RPV) and its internal components support and align the fuel assemblies that make up the reactor core and provide a flow path to ensure adequate heat removal from the fuel assemblies. The RPV also provides containment and a floodable volume to maintain core cooling in the event of an accident causing loss of the primary coolant. Maintenance of the structural integrity of the RPV is essential in ensuring plant safety, since there is no redundant system that can maintain core cooling in the event of vessel failure.

The reactor vessel comprises a cylindrical shell with a welded hemispherical bottom head and a removable hemispherical upper head (see Figs. 2.3 and 2.4). Some RPV shells were fabricated from curved plates that were joined by longitudinal and circumferential welds. Others were manufactured using forged rings and therefore have only circumferential welds that join the rings. The cylindrical portion of the RPV typically ranges from about 4.57 meters to 6.10 meters (15 feet to 20 feet) in diameter, 12.19 meters to 21.34 meters (40 feet to 70 feet) in height, and has a wall thickness ranging from 12.70 cm to 22.86 cm (5 inches to 9 inches).

There are two types of commercial light water nuclear power reactors: pressurized water reactors (PWRs) and boiling water reactors (BWRs). PWR vessels operate in a water solid condition with sufficient pressure to prevent steaming in the reactor vessel (e.g., 15.51 megapascal (MPa) at 288 °C or 2250 pounds per square inch (psi) at 550 °F). BWR vessels operate with a large portion of water inventory at saturated steam conditions (e.g., 8.62 MPa at 288 °C or 1050 psi at 550 °F).

RPVs are fabricated from low-alloy steels. An important characteristic of these steels is that their properties change as a function of temperature and as a function of irradiation. The material property of particular importance in assessing RPV integrity is fracture resistance. (The technical term for fracture resistance, as used in the regulations, is fracture toughness.) Fracture resistance is a measure of the material's capability to resist sudden failure caused by the propagation of a crack. Fracture resistance of RPV materials increases with increasing temperature and decreases with increasing irradiation from the reactor. Charpy V-notch impact energy tests, which measure the amount of energy required to fail a small material specimen, can be related to the change in fracture resistance. Figure 3.1 shows how the impact energy and fracture resistance increase with temperature until a plateau or "upper-shelf" value is reached. Figure 3.2(a) illustrates the effects of irradiation. Two major effects are apparent. First, as the level of irradiation increases it is necessary to go to higher temperatures to achieve the same level of fracture resistance. This change is referred to as the "shift in transition temperature." Second, the upper-shelf value, that is, the maximum fracture resistance, decreases.

A few observations using Figure 3.2(a) can give some important insights with regard to RPV integrity issues. Normal RPV operating temperature when a plant is at full power is about 288 °C (550 °F). However, during certain transients or under postulated accident conditions, the reactor temperature can decrease. At the beginning of service, when the fluence levels are low, the RPV is typically operating with its material properties at a few hundred degrees above the temperature at which the upper-shelf energy (USE) is reached. This provides a large temperature range in which the reactor temperature can decrease without any decrease in material properties. However, with increasing levels of fluence, this temperature range becomes smaller and a drop in temperature could result in a significant decrease in the fracture resistance of the reactor vessel. This observation is extremely important in assessing the implications of system transients referred to as "pressurized thermal shock (PTS) transients." These types of transients are characterized by a rapid and significant decrease in reactor temperature concurrent with or followed by repressurization in the reactor vessel. To ensure that PTS transients do not pose a serious threat to RPV integrity, Section 50.61 of Title 10 of the *Code of Federal Regulations* (10 CFR), the PTS rule, limits the shift in transition temperature caused by irradiation. The specific parameter defined in the rule that is related to the shift in transition temperature is referred to as the RT_{PTS} value, and it is limited to 132 °C (270 °F) for axial welds, plates, or forgings and 149 °C (300 °F) for circumferential welds in the RPV. The RT_{PTS} value increases with irradiation and the rate of increase is higher for materials with greater copper and nickel contents. The PTS rule gives guidance on how the RT_{PTS} value is to be determined on the basis of irradiation levels and material chemistries. PWRs must meet the requirements of the PTS rule. However, BWRs do not need to meet these requirements because they operate with a large portion of water inventory inside the pressure vessel at saturated steam conditions. Since any sudden cooling will condense steam and result in a pressure decrease, simultaneous creation of high pressure and low temperature is improbable.

The staff of the U.S. Nuclear Regulatory Commission (NRC) has reviewed the responses to Generic Letter (GL) 92-01 and other docketed information and assessed the RT_{PTS} values for all the PWR RPVs in U.S. plants. On the basis of this review, the NRC staff has concluded that the RPVs in most plants will be well below the PTS screening limits at the end of their current operating licenses. On the basis of the currently docketed information, Beaver Valley 1 and Palisades are the only plants projected to exceed their PTS screening limits when their licenses expire. Beaver Valley 1 and Palisades are projected to exceed the PTS screening limits in 2012 and 2004, respectively, before the end of their operating licenses in 2016 and 2007. It is important to note that these results are based on the information currently reported by the licensees and are subject to change. The dates when the plants are projected to reach the screening criteria may change as a result of new surveillance data and additional analyses. Also, by implementing different fuel management techniques and inserting special neutron-absorbing materials in the reactor core, licensees may be able to reduce the irradiation levels sufficiently to stay below the screening criteria. In addition, licensees may anneal the RPV to restore the material properties to near the original unirradiated condition. The industry has already informed the NRC staff that it will submit additional data and information. It is expected, therefore, that the above information will change. These changes will be reported in yearly updates of this report.

BWRs do not have RT_{PTS} values because they are not subject to PTS events. However, both PWRs and BWRs must also meet the pressure-temperature limits of Appendix G, 10 CFR Part 50. These limits depend on the amount of increase in the transition temperature, which is included in the adjusted reference temperature (ART). The RT_{PTS} is the ART at the end-of-life fluence at the inside surface of the vessel.

The second issue that needs to be assessed relative to RPV integrity is the drop in USE. This drop is a function of fluence and copper content, and methods for calculating it are specified in NRC Regulatory Guide 1.99, Revision 2. Appendix G of 10 CFR Part 50 specifies that USE as measured in Charpy impact tests must be greater than 102 joules (75 ft-lb) in the unirradiated condition and should remain above 68 joules (50 ft-lb) during the operating lifetime unless analyses are performed to demonstrate that margins of safety equivalent to those specified in Appendix G of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) exist for lower energies. The industry, through owners groups programs, has performed analyses to demonstrate what USEs are necessary to satisfy the ASME Code margins for generic groupings of plants. In addition, licensees of some plants have performed plant-specific analyses.

The NRC staff has reviewed the responses to GL 92-01 and other docketed information and the industry's equivalent margins analyses and has assessed the USE values for all the RPVs in U.S. plants. In some cases, because of limitations in the available data, the NRC staff could not reliably conclude that the material in all RPVs would remain above 68 joules (50 ft-lb) throughout their operating life. In these cases the NRC staff used conservative values for the drop in USE and evaluated the resulting USE values to ensure that equivalent margins would exist. An NRC report, NUREG/CR-6023, "Generic Analyses for Evaluation of Low Charpy USE Effects on Safety Margins Against Fracture of RPV Materials" (July 1993), states that RPV materials for PWR and BWR vessels can have USE values of less than 68 joules (50 ft-lb) and still meet the margins of safety against fracture equivalent to those required by Appendix G of the ASME Code. On the basis of the review of the industry's equivalent margins analyses and the generic study conducted by the NRC, the staff has concluded that all RPVs will have adequate upper-shelf toughness throughout their current licensed operating life.

The assessment of RPV integrity must be a continuing proactive effort. As noted above it is expected that additional information and analyses and licensee programs to reduce neutron flux will result in changes in the currently predicted RT_{PTS} and USE values. The NRC staff will continue to assess new information as it becomes available and will provide yearly updates of this report based on this information. This effort will be facilitated by the use of a computerized reactor vessel integrity database (RVID). This database developed by the NRC includes summary tables containing necessary input for evaluating RPV structural integrity in accordance with the requirements of 10 CFR Part 50, Appendix G, and the PTS rule. The data come from licensee responses to GL 92-01 and staff's request for additional information, documents referenced in the GL 92-01 submittals, PTS submittals, and surveillance capsule reports. The data have been verified by the licensees. It is anticipated that the RVID will be made available for public access in late 1994 or early 1995. The RVID will also be updated periodically on the basis of NRC assessments of new information from the industry and licensees. Appendix B contains summary files for all plants. Each file contains ART (for BWRs) or RT_{PTS} (for PWRs) and USE data for the limiting material or materials of a plant.

ACKNOWLEDGMENTS

We wish to thank the many persons who contributed immeasurably to the work reported in this report and to the quality of the final product. In particular, we acknowledge the effort of Mr. A. Hiser, who carefully reviewed this report and gave us many valuable suggestions. We also acknowledge the effort of Mr. D. McDonald, who coordinated the staff review activities with licensees and owners groups. In addition, we would like to thank Ms. A. Lowery, Ms. J. Howard, and Ms. T. Courts for preparing the plant summary sheets in Appendix B.

NOMENCLATURE

a crack depth

CF chemistry factor used in calculating material reference temperature shift

d/da differentiate with respect to crack depth

f fluence

$F(x, t)$ neutron flux

I unirradiated or initial reference temperature

$J_{applied}$ applied J-integral fracture parameter

$J_{material}$ J-integral fracture toughness

K_{Ia} crack arrest fracture toughness

K_{IC} plane strain fracture toughness

K_{Id} dynamic fracture toughness

K_{IR} reference fracture toughness

M margin added to cover uncertainties in initial properties, copper and nickel contents, fluence, and the calculational procedures in predicting the reference temperature after irradiation

P_3 third-degree Legendre polynomial representation of the scattering angular distribution

RT_{NDT} the reference temperature as adjusted for the effect of neutron radiation

RT_{PTS} the adjusted reference temperature at the end-of-license fluence at the inside surface of the vessel

ΔRT_{PTS} increase in RT_{PTS} caused by irradiation

S_8 function representing the division of the solid angle of scattering into eight segments

ABBREVIATIONS

AEC	Atomic Energy Commission
ART	adjusted reference temperature
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
B&W	Babcock and Wilcox
B&WOG	Babcock and Wilcox Owners Group
BTP	branch technical position
BWR	boiling water reactor
BWROG	Boiling Water Reactor Owners Group
CB&I	Chicago Bridge and Iron Company
CE	Combustion Engineering
CEOG	Combustion Engineering Owners Group
CF	chemistry factor
CFR	*Code of Federal Regulations*
CVN	Charpy V-notch
2–D	two-dimensional
3–D	three-dimensional
EFPY	effective full-power year
EOL	end of license
EPFM	elastic-plastic fracture mechanics
EPRI	Electric Power Research Institute
GDC	general design criterion
GE	General Electric
GL	generic letter
HSST	Heavy Section Steel Technology
ID	inner diameter
LEFM	linear elastic fracture mechanics
LTOP	low-temperature overpressure protection
NRC	Nuclear Regulatory Commission
NSSS	nuclear steam supply system
ORNL	Oak Ridge National Laboratory
P–T	pressure-temperature
PTS	pressurized thermal shock
PVRC	Pressure Vessel Research Committee
PWR	pressurized water reactor
RCS	reactor coolant system
RG	regulatory guide
RPV	reactor pressure vessel
RVID	reactor vessel integrity database
SER	safety evaluation report
USE	upper-shelf energy
W	Westinghouse
WGFE	Working Group on Flaw Evaluation
WGOPC	Working Group on Operating Plant Criteria
WOG	Westinghouse Owners Group
WRC	Welding Research Council

1 INTRODUCTION

This report documents the results of the reassessment of reactor pressure vessel (RPV) integrity for all domestic commercial nuclear power plants performed by the staff of the U.S. Nuclear Regulatory Commission (NRC). This reassessment had two major objectives. The first objective was to complete a comprehensive evaluation of the current status of licensees' compliance with fracture toughness requirements and material surveillance program requirements. The second objective was to establish a systematic, comprehensive review and tracking process to ensure future timely assessment of RPVs as new data and information become available.

The first objective consisted primarily of reviewing licensee responses to Generic Letter (GL) 92-01, Revision 1, supplemented with other currently available docketed information. The staff evaluated the data to assess the changes in important material properties caused by neutron irradiation damage and to ensure that criteria established to ensure RPV integrity will be satisfied throughout the licensed life of the plant. Results of these evaluations are presented in this report.

The purpose of material surveillance programs is to monitor changes in the fracture toughness properties of materials in the beltline region of the RPV and to analyze RPV integrity. Surveillance programs are designed not only to examine the current status of RPV material properties, but also to predict, well in advance, the changes in these properties resulting from the cumulative effects of irradiation.

Assessing the status of all RPVs in the United States was a major undertaking. Numerous engineering analyses had to be performed and reviewed by the NRC staff, and an extensive amount of data had to be analyzed. One of the more difficult aspects of the review process was that licensees of plants with reactor vessels constructed to early editions of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code were not required to determine all the initial material properties of interest and did not necessarily include the material of most interest in their surveillance programs. These licensees had to provide fracture toughness data and analyses to demonstrate equivalence with the fracture toughness requirements. Discussions and reviews of these analyses are included in this report.

The second objective emphasized the fact that the assessment of RPV integrity is a continuing, proactive effort. To facilitate this ongoing effort, the NRC staff developed the computerized reactor vessel integrity database (RVID) from data and information gathered during this RPV integrity review effort. The RVID will be maintained and updated annually to incorporate additional new data and information.

This report includes an executive summary, which gives some basic technical background information necessary to understand RPV integrity issues, presents the current status of domestic RPVs, and describes future RPV-related activities, including the maintenance and updating of the RVID. Sections 2 through 4 of the report give detailed descriptions of the engineering and regulatory requirements associated with RPV technology and describe the review processes and analyses that were performed to assess the current status of domestic RPVs. Section 5 discusses the RVID. Section 6 presents conclusions. Appendix A is a list of owners groups and their plants. Appendix B contains RVID summary files for all domestic RPVs.

2 REACTOR PRESSURE VESSEL – GENERAL DESCRIPTION

2.1 Background

For current commercial nuclear plants in the United States, either the pressurized water reactor (PWR) or the boiling water reactor (BWR) design is used. PWR vessels operate in a water solid condition with sufficient pressure to prevent steaming in the reactor vessel. BWR vessels operate with a large portion of water inventory at saturated steam conditions. There are 74 PWR units and 37 BWR units that are either operating or about to obtain an operating license. Identified by nuclear steam supply system (NSSS) vendors, there are 52 Westinghouse (W) units, 15 Combustion Engineering (CE) units, 7 Babcock and Wilcox (B&W) units, and 37 General Electric (GE) units. Figures 2.1 and 2.2 show the typical arrangements for the PWR and BWR internal components. Table 2.1 gives the approximate RPV dimensions for PWRs and BWRs.

In the United States, RPV fabricators were primarily CE, B&W, and Chicago Bridge and Iron Company (CB&I). Some of the RPVs were fabricated in the Netherlands and one in Japan. The Lukens Steel Company was the predominant supplier of steel plates used for RPV fabrication, and Bethlehem Steel Corporation and U.S. Steel Company were the predominant suppliers of forgings for the RPV (Whitman et al., 1967). It is interesting to note that a plant's NSSS supplier (i.e., W, B&W, CE, or GE) did not necessarily fabricate the reactor vessel in the plant.

2.2 Reactor Vessel Design

The *Code of Federal Regulations*, Title 10, Part 50 (10 CFR Part 50), "Domestic Licensing of Production and Utilization Facilities," contains rules for the design, construction, operation, and inspection of nuclear power plants. The specific requirements in 10 CFR Part 50 are that the RPV design must conform to the Boiler and Pressure Vessel Code of the American Society of Mechanical Engineers (ASME Code). The applicable portions of the ASME Code are Section III, "Rules for Construction of Nuclear Power Plant Components," and Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components." Before 1963 and the issuance of ASME Code, Section III, reactor vessels were designed according to ASME Code, Section I, and/or ASME Code, Section VIII. For the design of the early vessels, various nuclear code cases, the Navy code at that time, and supplementary requirements of the vessel vendors or fabricators were also used.

2.3 Reactor Vessel Fabrication

Most reactor vessels that are in service in U.S. nuclear plants were fabricated in the 1960s. Reactor vessel fabrication has been an evolving technology, and later RPVs were fabricated using knowledge gained from the RPV surveillance programs and more modern fabrication methods. They were fabricated using large forgings, thereby reducing the number of welds in the beltline of the reactor, and materials less susceptible to radiation damage.

Steels used to fabricate the RPV were selected on the basis of stringent design and quality requirements. In the 1950s and 60s, RPVs were fabricated from American Society for Testing and Materials (ASTM) A 302 Grade B steel, a manganese-molybdenum steel plate with low nickel content, and ASTM A 302 Grade B Modified steel, the same steel with extra nickel. The A 302 Grade B Modified steel is now identified as ASTM A 533 Grade B, which has been used in RPVs since the 1960s. Another RPV material is ASTM A 508 Class 2 steel, a nickel-chromium-molybdenum forging steel for shell courses, flanges, and nozzles. Nickel is added to steel to increase toughness or resistance to impact. Molybdenum improves high-temperature tensile strength and reduces a steel's susceptibility to temper brittleness. Chromium is added to the RPV steel to improve hardenability. Manganese prevents the formation of iron sulfide at the grain boundaries and consequently minimizes surface ruptures at rolling temperatures, resulting in improvement in surface quality after rolling. Manganese also improves the tensile strength of steels.

The reactor vessel was made up of cylindrical shell courses with a welded hemispherical bottom head and a removable hemispherical upper head (see Figs. 2.3 and 2.4). Some RPV shell courses were fabricated from curved plates joined by longitudinal and circumferential welds. Some RPV shell courses were manufactured from ring forgings, which eliminated the longitudinal welds and sometimes reduced the number of circumferential welds. The welding processes were mostly the submerged-arc and shielded-metal-arc processes. Before the early 1970s, copper-coated weld wire was used in fabricating vessels to improve weldability and reduce corrosion during storage of the weld wire. When it was discovered that copper and phosphorous increased the weld's sensitivity to radiation embrittlement, RPV fabricators imposed strict limits on the percentage of copper and phosphorous in reactor vessel welds as well as plates (Whitman et al., 1967, Griesbach and Server, 1993, and Griesbach, 1994).

The beltline of the reactor vessel is defined in Appendix G, 10 CFR Part 50, as the region of the reactor vessel that directly surrounds the effective height of the active core and the adjacent regions of the reactor vessel that are predicted to experience sufficient neutron damage to be considered in the selection of the limiting material with regard to radiation damage. The NRC staff considered materials with a projected neutron fluence of greater than 1.0E17 neutrons per square centimeter (n/cm^2) at end of license (EOL) to experience sufficient neutron damage to be included in the beltline. This neutron fluence is based on the surveillance requirements in Appendix H, 10 CFR Part 50, and Figure 1 in Regulatory Guide 1.99, Revision 2.

The inside surface (shell courses, heads, nozzles, and other openings) was clad with corrosion-resistant material (e.g., austenitic stainless steel) to prevent general corrosion of the materials by reactor coolant. The RPVs underwent various nondestructive examinations during the fabrication process to ensure structural integrity. Whitman et al. (1967) and Griesbach and Server (1993) describe fabrication methods in detail; the Electric Power Research Institute (1993) gives additional references.

Table 2.1 General Reactor Vessel Dimensions

	Boiling water reactors	Pressurized water reactors		
	General Electric	Westinghouse	Combustion Engineering	Babcock and Wilcox
Inside diameter, cm	470–645	277–439	356–462	434
(in.)	(185–254)	(109–173)	(140–182)	(171)
Beltline thickness, cm	11.35–18.11	16.51–27.00	18.64–23.11	21.44–21.74
(in.)	(4.47–7.13)	(6.5–10.63)	(7.34–9.10)	(8.44–8.56)
Cladding thickness, cm	0.33	0.28–0.41	0.33–0.84	0.48
(in.)	(0.13)	(0.11–0.16)	(0.13–0.33)	(0.19)

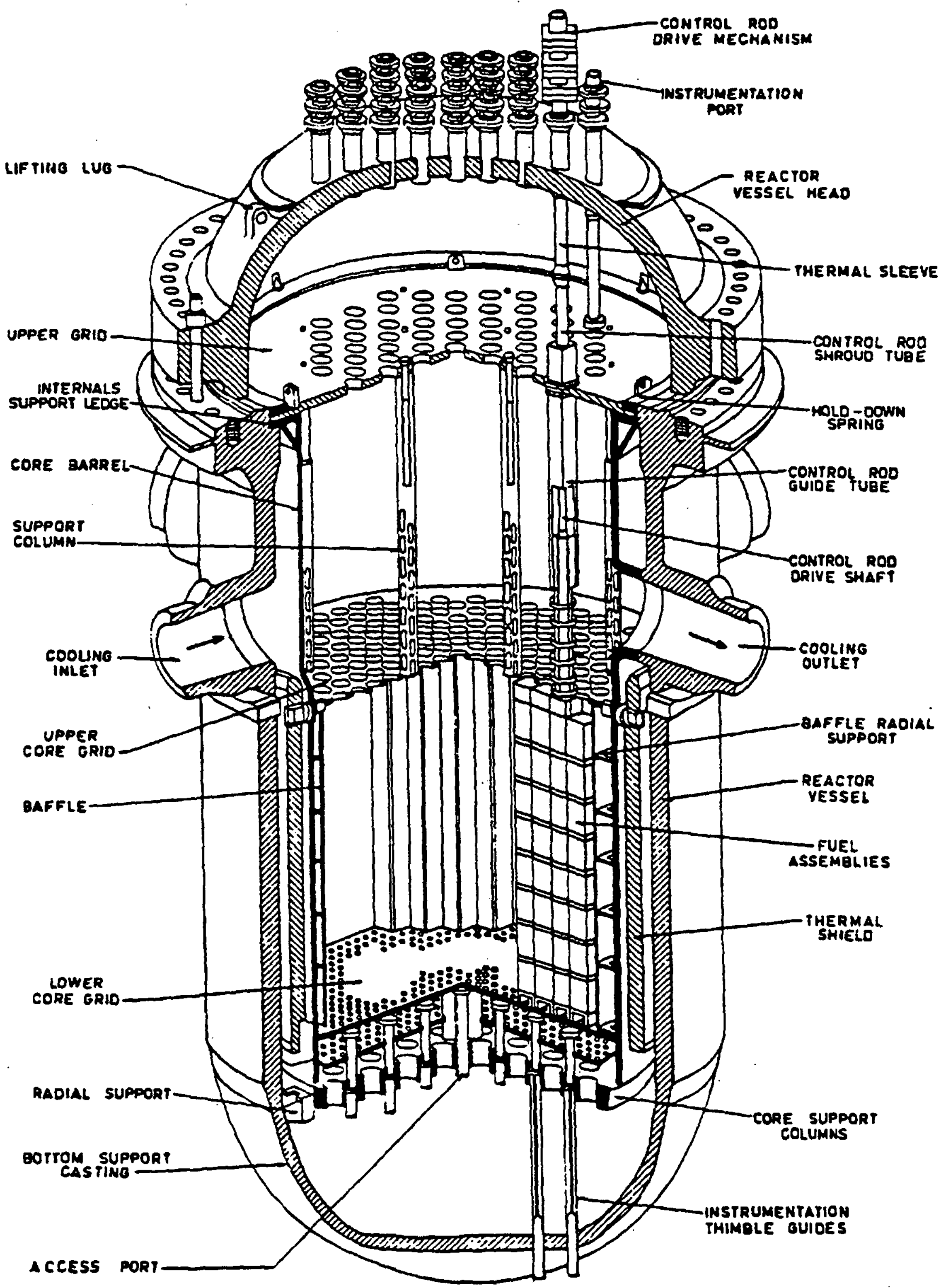


Figure 2.1 General Arrangement of a Presurized Water Reactor Vessel and Internal Components

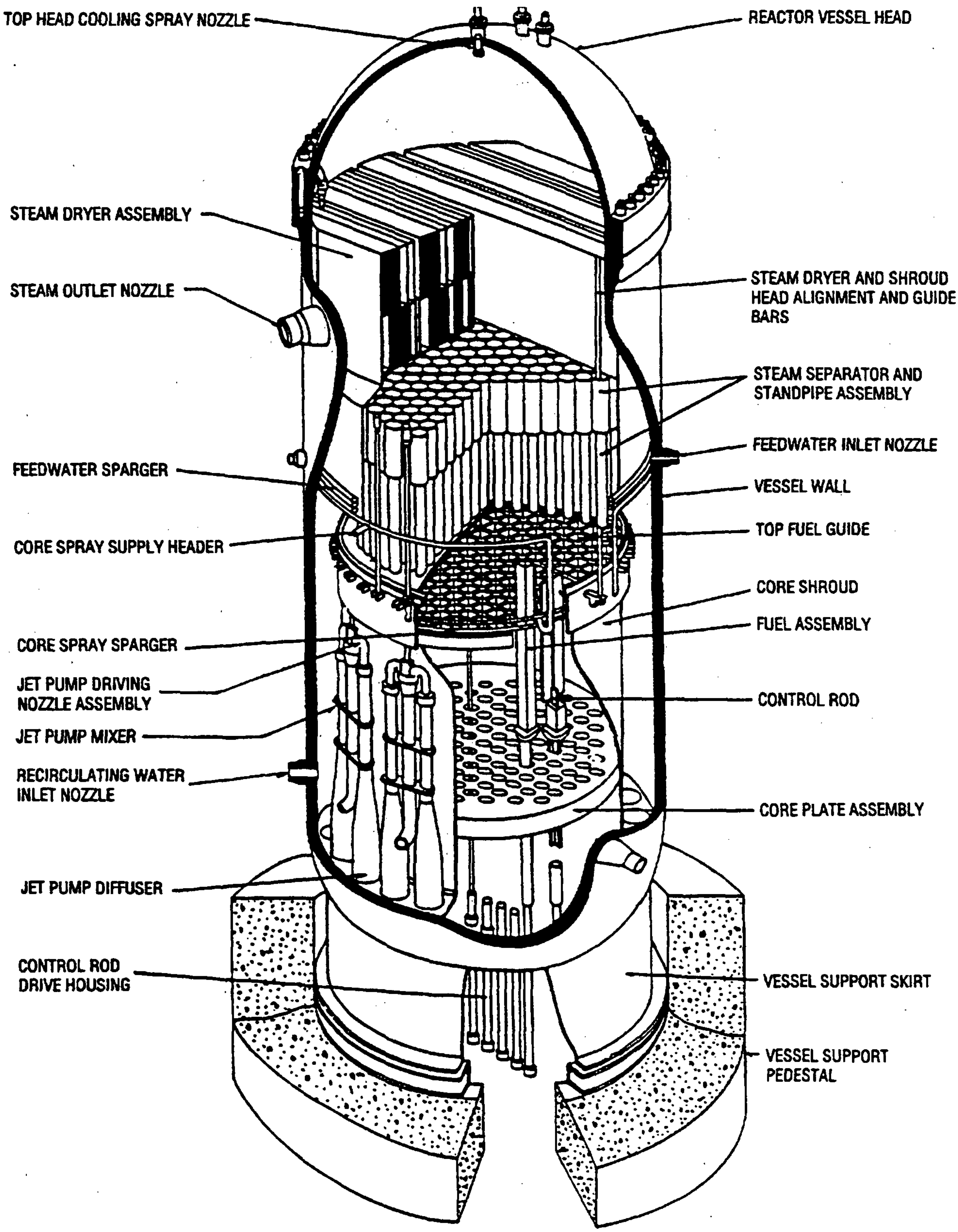


Figure 2.2 General Arrangement of a Boiling Water Reactor Vessel and Internal Components

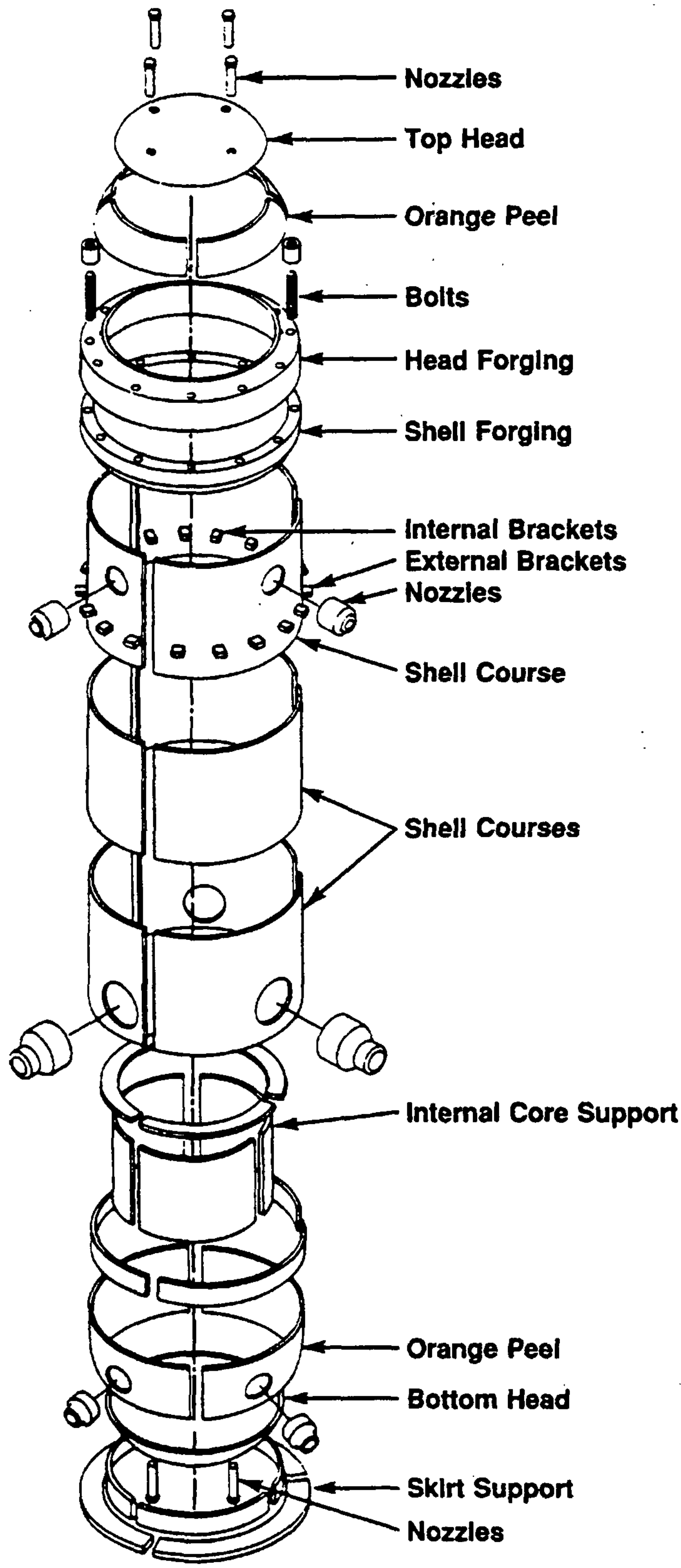


Figure 2.3 Exploded Schematic View of a Typical Boiling Water Reactor Pressure Vessel

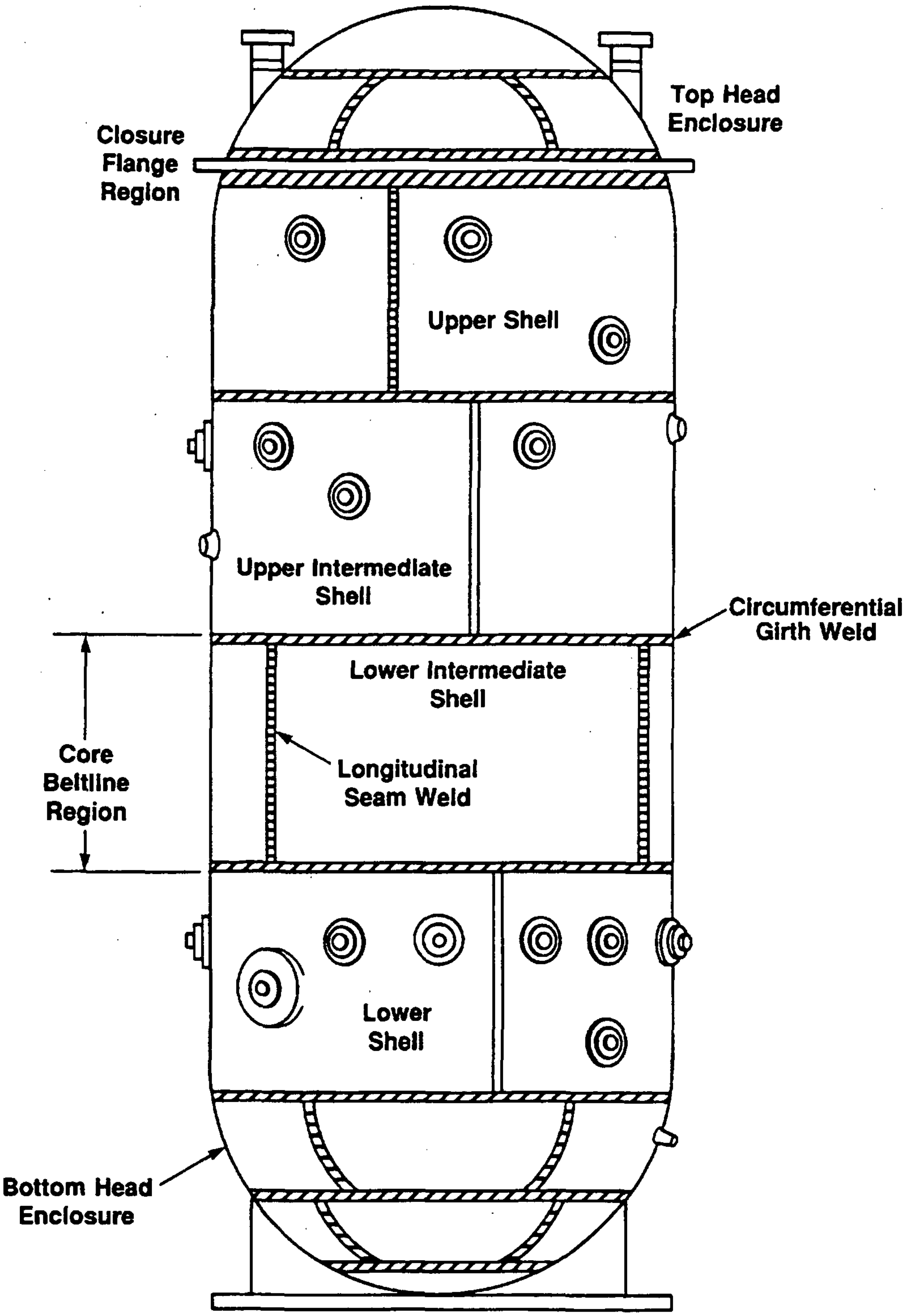


Figure 2.4 Typical Arrangement of Reactor Vessel Plates and Welds

3 RADIATION EMBRITTLEMENT OF REACTOR PRESSURE VESSEL MATERIALS

3.1 Background

Radiation embrittlement of RPV materials is a concern because it causes a reduction in the capability of the material to withstand the presence of cracks and a reduction in the fracture toughness of the material. The fracture toughness of metals increases in an essentially exponential manner with increasing temperature. Radiation embrittlement effectively shifts the fracture toughness trend curve, determined using unirradiated materials, to higher temperatures.

In the early 1970s, the Pressure Vessel Research Committee (PVRC) convened an ad hoc committee to study RPV fracture and to recommend analysis methods for ensuring the structural integrity of pressure vessels, piping, and other materials. The results of the committee's efforts were documented in Welding Research Council (WRC) Bulletin 175 (PVRC Ad Hoc Group on Toughness Requirements, 1972). The committee recommendations were based on linear elastic fracture mechanics (LEFM) concepts, and incorporated a relatively large reference crack size, a conservative estimate of fracture toughness, and margins of safety. The basic approach is to compare a computed crack driving force, called the stress intensity factor, with the material's fracture toughness. If the stress intensity factor exceeds the fracture toughness, the crack will extend unstably, resulting in "failure" of the structure. The analysis approach recommended in WRC Bulletin 175 was designed to protect against this kind of failure. The structural integrity concepts recommended in WRC Bulletin 175 were subsequently incorporated into Appendix G of Section III (and eventually Appendix G of Section XI) of the ASME Code, and by reference into 10 CFR Part 50, Appendix G.

The LEFM technology endorsed in WRC Bulletin 175 assumes linear elastic behavior in the component being analyzed, but recognizes that there will be plastic deformation in the vicinity of the crack tip. However, the size of the plastically deformed region, called the plastic zone size, is limited in comparison to the crack size and other structural dimensions. This effectively limits the applicability of LEFM to those situations in which fracture toughness is relatively low, because to attain high values of fracture toughness necessarily results in large plastic zone sizes, thus violating the LEFM assumptions. Limiting the applicability of LEFM to low values of fracture toughness also generally limits its applicability to relatively low temperatures, since fracture toughness and temperature are related. This limitation restricts the use of LEFM to situations in which the vessel wall temperature is less than the normal operating temperature, for example, startup and shutdown conditions and certain accident conditions. Appendix G of 10 CFR Part 50 requires that the methodology in Appendix G of Section III of the ASME Code be used to establish pressure-temperature (P-T) limits for heatup, cooldown, and hydrostatic tests of the RPV, before core critical operation.

Appendix G of 10 CFR Part 50 also includes a minimum requirement on the Charpy upper-shelf energy (USE) of the beltline materials to protect against low energy ductile failure of the pressure vessel. However, LEFM is not applicable for "upper-shelf" conditions because the fracture toughness is too high and the basic tenets of the technology are violated. In the 1980s this problem was overcome with the advent of elastic-plastic fracture mechanics (EPFM). Although the basic concept is the same—a computed crack driving force is compared to the material's capability to tolerate the presence of cracks—the computational and material testing methods are more complicated than for LEFM.

After several years of review, the criteria for the EPFM-based methodology for evaluating low USE materials were published in ASME Code, Section XI, Code Case N-512 and were subsequently incorporated into Appendix K of the 1993 Addenda to Section XI of the ASME Code. Further details, including the resolution of Generic Safety Issue A-11 on low upper-shelf toughness, are given in Section 4.3.3. The *Code of Federal Regulations* is currently being revised to incorporate the 1993 Addenda to the Code.

The NRC and its predecessor, the Atomic Energy Commission (AEC), in conjunction with the National Laboratories and industry subcontractors, have conducted research addressing pressure vessel integrity since 1967. The work resulted in the development of much of the linear elastic and elastic-plastic fracture mechanics technologies used in evaluating RPVs and significant experimental validation of the fracture analysis methods. The NRC research was performed as part of the Heavy Section Steel Technology (HSST) program. A historical summary report of the HSST program is contained in NUREG/CR-4489.

3.2 Measures of Radiation Embrittlement

Neutron radiation from the reactor core causes embrittlement of the RPV beltline materials. Embrittlement refers to a decrease in the fracture toughness of reactor vessel materials. Embrittlement may be measured from Charpy impact tests or more sophisticated fracture toughness tests. Section XI, Appendix G of the ASME Code defines a relationship between Charpy impact energy and fracture toughness.

In a Charpy impact test, a notched test specimen (1 x 1 x 5.5 cm or 0.394 x 0.394 x 2.17 inches) is struck by a hammer, which impacts each test specimen with the same energy. The hammer fractures the test specimen. The amount of energy absorbed by the test specimen is recorded and the fracture surface appearance is reported. The Charpy impact test is performed at different temperatures to determine the amount of energy absorbed and the fracture surface appearance as a function of temperature. Charpy impact testing of RPV materials at different temperatures results in curves (Fig. 3.1) that are characterized by three regions: (1) the lower-shelf region, (2) the upper-shelf region, and (3) the transition region, which is between the upper-shelf and lower-shelf regions. At temperatures corresponding to the upper-shelf region, RPV materials behave in a ductile manner; at temperatures corresponding to the lower-shelf region, the RPV materials behave in a brittle manner; at temperatures corresponding to the transition region, the behavior changes from brittle to ductile with increasing temperature.

Material toughness can be defined as the material's capability to carry load or deform plastically in the presence of a notch (Rolfe and Barsom, 1977). Fracture toughness is a material property that defines a component's capability to carry load in the presence of a crack. When the fracture toughness is exceeded, an existing crack is predicted to propagate. As a result, fracture toughness is often described as a material's fracture resistance to crack propagation. In LEFM, the material's fracture toughness depends on the rate of loading and the amount of constraint. For reactor vessel materials, the plane strain fracture toughness, K_{IC}, corresponds to a material's fracture toughness under conditions of plane strain and slow loading (static). The dynamic fracture toughness, K_{Id}, corresponds to a material's fracture toughness under conditions of plane strain and rapid loading. The arrest fracture toughness, K_{Ia}, corresponds to a material's fracture toughness under conditions where a rapidly propagating fracture is arrested. As the rate of loading increases, the dynamic fracture toughness decreases and approaches the arrest fracture toughness. The reference fracture toughness, K_{IR}, is the lower bound of all static, dynamic, and arrest fracture toughness data on RPV materials. K_{IR} is used in the Appendix G, 10 CFR Part 50, P–T limit analysis. Tests of reactor vessel materials indicate that as the metal temperature rises its fracture toughness increases. The NRC and the nuclear industry have performed extensive tests of RPV materials to determine the relationship between the fracture toughness properties K_{IC}, K_{Id}, K_{Ia}, K_{IR} and the reference nil-ductility temperature (the unirradiated reference temperature). The unirradiated reference temperature is defined in Section III of the ASME Code and is determined from Charpy impact and drop weight tests of unirradiated reactor vessel material.

NRC regulations (10 CFR Part 50, Appendix H) require that the Charpy impact test specimens be placed in capsules in vessels and periodically removed to monitor radiation embrittlement. The linear elastic fracture toughness properties of RPV materials must be determined on large test specimens because of the need to have plane strain conditions for the specimen. Therefore, these test specimens cannot be placed in commercial reactors. However, they have been placed in test reactors. The test results from test reactors have been used to determine the relationship between irradiated Charpy test results and linear elastic fracture toughness. Hence, the Charpy impact properties determined from specimens irradiated in commercial reactors can be used to infer the linear elastic fracture toughness of the material. Use of EPFM enables valid fracture toughness data to be obtained from smaller specimens (1.27–2.54 cm or 0.5–1.0 inch thick). Some power reactor surveillance programs make use of specimens to monitor radiation effects on USE.

The embrittlement caused by neutron irradiation results in an increase in transition temperature [Fig. 3.2(a)] and a drop in Charpy USE [Fig. 3.3(a)]. The NRC, as part of its research effort, has performed tests to demonstrate that the increase in the 41 joule (30 ft-lb) transition temperature corresponds to an equivalent decrease in the material's linear elastic fracture toughness [Fig. 3.2(b)] and that as the Charpy USE decreases, the EPFM fracture resistance (J-integral) decreases [Fig. 3.3(b)].

The neutron fluence used to determine the increase in transition temperature or the decrease in the USE is the time-integrated number of neutrons with energy greater than 1 million electron volts (1 MeV). Although neutrons with energies less than 1 MeV also contribute to radiation embrittlement, the NRC recommends this indexing procedure because the surveillance data that were used to determine the embrittling effect of neutron radiation were correlated to neutron energies greater than 1 MeV. The surveillance data were determined from material irradiated in capsules that are at or near the inside surface of U.S. commercial reactor vessels and have neutron spectrums equivalent to the vessel's inside surface. Thus, the embrittlement of the surveillance material is equivalent to that of the inside surface and includes the effect of neutrons with energies less than 1 MeV. The embrittlement correlations are valid for U.S. commercial reactors because they all have similar neutron spectrums at the vessel's inside surface.

3.3 Regulations and Guidelines

The NRC has issued regulations and guidelines on radiation embrittlement of reactor vessel materials to ensure that the integrity of the reactor vessel is maintained until the licenses of nuclear facilities expire. General Design Criterion (GDC) 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Licensing of Production and Utilization Facilities," requires, in part, that the reactor coolant pressure boundary be designed with sufficient margin to ensure that, when stressed under operating, maintenance, testing, and postulated accident conditions, (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized.

The reactor vessel, which is part of the reactor coolant pressure boundary, requires special consideration because its fracture is not considered in designing systems and components for safe shutdown of the nuclear facility. The regulatory requirements that the reactor vessel must meet to satisfy GDC 31 are contained in 10 CFR 50.60, 10 CFR 50.61, and Appendices G and H of 10 CFR Part 50.

The regulations in 10 CFR 50.60, "Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation," require licensees to meet the fracture toughness requirements of Appendix G of 10 CFR Part 50 and the material surveillance program requirements in Appendix H of 10 CFR Part 50. Appendix G, "Fracture Toughness Requirements," of 10 CFR Part 50 requires that reactor vessel materials meet the fracture toughness requirements of the ASME Code and that the reactor vessel beltline materials have an unirradiated Charpy USE no less than 102 joules (75 ft-lb) and maintain the USE throughout the life of the vessel of no less than 68 joules (50 ft-lb), unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that lower values of USE will provide margins of safety against fracture equivalent to those required by Appendix G of the ASME Code. The ASME Code requirements for fracture toughness are contained in Appendices G and K of Sections III and XI of the ASME Code. The Charpy USE is defined in American Society for Testing and Materials (ASTM) E 185–79 and –82, which are incorporated by reference in Appendix H of 10 CFR Part 50. Appendix H, "Reactor Vessel Material Surveillance Program Requirements," also requires licensees to establish a material surveillance program to monitor the effect of neutron irradiation and the thermal environment on the fracture toughness of the reactor vessel beltline materials.

The ASME Code fracture toughness requirements for determining the unirradiated reference temperature were initially defined in the Summer 1972 Addenda to the 1971 Edition. Hence, sufficient test data are generally not available for reactor vessels fabricated to ASME Code addenda and editions earlier than the Summer 1972 Addenda to determine the unirradiated reference temperature in accordance with the ASME Code. In Branch Technical Position MTEB 5–2, "Fracture Toughness Requirements," in Standard Review Plan Section 5.3.2 (NUREG–0800), the NRC has provided criteria for determining the unirradiated reference temperature from results of tests performed to ASME Code addenda and editions earlier than the Summer 1972 Addenda.

The increase in the 41 joule (30 ft-lb) transition temperature and the decrease in USE in irradiated reactor vessel materials may be determined by following the methodologies documented in Regulatory Guide (RG) 1.99, Revision 2. In this RG, the NRC recommends two methods for determining the increase in transition temperature and the drop in USE. In the first method, the amount of increase in transition temperature depends on the amount of neutron fluence and the amounts of copper and nickel in the reactor vessel material, and the decrease in USE depends on the amount of neutron fluence and the amount of copper in the reactor vessel materials. The increase in transition temperature was empirically derived from surveillance data on material irradiated in capsules in U.S. commercially operated nuclear reactor vessels. The decrease in USE was determined by bounding the data from test reactors. The second method of determining the increase in transition temperature and the decrease in USE that is documented in RG 1.99, Revision 2, is based on the test results from credible surveillance data. RG 1.99, Revision 2, defines the adjusted reference temperature (ART) as the sum of (1) the unirradiated reference temperature, initial RT_{NDT}; (2) the increase in the 41 joule (30 ft-lb) transition temperature resulting from neutron irradiation; and (3) margin to account for uncertainty in the unirradiated reference temperature and the increase in transition temperature. The ART is used in regulatory screening criteria such as the pressurized thermal shock (PTS) screening criteria and in calculating the material fracture toughness used in numerous RPV integrity calculations such as that of P-T limits. The ART is related to the RT_{PTS} value in the PTS rule, 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events." This rule establishes screening criteria that ensure high reliability of the reactor vessel during postulated PTS events. It is discussed in greater detail in Section 4.2. The RT_{PTS} value is the ART at the projected end-of-license fluence at the inside surface of the reactor vessel. The RT_{PTS} value is used to determine whether the reactor vessel is above or below the PTS screening criteria in 10 CFR 50.61.

Additional guidance on RPV integrity is given in RG 1.154 and draft RG DG-1023. RG 1.154 describes analysis procedures to assess the probability of through-wall cracks as a result of postulated PTS events for proposed operation of a plant with reactor vessels above the screening criteria in 10 CFR 50.61. Draft RG DG-1023 recommends a method for demonstrating that reactor vessels with Charpy USE less than 68 joules (50 ft-lb) have margins of safety against fracture equivalent to those required by Appendix G of the ASME Code. This guide contains formulae for determining a reactor vessel material's J-integral fracture resistance. The J-integral fracture resistance may be inferred using Charpy models, copper-fluence models, or pre-irradiation models, which were determined by statistical analyses of generic commercial and test reactor data. The relationship between J-integral fracture resistance and the models is documented in NUREG/CR-5729.

3.4 Neutron Fluence Calculation

The rate at which neutrons above 1 MeV (of all directions) reach a specific location of the pressure vessel is measured by the neutron flux $F(x,t)$ in neutrons per square centimeter second (n/cm^2 sec). The integral of the flux over time is the neutron fluence. The energy distribution of the flux at a specified location and time is the neutron spectrum. The product of the neutron spectrum and a cross section determines the particular reaction rate.

Estimating the fast neutron flux (fluence) to the pressure vessel for energies greater than 1 MeV has proven to be very difficult and time consuming. Some of the reasons for this difficulty are (1) the large attenuation of the flux from the edge of the core to the pressure vessel—from 4 to 6 orders of magnitude; (2) the difficulty in estimating the correct neutron spectrums; (3) the importance of using the exact geometrical dimensions, which may not always be accurately known; and (4) cross-section uncertainties.

In the United States, the most frequently used computational tools for estimating $F(x,t)$ are computer codes based on finite discrete ordinates techniques. DOT and TORT are two such codes for two-dimensional (2-D) and three-dimensional (3-D) geometries, respectively. Experience has shown that combinations of 2-D fluxes in (r,θ) and (r,z) geometries are adequate for representing 3-D fluxes in most practical situations. There are, however, instances, such as partial-length fuel assemblies or partially shielded assemblies, which are best treated with 3-D calculations using codes such as TORT. Both 2-D and 3-D calculations are simplified by taking advantage of the eightfold reactor core symmetry.

Rules have been developed for the use of radial and azimuthal mesh spacing for an accurate mathematical representation of the details of a problem. The minimum geometrical quadrature approximation and the scattering cross-section representation that yield acceptable results are designated as S_8 and P_3 respectively. (S_8 represents the division of the solid angle of scattering into eight segments, and P_3 represents a third-degree Legendre polynomial representation of the scattering angular distribution.) The most effective neutron source is from the two outer peripheral rows of assemblies, where there is a large radial power (neutron source) gradient. Therefore, a pin-by-pin representation of the power distribution in the outer rows of assemblies is necessary. Similarly, the plutonium buildup in the outer rows of assemblies must be accurately represented, because plutonium fission results in a higher number of neutrons per fission at higher energies.

Regardless of the approximation, convergence criteria, and mesh spacing, experimental benchmarking is necessary to ensure codes yield reliable flux (fluence) values at the pressure vessel. The NRC has funded two experiments at Oak Ridge National Laboratory (ORNL) simulating fast neutron flux transmission from the core to the pressure vessel. The results of these experiments have been widely used to benchmark fluence calculations. A properly benchmarked code is expected to yield an accuracy of ± 20 percent (1σ). Flux (fluence) to the reactor cavity can also be calculated using the same techniques as described above. Although the NRC has used only finite difference techniques for RPV work, Monte Carlo-based methods also can be used to perform similar calculations to the same accuracy.

Whatever method is used in calculating the neutron flux to the pressure vessel, the accuracy must be shown to be at least ± 20 percent (1σ) because this value has been included in the estimation of the margin term in 10 CFR 50.61. Thus, the ± 20 percent in the fluence is required to maintain the validity of the PTS rule.

The NRC issued a draft RG (DG-1025), outlining input approximations and code benchmarking, for public comment in October 1993.

3.5 Fluence Reduction Plan

There have been two different reasons for reducing fluence in U.S. RPVs. The first is fuel cycle economics and is related to increased fuel cycle length, which is achieved by increased enrichment and reduced neutron leakage. The other is to reduce the irradiation rate of the pressure vessel in general or specific portions such as longitudinal or circumferential welds. The means for reducing the neutron flux that reaches a specific point of the pressure vessel is to reduce the neutron source in the outer assemblies that are close to the designated point of the pressure vessel.

The original (1960s and 70s) core loading schemes, the out-in arrangements, called for loading fresh fuel in the outer core periphery, moving the fuel from these locations further into the core, and removing the fuel from the innermost locations that already had a three-cycle exposure. This method has the advantage of producing uniform power distributions, thus maintaining low power peaking factors at full-power operation. However, it has the disadvantage of high neutron leakage caused by relatively high powers at the peripheral assemblies, resulting in poor neutron utilization. As fuel enrichment increased and fuel fabrication techniques improved, the trend toward longer fuel cycles continued. Increased fuel enrichment, fuel poisons, and low-leakage arrangements are used to increase fuel cycle lifetime and fuel utilization (megawattdays per metric ton). These low neutron leakage arrangements are designated in-out or in-out-out, indicating the use of once or twice irradiated assemblies in the core periphery.

The other type of low-leakage core is designed specifically to lower exposure of the entire RPV beltline region or of specific portions of it. In instances where the base metal is the critical element, the entire core leakage and the peak flux location in particular are lowered. This can be achieved by using twice- or thrice-burned assemblies in the periphery, as described above. Fluence reduction factors of 2 to 3 can be achieved using this method. If a longitudinal weld is the critical element, it can be protected by lowering power in the peripheral assemblies directly opposite the weld. This requires using lower power assemblies in these locations. For greater reductions, the outer fuel pins in the partially shielded assemblies can be replaced with steel pins; for most drastic reductions, whole assemblies can be replaced with assemblies consisting of steel pins in place of fuel pins. In this manner reduction factors up to 6 or 7 can be achieved. If a circumferential weld is the critical element, some of the circumferential assemblies need to be shielded for a length of 91.4 to 121.9 cm (3 to 4 feet) at a height corresponding to the height of the weld. This can be accomplished in a similar manner by creating partially shielded assemblies or using hafnium or other suitable poisons to suppress power production in the desired location. Flux reductions of up to a factor of 9 or 10 at the peak azimuthal location can be accomplished in this manner. Lowering fluence to specific longitudinal or circumferential welds requires location-specific assemblies.

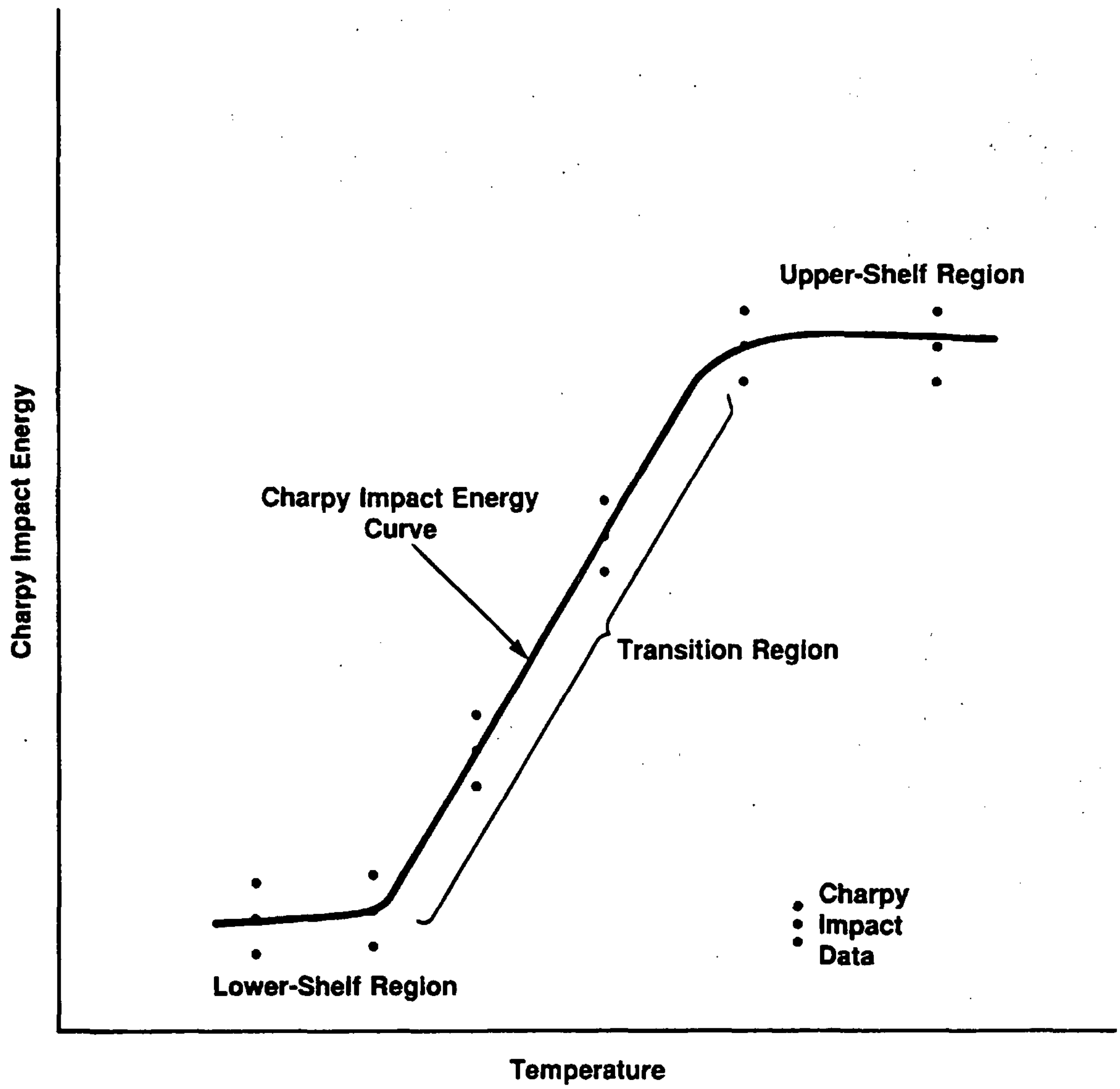
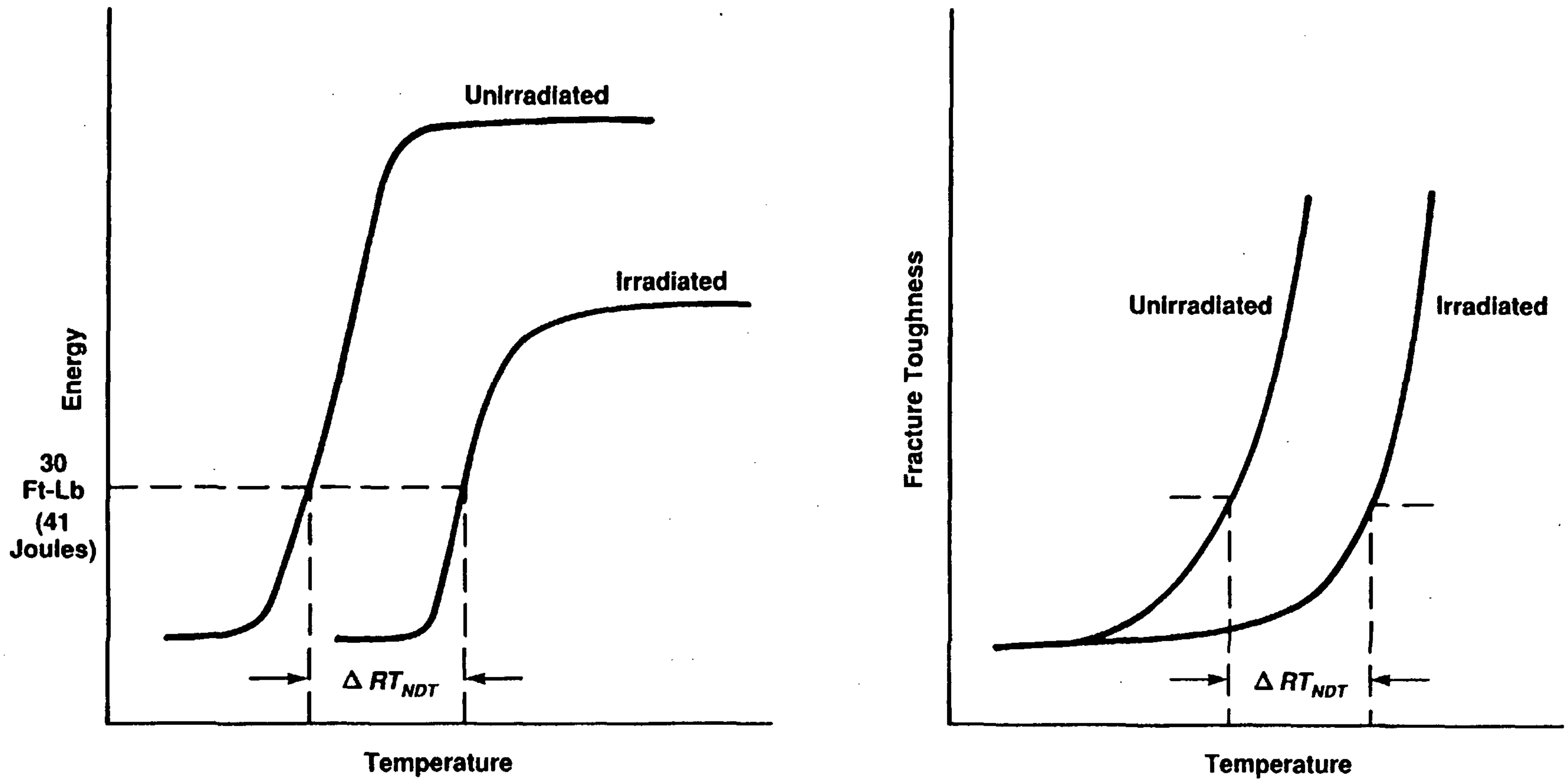


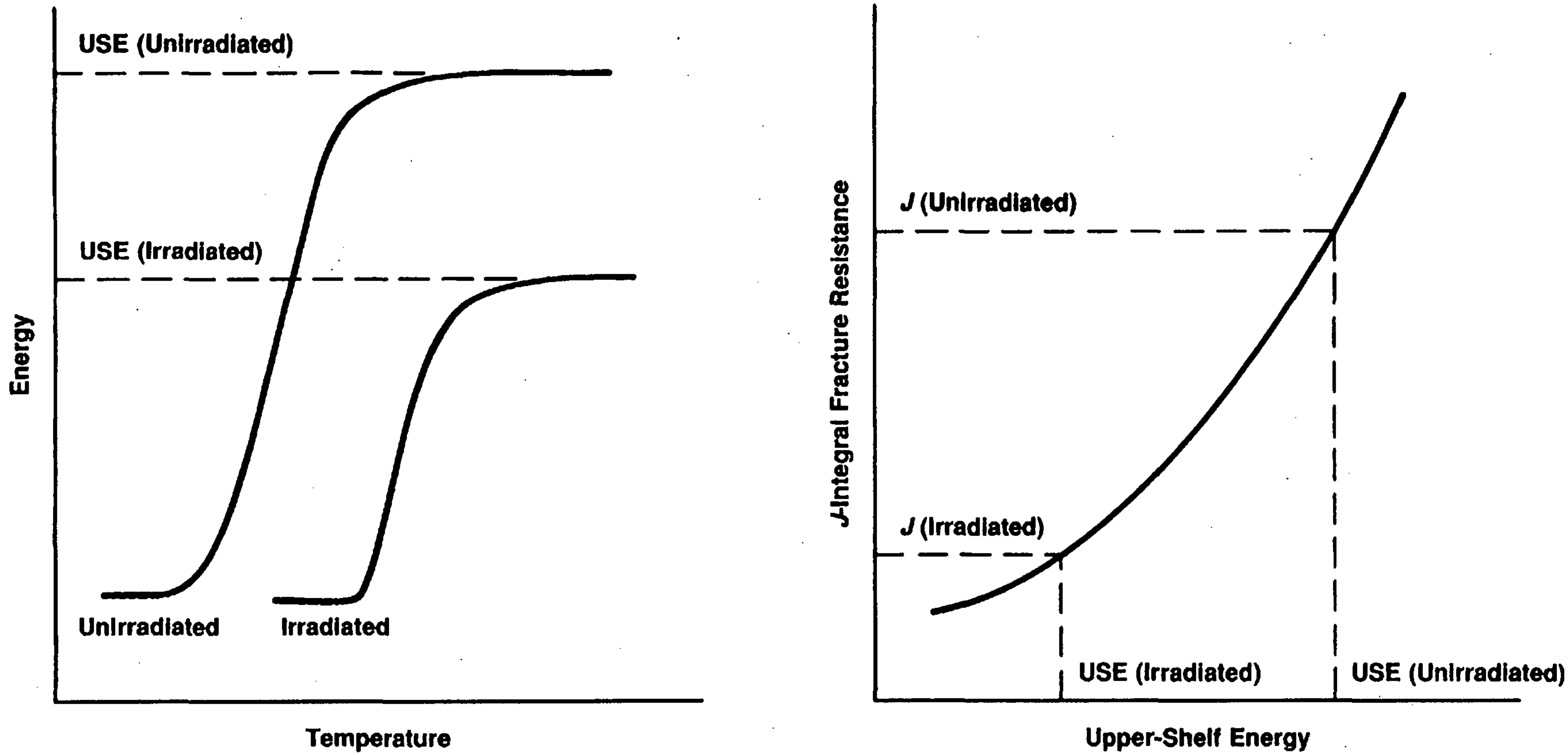
Figure 3.1 Effect of Temperature on Impact Properties



(a) Transition Energy/Temperature

(b) Fracture Toughness/Temperature

Figure 3.2(a) and (b) Effect of Neutron Radiation on Transition Region (41 Joules or 30 Ft-Lb Energy) and Fracture Toughness Properties



(a) Upper-Shelf Energy/Temperature

(b) J-Integral Fracture Resistance/ Upper Shelf Energy

Figure 3.3(a) and (b) Effect of Neutron Irradiation on Upper–Shelf Energy and J–Integral Fracture Resistance

4 REACTOR PRESSURE VESSEL INTEGRITY ISSUES AND REQUIREMENTS

4.1 Industry Surveillance Programs (Appendix H)

4.1.1 Background

Appendix H of 10 CFR Part 50 requires licensees to maintain material surveillance programs. The purpose of these programs is to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline regions that result from the exposure of these materials to neutron irradiation and to the thermal environment. Under the program, surveillance capsules are located in each vessel between the core and the RPV in close proximity to the beltline region. The surveillance capsules contain material specimens that are consequently exposed to the neutron irradiation and thermal environment of the beltline region to simulate the exposure conditions of the actual vessel beltline materials. The types of material specimens contained in the surveillance capsules include Charpy specimens, tensile specimens, and, in a few cases, compact fracture toughness specimens.

The surveillance capsules are withdrawn periodically and tested in accordance with the approved withdrawal schedule and the requirements of Appendix H of 10 CFR Part 50. The resultant data are analyzed, in addition to prior results from the surveillance program, to determine the adequacy of the fracture toughness of the RPV materials as required by Appendix G of 10 CFR Part 50, Sections IV ("Fracture Toughness Requirements") and V ("Inservice Requirements—Reactor Vessel Beltline Material"), and 10 CFR 50.61, the PTS rule.

4.1.2 Surveillance Program Criteria

Appendix H of 10 CFR Part 50 states that material surveillance programs are not required when it can be conservatively demonstrated that the peak neutron fluence at the end of the design life of the vessel will not exceed 1.0E17 n/cm^2 (E > 1 MeV). Licensees of reactor vessels with end-of-design-life fluences that exceed this value must monitor their beltline materials through a surveillance program.

Surveillance programs are designed to meet the requirements of ASTM E 185–82, which covers all aspects of the design of the surveillance program, including the following:

(1) selection of the materials to be included in the surveillance capsules

(2) type, number, and orientation of the test specimens

(3) specimen encapsulation requirements

(4) number and location of surveillance capsules

(5) neutron dosimetry

(6) temperature monitoring

(7) surveillance capsule withdrawal schedule

Requirements for surveillance programs implemented before the first capsule is withdrawn must meet the requirements of the edition of ASTM E 185 that is current on the issue date of the ASME Code to which the reactor vessel was purchased. For each capsule withdrawal before July 26, 1983, either the 1973, the 1979, or the 1982 Edition of ASTM E 185 may be used. Capsule withdrawals after July 26, 1983, must meet the test procedures and reporting requirements of the 1982 Edition of ASTM E 185 to the *extent practicable for the configuration of the specimens* in the capsule.

Material surveillance programs that were designed to meet either the 1973, the 1979, or the 1982 Edition of ASTM E 185 were required to include one heat of base metal (plate or forging), weld metal, and heat-affected-zone material that were predicted to be most limiting for operation of the reactor vessel. Before the 1973 Edition, ASTM E 185 was not as explicit as to the selection of material to be placed in the surveillance capsules. The statistical analyses of surveillance data conducted by the NRC in the 1980s showed that copper and nickel caused an increase in the transition temperature. Before these analyses, it was believed that copper and phosphorous were the major contributors. Because the earlier standards did not contain explicit directions and knowledge about the chemical elements that cause the increase in transition temperature was lacking, many surveillance programs do not specify the materials that will limit operation of the reactor vessel. Because of this deficiency, many licensees needed to use generic data and perform equivalent margins analyses to document compliance with the fracture toughness requirements of Appendix G, 10 CFR Part 50.

The surveillance capsules must be located near the inside vessel wall in the beltline region so that the material specimens duplicate, to the greatest degree possible, the neutron spectrum, temperature history, and maximum neutron fluence as experienced at the reactor vessel's inner surface. The surveillance capsules are designed and located to permit insertion of replacement capsules. Accelerated-irradiation capsules may be used in addition to the required number of surveillance capsules specified in ASTM E 185.

An integrated surveillance program may be accepted for a set of reactors that have similar designs and operating features. An acceptable integrated surveillance program would include material specimens that are representative of the beltline materials from each reactor in an integrated program. These specimens may then be irradiated in one or more of the reactors in the integrated program. There must, however, be an adequate dosimetry program for each reactor. Integrated surveillance programs do not permit reduction in the requirements for the number of materials to be irradiated, types of specimens, or number of specimens per reactor; however, the amount of testing may be reduced if the initial results agree with predictions.

Integrated surveillance programs must be approved, on a case-by-case basis, by the Director of the Office of Nuclear Reactor Regulation. Criteria for approval are the following:

(1) The design and operating features of the reactors in the set must be sufficiently similar to permit accurate comparisons of the predicted amount of radiation damage as a function of the total power output.

(2) There must be an adequate arrangement for data sharing among the set of plants.

(3) There must be a contingency plan to ensure that the surveillance plan for each reactor will not be jeopardized by operation at reduced power level or by an extended outage of another reactor from which data are expected.

(4) Implementation of an integrated surveillance program must provide substantial advantages, such as reduced power outages or reduced personnel exposure to radiation, as a direct result of not requiring surveillance capsules in all reactors in the set.

4.1.3 Report of Test Results

Test results from the withdrawal of each capsule must be reported in a summary technical report, which must be submitted within 1 year unless the Director of the Office of Nuclear Reactor Regulation grants an exemption. The report must include the data as required by ASTM E 185 and the results of all fracture toughness tests conducted on the surveillance material specimens under irradiated and unirradiated conditions.

4.2 Pressurized Thermal Shock

4.2.1 Description of Issue and Background

Unanticipated transients or design-basis postulated accidents in PWRs could result in a rapid and significant decrease in the reactor coolant temperature, concurrent with or followed by repressurization. These events are often referred to as "overcooling or PTS events." In these PTS events, rapid cooling of the reactor vessel's internal surface results in thermal stress with a maximum tensile stress at the inside surface of the vessel. The magnitude of the thermal stress depends on the temperature profile across the reactor vessel wall as a function of time. The effects of this thermal stress are compounded by pressure stresses if the vessel is pressurized.

Severe reactor system overcooling events that could be accompanied by pressurization or repressurization of the reactor vessel can result from a variety of causes. These include system transients, some of which could be initiated by instrumentation and control system malfunctions, including stuck-open valves in either the primary or secondary systems, and postulated accidents such as small-break loss-of-coolant accidents, main steam line breaks, and feedwater line breaks. Eight actual overcooling events, during which the primary coolant system water temperature rapidly decreased by 111 °C (200 °F) or more, were identified in NRC Commission Paper SECY 82–465.

As long as the fracture toughness of the reactor vessel material is relatively high, such events will not threaten RPV integrity. However, as discussed in Section 3 of this report, the fracture toughness of reactor vessel materials decreases with exposure to fast neutrons during the life of a nuclear power plant. If the fracture toughness of the vessel material has been reduced sufficiently, severe PTS events could cause propagation of small flaws that might exist near the inner surface of the vessel. The assumed initial flaw might propagate into a crack through the vessel wall of sufficient extent to threaten vessel integrity and, therefore, core cooling capability.

The PTS issue is evaluated only for PWRs. BWRs are not subject to PTS concerns because they operate with a large portion of water inventory inside the pressure vessel at saturated steam conditions. Since any sudden cooling will condense steam and result in a pressure decrease, simultaneous creation of high pressure and low temperature is improbable. Also contributing to the lack of PTS concerns for BWRs is the lower fluence at the vessel's inner wall, and the use of a thinner vessel wall, resulting in a lower thermal stress intensity for postulated cracks.

As a result of its evaluation of the PTS issue in the spring of 1981, the NRC staff concluded that no immediate licensing actions were required, but since the consequences of overcooling events increase as the vessels accumulate additional neutron irradiation, extensive further investigations were necessary to determine whether and when corrective action would be needed to provide assurance of vessel integrity throughout the intended service life of a reactor vessel. Subsequently, the NRC staff held meetings with licensees, reactor manufacturers, and owners groups to discuss PTS concerns and exchange technical information. The NRC staff and industry analyses are summarized in NRC Commission Paper SECY-82-465. As a result of these analyses, the NRC staff concluded that the risk from PTS events is acceptably low for (1) forgings, plates, and axial welds with RT_{PTS} values less than 132 °C (270 °F) and (2) circumferential welds with RT_{PTS} values less than 149 °C (300 °F). RT_{PTS} is a parameter related to the brittle-to-ductile transition temperature of the material. Methods for determining RT_{PTS} are discussed in the following section.

4.2.2 Applicable Requirements and Guidelines

The PTS rule, 10 CFR 50.61, was initially incorporated into the regulations on July 23, 1985, and was amended on May 15, 1991. The rule establishes screening criteria that define a limiting level of embrittlement beyond which operation cannot continue without further plant-specific evaluation. The screening criteria are given in terms of reference temperature, RT_{PTS}. The screening criteria are 132 °C (270 °F) for plates, forgings, and axial welds and 149 °C (300 °F) for circumferential welds. The RT_{PTS} is defined as:

$$RT_{PTS} = I + \Delta RT_{PTS} + M$$

where I is the unirradiated or initial reference temperature (RT_{NDT}), ΔRT_{PTS} is the increase in RT_{PTS} caused by irradiation, and M is a margin added to cover uncertainties in the initial properties, copper and nickel contents, fluence, and calculation procedures. The amount of increase in RT_{PTS} is based on the amount of neutron irradiation and the amount of copper and nickel in the material. The greater the amounts of copper, nickel, and neutron fluence, the greater the increase in RT_{PTS} for the material and the lower its fracture resistance. The PTS rule requires that the unirradiated reference temperature be determined from measurements as defined in the ASME Code, Section III, paragraph NB-2331. The amount of margin depends on whether (1) the material is a weld or a base metal, (2) the unirradiated reference temperature is a generic value or a measured value, and (3) the increase in RT_{PTS} is derived from credible surveillance material or from the chemistry factor tables in the PTS rule.

The PTS rule amendment of May 15, 1991, changed the method of calculating embrittlement to the method recommended in RG 1.99, Revision 2, when surveillance data are not available. The amended rule requires licensees to consider the effect of reactor vessel operating temperature and surveillance results on the calculated RT_{PTS} value.

If an RT_{PTS} value for any material in an RPV is projected to exceed the PTS screening criteria, the PTS rule provides licensees with alternative and required actions. First, the PTS rule requires licensees to submit an analysis and an implementation schedule for flux reduction programs that are "reasonably practicable to avoid exceeding the PTS screening criterion." If no reasonably practicable flux reduction program will prevent the RT_{PTS} value from exceeding the PTS screening criteria before an operating license expires, the PTS rule requires licensees to submit a safety analysis to determine what, if any, modifications to equipment, systems, and operations are necessary to prevent potential failure of the reactor vessel as a result of postulated PTS events. This safety analysis must be submitted at least 3 years before the value of RT_{PTS} is projected to exceed the PTS screening criteria. RG 1.154 describes an analysis procedure acceptable to the NRC staff for licensees to determine whether modifications to equipment, systems, or operations will provide adequate assurance that reactor vessels will have sufficient fracture toughness to prevent reactor vessel failure from postulated PTS events when the RT_{PTS} value exceeds the PTS screening criteria. As specified in paragraph (b)(6) of the rule, the NRC may, on a case-by-case basis, approve operation of the plant at RT_{PTS} values in excess of the PTS screening criteria.

4.2.3 Status of Pressurized Thermal Shock Reviews

As a result of information it received during its review of licensee responses to Generic Letter (GL) 92-01, Revision 1, the staff was able to evaluate the RT_{PTS} value for each PWR plant. The staff's evaluation shows that all plants will be below the PTS screening criteria at the end of their licenses (EOLs) except for Beaver Valley 1 and Palisades. A discussion of these plants and other plants whose licensees responded to the amended PTS rule follows.

The amendment to the PTS rule dated May 15, 1991, required licensees with reactor vessel beltline materials that are projected to exceed the PTS screening criteria before their operating licenses expire to submit an assessment by December 16, 1991. All other licensees are required to submit their assessments by June 14, 1996, or with the next update of their P-T limits or the next reactor vessel material surveillance report, whichever comes first.

Licensees for Palisades, Calvert Cliffs 1 and 2, Beaver Valley 1, and Fort Calhoun submitted PTS assessments on or before December 16, 1991.

The NRC staff in a letter dated July 12, 1994, issued an interim safety evaluation for Palisades. Because the limiting weld in the Palisades RPV is not included in the Palisades surveillance program, the licensee used chemical composition data from other facilities to demonstrate that its reactor vessel will be below the screening criteria in the PTS rule. The data submitted by the licensee show that the Palisades reactor vessel will exceed the PTS screening criteria in 2004. To demonstrate that the Palisades reactor vessel will be below the PTS screening criteria at EOL in 2007, the licensee is (1) gathering additional materials properties data on its retired steam generators; (2) instituting a surveillance program, which would include the limiting weld; (3) evaluating annealing of the reactor vessel; and (4) considering instituting ultra low leakage fuel strategy. The retired steam generators have shell welds fabricated using the same heat number of weld wire and flux type that were used in fabricating the RPV limiting weld. The materials properties tests on the welds from the retired steam generators are expected to be completed in late 1994. The date that the Palisades reactor vessel is projected to exceed the PTS screening criteria will be affected by the results from these tests.

The licensee for Calvert Cliffs 1 and 2 has submitted assessments that both units will be below the PTS screening criteria at the expiration of their licenses. The NRC staff in a letter dated July 29, 1994, has reviewed the assessments and agrees with their conclusions.

In its assessment of the Beaver Valley 1 reactor vessel, the licensee recommended that the standard deviation for the increase in reference temperature be reduced by 50 percent because it had used surveillance data to determine the RT_{PTS}. RG 1.99, Revision 2, permits the standard deviation for the increase in reference temperature to be reduced by 50 percent if the scatter in the surveillance data meets the credibility criteria in the RG. However, since the scatter in the Beaver Valley 1 surveillance data exceeded the amount permitted by the credibility criteria in RG 1.99, Revision 2, the NRC staff in a letter dated April 20, 1993, concluded that the standard deviation should not be reduced to determine the RT_{PTS}. Using 100 percent of the standard deviation as recommended in RG 1.99, Revision 2, the Beaver Valley 1 reactor vessel will be 3.3 °C (6 °F) above the screening criterion at 32 effective full-power years of operation and will reach the PTS screening criteria in 2012. The Beaver Valley 1 license expires on January 29, 2016. The date for Beaver Valley 1 to reach the PTS screening limit is a preliminary estimate that is based on the installation of hafnium power suppression assemblies during the next refueling outage. The licensee for Beaver Valley 1 is considering the following: (1) replacing the thermal shields with neutron pads, (2) replacing internals to add radial reflectors, (3) increasing baffle plate thickness, and (3) annealing the reactor vessel. The licensee for Beaver Valley 1 will submit a reassessment of its RT_{PTS} values for staff review when the revised fuel strategy is instituted.

The licensee for Fort Calhoun performed an assessment of the RT_{PTS} value for its reactor vessel on August 9, 2013, the date corresponding to 40 years after the date its operating license was issued. The licensee's assessment shows that the Fort Calhoun reactor vessel will be below the screening criterion on August 9, 2013. The NRC staff in a letter dated December 3, 1993, has reviewed the assessment and agrees with this conclusion.

The NRC staff has received assessments from other licensees. The most significant are the responses for Indian Point 3 and Zion 1 and 2.

In its assessment for Indian Point 3, the licensee used (1) only transversely oriented surveillance data and (2) reduced the standard deviation for the increase in reference temperature by 50 percent from the value that is reported in RG 1.99, Revision 2. The Indian Point 3 surveillance program contains transversely oriented and longitudinally oriented test data from the limiting beltline plate. RG 1.99, Revision 2, permits the licensee to reduce the standard deviation by 50 percent, if all the surveillance data are used to determine the effect of irradiation and the data meet the credibility criteria in the RG. In addition, recent analysis by the NRC of generic surveillance data on plate material similar to that used at Indian Point 3 shows that the standard deviation for plate material that is documented in RG 1.99, Revision 2, may need to be increased. This RG identifies the standard deviation for the increase in reference temperature for plate material as 9.44 °C (17 °F). The revised analysis of the surveillance data shows that a standard deviation for the increase in reference temperature for plate material of 14.44 °C (26 °F) may be more appropriate. Hence, the NRC staff in a letter dated August 12, 1993, stated that the licensee should determine the effect of irradiation on the limiting beltline plate by using all the surveillance data (both transversely and longitudinally oriented surveillance data) and should consider the effect of increasing the standard deviation for the increase in reference temperature for the limiting beltline plate. Using all of the surveillance data and a standard deviation of 14.44 °C (26 °F) results in an RT_{PTS} value that is 0.56 °C (1 °F) below the screening criterion at the expiration of the Indian Point 3 license. This conclusion could change on the basis of future surveillance data.

Charpy impact tests on welds representing the limiting weld at Zion 1 and 2 showed that the unirradiated RT_{PTS} could be much greater than that for other similar welds when the RT_{PTS} is calculated using the methodology required by the PTS rule because of a large variance in impact properties. The large variance in impact properties is due to the low Charpy USE of this weld. The licensee has proposed an alternative methodology for determining the unirradiated RT_{PTS} value. However, it needed an exemption from the PTS rule to allow the use of an alternative methodology to determine the unirradiated RT_{PTS} value. The NRC staff in a letter dated February 22, 1994, concluded that the licensee's methodology was more appropriate than that described in the PTS rule for determining the unirradiated RT_{PTS} for the limiting welds.

at Zion 1 and 2; consequently, it issued an exemption to the licensee. When the unirradiated RT_{PTS} value is calculated using the licensee's alternative methodology, the Zion reactor vessels will be below the PTS screening criteria at EOL. The alternative methodology is applicable to other vessels that were fabricated with the same heat of material as the limiting Zion welds. The NRC staff will review the unirradiated RT_{PTS} value for these welds when it is submitted and will include it in the reactor vessel integrity database.

Summary and Conclusion

The NRC staff has reviewed licensee responses to GL 92–01 and other docketed information and assessed the RT_{PTS} values for all the PWR RPVs in U.S. plants. On the basis of this review, it has concluded that the RPVs in most plants will be well below the PTS screening limits at the end of their current operating licenses. On the basis of the currently docketed information, Beaver Valley 1 and Palisades are the only plants projected to exceed their PTS screening limits when their licenses expire. Beaver Valley 1 and Palisades are projected to exceed the PTS screening limits in 2012 and 2004, respectively, before the end of their operating licenses in 2016 and 2007. It is important to note that these results are based on the information currently reported by the licensees and are subject to change. The dates when the plants are projected to reach the screening criteria may change as a result of new surveillance data and additional analyses. Also, by implementing different fuel management techniques and inserting special neutron-absorbing materials in the reactor core, licensees may be able to reduce the irradiation levels sufficiently to stay below the screening criteria. In addition, licensees may anneal the RPV to restore the material properties to near the original unirradiated condition. The industry has already informed the NRC staff that it will submit additional data and information. Therefore, it is expected that the above information will change. These changes will be reported in yearly updates of this report.

4.3 Low Upper-Shelf Toughness Safety Margin Issue

4.3.1 Description of Issue

As discussed in Section 3, steels used in the construction of nuclear RPVs typically exhibit a transition in their fracture behavior from brittle to ductile over the range of temperatures from ambient (21.11 °C or 70 °F) to operating (288 °C or 550 °F). Fracture characteristics of the steels over this temperature range have historically been linked to their performance in the Charpy V-notch (CVN) test, which is described in Section 3.2. The fracture performance of an RPV steel, determined by using the CVN test, is typically divided into three regions—(1) lower shelf, (2) transition, and (3) upper shelf—as illustrated in Figure 3.1. Upper-shelf toughness refers to a measure of the energy required to completely fracture a CVN specimen when the material is behaving in a ductile fashion.

For most RPV steels and weldments, the USE is high (>102 joules or 75 ft-lb). Fracture of the steel on the upper shelf under these conditions is typically associated with significant plastic deformation and a large amount of absorbed energy. The fracture mechanism for ductile fracture is called microvoid coalescence. This fracture process for a material with a high USE at upper-shelf temperatures is characteristically slow and stable under a slowly rising load. The USE has been shown to decrease as a result of irradiation damage with increasing time of operation as shown in Figure 3.2a. If the USE is low enough, the loading on a potential flaw in the RPV beltline may provide enough energy to cause fracture to proceed by low energy ductile tearing. With a high enough fracture driving force, the ductile tearing may proceed in an unstable fashion. It is this potential for unstable ductile fracture in RPV materials that is the primary cause of concern for low upper-shelf materials. However, the low upper-shelf safety margin issue encompasses conditions for both ductile initiation and growth.

The principal means of demonstrating adequate USE for RPV steels is through materials test programs that provide data for both the unirradiated and irradiated conditions. However, the materials test programs that were instituted when many of the commercial nuclear plants began operating (1) did not necessarily cover all material conditions (welds in particular) and (2) lacked data for specific properties (e.g., USE) that were not required by the codes and standards in force at the time. If USE data were lacking for specific plates or welds, statistical analyses of the material within the same plate or weld family have been used to demonstrate adequate upper-shelf toughness.

If data were lacking and statistical analyses were not supportable, fracture mechanics-based equivalent margins analyses, as described in the following section, have been used to facilitate resolution of the low upper-shelf safety margin issue. The industry has performed bounding equivalent margins analyses on an owners group basis to provide a more efficient use of industry and NRC staff resources. In lieu of, or in addition to, the industry analyses, licensees of some plants have submitted plant-specific analyses. The equivalent margins analyses have been used by the industry and individual licensees to demonstrate adequate safety margins for USE values below the NRC screening criterion of 68 joules (50 ft-lb) at EOL.

4.3.2 Applicable Requirements and Guidelines

The regulatory requirements for the low upper-shelf safety margin issue are contained in Appendix G of 10 CFR Part 50. Appendix G requires that the initial unirradiated USE at the start of vessel life be no less than 102 joules (75 ft-lb) and that the vessel maintain a USE level of no less than 68 joules (50 ft-lb) throughout the service life. If it is anticipated that a vessel might fall below 68 joules (50 ft-lb) before EOL, an analysis must be submitted that demonstrates "margins of safety against fracture equivalent to those required by Appendix G of the ASME Code." This analysis is subject to the approval of the Director, Office of Nuclear Reactor Regulation.

Guidelines that the NRC staff finds acceptable for conducting equivalent margins analyses are contained in ASME Code Case N-512 and Appendix K and draft RG DG-1023. DG-1023 incorporates the criteria of Code Case N-512 and contains additional guidance on material properties and transient selection. The code case and the draft RG recommend that the analysis be performed by comparing the material fracture resistance with the applied fracture driving force. These documents contain criteria for ASME Code, Section XI service levels A, B, C, and D that must be satisfied to demonstrate that the material meets the margins of safety required by Appendix G of the ASME Code. Service levels A, B, C, and D are defined as normal operating, upset, emergency, and faulted conditions, respectively. Typically, the limiting condition for levels A and B is the 55.56 °C/hr (100 °F/hr) heatup/cooldown transient. The governing transients for levels C and D are typically associated with postulated accident conditions and are either vendor or plant specific. The NRC staff has found that levels A and B are "controlling" in equivalent margins analyses.

As the majority of plants do not have fracture toughness information for their limiting vessel materials, CVN or chemical composition and fluence data are typically used to infer the fracture toughness. For the approach based on CVN data, RG 1.99, Revision 2, contains a procedure for estimating the drop in CVN USE with increasing irradiation damage. NUREG/CR-5729 contains empirically derived models for predicting material J-R curves from CVN data or chemical content and fluence. The NUREG/CR-5729 models are applicable to most RPV materials. In addition, for plates, NRC Branch Technical Position (BTP) MTEB 5-2 (NUREG-0800) contains criteria for estimating the CVN toughness for the transverse orientation from longitudinal data.

4.3.3 Background

Metallurgical Aspects

Upper-shelf toughness in RPV steels is primarily controlled by the size, shape, and distribution of nonmetallic inclusions [e.g., manganese sulfide (MnS) particles]. Dislocation models for ductile fracture predict that the local stress concentration around an inclusion increases with decreasing particle size; hence, microvoid nucleation should be facilitated by a fine particle size (Goods and Brown, 1979). However, the dislocation models consider microvoid nucleation due only to inclusion-matrix debonding (Anderson, 1991). Experimentally, it has also been shown that void nucleation can occur more readily for larger inclusion sizes. This has been attributed to the greater tendency of the larger inclusions to crack or to exhibit partial matrix debonding as a result of fabrication stresses. At present, therefore, the dislocation microvoid nucleation models are not capable of adequately accounting for all experimental observations. Regarding the effect on USE, the dislocation models would predict a lower USE with smaller inclusions.

For RPV fabrication in the 1960s and 1970s, a fine inclusion size and dispersion were considered desirable for producing good radiographs and minimizing subsequent weld repairs. The Linde 80 flux used for the welding of B&W RPVs during this period was formulated with the intent of producing a fine inclusion size and spacing. The resulting weldments entered service with lower USEs when compared with plates or other weldments. This result is consistent with predictions from the dislocation models described above.

A deleterious characteristic of the welding procedures used for RPV fabrication at that time (1960s–1970s) was the use of copper-coated weld wire. The copper coatings were intended to minimize oxidation of the weld wires and to enhance electrical conductivity for arc stability during welding. The weldments produced with these wires had relatively high levels of copper, which subsequently was discovered to make them particularly susceptible to damage by neutron irradiation embrittlement in the beltline of the RPV (Steele, 1975). Neutron irradiation embrittles or hardens the steel matrix by displacing atoms from their normal crystal lattice positions. Copper is considered to enhance the hardening by forming precipitates in the matrix. One manifestation of the irradiation damage is a lowering of the USE with increasing irradiation (see Fig. 3.2a). A further complicating feature was that the thickness of the copper coating on the weld wires was highly variable, resulting in significant variations in copper content at different locations within a given weld.

The combination of the original metallurgical condition, which produced an initially low USE, and the copper-coated weld wire, which caused further reductions in USE with increasing time of operation, provided the impetus for examining margins of safety against fracture of RPV materials with low USEs. This issue was originally focused on B&W weldments fabricated using Linde 80 flux. However, the issue has since been broadened to include other weldments and plates and forgings.

In the case of plate, minimal cross rolling combined with a large percentage of MnS inclusions has been shown to produce a low USE in an A302 Grade B plate (NUREG/CR-5265). The minimal cross rolling resulted in MnS inclusions that were highly elongated in the principal rolling direction (stringers). The low interfacial energy required to separate the MnS stringers from the matrix, coupled with cracking of the stringers themselves, was considered to be the primary reason for the low USE for fracture along the principal rolling direction. It should be noted that the plate used for this study is considered to be an "irregular" A302 Grade B plate. However, in the absence of other A302 Grade B J-R curve data, the NRC staff has used the plate as a bounding case for RPVs fabricated with A302 Grade B. Recent (1994) preliminary fracture test results from Oak Ridge National Laboratory (ORNL) on what is considered a more representative A302 Grade B plate indicate significantly better fracture behavior in comparison with the material discussed in NUREG/CR-5265.

For plate materials in general, the crack plane orientation can have a significant effect on toughness. The principal rolling direction is typically considered to be the weak orientation for fracture (ASME Transverse, ASTM T-L). The fracture toughness in the strong orientation perpendicular to the principal rolling direction (ASME Longitudinal, ASTM L-T) is typically substantially higher in comparison with the weak orientation. For vessel flaw evaluation the weak orientation corresponds to a circumferential flaw while the strong orientation corresponds to an axial flaw (see Fig. 4.1). If data exist only for the strong orientation, a conversion to the weak orientation can be obtained by multiplying the USE for the strong orientation by 0.65 [BTP MTEB 5-2 (NUREG-0800)].

As a general category, forgings typically exhibit mechanical and fracture properties that are superior to those of either plates or weldments. This is attributable to the high degree of working of the forging at elevated temperatures in the process of forming the desired shape of the component. However, even forgings have not been exempt from low upper-shelf toughness concerns. An A508 reactor vessel ring forging fabricated by the Rotterdam Shipbuilding and Drydock Co. has been shown to have a low unirradiated upper-shelf toughness. The metallurgical reasons for this condition are not known with certainty but are likely related to the size, shape, and spacing of nonmetallic inclusions.

It should be noted that most materials used in RPV construction have exhibited initially high levels of upper-shelf toughness in the unirradiated condition. Accumulated damage from irradiation embrittlement is not expected to reduce the upper-shelf toughness of these materials below acceptable levels before the EOL dates for the plants. However, for plants that are below the screening criteria by EOL, fracture mechanics equivalent margins analyses (described below) have been able to demonstrate adequate margins of safety to USE levels below 54 joules (40 ft-lb) (see Section 4.3.4). ORNL has performed independent confirmation of the safety margins obtained using such analyses for the NRC Office of Research (NUREG/CR-6023).

Fracture Criteria

During the 1960s and 1970s when most of the currently operating RPVs were fabricated, it was widely believed that the only possible modes of failure for RPV materials on the upper shelf were plastic collapse or stable ductile tearing. However, several subsequent unexpected ductile fractures of large steel components focused attention on the possibility of low energy upper-shelf fractures (Babecki et al., 1959, and Pellini and Puzak, 1964). The recovery of the fracture properties of one of these components via heat treatment showed that elevating the USE to the range of 68–81 joules (50–60 ft-lb) was sufficient to prevent additional low energy ductile fractures. This led to a draft Atomic Energy Commission (AEC) document, "Material Fracture Toughness Requirements," in 1969 that required annealing for any vessel predicted to fall below 68 joules (50 ft-lb) during service. The overall technical basis for the 68 joule (50 ft-lb) criterion was given in the AEC report as the "NRL ratio analysis diagram, a CVN upper shelf-fracture toughness correlation attributed to Rolfe and Novak (1970), and a leak-before-break calculation for a 25.4-cm-thick (10-inch-thick) vessel wall."

Consideration of upper-shelf toughness resulted in paragraph IV.A.A. of Appendix G of 10 CFR Part 50, which states that if a vessel were predicted to fall below 68 joules (50 ft-lb) during the service life, it would have to be "demonstrated...that lower values of USE will provide margins of safety against fracture equivalent to those required by Appendix G of the ASME Code." The interpretation of "equivalent margins" has been the subject of considerable technical controversy. To settle the controversy, the NRC formally opened Generic Safety Issue A-11 and formed a task group to develop criteria for safety margins for low upper-shelf toughness materials in 1980. The focus of the group was to develop elastic-plastic fracture mechanics (EPFM) analytical criteria for the fracture safety evaluation. This issue was closed out with the publication of NUREG-0744 in 1982.

However, the report only endorsed J-integral tearing instability (J-T) analysis as an appropriate method for resolving the issue. Actual development of the criteria was transferred to the ASME Section XI Working Group on Flaw Evaluation (WGFE).

From 1983 to 1992 the WGFE and the Working Group on Operating Plant Criteria (WGOPC) worked on developing criteria for the low upper-shelf safety margin issue. This effort required a relatively long time to complete because of the state-of-the-art issues in EPFM that were being addressed. Further details can be found in the historical overview by Merkle (1991). In 1992, the WGFE officially transmitted criteria for assessing reactor vessels with low upper-shelf Charpy energy levels to the NRC. The criteria were formally published in 1993 in ASME Code, Section XI, as Code Case N-512 and are currently incorporated as Appendix K in the 1993 Addenda to Section XI of the ASME Code. Code Case N-512 and Appendix K describe a fracture safety assessment procedure that is based on J-integral tearing instability analysis. Simply stated, the methodology involves evaluating the ductile fracture initiation and stability of a cracked component by comparing the amount of energy available to drive the fracture process with the amount of energy required to extend the crack. The parameters used in the analysis are the $J_{material}$, which consists of a conservative representation of the material J-R curve (material fracture resistance), and the $J_{applied}$ (the driving force for fracture). The $J_{applied}$ is calculated for a postulated flaw in the vessel wall with loading conditions and safety factors that vary for normal operating, upset, emergency, and faulted conditions (levels A, B, C, and D). To demonstrate equivalent margins, the $J_{applied}$ must be less than or equal to the $J_{material}$ at 0.254 cm (0.1 inch) of crack extension and the $dJ/da_{applied}$ must be less than the J-R curve slope (dJ/da) evaluated at $J_{applied} = J_{material}$. The NRC staff and contractors were involved in the development of the criteria, and the ASME Code procedures describe a methodology and criteria that the NRC staff finds acceptable for performing analyses of this type. The NRC staff subsequently developed draft RG DG-1023, which incorporates the methodology and criteria recommended in the ASME Code procedures and also contains guidance on transient selection and material properties.

4.3.4 Status of Low Upper-Shelf Energy Reviews

The NRC staff published the results of a preliminary GL 92-01, Revision 1, review in February 1993 in SECY-93-048. All licensees responded to GL 92-01, Revision 1, that, based on plant-specific data and evaluations, their reactor vessels satisfied the 68 joule (50 ft-lb) minimum USE criterion. In the NRC review, the staff used RG 1.99, Revision 2, to evaluate all RPV materials. For plates or forgings with only unirradiated USE data in the strong orientation, the staff used the correction factor of 0.65 recommended in BTP MTEB 5-2 (NUREG-0800) to determine EOL USEs. Using the conservative NRC generic criteria, the NRC staff identified, in SECY-93-048, 18 plants having EOL USE values below 68 joules (50 ft-lb) for their beltline materials. In addition, many BWR plants were identified as lacking the unirradiated USE and material data necessary to perform plant-specific analyses. Since then, all licensees with plants having EOL USE values below 68 joules (50 ft-lb) and all BWR licensees that lacked unirradiated USE data have submitted equivalent margins analyses, either through an owners group or on a plant-specific basis. These analyses are intended to demonstrate through fracture mechanics analysis that there exist margins of safety against fracture equivalent to those required by Appendix G of ASME Code, Section III, for beltline materials having USE values below 68 joules (50 ft-lb).

The NRC staff has completed the review of all these analyses. All four owners groups assisted their member plants in demonstrating that they meet Appendix G requirements on USE by submitting equivalent margins analyses. The BWR Owners Group and the Babcock and Wilcox Owners Group submitted their analyses as topical reports. The Westinghouse Owners Group and the Combustion Engineering Owners Group submitted their reports for information only. In addition to the owners group efforts, some licensees still chose to submit plant-specific equivalent margins analyses. The owners group generic analyses and licensee plant-specific analyses are discussed in the following section.

4.3.5 Equivalent Margins Analyses

4.3.5.1 Owners Groups Analyses

Boiling Water Reactor Owners Group (BWROG)

The NRC staff verified that the BWROG analyses in topical report NEDO-32205 (GE Nuclear Energy, 1993) complied with the analytical procedures and the acceptance criteria in ASME Code, Section XI, Code Case N-512 in calculating the minimum permissible USE for each type of beltline material (the 29 BWROG plants are listed in Appendix A). The NRC staff identified some minor deviations from the code case procedures and some unique approaches used by the BWROG because of lack of guidance in the code case. They were evaluated and found to be acceptable. Most BWROG plants do not have complete heat-specific initial USE values for their beltline materials. To overcome this deficiency, the BWROG performed a statistical analysis of the BWROG database, which consists of USE test data on some beltline materials (plates, forgings, and welds) from about 31 BWR reactor plants, to derive statistically the initial USE values for materials that originally did not have documented USE values. In this part of the evaluation, the NRC staff established the generic initial USE values using the lower tolerance limit, so that there is 95-percent confidence that 95 percent of the population will be above that value. This is a conservative approach, and future evaluation using more sophisticated statistical methods may allow an alternative evaluation.

After the initial USE values were determined, the BWROG then predicted the EOL USE values in accordance with RG 1.99, Revision 2. The EOL USE value for each type of beltline material is higher than the minimum permissible USE calculated by using the methodology of Code Case N-512. Therefore, the topical report demonstrates that the materials evaluated will have the margins of safety against fracture equivalent to those required by Appendix G of the ASME Code, Section III and Section XI, in accordance with Appendix G of 10 CFR Part 50, at least through their currently projected EOLs. The details of the evaluation of this topical report can be found in the NRC safety evaluation report (SER) forwarded to the BWROG by letter dated December 8, 1993. However, to administratively close the issue of USE for BWROG plants, licensees using the topical report to demonstrate compliance for their plants have been asked to make submittals to the NRC demonstrating the applicability of NEDO-32205 to their plants and requesting NRC review and approval.

Babcock and Wilcox Owners Group (B&WOG)

The NRC staff reviewed the B&WOG equivalent margins analyses in two topical reports, BAW-2192P* and BAW-2178P**. The former report (service levels A and B) was reviewed by the NRC staff, and the latter report (service levels C and D) was reviewed by the NRC staff with assistance from ORNL. From both reviews, the NRC staff concluded that the RPVs in the 16 B&WOG plants listed in Appendix A have been shown to have adequate margins of safety against ductile tearing in low upper-shelf welds, at currently projected EOL, for all service levels, by analysis results meeting the criteria in ASME Code Case N-512. It should be noted that only Linde 80 welds were evaluated in those reports. In two NRC SERs (forwarded by letters both dated March 29, 1994), the NRC staff compared the J-R curves of Linde 80 welds with those of Rotterdam welds, which also exist in some B&W vessels, and confirmed that Linde 80 welds are limiting. Extension of the reports to include plates and forgings has been verified in BAW-2222 (B&W Nuclear Service Company, 1994) in which B&WOG developed generic USE values using the lower tolerance limit, so that there is 95-percent confidence that 95 percent of the population will be above that value. This method of developing generic values is acceptable to the NRC staff for all B&WOG plates and forgings without documented USE values. Consequently, the topical reports have demonstrated that all beltline materials of the B&WOG RPVs will have adequate margins of safety against fracture at least through their currently projected EOLs. However, to administratively close the issue of USE for B&WOG plants, licensees using either or both topical reports to demonstrate compliance for their plants have been asked to make submittals to the NRC demonstrating the applicability of BAW-2192P and BAW-2178P to their plants and requesting NRC review and approval.

Westinghouse Owners Group (WOG)

WOG submitted a bounding equivalent margins analysis for its 42 member plants (listed in Appendix A) in WCAP-13587, Revision 1 (Westinghouse Electric Corporation, 1993). This report was submitted for information only; review and approval were not requested. However, the NRC staff evaluated the report and forwarded a safety assessment in a letter to the Nuclear Management and Resources Council dated April 21, 1994. The NRC staff found that the methodology used and the analysis performed for the WOG were acceptable and that all WOG plants will have adequate margins of safety against fracture at least through their currently projected EOLs. The equivalent margins analysis was performed using lower bound properties applicable to plates and forgings. The WOG has reported that the lower bound properties for the plates and forgings are more limiting than the properties of the welds. Consequently, WCAP-13587, Revision 1, demonstrates that the plates and forgings evaluated have margins of safety against fracture equivalent to those required by Appendix G of the ASME Code. Since WCAP-13587, Revision 1, was not submitted as a topical report for formal review and approval by the NRC, in order to administratively close the issue of USE for WOG plants, licensees desiring to use the WOG analysis to demonstrate compliance for their plants have been asked to make submittals to the NRC demonstrating the applicability of WCAP-13587 to their plants and requesting NRC review and approval.

Combustion Engineering Owners Group (CEOG)

The CEOG provided a bounding equivalent margins analysis for their 15 member plants (listed in Appendix A) in CEN-604, Revision 1 (ABB Combustion Engineering Nuclear Power, 1993), which included responses to an NRC staff request for additional information. The NRC staff found that the methodology used and the analysis performed by the CEOG were acceptable and that all CEOG plants will have adequate margins of safety against

*B&W Nuclear Service Company, "Low Upper-Shelf Toughness Fracture Analysis of Reactor Vessels of B&W Owners Group Reactor Vessel Working Group for Load Level A & B Conditions," Lynchburg, Virginia, December 1993 (proprietary).

**———, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Level C & D Service Loads," Lynchburg, Virginia, February 1993 (proprietary).

fracture at least through their currently projected EOLs. The NRC staff safety assessment has been completed (letter to CEOG, September 15, 1994). The equivalent margins analysis was performed using bounding plate properties and is thus applicable to plates and forgings but not to weldments. The CEOG has reported that the analysis for the plate and forging materials is limiting with regard to USE and bounds the weldments. Since CEN-604, Revision 1, was not submitted as a topical report for formal review and approval by the NRC, in order to administratively close the issue of USE for CEOG plants, licensees desiring to use the CEOG analysis to demonstrate compliance for their plants have been asked to make submittals to the NRC demonstrating the applicability of CEN-604 to their plants and requesting NRC review and approval. The methodology used by the CEOG coupled with plant-specific information on the welds was applied in the analyses performed for Fort Calhoun and Big Rock Point.

4.3.5.2 Plant-Specific Analyses

As discussed in Section 4.3.4, some licensees chose to submit plant-specific equivalent margins analyses. These plants were Turkey Point 3 and 4 (service levels A and B), Zion 1 and 2 (service levels A and B), Nine Mile Point 1, Fort Calhoun, Watts Bar 1, Robinson 2, and Big Rock Point. Currently, the service levels C and D analyses for Turkey Point 3 and 4 and Zion 1 and 2 are covered by BAW-2178P. The NRC staff has reviewed all plant-specific equivalent margins analysis reports and has found that the licensees for the subject plants have demonstrated that their plants' beltline materials will have margins of safety against fracture equivalent to those required by Appendix G of the ASME Code at least through their currently projected EOLs. The NRC staff has issued the SERs for these plants in letters dated October 19 and December 2, 1993, and April 20, May 11, September 14, and September 19, 1994.

4.3.5.3 Summary and Conclusion

The NRC staff has reviewed the licensees' responses to GL 92-01 and other docketed information and the industry's equivalent margins analyses and has assessed the USE values for all the RPVs in U.S. plants. In some cases, because of limitations in the available data, the NRC staff could not reliably conclude that the material in all RPVs would remain above 68 joules (50 ft-lb) throughout their operating life. In these cases the NRC staff used conservative values for the drop in USE and evaluated the resulting USE values to ensure that equivalent margins would exist. An NRC report, NUREG/CR-6023, indicates that RPV materials for PWR and BWR vessels can have USE values of less than 68 joules (50 ft-lb) and still meet the margins of safety against fracture equivalent to those required by Appendix G of the ASME Code. On the basis of the review of the industry's equivalent margins analyses and the generic study conducted by the NRC, the staff has concluded that all RPVs will have adequate upper-shelf toughness throughout their current licensed operating life.

4.4 Pressure-Temperature Limits

The purpose of pressure-temperature (P-T) limits is to ensure the structural integrity of the reactor vessel through the imposition of limits on reactor coolant system (RCS) operation. The P-T limits are derived on the basis of linear elastic fracture mechanics (LEFM) analyses. In these analyses, the minimum temperature necessary to ensure adequate margins against RPV failure is determined as a function of pressure, and a set of pressure versus temperature or P-T curves is developed.

The temperature and pressure of the RCS are limited to below and to the right of the P-T curves for all modes of RCS operation (Figs. 4.2 and 4.3). The P-T curves must be shifted to the right as the RPV material loses toughness as a result of irradiation. Therefore, licensees periodically revise the P-T limits because either the test results from surveillance capsules require changes to the limits, or the plant is about to exceed the effective full-power years (EFPYs) used to develop the P-T limits. The P-T limits apply to heatup, cooldown, criticality, hydrostatic test, and leak test conditions.

The P-T limits are increased to higher temperatures by the amount of increase in the 41 joule (30 ft-lb) transition temperature that is determined from surveillance data or from chemistry tables in RG 1.99, Revision 2. The adjusted reference temperature (ART), which is discussed in Section 3.3, is the material property that is used to determine the effect of radiation on each plant's P-T limits.

4.5 Low-Temperature Overpressure Protection Limits

4.5.1 Description of Issue and Background

Low-temperature overpressure protection (LTOP) limits are used in nuclear plants to ensure that pressure transients at low temperatures do not cause fracture of the reactor vessel. The NRC staff reviewed the LTOP issue in NUREG-0224 dated September 1978. In the report, it identified operating transients during low-temperature operation that exceeded the P-T limits for the reactor vessel. The NRC staff reported that reactor vessel pressure transients during low-temperature operation had been initiated by a variety of causes, grouped into the following categories: personnel error, procedural deficiencies, component random failures, and spurious valve actuation. The resultant pressure transients are two types: a mass input type from charging pumps, safety injection pumps, or safety injection accumulators and a thermal expansion type caused by the feedback of heat from the secondary side of the steam generators. These transients are of particular concern during low-temperature operation because many PWRs operate at low temperatures with the RCS water solid. When the RCS is water solid, mass input and thermal expansion type transients cause an instantaneous increase in the system pressure. As a result of its review, the NRC staff issued Branch Technical Position RSB 5-2, "Overpressure Protection of Pressurized Water Reactors While at Low Temperatures," which is contained in Standard Review Plan Section 5.2.2 (NUREG-0800). This branch technical position states that a protection system should be designed and installed that will prevent the RCS from exceeding the limits specified in Appendix G of 10 CFR Part 50 while at low temperatures. Appendix G of 10 CFR Part 50 requires that the reactor vessel be operated with P-T limits at least as conservative as those obtained by following the analysis methods and the required margins of safety of Appendix G of the ASME Code.

4.5.2 Status of Low-Temperature Overpressure Protection Limits Issue

To prevent LTOP transients that would produce excursions exceeding the limits in Appendix G of 10 CFR Part 50, PWRs have a system to relieve pressure in the reactor vessel. The actuation setpoint of such an LTOP system must be set at a pressure low enough to keep the pressure in the reactor vessel from the Appendix G limits during an LTOP transient. However, to prevent actuating the LTOP system during pressure surges that occur during low-temperature operations, such as surges resulting from starts of a reactor coolant pump or a charging pump, the pressure in the reactor vessel must be kept below the LTOP system actuation setpoint pressure. In addition, in order to start a reactor coolant pump, there must be a differential pressure across the reactor coolant pump seals. The difference between these pressure values establishes an operating window for heatup of the reactor vessel.

As a plant ages, the operating window is reduced because of the embrittling effects of neutron irradiation on the reactor vessel. For a sufficiently small operating window, an expected pressure surge during low-temperature operations could actuate the LTOP system and cause a loss of reactor coolant inventory through the LTOP pressure relief valves. The loss would be exacerbated if the pressure relief valves were to remain open for any reason. Unnecessary operation of the pressure relief valve is undesirable.

To increase the operating window, the ASME Code Committee has approved ASME Code, Section XI, Code Case N-514, which would permit recalculation of the LTOP system setpoint so as to limit the maximum pressure in the reactor vessel to 110 percent, rather than 100 percent, of the pressure determined to satisfy ASME Code, Section III and Section XI, Appendix G. The content of Code Case N-514 has been incorporated into Appendix G of Section XI of the ASME Code and published in the 1993 Addenda to Section XI. The NRC staff is currently developing a revision to 10 CFR 50.55a that will endorse the 1993 Addenda and Appendix G of Section XI in the regulations.

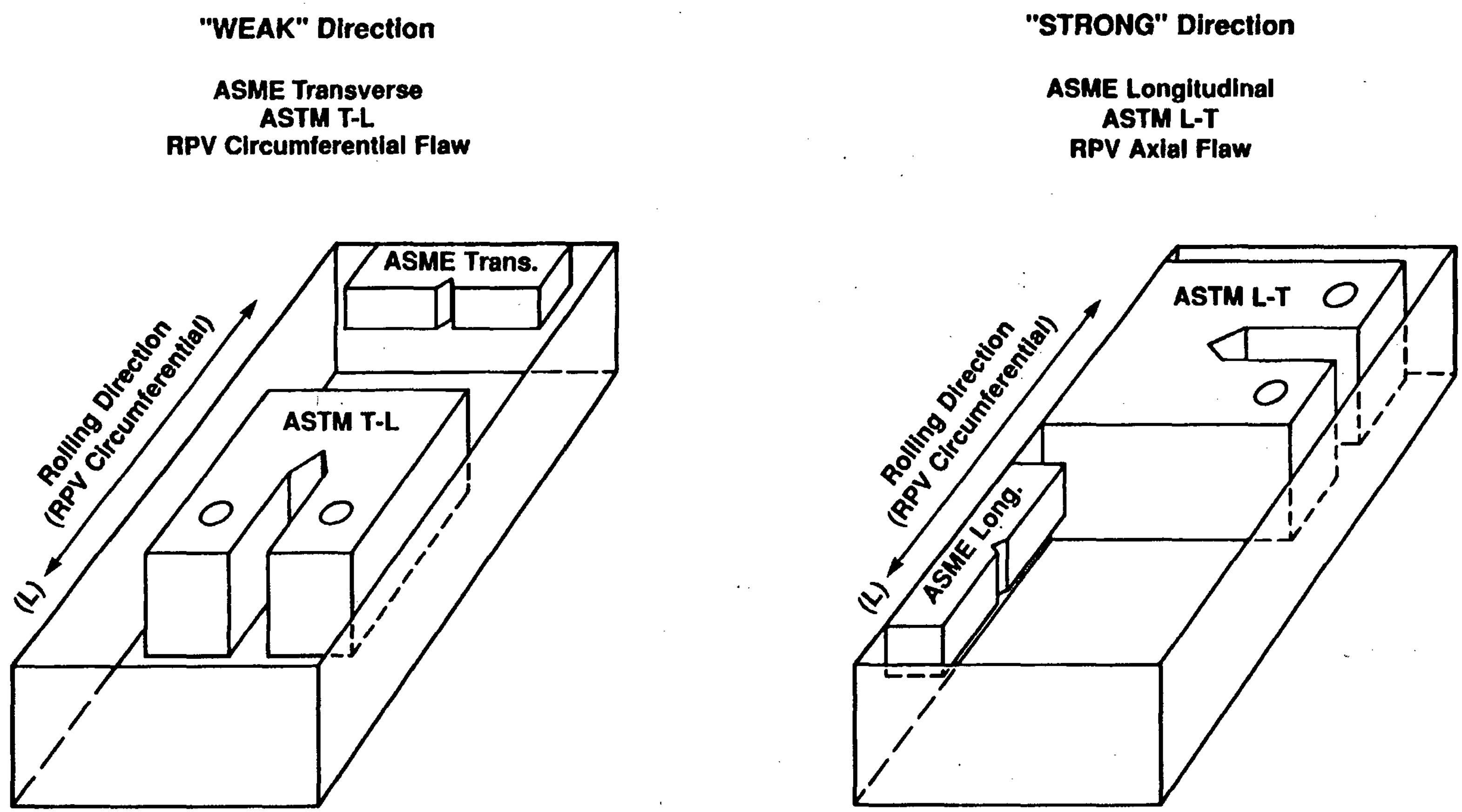
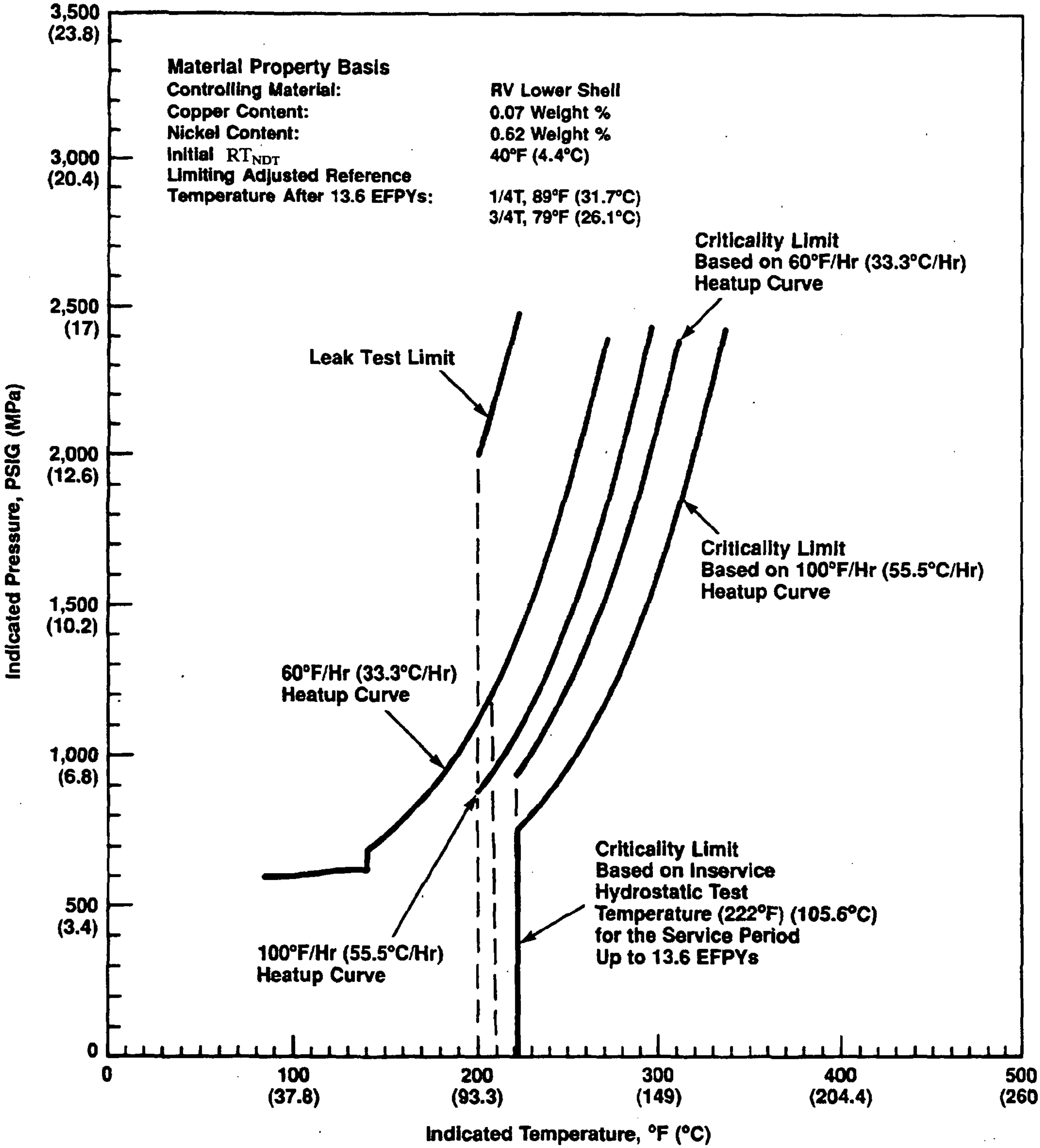
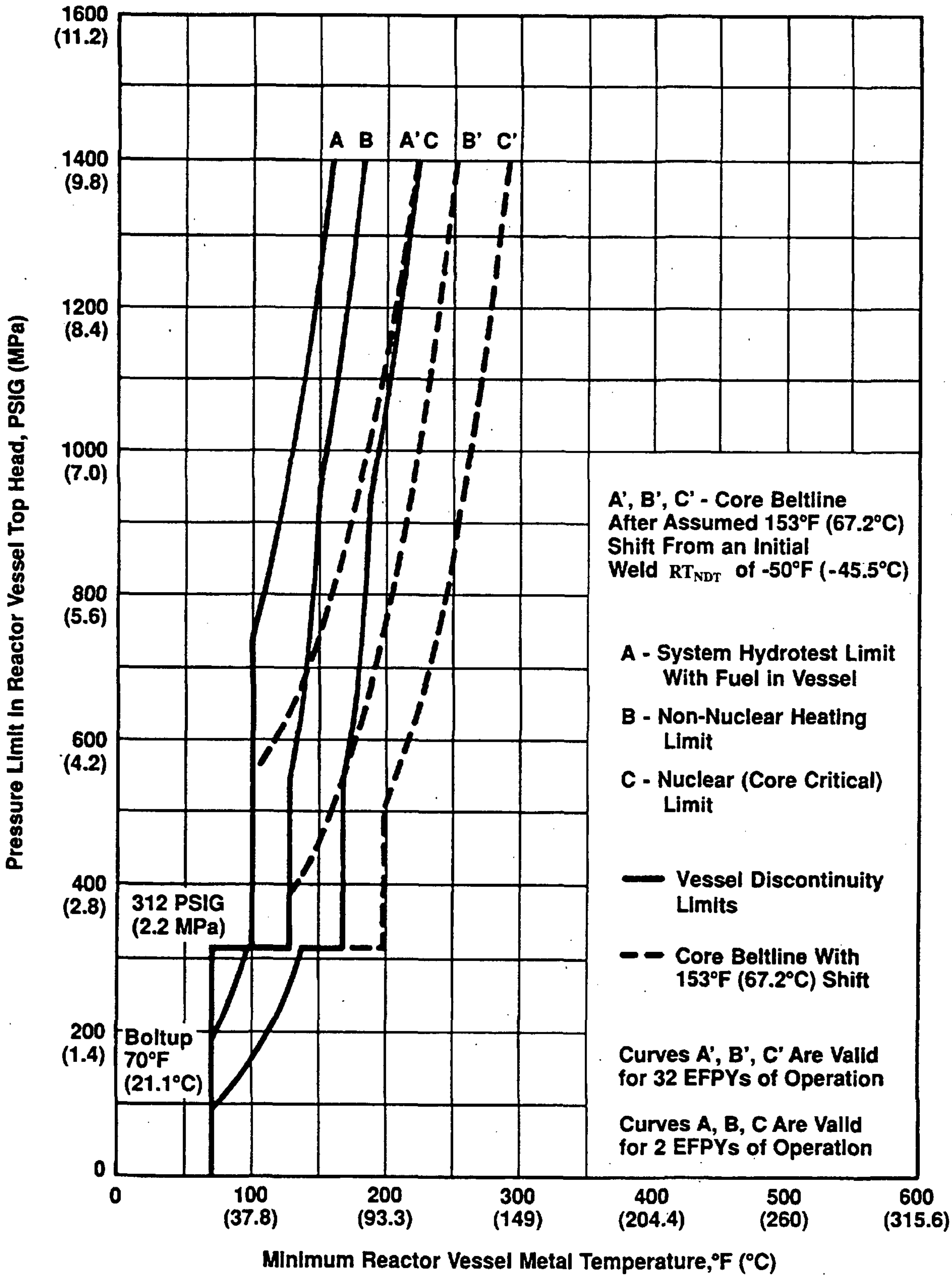


Figure 4.1 Definition of the ASME and ASTM Flaw Orientations in an RPV



Note: EFPY = effective full-power year

Figure 4.2 Typical Pressurized Water Reactor Pressure-Temperature Operating Limits



Note: EPFY = effective full-power year

Figure 4.3 Typical Boiling Water Reactor Pressure-Temperature Operating Limits

5 REACTOR VESSEL INTEGRITY DATABASE

5.1 Reactor Vessel Integrity Database Tables

As a result of its review of licensee responses to Generic Letter (GL) 92–01, the NRC staff has developed a comprehensive database to compile and record summaries of the material properties for the reactor vessel beltline materials for each plant. This database is known as the reactor vessel integrity database (RVID).

In addition to the licensee responses to GL 92–01, the following documents were included in the review process and development of the RVID: surveillance capsule reports, documents referenced in the GL 92–01 submittals, and as applicable, pressurized thermal shock (PTS) submittals and responses to the staff's requests for additional information. The staff reviewed the data from these source documents and documented them in the RVID tables.

The format of the RVID has been set up so that there are four tables for each plant: a background information table, a chemistry data table, an upper-shelf energy (USE) table, and a pressure-temperature (P-T) limits table (BWRs) or a PTS table (PWRs). Information headings for each of the tables are as follows:

- Background Information Table

  Plant Name
  Docket Number
  NSSS [Nuclear Steam Supply System] Vendor
  Vessel Manufacturer
  Edition of ASME Code for Design
  Date of Commercial Operation
  Date of License Expiration
  Vessel Beltline Thickness
  Vessel Inside Radius

- Chemistry Data Table

  Beltline Identification
  Heat Number
  Copper (Cu) %
  Nickel (Ni) %
  Sulfur (S) %
  Phosphorous (P) %

- USE Table

  Beltline Identification
  Heat Number
  Material Type
  USE at End of License
  1/4T Neutron Fluence at End of License
  Unirradiated USE
  Method of Determining Unirradiated USE
  Percent Drop of USE at End of License
  Method of Determining Percent Drop

- P–T Limits Table (BWRs) or PTS Table (PWRs)

  Beltline Identification
  Heat Number
  RT_{PTS} at End of License
  ID [Inner-Diameter] Fluence at End of License
  Initial RT_{NDT}
  Method of Determining Initial RT_{NDT}
  ART [Adjusted Reference Temperature] at End of License
  Fluence Factor at End of License
  Chemistry Factor
  Method of Determining Chemistry Factor
  Margin
  Method of Determining Margin
  Cu %
  Ni %

References and notes follow each table to document the original sources of data and to provide additional information.

The RVID includes sort and data search capabilities. The user can select a desired grouping of plants (i.e., all plants) or limit the search to either PWRs, BWRs, specific owners groups, or selected individual plants, and then specify up to ten of the above categories to search and list. The NRC staff will update the RVID periodically to reflect the latest information available.

5.2 Summary Files – Plant-Specific Evaluations

Plant summaries, including RT_{PTS} and USE data for the limiting material or materials for each plant, were derived from the staff review of licensee responses to GL 92–01. The staff requested licensees to confirm the applicability of the unirradiated, neutron fluence and chemical composition data that provide the basis for plant summaries. The majority of licensees have confirmed that the data are applicable to their plants. Since the RVID is too extensive to reproduce in its entirety, only embrittlement information for the limiting beltline material or materials is presented in the summary file. Appendix B contains summary files for all 111 plants. The entries are explained below in the order they appear in the file.

Plant Name: (The plant-specific evaluations are ordered alphabetically.)

Docket Number: An NRC identification number

NSSS Vendor: Nuclear steam system supplier and design type vendor (i.e., General Electric, Westinghouse, Combustion Engineering, or Babcock and Wilcox)

Vessel Manufacturer: Identification of vessel manufacturer

Edition of ASME Code for Design: Edition and addenda of the ASME Code

Date of Commercial Operation: Date when the plant began operating commercially

Date of License Expiration: Date when license expires, also referred to as EOL [end-of-license] date

For PWRs, the RT_{PTS} for the Limiting Beltline Material:

or

For BWRs, the ART for the Limiting Beltline Materials:

Limiting Beltline Material: The PTS screening criterion for PWRs is 132 °C (270 °F) for plates, forgings, and axial weld materials or 149 °C (300 °F) for circumferential weld materials. The limiting material in a PWR is the material with an RT_{PTS} value at EOL that most closely approaches or exceeds these values. The limiting material in a BWR is the material with the greatest ART at EOL. The ART and the RT_{PTS} value are equivalent material properties that are the sum of (1) unirradiated reference temperature (initial RT_{NDT}); (2) margin to be added to cover uncertainties in the initial properties, copper and nickel contents, fluence, and calculation procedures; and (3) increase in RT_{NDT} caused by radiation.

ID Fluence at EOL: The fluence value at the ID of the reactor vessel at EOL, cited directly from ID value or calculated using RG 1.99, Revision 2, neutron fluence attenuation methodology from the quarter thickness reported in the most recent submittal (GL 92-01, PTS, or P-T limits submittals).

Initial RT_{NDT}: The initial RT_{NDT} of the unirradiated material measured as defined in ASME Code, Section III, paragraph NB-2331, or its equivalent, in accordance with Branch Technical Position MTEB 5-2 in Standard Review Plan Section 5.3.2 (NUREG-0800). If measured values are not available, generic mean values must be used. The PTS rule specifies a generic mean value of -17.78 °C (0 °F) for welds made with Linde 80 flux, and -48.89 °C (-56 °F) for welds made with Linde 0091, 1092 and 124 and ARCOS B-5 weld fluxes. Other values were reported where sufficient data were provided.

Method of Determining Chemistry Factor: The chemistry factor (CF) is a function of copper and nickel content. "Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2" indicates that the CF was determined from the CF tables in RG 1.99, Revision 2. "Surveillance data per Paragraph C.2.1 of RG 1.99, Rev. 2" indicates that the CF was determined from surveillance data by means of procedures described in RG 1.99, Revision 2.

Increase in RT_{NDT}: The mean value of the adjustment in reference temperature caused by irradiation and calculated as follows: Increase in $RT_{NDT} = (CF)f^{(0.28-0.10 \log f)}$, where CF is the chemistry factor and f (10E19 n/cm^2, E > 1 MeV) is the fluence.

Margin: A numerical quantity added to cover uncertainties in the values of initial RT_{NDT}, copper and nickel contents, fluence, and the calculational procedures involved in determining RT_{NDT}. The PTS rule states that (1) when tables are used to determine the CF and generic values are used for the initial RT_{NDT}, the margin values are 36.67 °C (66 °F) for welds and 26.67 °C (48 °F) for base metal and (2) when tables are used to determine the CF and measured values are used for the initial RT_{NDT}, the margin values are 31.11 °C (56 °F) for welds and 18.89 °C (34 °F) for base metal. Other margin values have also been calculated using the methodology in RG 1.99, Revision 2, when credible surveillance data were used to determine the CF, or generic mean values of initial RT_{NDT} for welds were different from the values in the PTS rule.

ART or RT_{PTS} at EOL: The reference temperature as adjusted for the effects of neutron irradiation at EOL, which is a summation of the initial RT_{NDT}, the margin, and the increase in RT_{NDT}.

USE for the Limiting Beltline Material:

Limiting Beltline Material: The material with the lowest USE value at EOL or the material with insufficient data to determine the initial (unirradiated) USE.

1/4T Fluence at EOL: The fluence value at the quarter thickness of the reactor vessel at EOL, cited directly or calculated using RG 1.99, Revision 2 neutron fluence attenuation methodology from the ID reported in the most recent submittal (GL 92-01, PTS, or P-T limits submittals).

Initial USE: Unirradiated USE reported if sufficient data exist. The USE is rounded to the nearest whole number. If insufficient data exist to report an initial USE, the USE issue may be covered by either owners group or plant-specific equivalent margins analyses, as explained in Section 4.3.5, and will be highlighted by stating, "Heat specific value not reported."

Percent Drop at EOL: The percentage drop of USE from the initial unirradiated value to the projected EOL value resulting from radiation embrittlement. The percent drop is calculated using the methodology in RG 1.99, Revision 2.

USE at EOL: The calculated USE value at EOL if an initial USE value is reported. The USE is rounded to the nearest whole number. "Not applicable since heat specific value not reported" is stated if an initial USE is not reported.

Date USE Screening Limit Will Be Exceeded: Appendix G of 10 CFR Part 50 requires that reactor vessel beltline materials maintain their USE throughout the life of the vessel of no less than 68 joules (50 ft-lb), unless it is demonstrated in a manner approved by the Director of the Office of Nuclear Reactor Regulation that lower values of USE will provide margins of safety against fracture equivalent to those required by Appendix G of the ASME Code. The file states, "after EOL" if the USE at EOL is projected to be equal to or greater than 68 joules (50 ft-lb). The file states, "before EOL" if the USE at EOL is predicted to be below 68 joules (50 ft-lb). "Not applicable since heat specific value not reported" is stated if a heat specific initial USE value was not reported.

Bases for Accepting the USE at EOL: The specific basis by which the USE at EOL was found to be acceptable in accordance with the requirements of Appendix G of 10 CFR Part 50. If the USE at EOL is greater than 68 joules (50 ft-lb), either surveillance data or chemistry data were used to determine the percent drop. If the USE is less than 68 joules (50 ft-lb) or if an initial USE value was not reported, an equivalent margins analysis was performed to demonstrate that the material will provide margins of safety against fracture equivalent to those required by Appendix G of the ASME Code.

6 SUMMARY AND CONCLUSIONS

The NRC staff has reviewed the licensee responses to Generic Letter (GL) 92–01 and other docketed information and assessed the RT_{PTS} values for all the PWR reactor pressure vessels (RPVs) in U.S. plants. On the basis of this review, the NRC staff has concluded that the RPVs in most plants will be well below the pressurized thermal shock (PTS) screening limits at the end of their current operating licenses. On the basis of the currently docketed information, Beaver Valley 1 and Palisades are the only plants projected to exceed their PTS screening limits before their licenses expire. Beaver Valley 1 and Palisades are projected to exceed the PTS screening limits in 2012 and 2004, respectively, before the end of their operating licenses in 2016 and 2007. It is important to note that these results are based on the information currently reported by the licensees and are subject to change. The dates when the plants are projected to reach the screening criteria may change as a result of new surveillance data and additional analyses. Also, by implementing different fuel management techniques and inserting special neutron-absorbing materials in the reactor core, licensees may be able to reduce the irradiation levels sufficiently to stay below the screening criteria. In addition, licensees may anneal the RPV to restore the material properties to near the original unirradiated condition. Industry has already informed the NRC staff that additional data and information will be submitted. Therefore, it is expected that the above information will change. These changes will be reported in periodic updates of this report.

On the basis of the responses to GL 92–01 and other docketed information and the industry's equivalent margins analyses, the staff has assessed the upper-shelf energy (USE) values for all the RPVs in U.S. plants. In some cases, because of limitations in the available data, the NRC staff could not reliably conclude that the material in all RPVs would remain above 68 joules (50 ft-lb) throughout their operating life. In these cases, the NRC staff used conservative values for the drop in USE and evaluated the resulting USE values to ensure that equivalent margins would exist. An NRC report, NUREG/CR–6023, indicates that RPV materials for PWR and BWR vessels can have USE value of less than 68 joules (50 ft-lb) and still meet the margins of safety against fracture equivalent to those required by Appendix G of the ASME Code. On the basis of this generic study and the GL 92–01 review conducted by the NRC, the staff has concluded that all RPVs will have adequate upper-shelf toughness throughout their current licensed operating life.

The assessment of RPV integrity must be a continuing proactive effort. As noted above it is expected that additional information and analyses and licensee programs to reduce neutron flux will result in changes in the currently predicted RT_{PTS} and USE values. The NRC staff will continue to assess new information as it becomes available and will periodically update this report on the basis of this information. This effort will be facilitated through the use of the computerized reactor vessel integrity database (RVID). This database developed by the NRC includes summary tables containing necessary input for evaluating RPV structural integrity in accordance with the requirements of 10 CFR Part 50, Appendix G, and the PTS rule (10 CFR 50.61). The data come primarily from licensee responses to GL 92–01 and have been verified by the licensees. It is anticipated that the RVID will be made available for public access in late 1994 or early 1995. The RVID will also be updated periodically on the basis of NRC assessments of new information from the industry and licensees. Appendix B contains summary files for all plants. Each file contains the adjusted reference temperature (ART) (for BWRs) or RT_{PTS} (for PWRs) and USE data for the limiting material or materials of a plant.

7 REFERENCES

ABB Combustion Engineering Nuclear Power, CEN-604, "Evaluation of Low Upper Shelf Energy for Combustion Engineering Nuclear Steam Supply Systems Reactor Pressure Vessels," Revision 1, Windsor, Connecticut, September 1993.

American Society for Testing and Materials, E 185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," Philadelphia, Pennsylvania, September 1982. Last previous edition E 185-79, July 1979.

American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Code Case N-512, "Assessment of Reactor Vessels With Low Upper Shelf Charpy Impact Energy Levels," Section XI, Division 1, New York, February 12, 1993.

—, Boiler and Pressure Vessel Code, Code Case N-514, "Pressure Temperature Limits for Low Temperatures Overpressure Events," Section XI, Division 1, New York, December 1992.

Anderson, T.L., *Fracture Mechanics, Fundamentals and Applications,* CRC Press, Boca Raton, Florida, 1991.

Atomic Energy Commission, Draft Section 51.123, "Material Fracture Toughness Requirements," Washington, D.C., June 18, 1969.

Babecki, A.J., P.P. Puzak, and W.S. Pellini, "Report of Anomalous Brittle Failures of Heavy Steel Forgings at Elevated Temperatures," ASME Paper 59-MET-6, American Society of Mechanical Engineers, New York, May 1959.

B&W Nuclear Service Company, BAW-2222, "Reactor Vessel Working Group Response to Closure Letters to NRC Generic Letter 92-01, Revision 1," Lynchburg, Virginia, June 1994.

Code of Federal Regulations, Title 10, "Energy," U.S. Government Printing Office, Washington, D.C., updated periodically.

Electric Power Research Institute, TR-100251, "White Paper on Reactor Vessel Integrity Requirements for Level A and B Conditions," Palo Alto, California, January 1993.

GE Nuclear Energy, NEDO-32205, "10 CFR 50 Appendix G Equivalent Margin Analysis for Low Upper Shelf Energy in BWR/2 Through BWR/6 Vessels," Licensing Topical Report for BWR Owners Group, San Jose, California, April 1993.

Goods, S.H., and L.M. Brown, "The Nucleation of Cavities by Plastic Deformation," *Acta. Met.*, 21, 1-15, 1979.

Griesbach, T.J., and W.L. Server, "Reactor Vessel Embrittlement Management Handbook," Electric Power Research Institute (EPRI) TR-101975-T2, Palo Alto, California, December 1993.

Griesbach, T.J., "Reactor Pressure Vessel Design and Fabrication," Electric Power Research Institute (EPRI) TR-101975-V6, Palo Alto, California, September 1994.

Merkle, J.G., "A Summary of the Low Upper Shelf Toughness Safety Margin Issue," PVP Vol. 213/MPC Vol. 32, Pressure Vessel Integrity, American Society of Mechanical Engineers, New York, 1991.

Pellini, W.S., and P.P. Puzak, "Practical Considerations in Applying Laboratory Fracture Test Criteria to the Fracture-Safe Design of Pressure Vessels," *J. Eng. Power*, 429-443, October 1964.

PVRC Ad Hoc Group on Toughness Requirements, "PVRC Recommendations on Toughness Requirements for Ferritic Materials," *Weld. Res. Coun. Bull.* No. 175, August 1972.

Rolfe, S.T., and J.M. Barsom, *Fracture and Fatigue Control in Structures—Applications of Fracture Mechanics,* Prentice-Hall, Englewood Cliffs, New Jersey, 1977.

Rolfe, S.T, and S.R. Novak, "Slow Bend K_{IC} Testing of Medium Strength High Toughness Steels," ASTM STP 463, American Society for Testing and Materials, Philadelphia, Pennsylvania, 1970, pp. 124-159.

Steele, L.E., "Neutron Irradiation Embrittlement of Reactor Pressure Vessel Steels," Technical Reports Series, No. 163, International Atomic Energy Agency, Vienna, Austria, 1975.

U.S. Nuclear Regulatory Commission, Commission Paper SECY-82-465, "Pressurized Thermal Shock (PTS)," November 23, 1982.

—, Commission Paper SECY-93-048, "Status of Reactor Pressure Vessel Issues Including Compliance With 10 CFR Part 50, Appendices G and H (WITS 9100161)," February 25, 1993.

—, Draft Regulatory Guide DG-1023, "Evaluation of Reactor Pressure Vessels With Charpy Upper Shelf Energy Less Than 50 ft-lb," August 1993.

—, Draft Regulatory Guide DG-1025, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," October 1993.

—, Generic Letter 92-01, Revision 1, "Reactor Vessel Structural Integrity," March 6, 1992.

—, letter from L. Raghavan (NRC) to J. H. Goldberg (Florida Power and Light Company), "Turkey Point Units 3 and 4—Review of Babcox & Wilcox Owners Group Materials Committee Report—Upper Shelf Energy," October 19, 1993.

—, letter from S. Bloom (NRC) to T. L. Patterson (Omaha Public Power District), "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 158 to Facility Operating License No. DPR-40," December 3, 1993.

—, letter from James T. Wiggins (NRC) to Lesley A. England (Boiling Water Reactor Owners Group), "Safety Evaluation by the Office of Nuclear Reactor Regulation of BWR Owners' Group Topical Report NEDO-32205, Revision 1, on Upper Shelf Energy Equivalent Margins Analysis," December 8, 1993.

—, letter from Brian W. Sheron (NRC) to George L. Lehmann (Boiling Water Reactor Owners Group), "Safety Evaluation by the Office of Nuclear Reactor Regulation of B&W Owners' Group Topical Report BAW-2178P, on Upper Shelf Energy Equivalent Margins Analysis," March 29, 1994.

—, letter from Brian W. Sheron (NRC) to George L. Lehmann (Boiling Water Reactor Owners Group), "Safety Evaluation Report Low Upper-Shelf Toughness Fracture Analysis of Reactor Vessels of B&W Owners Group Reactor Vessel Working Group for Load Level A & B Conditions, BAW-2192P, Revision 1," March 29, 1994.

—, letter from R. Capra (NRC) to B. Ralph Sylvia (Niagara Mohawk Power Corporation), "Elastic-Plastic Fracture Mechanics Assessment of Nine Mile Point Nuclear Station Unit No. 1 Reactor Vessel Beltline Plates," April 20, 1994.

—, letter from Wayne Hodges (NRC) to William Rasin (Nuclear Management and Resources Council), "Safety Assessment of Report WCAP-13587, Revision 1, 'Reactor Vessel Upper Shelf Energy Bounding Evaluation for Westinghouse Pressurized Water Reactors,'" April 21, 1994.

—, letter from C. Shiraki (NRC) to D. L. Farrar (Commonwealth Edison Company), "Acceptance of the Submittal Entitled Low Upper-Shelf Toughness Fracture Analysis of Reactor Vessels of Zion Nuclear Power Station, Units 1 and 2 for Load Level A & B Conditions," May 11, 1994.

—, letter from L. Olshan (NRC) to P. M. Donnelly (Consumers Power Company), "Big Rock Point Plant—Response to Generic Letter (GL) 92-01, Revision 1, 'Reactor Vessel Structural Integrity, 10 CFR 50.54 (f),'" September 14, 1994.

—, letter from Brian W. Sheron (NRC) to Raymond Burski (Combustion Engineering Owners Group), "Safety Assessment of Report CEN-604, Revision 01, 'Final Evaluation of Low Upper Shelf Energy for Combustion Engineering Nuclear Steam Supply Systems Reactor Pressure Vessels, Final Report,'" September 15, 1994.

—, letter from B. Mozafari (NRC) to C. S. Hinnant (Carolina Power & Light Company), "Safety Evaluation of Low Upper Shelf Energy Equivalent Margins Analysis for the H.B. Robinson Steam Electric Plant, Unit No. 2," September 19, 1994.

—, NUREG-0224, "Reactor Vessel Pressure Transient Protection for Pressurized Water Reactors," September 1978.

—, NUREG-0744, "Resolution of the Reactor Vessel Materials Toughness Safety Issue," October 1982.

—, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," 1981.

—, NUREG/CR-4489, "Historical Summary of the Heavy-Section Steel Technology Program and Some Related Activities in Light-Water Reactor Pressure Vessel Safety Research," Oak Ridge National Laboratory, March 1986.

—, NUREG/CR-5265, "Size Effects on J-R Curves for A302-B Plate," Materials Engineering Associated, Inc., January 1989.

—, NUREG/CR-5729, "Multivariable Modeling of Pressure Vessel and Piping J-R Data," Modeling and Computing Services, May 1991.

—, NUREG/CR-6023, "Generic Analyses for Evaluation of Low Charpy Upper Shelf Energy Effects on Safety Margins Against Fracture of Reactor Pressure Vessel Materials," Oak Ridge National Laboratory, July 1993.

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

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

Westinghouse Electric Corporation, WCAP-13587, "Reactor Vessel Upper Shelf Energy Bounding Evaluation for Westinghouse Pressurized Water Reactors", Revision 1, Pittsburgh, Pennsylvania, September 1993.

Whitman, G. D., et al., "Technology of Steel Pressure Vessels for Water-Cooled Nuclear Reactors," ORNL-NSIC-21, Oak Ridge National Laboratory, Oak Ridge, Tennessee, December 1967.

APPENDIX A

OWNERS GROUPS AND THEIR PLANTS

BWROG Plants

Browns Ferry 1, 2, and 3
Brunswick 1 and 2
Cooper
Dresden 2 and 3
Duane Arnold
FitzPatrick
Grand Gulf 1
Hatch 1 and 2
Hope Creek
La Salle 1 and 2
Limerick 1 and 2
Millstone 1
Monticello
Oyster Creek
Peach Bottom 2 and 3
Quad Cities 1 and 2
Susquehanna 1 and 2
Vermont Yankee
Washington Nuclear Project 2

B&WOG Plants

Arkansas 1
Crystal River 3
Davis-Besse
Ginna
Oconee 1, 2, and 3
Point Beach 1 and 2
Surry 1 and 2
Three Mile Island 1
Turkey Point 3 and 4
Zion 1 and 2

WOG Plants

Beaver Valley 1 and 2
Callaway
Catawba 1 and 2
Comanche Peak 1 and 2
D.C. Cook 1 and 2
Diablo Canyon 1 and 2
Farley 1 and 2
Haddam Neck
H.B. Robinson 2
Indian Point 2 and 3
Kewaunee
McGuire 1 and 2
Millstone 3
North Anna 1 and 2
Prairie Island 1 and 2
Salem 1 and 2
Seabrook
Sequoyah 1 and 2
Shearon Harris
South Texas 1 and 2
Surry 1 and 2
Trojan
V.C. Summer
Vogtle 1 and 2
Watts Bar 1 and 2
Wolf Creek

CEOG Plants

Arkansas Nuclear One 2
Calvert Cliffs 1 and 2
Fort Calhoun
Maine Yankee
Millstone 2
Palisades
Palo Verde 1, 2, and 3
San Onofre 2 and 3
St. Lucie 1 and 2
Waterford 3

APPENDIX B

REACTOR VESSEL INTEGRITY DATABASE SUMMARY FILES

Plant Name: ANO-1

Docket Number: 50-313

NSSS Vendor: Babcock and Wilcox

Vessel Manufacturer: Babcock and Wilcox

Edition of ASME Code for Design: Summer 1967 Addenda to the 1965 ASME Code

Date of Commercial Operation: December 19, 1974

Date of License Expiration: May 20, 2014

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Upper shell axial welds WF-18, heat 8T1762
ID Fluence at EOL: 7.05E18 n/cm^2
Initial RT_{NDT}: -5°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 137°F
Margin: 68°F
RT_{pts} at EOL: 200°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Upper to lower shell circumferential weld WF-112, heat 406L44 and Upper shell axial welds WF-18, heat 8T1762
1/4T Fluence at EOL: 5.26E18 n/cm^2 for WF-112 and 3.95E18 n/cm^2 for WF-18
Initial USE: Heat-specific value not reported
Percent Drop at EOL: 33% for WF-112 and 28% for WF-18
USE at EOL: Not applicable since heat-specific value not reported
Date USE Screening Limit will be Exceeded: Not applicable since heat specific value not reported
Bases for Accepting the USE at EOL: Equivalent margin analyses were performed in Topical Reports BAW-2178P and BAW-2192P[1]

REFERENCES:

July 1, 1992, letter from J. J. Fisicaro (EO) to USNRC Document Control Desk, Subject: Response to Generic Letter 92-01, Revision 1, "Reactor Vessel Structural Integrity"

Babcock and Wilcox Report BAW 1803, Rev. 1, and BAW-2222

[1]Staff will review applicability of Topical Reports BAW-2178P and BAW-2192P.

Plant Name: ANO-2

Docket Number: 50-366

NSSS Vendor: Combustion Engineering

Vessel Manufacturer: Combustion Engineering

Edition of ASME Code for Design: Summer 1970 Addenda to the 1968 ASME Code

Date of Commercial Operation: March 26, 1980

Date of License Expiration: July 17, 2018

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate shell plate, heat C-8009-1
ID Fluence at EOL: 5.26E19 n/cm^2
Initial RT_{NDT}: -26°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 118°F
Margin: 34°F
RT_{pts} at EOL: 126°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Lower shell plate, heat C-8010-1
1/4T Fluence at EOL: 2.73E19 n/cm^2
Initial USE: 90 ft-lb
Percent Drop at EOL: 24.1%
USE at EOL: 68 ft-lb
Date USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Chemistry data per Paragraph C.1.2 of RG 1.99, Rev. 2

REFERENCES:

Chemical composition and Initial RT_{NDT}, are from July 1, 1992, letter from J. J. Fisicaro (EO) to USNRC Document Control Desk, Subject: Response to Generic Letter 92-01, Revision 1, "Reactor Vessel Structural Integrity"

Fluence data are from June 18, 1991, letter from J. W. Yelverton (EO) to USNRC

Initial USE data are from Table 5.2-5 and 5.2-16 of ANO-2 FSAR

Plant Name: Beaver Valley 1

Docket Number: 50-334

NSSS Vendor: Westinghouse

Vessel Manufacturer: Combustion Engineering

Edition of ASME Code for Design: Winter 1968 Addenda to the 1968 ASME Code

Date of Commercial Operation: October 1, 1976

Date of License Expiration: January 29, 2016

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Lower shell plate, heat B6903-1
ID Fluence at EOL: 2.68E19 n/cm^2
Initial RT_{NDT}: 27°F
Method of Determining Chemistry Factor: Surveillance data per Paragraph C.2.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 212°F
Margin: 34°F
RT_{pts} at EOL: 273°F
Date at which PTS Screening Limit will be exceeded: 2012

USE for the Limiting Beltline Material:

Limiting Beltline Material: Lower shell plate, heat B6903-1
1/4T Fluence at EOL: 1.86E19 n/cm^2
Initial USE: 80 ft-lb
Percent Drop at EOL: 30%
USE at EOL: 56 ft-lb
Date USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Surveillance data per Paragraph C.2.2 of RG 1.99, Rev. 2

REFERENCES:

Chemical composition, initial RT_{NDT} and initial USE data are from July 8, 1992, letter from J. D. Sieber (DQL) to USNRC Document Control Desk, Subject: Beaver Valley Power Station, Unit No. 1 and No. 2, Response to Generic Letter 92-01

PTS data reported in a September 6, 1994 letter from G.S. Thomas to USNRC - Fluence were calculated from data in September 6, 1994 letter

Plant Name: Beaver Valley 2

Docket Number: 50-412

NSSS Vendor: Westinghouse

Vessel Manufacturer: Combustion Engineering

Edition of ASME Code for Design: Summer 1972 Addenda to the 1971 ASME Code

Date of Commercial Operation: November 17, 1987

Date of License Expiration: May 27, 2027

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate shell plate, identification B9004-1 (no heat number)
ID Fluence at EOL: 6.20E19 n/cm^2
Initial RT_{NDT}: 60°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 63°F
Margin: 34°F
RT_{pts} at EOL: 157°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate shell plate, identification B9004-2
1/4T Fluence at EOL: 3.87E19 n/cm^2
Initial USE: 76 ft-lb
Percent Drop at EOL: 26.1%
USE at EOL: 56 ft-lb
Date USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Chemistry data per Paragraph C.1.2 of RG 1.99, Rev. 2

REFERENCES:

July 8, 1992, letter from J. D. Sieber (DQL) to USNRC Document Control Desk, Subject: Beaver Valley Power Station, Unit No. 1 and No. 2, Response to Generic Letter 92-01.

Table A-2 of WCAP-12406.

Plant Name: Big Rock Point

Docket Number: 50-155

NSSS Vendor: General Electric

Vessel Manufacturer: Combustion Engineering

Edition of ASME Code for Design: Section I of the 1959 ASME Code Utilizing Code Cases 1270N, 1271N, and 1273N

Date of Commercial Operation: March 29, 1963

Date of License Expiration: May 31, 2000

ART for the Limiting Beltline Material:

Limiting Beltline Material: Welds, no heat numbers
ID Fluence at EOL: 5.011E19 n/cm^2
Initial RT_{NDT}: 56°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 192°F
Margin: 66°F
ART at EOL: 202°F

USE for the Limiting Beltline Material:

Limiting Beltline Material: Welds, no heat numbers
1/4T Fluence at EOL: 3.623E19 n/cm^2
Initial USE: Heat-specific value not reported
Percent Drop at EOL: 51.0%
USE at EOL: Not applicable since heat-specific value not reported
Date USE Screening Limit will be Exceeded: Not applicable since heat specific value not reported
Bases for Accepting the USE at EOL: Plant-specific equivalent margin analyses submitted on May 20, 1994 was approved by the NRC

References:

The fluence and USE data are from February 25, 1993, letter to NRC.

Chemical composition and initial RT data are from July 1, 1992, letter from W. L. Beckman (CPCo) to USNRC Document Control Desk, Subject: Big Rock Point Response to GL 92-01, Rev. 1.

Base metal initial USE and orientation, and weld initial USE are from C. Z. Serpan, Jr. and H. E. Watson, "Mechanical Property and Neutron Spectral Analyses of the Big Rock Point Reactor Pressure Vessel," Nuclear Engineering and Design, 11 (1970), pp. 393-415

Plant Name: Braidwood 1

Docket Number: 50-456

NSSS Vendor: Westinghouse

Vessel Manufacturer: Babcock and Wilcox

Edition of ASME Code for Design: Summer 1973 Addenda to the 1971 ASME Code

Date of Commercial Operation: July 29, 1988

Date of License Expiration: October 17, 2028

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Middle circumferential weld, WF-562, heat 442011
ID Fluence at EOL: 3.03E19 n/cm^2
Initial RT_{ndt}: 40°F
Method of Determining Chemistry Factor: Chemistry data per paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{ndt} at EOL: 53°F
Margin: 53°F
RT_{pts} at EOL: 146°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Middle circumferential weld, WF-562, heat 442011
1/4T Fluence at EOL: 1.66E19 n/cm^2
Initial USE: 70 ft-lb
Percent Drop at EOL: 21.2%
USE at EOL: 55 ft-lb
Date at which USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Chemistry data per paragraph C.1.2 of RG 1.99, Rev. 2

References:

July 2, 1992, letter from M. A. Jackson (CECo) to T. E. Murley (USNRC), subject: Braidwood Station, Units 1 and 2; Byron Station, Units 1 and 2; Zion Station, Units 1 and 2

Plant Name: Braidwood 2

Docket Number: 50-457

NSSS Vendor: Westinghouse

Vessel Manufacturer: Babcock and Wilcox

Edition of ASME Code for Design: Summer 1973 Addenda to the 1971 ASME Code

Date of Commercial Operation: October 17, 1988

Date of License Expiration: December 18, 2027

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Middle circumferential weld, WF-562, heat 442011
ID Fluence at EOL: 3.03E19 n/cm^2
Initial RT_{ndt}: 40°F
Method of Determining Chemistry Factor: Chemistry data per paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{ndt} at EOL: 53°F
Margin: 53°F
RT_{pts} at EOL: 146°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Middle circumferential weld, WF-562
1/4T Fluence at EOL: 1.66E19 n/cm^2
Initial USE: 70 ft-lb
Percent Drop at EOL: 21.2%
USE at EOL: 55 ft-lb
Date at which USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Chemistry data per paragraph C.1.2 of RG 1.99, Rev. 2

References:

July 2, 1992, letter from M. A. Jackson (CECo) to T. E. Murley (USNRC), subject: Braidwood Station, Units 1 and 2; Byron Station, Units 1 and 2; Zion Station, Units 1 and 2

Plant Name: Browns Ferry 1

Docket Number: 50-259

NSSS Vendor: General Electric

Vessel Manufacturer: Babcock and Wilcox

Edition of ASME Code for Design: Summer 1965 Addenda to the 1965 ASME Code

Date of Commercial Operation: August 1, 1974

Date of License Expiration: December 20, 2013

ART for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate to lower shell circumferential weld, WF-154, heat 406L44
ID Fluence at EOL: 1.24E18 n/cm^2
Initial RT_{NDT}[1]: 20°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 91°F
Margin: 56°F
ART at EOL: 167°F

USE for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate to lower shell circumferential weld, WF-154, heat 406L44 and axial electroslag welds (ESW).
1/4T Fluence at EOL: 8.6E17 n/cm^2
Initial USE: Heat-specific value not reported
Percent Drop at EOL: 27% for WF-154 and 22.2% for ESW
USE at EOL: Not applicable since heat-specific value not reported
Date at which USE Screening Limit will be Exceeded: Not applicable since heat-specific value not reported
Bases for Accepting the USE at EOL: Equivalent margins analysis was performed in Topical Report NEDO-32205, Revision 1[2]

References:

Initial USE, fluence, initial RT, and chemical composition data are from July 7, 1992, letter from R. H. Shell (TVA) to USNRC Document Control Desk, Subject: Browns Ferry Nuclear Plant (BFN), Sequoyah Nuclear Plant (SQN), and Watts Bar Nuclear Plant (WBN)--Response to NRC Generic Letter 92-01 (Reactor Vessel Structural Integrity)

---

[1]Staff will review the forthcoming GE Topical Report on initial RT_{NDT} determination

[2]Staff will review applicability of Topical Report NEDO-32205, Rev. 1

Plant Name: Browns Ferry 2

Docket Number: 50-260

NSSS Vendor: General Electric

Vessel Manufacturer: Babcock and Wilcox, and Ishikawajima-Harima Heavy Industries

Edition of ASME Code for Design: Summer 1965 addenda to the 1965 ASME Code

Date of Commercial Operation: March 1, 1975

Date of License Expiration: June 28, 2014

ART for the Limiting Beltline Material:

Limiting Beltline Material: Axial electroslag welds
ID Fluence at EOL: 1.06E18 n/cm^2
Initial RT_{NDT}: 10°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 61°F
Margin: 56°F
ART at EOL: 127°F

USE for the Limiting Beltline Material:

Limiting Beltline Material: Axial eledctrolag welds
1/4T Fluence at EOL: 7.3E17 n/cm^2
Initial USE: Heat-specific value not reported
Percent Drop at EOL: 21.4%
USE at EOL: Not applicable since heat-specific value not reported
Date at which USE Screening Limit will be Exceeded: Not applicable since heat-specific value not reported
Bases for Accepting the USE at EOL: Equivalent margins analysis was performed in Topical Report NEDO-32205, Revision 1[1]

References:

Initial USE, fluence, initial RT, and chemical composition data are from July 7, 1992, letter from R. H. Shell (TVA) to USNRC Document Control Desk, Subject: Browns Ferry Nuclear Plant (BFN), Sequoyah Nuclear Plant (SQN), and Watts Bar Nuclear Plant (WBN)--Response to NRC Generic Letter 92-01 (Reactor Vessel Structural Integrity)

[1]Staff will review applicability of Topical Report NEDO-32205, Rev. 1

Plant Name: Browns Ferry 3

Docket Number: 50-296

NSSS Vendor: General Electric

Vessel Manufacturer: Babcock and Wilcox, and Ishikawajima-Harima Heavy Industries

Edition of ASME Code for Design: Summer 1966 Addenda to the 1965 ASME Code

Date of Commercial Operation: March 1, 1977

Date of License Expiration: July 2, 2016

ART for the Limiting Beltline Material:

Limiting Beltline Material: Axial electroslag welds
ID Fluence at EOL: 1.04E18 n/cm^2
Initial RT_{NDT}: 10°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 61°F
Margin: 56°F
ART at EOL: 127°F

USE for the Limiting Beltline Material:

Limiting Beltline Material: Axial electroslag welds
1/4T Fluence at EOL: 7.2E17 n/cm^2
Initial USE: Heat-specific value not reported
Percent Drop at EOL: 21.3%
USE at EOL: Not applicable since heat-specific value not reported
Date at which USE Screening Limit will be Exceeded: Not applicable since heat-specific value not reported
Bases for Accepting the USE at EOL: Equivalent margins analysis was performed in Topical Report NEDO-32205, Revision 1[1]

References:

Initial USE, fluence, initial RT, and chemical composition data are from July 7, 1992, letter from R. H. Shell (TVA) to USNRC Document Control Desk, Subject: Browns Ferry Nuclear Plant (BFN), Sequoyah Nuclear Plant (SQN), and Watts Bar Nuclear Plant (WBN)--Response to NRC Generic Letter 92-01 (Reactor Vessel Structural Integrity)

[1]Staff will review applicability of Topical Report NEDO-32205, Rev. 1

Plant Name: Brunswick 1

Docket Number: 50-325

NSSS Vendor: General Electric

Vessel Manufacturer: Chicago Bridge and Iron

Edition of ASME Code for Design: Summer 1967 Addenda to the 1965 ASME Code

Date of Commercial Operation: March 18, 1977

Date of License Expiration: September 8, 2016

ART for the Limiting Beltline Material:
- Limiting Beltline Material: Lower Intermediate Shell Course Plate, heat B8496-1
- ID Fluence at EOL: 1.5E18 n/cm^2
- Initial RT_{NDT}[1]: 10°F
- Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
- Increase in RT_{NDT} at EOL: 70°F
- Margin: 34°F
- ART at EOL: 114°F

USE for the Limiting Beltline Material:
- Limiting Beltline Material: Lower Intermediate Shell Course Plate, heat B8496-1
- 1/4T Fluence at EOL: 1.09E18 n/cm^2
- Initial USE: Heat-specific value not reported
- Percent Drop at EOL: 17%
- USE at EOL: Not applicable since heat-specific value not reported
- Date at which USE Screening Limit will be Exceeded: Not applicable since heat-specific value not reported
- Bases for Accepting the USE at EOL: Equivalent margins analysis was performed in Topical Report NEDO-32205, Revision 1[2]

References:

Fluence data are from May 13, 1994, letter from R.A. Anderson to the NRC. These data will appear in NEDO-24161, Rev. 1 and NEDO-24157, Rev. 2, which will be submitted to the NRC.

Chemical composition and initial RT data are from NEDO-24157.

---

[1] Staff will review the forthcoming GE Topical Report on initial RT_{NDT} determination

[2] Staff will review applicability of Topical Report NEDO-32205, Rev. 1

Plant Name: Brunswick 2

Docket Number: 50-324

NSSS Vendor: General Electric

Vessel Manufacturer: Chicago Bridge and Iron

Edition of ASME Code for Design: Summer 1967 Addenda to the 1965 ASME Code

Date of Commercial Operation: November 3, 1975

Date of License Expiration: December 27, 2014

ART for the Limiting Beltline Material:

Limiting Beltline Material: Lower Shell Course Plate, heat C4500-2
ID Fluence at EOL: 1.27E18 n/cm^2
Initial RT_{NDT}[1]: 10°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 50°F
Margin: 34°F
ART at EOL: 94°F

USE for the Limiting Beltline Material:

Limiting Beltline Material: Lower Shell Course Plate, heat C4500-2
1/4T Fluence at EOL: 9.25E17 n/cm^2
Initial USE: Heat-specific value not reported
Percent Drop at EOL: 13.5%
USE at EOL: Not applicable since heat-specific value not reported
Date at which USE Screening Limit will be Exceeded: Not applicable since heat-specific value not reported
Bases for Accepting the USE at EOL: Equivalent margins analysis was performed in Topical Report NEDO-32205, Revision 1[2]

References:

Fluence data are from May 13, 1994, letter from R.A. Anderson to the NRC. These data will appear in NEDO-24161, Rev. 1 and NEDO-24157, Rev. 2, which will be submitted to the NRC.

Chemical composition and initial RT data are from NEDO-24161.

---

[1]Staff will review the forthcoming GE Topical Report on initial RT_{NDT} determination

[2]Staff will review applicability of Topical Report NEDO-32205, Rev. 1

Plant Name: Byron 1

Docket Number: 50-454

NSSS Vendor: Westinghouse

Vessel Manufacturer: Babcock and Wilcox

Edition of ASME Code for Design: 1972 Summer Addenda to the 1971 ASME Code

Date of Commercial Operation: September 15, 1985

Date of License Expiration: October 31, 2024

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Upper shell forging, heat 5P-5933
ID Fluence at EOL: 2.159E19 n/cm^2
Initial RT_{ndt}: 40°F
Method of Determining Chemistry Factor: Chemistry data per paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{ndt} at EOL: 38°F
Margin: 34°F
RT_{pts} at EOL: 112°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Middle circumferential weld, WF-336, heat 442002
1/4T Fluence at EOL: 1.179E19 n/cm^2
Initial USE: 73 ft-lb
Percent Drop at EOL: 20%
USE at EOL: 58 ft-lb
Date at which USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Chemistry data per paragraph C.1.2 of RG 1.99, Rev. 2

References:

July 2, 1992, letter from M. A. Jackson (CECo) to T. E. Murley (USNRC), subject: Braidwood Station, Units 1 and 2; Byron Station, Units 1 and 2; Zion Station, Units 1 and 2.

Plant Name: Byron 2

Docket Number: 50-455

NSSS Vendor: Westinghouse

Vessel Manufacturer: Babcock and Wilcox

Edition of ASME Code for Design: 1972 Summer Addenda to the 1971 ASME Code

Date of Commercial Operation: August 21, 1987

Date of License Expiration: November 6, 2026

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Middle circumferential weld WF-447, heat 442002
ID Fluence at EOL: 3.03E19 n/cm^2
Initial RT_{ndt}: 10°F
Method of Determining Chemistry Factor: Chemistry data per paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{ndt} at EOL: 88°F
Margin: 56°F
RT_{pts} at EOL: 154°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Middle circumferential weld WF-447, heat 442002
1/4T Fluence at EOL: 1.66E19 n/cm^2
Initial USE: 67 ft-lb
Percent Drop at EOL: 21.9%
USE at EOL: 52 ft-lb
Date at which USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Chemistry data per paragraph C.1.2 of RG 1.99, Rev. 2

References:

Chemical compositions, IRT_{ndt}, initial USE data are from July 2, 1992, letter from M. A. Jackson (CECo) to T. E. Murley (USNRC), subject: Braidwood Station, Units 1 and 2; Byron Station, Units 1 and 2; Zion Station, Units 1 and 2.

Plant Name: Callaway 1

Docket Number: 50-483

NSSS Vendor: Westinghouse

Vessel Manufacturer: Combustion-Engineering

Edition of ASME Code for Design: Winter 1972 Addenda to the 1971 ASME Code

Date of Commercial Operation: December 19, 1984

Date of License Expiration: October 18, 2024

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Plate R2708-1, heat C4499-2
ID Fluence at EOL: 2.39E19 n/cm^2
Initial RT_{ndt}: 50°F
Method of Determining the Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{ndt} at EOL: 29°F
Margin: 34°F
RT_{pts} at EOL: 113°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Plate R2707-1, heat C4344-1
1/4T Fluence at EOL: 1.31E19 n/cm^2
Initial USE: 78 ft-lb
Percent Drop at EOL: 20%
USE at EOL: 62 ft-lb
Date at which USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Chemistry data per paragraph C.1.2 of RG 1.99, Rev. 2

References:

Table A-3 of WCAP-11374 and December 18, 1991, letter from D. F. Schnell (UECo) to USNRC Document Control Desk, subject: Revision to Technical Specification 3/4.4.9 Pressure Temperature Limits

June 13, 1994 letter from D. F. Schnell (Union Electric) to USNRC, subject: Callaway Plant Reactor Vessel Structural Integrity.

Plant Name: Calvert Cliffs 1

Docket Number: 50-317

NSSS Vendor: Combustion Engineering

Vessel Manufacturer: Combustion Engineering

Edition of ASME Code for Design: Winter 1967 Addenda to the 1965 ASME Code

Date of Commercial Operation: May 8, 1975

Date of License Expiration: July 31, 2014

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Lower shell axial weld 3-203A, B, C heat 21935
ID Fluence at EOL: 4.56E19 n/cm^2
Initial RT_{NDT}: -56°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 247°F
Margin: 66°F
RT_{pts} at EOL: 257°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Lower shell course plate D-7207-1, heat C-4420-1
1/4T Fluence at EOL: 2.72E19 n/cm^2
Initial USE: 77 ft-lb
Percent Drop at EOL: 28%
USE at EOL: 55 ft-lb
Date USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Chemistry data per Paragraph C.1.2 of RG 1.99, Rev. 2

REFERENCES:

December 12, 1991; June 30, 1992; February 16, 1993; and October 8, 1993, letters from the licensee (BG&E) to USNRC

Plant Name: Calvert Cliffs 2

Docket Number: 50-318

NSSS Vendor: Combustion Engineering

Vessel Manufacturer: Combustion Engineering

Edition of ASME Code for Design: Winter 1967 Addenda to the 1965 ASME Code

Date of Commercial Operation: April 1, 1977

Date of License Expiration: August 13, 2016

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate Shell Plate D-8907-2, heat C-5286-1
ID Fluence at EOL: 4.28E19 n/cm^2
Initial RT_{NDT}: 20°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 139°F
Margin: 34°F
RT_{pts} at EOL: 193°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material[1]:

Limiting Beltline Material: Intermediate shell courses D-8906-1, heat A-4463-1 and D-8906-3, heat A-4463-2
1/4T Fluence at EOL: 2.55E19 n/cm^2
Initial USE: 77 ft-lb for D-8906-1 and 75 ft-lb for D-8906-3
Percent Drop at EOL: 30% for D-8906-1 and 28% for D-8906-3
USE at EOL: 54 ft-lb
Date Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Chemistry data per Paragraph C.1.2 of RG 1.99, Rev. 2

REFERENCES:

December 12, 1991; June 30, 1992; February 16, 1993; October 8, 1993; and May 11, 1994, letters from the licensee (BG&E) to USNRC.

[1]Additional information will be provided by the CEOG in 1995 to further demonstrate the weld heat A-8746 has irradiated USE of greater than 50 ft-lb at EOL.

Plant Name: Catawba 1

Docket Number: 50-413

NSSS Vendor: Westinghouse

Vessel Manufacturer: Rotterdam Dockyard

Edition of ASME Code for Design: Winter 1971 Addenda to the 1971 ASME Code

Date of Commercial Operation: June 29, 1985

Date of License Expiration: December 6, 2024

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate Shell Forging 05, heat 411343
ID Fluence at EOL: 2.52E19 n/cm^2
Initial RT_{ndt}: -8°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{ndt} at EOL: 72°F
Margin: 34°F
RT_{pts} at EOL: 98°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Circumferential Weld W05, heat 895075
1/4T Fluence at EOL: 1.52E19 n/cm^2
Initial USE: 128 ft-lb
Percent Drop at EOL: 21%
USE at EOL: 101 ft-lb
Date at which USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Chemistry data per Paragraph C.1.2 of RG 1.99, Rev. 2

References:

Chemical composition and initial USE data are from July 3, 1992, letter from H. B. Tucker (DPCo) to USNRC Document Control Desk, subject: McGuire Nuclear Station, Units 1 and 2; Catawba Nuclear Station Units 1 and 2; Oconee Nuclear Station, Units 1, 2, and 3: Response to Generic Letter 92-01, Revision 1, Reactor Vessel Structural Integrity

Fluence data are from Table 6-11 of WCAP-11527

Plant Name: Catawba 2

Docket Number: 50-414

NSSS Vendor: Westinghouse

Vessel Manufacturer: Combustion Engineering

Edition of ASME Code for Design: Winter 1972 Addenda to the 1971 ASME Code

Date of Commercial Operation: August 19, 1986

Date of License Expiration: February 24, 2026

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate shell plate identification B8605-2 (no heat numbers)
ID Fluence at EOL: 2.46E19 n/cm^2
Initial RT_{ndt}: 33°F
Method of Determining Chemistry factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{ndt} at EOL: 55°F
Margin: 34°F
RT_{pts} at EOL: 122°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate shell plate identification B8605-2 (no heat numbers)
1/4T Fluence at EOL: 1.48E19 n/cm^2
Initial USE: 82 ft-lb
Percent Drop at EOL: 21%
USE at EOL: 65 ft-lb
Date at which USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Chemistry data per Paragraph C.1.2 of RG 1.99, Rev. 2

References:

Chemical composition and USE data are from Table A-1 of WCAP-11941.

Fluence data are from Table 6-13 of WCAP-11941.

Plant Name: Clinton

Docket Number: 50-461

NSSS Vendor: General Electric

Vessel Manufacturer: Chicago Bridge and Iron

Edition of ASME Code for Design: Summer 1973 Addenda to the 1971 ASME Code

Date of Commercial Operation: November 24, 1987

Date of License Expiration: September 29, 2026

ART for the Limiting Beltline Material:

Limiting Beltline Material: Welds, heat 76492
ID Fluence at EOL: 6.9E18 n/cm^2
Initial RT_{NDT}: -30°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 120°F
Margin: 56°F
ART at EOL: 146°F

USE for the Limiting Beltline Material:

Limiting Beltline Material: Shell course 1 plate, heat A2740-1
1/4T Fluence at EOL: 4.65E17 n/cm^2
Initial USE: 67 ft-lb
Percent Drop at EOL: 9.2%
USE at EOL: 61 ft-lb
Date at which USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Chemistry data per Paragraph C.1.2 of RG 1.99, Rev. 2

References:

Fluence and Ni contents for plate A2740-1 are from October 18, 1990, letter from J. B. Hickman (USNRC) to F. A. Spangenberg (Illnois Power), Subject: Issuance of Amendment No. 51

All other data are from Clinton USAR

July 2, 1992 letter from J. S. Perry (Illinois Power) to the USNRC, subject: Response to Generic Letter 92-01.

Plant Name: Comanche Peak 1

Docket Number: 50-445

NSSS Vendor: Westinghouse

Vessel Manufacturer: Combustion Engineering

Edition of ASME Code for Design: Summer 1972 Addenda to the 1971 ASME Code

Date of Commercial Operation: August 13, 1990

Date of License Expiration: February 8, 2030

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Lower shell plate R1108-1, heat C4464-1
ID Fluence at EOL: 3.04E19 n/cm^2
Initial RT_{NDT}: 0°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 66°F
Margin: 34°F
RT_{pts} at EOL: 100°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Lower shell plate R1108-2, heat C4533-2
1/4T Fluence at EOL: 1.812E19 n/cm^2
Initial USE: 78 ft-lb
Percent Drop at EOL: 21.8%
USE at EOL: 61 ft-lb
Date at which USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Chemistry data per Paragraph C.1.2 of RG 1.99, Rev 2

References:

Initial USE data for beltline materials other than the surveillance materials are from Comanche Peak Unit 1 FSAR

The rest of the information on beltline materials is from WCAP-13422.

Plant Name: Comanche Peak 2

Docket Number: 50-446

NSSS Vendor: Westinghouse

Vessel Manufacturer: Combustion Engineering

Edition of ASME Code for Design: Summer 1972 Addenda to the 1971 ASME Code

Date of Commercial Operation: August 3, 1993

Date of License Expiration: February 2, 2033

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate shell plate R3807-2, heat 5522-2
ID Fluence at EOL: 3.04E19 n/cm^2
Initial RT_{NDT}: 10°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 48°F
Margin: 34°F
RT_{pts} at EOL: 92°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Circumferential weld, heat 89833
1/4T Fluence at EOL: 1.812E19 n/cm^2
Initial USE: 96 ft-lb
Percent Drop at EOL: 21.8%
USE at EOL: 75 ft-lb
Date USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Chemistry data per Paragraph C.1.2 of RG 1.99, Rev. 2

References:

Fluence data are from WCAP-13422.

The rest of the information on beltline materials is from WCAP-10684

Plant Name: Cook 1

Docket Number: 50-315

NSSS Vendor: Westinghouse

Vessel Manufacturer: Combustion Engineering

Edition of ASME Code for Design: Winter 1966 Addenda to the 1965 ASME Code

Date of Commercial Operation: August 28, 1975

Date of License Expiration: October 25, 2014

RT_{pts} for the Limiting Beltline Material:
  Limiting Beltline Material: Intermediate shell axial welds 2-442 and lower shell axial welds 3-442, heats 13253/12008
  ID Fluence at EOL: 9.5E18 n/cm^2
  Initial RT_{NDT}: -56°F
  Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
  Increase in RT_{NDT} at EOL: 204°F
  Margin: 66°F
  RT_{pts} at EOL: 214°F
  Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:
  Limiting Beltline Material: Intermediate to lower shell circumferential welds 9-442, heat 1P3571
  1/4T Fluence at EOL: 8.47E18 n/cm^2
  Initial USE: 105 ft-lb
  Percent Drop at EOL: 43%
  USE at EOL: 60 ft-lb
  Date at which USE Screening Limit will be Exceeded: After EOL
  Bases for Accepting the USE at EOL: Chemistry data per Paragraph C.1.2 of RG 1.99, Rev. 2

References:

Fluence data were revised based on the May 20, 1994, letter to the NRC.

Initial USE, and chemical composition data are from July 13, 1992, letter from E. E. FitzPatrick (IMPCo) to USNRC Document Control Desk, Subject: Donald C. Cook Nuclear Power Plant, Units 1 and 2, Generic Letter 92-01, Revision 1, Reactor Vessel Structural Integrity and from November 29, 1993 letter to USNRC.

Plant Name: Cook 2

Docket Number: 50-316

NSSS Vendor: Westinghouse

Vessel Manufacturer: Chicago Bridge and Iron

Edition of ASME Code for Design: Summer 1968 Addenda to the 1968 ASME Code

Date of Commercial Operation: July 1, 1978

Date of License Expiration: December 23, 2017

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate shell plate, heat C5556-2
ID Fluence at EOL: 1.71E19 n/cm^2
Initial RT_{NDT}: 58°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 122°F
Margin: 34°F
RT_{pts} at EOL: 214°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate shell plate, heat C5521-2
1/4T Fluence at EOL: 1.02E19 n/cm^2
Initial USE: 86 ft-lb
Percent Drop at EOL: 32.0%
USE at EOL: 58 ft-lb
Date at which USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Surveillance data per Paragraph C.2.2 of RG 1.99, Rev. 2

References:

Initial USE and chemical composition data are found in July 13, 1992, letter from E. E. Fitzpatrick (IMPCo) to USNRC Document Control Desk, Subject: Donald C. Cook Nuclear Power Plant, Units 1 and 2, Generic Letter 92-01, Revision 1, Reactor Vessel Structural Integrity.

Fluence and material data are from April 12, 1993, letter to the NRC.

Plant Name: Cooper

Docket Number: 50-298

NSSS Vendor: General Electric

Vessel Manufacturer: Combustion Engineering

Edition of ASME Code for Design: Winter 1966 Addenda to the 1965 ASME Code

Date of Commercial Operation: July 1, 1974

Date of License Expiration: December 18, 2014

ART for the Limiting Beltline Material:

Limiting Beltline Material: Lower shell plate, heat C2307-2
ID Fluence at EOL: 1.5E18 n/cm^2
Initial RT_{NDT} [1]: -20°F
Method of Determining Chemistry Factor: Surveillance data per Paragraph C.2.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 132°F
Margin: 17°F
ART at EOL: 129°F

USE for the Limiting Beltline Material:

Limiting Beltline Material: Lower shell axial welds 2-233 A, B, C, heat 12420
1/4T Fluence at EOL: 1.08E18 n/cm^2
Initial USE: Heat-specific value not reported
Percent Drop at EOL: 22%
USE at EOL: Not applicable since heat-specific value not reported
Date at which USE Screening Limit will be Exceeded: Not applicable since heat-specific value not reported
Bases for Accepting the USE at EOL: Equivalent margins analysis was performed in the Topical Report NEDO-32205, Revision 1[2]

References:

Chemical composition and initial RT are from February 25, 1993, letter from G.R. Horn (NPPD) to USNRC.

July 1, 1992, letter from G. R. Horn (NPPD) to USNRC Document Control Desk, Subject: Response to Generic Letter 92-01, Revision 1.

Plate initial USE data are from Table 7-1 of NDE-103-0986, which evaluated surveillance capsule 1.

---

[1]Staff will review the forthcoming GE Topical Report on initial RT_{NDT} determination.

[2]Staff will review applicability of Topical Report NEDO-32205, Rev. 1.

Plant Name: Crystal River 3

Docket Number: 50-302

NSSS Vendor: Babcock and Wilcox

Vessel Manufacturer: Babcock and Wilcox

Edition of ASME Code for Design: Summer 1967 Addenda to the 1965 ASME Code

Date of Commercial Operation: March 13, 1977

Date of License Expiration: December 3, 2016

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Upper shell axial welds WF-8 and WF-18, heat 8T1762
ID Fluence at EOL: 7.96E18 n/cm^2
Initial RT_{NDT}: -5°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 143°F
Margin: 68°F
RT_{pts} at EOL: 206°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Upper shell axial welds WF-8 and WF-18 heat 8T1762 and Upper to lower circumferential weld WF-70, heat 72105
1/4T Fluence at EOL: 4.60E18 n/cm^2 for WF-70 and 4.46E18 n/cm^2 for WF-8 and WF-18
Initial USE: Heat-specific value not reported
Percent Drop at EOL: 33% for WF-70 and 27% for WF-8 and WF-18
USE at EOL: Not applicable since heat-specific value was not reported
Date USE Screening Limit will be Exceeded: Not applicable since heat specific value was not reported
Bases for Accepting the USE at EOL: Equivalent Margin analyses were performed in Topical Reports BAW-2178P and BAW-2192P[1]

REFERENCES:

Fluence, chemistry composition and initial USE are from reports BAW-2166 and BAW-2222.

---

[1]Staff will review applicability of Topical Reports BAW-2178P and BAW-2192P.

Plant Name: Davis-Besse

Docket Number: 50-346

NSSS Vendor: Babcock and Wilcox

Vessel Manufacturer: Babcock and Wilcox

Edition of ASME Code for Design: Summer 1968 Addenda to the 1968 ASME Code

Date of Commercial Operation: July 31, 1978

Date of License Expiration: April 22, 2017

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Upper to lower shell circumferential weld, WF-182-1, heat 821T44
ID Fluence at EOL: 1.07E19 n/cm^2
Initial RT_{NDT}: 2°F
Method of Determining Chemistry Factor: Surveillance data per Paragraph C.2.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 165°F
Margin: 28°F
RT_{pts} at EOL: 195°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Nozzle belt to upper shell circumferential weld, WF-233, heat T29744
1/4T Fluence at EOL: 9.0E17 n/cm^2
Initial USE: Heat-specific value not reported
Percent Drop at EOL: 24%
USE at EOL: Not applicable since heat-specific value was not reported
Date USE Screening Limit will be Exceeded: Not applicable since heat specific value was not reported
Bases for Accepting the USE at EOL: Equivalent margin analyses were performed in Topical Reports BAW-2198P and BAW-2192P[1]

REFERENCES:

July 1, 1992, letter from D. C. Shelton (TECo) to USNRC Document Control Desk, Subject: Response to NRC Generic Letter 92-01, Revision 1, Reactor Vessel Structural Integrity, for the Davis-Besse Nuclear Power Station

Fluence, chemical composition and initial USE are from reports BAW-2166 and BAW-2222.

---

[1]Staff will review applicability of Topical Reports BAW-2178P and BAW-2192P.

Plant Name: Diablo Canyon 1

Docket Number: 50-275

NSSS Vendor: Westinghouse

Vessel Manufacturer: Combustion Engineering

Edition of ASME Code for Design: 1968 ASME Code

Date of Commercial Operation: May 17, 1985

Date of License Expiration: April 23, 2008, undergoing operating license recapture to 2021

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Lower shell axial weld 3-442C, heat 27204
ID Fluence at EOL: 1.32E19 n/cm^2
Initial RT_{NDT}: -56°F
Method of Determining Chemistry Factor: Surveillance data per Paragraph C.2.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 228°F
Margin: 44°F
RT_{pts} at EOL: 216°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate to lower shell circumferential weld 9-442, heat 21935
1/4T Fluence at EOL: 7.93E18 n/cm^2
Initial USE: 75 ft-lb
Percent Drop at EOL: 31%
USE at EOL: 52 ft-lb
Date at which USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Chemistry data per Paragraph C.1.2 of RG 1.99, Rev. 2

References:

Calculated chemistry factors are from July 25, 1994, letter to the NRC.

Initial USE for circ. weld 9-442 is from S. Bloom of USNRC to T.L. Patterson of Omaha Public Power District, dated December 3, 1993).

Fluence (based on the EOL date of 9/22/2021) and initial USEs are from December 4, 1992, letter from G. M. Rueger (PG&E) to USNRC, Subject: Response to Generic Letter 92-01, Revision 1, "Reactor Vessel Structural Integrity"--Supplemental Information.

Initial RT_{NDT} and chemical composition data are from June 30, 1992, letter from G. M. Rueger (PG&E) to USNRC, response to GL 92-01, Rev. 1.

Plant Name: Diablo Canyon 2

Docket Number: 50-323

NSSS Vendor: Westinghouse

Vessel Manufacturer: Combustion Engineering

Edition of ASME Code for Design: 1968 ASME Code

Date of Commercial Operation: March 13, 1986

Date of License Expiration: December 9, 2010, undergoing operating license recapture to 2025

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate shell plate, heat C5168-2
ID Fluence at EOL: 1.46E19 n/cm^2
Initial RT_{NDT}: 67°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 110°F
Margin: 34°F
RT_{pts} at EOL: 211°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Lower shell axial welds 3-201A, C, heat 33A277
1/4T Fluence at EOL: 6.08E18 n/cm^2
Initial USE: 88 ft-lb
Percent Drop at EOL: 36%
USE at EOL: 56 ft-lb
Date at which USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Chemistry data per Paragraph C.1.2 of RG 1.99, Rev. 2.

References:

Fluence (based on the EOL date of 4/26/2025) and initial USEs are from December 4, 1992, letter from G.M. Rueger (PG&E) to USNRC Document Control Desk, Subject: Response to Generic Letter 92-01, Revision 1, "Reactor Vessel Structural Integrity"--Supplemental Information.

Initial RT_{NDT} and chemical composition data are from June 30, 1992, letter from G. M. Rueger (PG&E) to USNRC Document Control Desk, Subject: Response to Generic Letter 92-01, Revision 1, Reactor Vessel Structural Integrity

Plant Name: Dresden 2

Docket Number: 50-237

NSSS Vendor: General Electric

Vessel Manufacturer: New York Shipbuilding

Edition of ASME Code for Design: Summer 1964 Addenda to the 1963 ASME Code

Date of Commercial Operation: June 9, 1970

Date of License Expiration: January 10, 2006

ART for the Limiting Beltline Material:
- Limiting Beltline Material: Lower intermediate shell plate, heat B4065-1
- ID Fluence at EOL: 3.6E17 n/cm^2
- Initial RT_{NDT}[1]: 20°F
- Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
- Increase in RT_{NDT} at EOL: 38°F
- Margin: 34°F
- ART at EOL: 92°F

USE for the Limiting Beltline Material:
- Limiting Beltline Material: Lower shell axial welds, heat 1P0815
- 1/4T Fluence at EOL: 2.5E17 n/cm^2
- Initial USE: Heat-specific value not reported
- Percent Drop at EOL: 17%
- USE at EOL: Not applicable since heat-specific value not reported
- Date at which USE Screening Limit will be Exceeded: Not applicable since heat-specific value not reported
- Bases for Accepting the USE at EOL: Equivalent margins analysis was performed in Topical Report NEDO-32205, Revision 1[2]

References:

Information on material types is from August 11, 1994, letter to the NRC.

Chemical composition, initial USE, initial RT, and fluence data are from July 1, 1992, letter from M. A. Jackson (CECo) to T. E. Murley (USNRC), Subject: Dresden Station Units 2 and 3; Quad Cities Station Units 1 and 2; LaSalle County Station Units 1 and 2.

---

[1]Staff will review the forthcoming GE Topical Report on initial RT_{NDT} determination

[2]Staff will review applicability of Topical Report NEDO-32205, Rev. 1

Plant Name: Dresden 3

Docket Number: 50-249

NSSS Vendor: General Electric

Vessel Manufacturer: Babcock and Wilcox

Edition of ASME Code for Design: Summer 1965 Addenda to the 1965 ASME Code

Date of Commercial Operation: November 16, 1971

Date of License Expiration: January 12, 2011

ART for the Limiting Beltline Material:

Limiting Beltline Material: Lower intermediate and lower shell axial welds, heat PQ-1300
ID Fluence at EOL: 5.1E17 n/cm^2
Initial RT_{NDT}: 40°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 47°F
Margin: 47°F
ART at EOL: 134°F

USE for the Limiting Beltline Material:

Limiting Beltline Material: Lower intermediate and lower shell axial welds, heat PQ-1300
1/4T Fluence at EOL: 3.5E17 n/cm^2
Initial USE: Heat-specific value not reported
Percent Drop at EOL: 20.5%
USE at EOL: Not applicable since heat-specific value not reported
Date at which USE Screening Limit will be Exceeded: Not applicable since heat-specific value not reported
Bases for Accepting the USE at EOL: Equivalent margins analysis was performed in Topical Report NEDO-32205, Revision 1[1]

References:

Information on material types is from August 11, 1994, letter to the NRC.

The initial RT for electroslag welds (heat PQ1300) and Linde 80 welds are from September 24, 1993, letter to the NRC.

Chemical composition, initial USE, initial RT, and fluence data are from July 1, 1992, letter from M. A. Jackson (CECo) to T. E. Murley (USNRC), Subject: Dresden Station Units 2 and 3; Quad Cities Station Units 1 and 2; LaSalle County Station Units 1 and 2

---

[1]Staff will review applicability of Topical Report NEDO-32205, Rev. 1

Plant Name: Duane Arnold

Docket Number: 50-331

NSSS Vendor: General Electric

Vessel Manufacturer: Chicago Bridge and Iron

Edition of ASME Code for Design: Summer 1967 Addenda to the 1965 ASME Code

Date of Commercial Operation: February 1, 1975

Date of License Expiration: February 21, 2014

ART for the Limiting Beltline Material:
- Limiting Beltline Material: Lower shell plate, heat B0402-1
- ID Fluence at EOL: 4.7E18 n/cm^2
- Initial RT_{NDT}[1]: 40°F
- Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
- Increase in RT_{NDT} at EOL: 68°F
- Margin: 34°F
- ART at EOL: 142°F

USE for the Limiting Beltline Material:
- Limiting Beltline Material: Lower shell plate, heat B0402-1 and circ. welds
- 1/4T Fluence at EOL: 3.6E18 n/cm^2
- Initial USE: Heat-specific value not reported
- Percent Drop at EOL: 17.5% (B0402-1), 15% (Circ. welds)
- USE at EOL: Not applicable since heat-specific value not reported
- Date at which USE Screening Limit will be Exceeded: Not applicable since heat-specific value not reported
- Bases for Accepting the USE at EOL: Equivalent margins analysis was performed in Topical Report NEDO-32205, Revision 1[2]

References:

Chemical composition, initial USE, initial RT, and fluence data are from July 6, 1992, letter from J. F. Franz, Jr. (IEL&P) to T. E. Murley (USNRC), response to NRC GL 92-01, Rev. 1.

---

[1]Staff will review the forthcoming GE Topical Report on initial RT_{NDT} determination

[2]Staff will review applicability of Topical Report NEDO-32205, Rev. 1

Plant Name: Farley 1

Docket Number: 50-348

NSSS Vendor: Westinghouse

Vessel Manufacturer: Combustion Engineering

Edition of ASME Code for Design: Summer 1970 Addenda to the 1968 ASME Code

Date of Commercial Operation: December 1, 1977

Date of License Expiration: June 25, 2017

RT_{PTS} for the Limiting Beltline Material:

Limiting Beltline Material: Lower shell plate B6919-2, heat C6940-1
ID Fluence at EOL: 3.75E19 n/cm^2
Initial RT_{NDT}: 5°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 131°F
Margin: 34°F
RT_{PTS} at EOL: 170°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Circ. weld, heat 6329637 and lower shell axial welds, heat 90099
1/4T Fluence at EOL: 2.34E19 n/cm^2 for circ. weld and 7.73E18 n/cm^2 for lower shell axial welds
Initial USE: 104 ft-lb for circ. weld and 83 ft-lb for lower shell axial welds
Percent Drop at EOL: 44.2% for circ. weld and 29.7% for lower shell axial welds
USE at EOL: 58 ft-lb for both circ. weld and lower shell axial welds
Date at which USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Chemistry data per Paragraph C.1.2 of RG 1.99, Rev. 2

References:

Fluence, initial RT_{NDT}, and chemistry values are from November 23, 1993, letter from D. Morey (SNOC)to USNRC Control Desk, Subject: Responses to Requests for Additional Information Regarding GL 92-01.

Initial USE values are from June 21, 1994, letter from J.D. Woodard to USNRC Control Desk, Subject: Responds to Open Issues GL 92-01.

Plant Name: Farley 2

Docket Number: 50-364

NSSS Vendor: Westinghouse

Vessel Manufacturer: Combustion Engineering

Edition of ASME Code for Design: Summer 1970 Addenda to the 1968 ASME Code

Date of Commercial Operation: July 30, 1981

Date of License Expiration: March 31, 2021

RT_{PTS} for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate shell plate B7212-1, heat C7466-1
ID Fluence at EOL: 3.8E19 n/cm^2
Initial RT_{NDT}: -10°F
Method of Determining Chemistry Factor: Surveillance data per Paragraph C.2.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 195°F
Margin: 17°F
RT_{PTS} at EOL: 202°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate shell plate B7212-1, heat C7466-1
1/4T Fluence at EOL: 2.369E19 n/cm^2
Initial USE: 100 ft-lb
Percent Drop at EOL: 38%
USE at EOL: 62 ft-lb
Date at which USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Surveillance data per Paragraph C.2.2 of RG 1.99, Rev. 2

References:

Fluence, initial USE, and initial RT_{NDT} are from November 23, 1993, letter from D. Morey (SNOC) to USNRC Document Control Desk, Subject: Responses to Requests for Additional Information Regarding GL 92-01.

Plant Name: Fermi 2

Docket Number: 50-341

NSSS Vendor: General Electric

Vessel Manufacturer: Combustion Engineering

Edition of ASME Code for Design: Summer 1969 Addenda to the 1968 ASME Code

Date of Commercial Operation: January 23, 1988

Date of License Expiration: March 20, 2025

ART for the Limiting Beltline Material:

Limiting Beltline Material: Lower shell axial welds 2-307A, B, C, heats 13253/12008
ID Fluence at EOL: 6.5E17 n/cm^2
Initial RT_{NDT}: -44°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 75°F
Margin: 56°F
ART at EOL: 87°F

USE for the Limiting Beltline Material:

Limiting Beltline Material: Lower shell axial welds 2-307A, B, C, heat 13253/12008
1/4T Fluence at EOL: 4.5E17 n/cm^2
Initial USE: 75 ft-lb
Percent Drop at EOL: 19.6%
USE at EOL: 60 ft-lb
Date at which USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Chemistry data per Paragraph C.1.2 of RG 1.99, Rev. 2

References:

Initial USE for the lower shell axial welds is a generic value for welds fabricated by Combustion Engineering using Linde 1092, 0091, 123 and Arcos B-5 fluxes. (Ref. Letter from S. Bloom (USNRC) to T.L. Patterson (OPPD) dated December 3, 1993).

Fluence, initial RT, and chemical composition data are from June 30, 1992, letter from W. S. Orser (DE) to USNRC Document Control Desk, Subject: Detroit Edison Response to Generic Letter 92-01, Rev. 1.

Plant Name: James A. FitzPatrick

Docket Number: 50-333

NSSS Vendor: General Electric

Vessel Manufacturer: Combustion Engineering

Edition of ASME Code for Design: Winter 1966 Addenda to the 1965 ASME Code

Date of Commercial Operation: July 28, 1975

Date of License Expiration: October 17, 2014

ART for the Limiting Beltline Material:
- Limiting Beltline Material: Lower shell axial welds 2-233A, B, C, heats 27204/12008
- ID Fluence at EOL: 1.96E18 n/cm^2
- Initial RT_{NDT}[1]: -22°F
- Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
- Increase in RT_{NDT} at EOL: 135°F
- Margin: 56°F
- ART at EOL: 169°F

USE for the Limiting Beltline Material:
- Limiting Beltline Material: Circumferential weld, 1-240, heat 305414 and Lower shell axial welds, 2-233 A, B, C, heats 27204/12008
- 1/4T Fluence at EOL: 1.7E18 n/cm^2 for welds 1-240 and 1.3E18 n/cm^2 for welds 2-233 A, B, C
- Initial USE: Heat-specific value not reported
- Percent Drop at EOL: 31.5% (1-240) and 24.7% (2-233 A, B, C)
- USE at EOL: Not applicable since heat-specific value not reported
- Date at which USE Screening Limit will be Exceeded: Not applicable since heat-specific value not reported
- Bases for Accepting the USE at EOL: Equivalent margins analysis was performed in Topical Report NEDO-32205, Revision 1[2]

References:

Fluence data were revised based on the April 29, 1994, letter to the NRC. Later correspondence with the licensee indicates that they were from GE report "Implementation of RG 1.99, Rev. 2 for James A. FitzPatrick Nuclear Power Plant," June, 1989.

Initial RT and chemical compositions are from July 9, 1992, letter from R. E. Beedle (PASNY) to USNRC, Subject: Response to GL 92-01, Rev. 1. Plate initial USE data are from May 15, 1978, letter to the NRC.

---

[1]Staff will review the forthcoming GE Topical Report on initial RT_{NDT} determination

[2]Staff will review applicability of Topical Report NEDO-32205, Rev. 1

Plant Name: Fort Calhoun

Docket Number: 50-285

NSSS Vendor: Combustion Engineering

Vessel Manufacturer: Combustion Engineering

Edition of ASME Code for Design: Winter 1967 Addenda to the 1965 ASME Code

Date of Commercial Operation: June 20, 1974

Date of License Expiration: August 9, 2013

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Lower Shell Axial Welds 3-410 A, B, C, heat 27204
ID Fluence at EOL: 1.49E19 n/cm^2
Initial RT_{NDT}: -56°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 254°F
Margin: 66°F
RT_{pts} at EOL: 264°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Lower Shell Axial Welds 3-410 A, B, C, heats 13253/12008
1/4T Fluence at EOL: 0.97E19 n/cm^2
Initial USE: 75 ft-lb
Percent Drop at EOL: 35%
USE at EOL: 49 ft-lb
Date USE Screening Limit will be Exceeded: Before EOL
Bases for Accepting the USE at EOL: Equivalent margin analyses approved by staff in a December 3, 1993, letter from S. Bloom (USNRC) to T.L. Patterson (OPPD)

REFERENCES:

Chemical composition and Initial RT_{NDT} are from July 6, 1992, letter from W. G. Gates (OPPD) to USNRC Document Control Desk, Subject: Response to NRC Generic Letter (GL) 92-01, Revision 1: Reactor Vessel Structural Integrity

Fluence data are from October 15, 1993, letter from W. G. Gates (OPPD) to USNRC

Initial USE for the Lower Shell Axial Welds is from letter from S. Bloom (USNRC) to T.L. Patterson (OPPD) dated December 3, 1993

Plant Name: Ginna

Docket Number: 50-244

NSSS Vendor: Westinghouse

Vessel Manufacturer: Babcock and Wilcox

Edition of ASME Code for Design: 1965 ASME Code

Date of Commercial Operation: July 1, 1970

Date of License Expiration: September 18, 2009

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate to lower shell circumferential weld SA-847, heat 61782
ID Fluence at EOL: 3.68E19 n/cm^2
Initial RT_{ndt}: -5°F
Method of Determining Chemistry Factor: Surveillance data per paragraph C.2.1 of RG 1.99, Rev. 2
Increase in RT_{ndt} at EOL: 179°F
Margin: 48°F
RT_{pts} at EOL: 222°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate to lower shell circumferential weld SA-847, heat 61782
1/4T Fluence at EOL: 2.268E19 n/cm^2
Initial USE: Heat specific value not reported
Percent Drop at EOL: 50%
USE at EOL: Not calculated since heat specific value was not reported
Date at which USE Screening Limit will be Exceeded: Not applicable since heat specific value was not reported
Bases for Accepting the USE at EOL: Equivalent margin analyses were performed in Topical Reports BAW-2178P and BAW-2192P[1].

References:

Fluence, initial USE, and chemical composition data are from July 2, 1992, letter from R. C. Mecredy (RG&E) to A. R. Johnson (USNRC), subject: Reactor Vessel Structural Integrity, 10CFR50.54(f), Response to Generic Letter 92-01, Revision 1, R. G. Ginna Nuclear Power Plant

[1] Staff will review applicability of Topical Reports BAW-2178P and BAW-2192P.

Plant Name: Grand Gulf 1

Docket Number: 50-416

NSSS Vendor: General Electric

Vessel Manufacturer: Chicago Bridge and Iron

Edition of ASME Code for Design: Winter 1972 Addenda to the 1971 ASME Code

Date of Commercial Operation: July 1, 1985

Date of License Expiration: June 16, 2022

ART for the Limiting Beltline Material:

Limiting Beltline Material: #2 shell axial welds, heat 627260
ID Fluence at EOL: 3.11E18 n/cm^2
Initial RT_{NDT}: -30°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 54°F
Margin: 54°F
ART at EOL: 78°F

USE for the Limiting Beltline Material:

Limiting Beltline Material: #2 shell axial welds, heat 5P6214B
1/4T Fluence at EOL: 2.11E18 n/cm^2
Initial USE: 91 ft-lb
Percent Drop at EOL: 13%
USE at EOL: 79 ft-lb
Date at which USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Chemistry data per Paragraph C.1.2 of RG 1.99, Rev. 2

References:

ID fluence was revised to 3.11E18 based on May 5, 1994, letter to NRC, referring the licensee's document "Internal Calculation, Design Mechanical Engineering Calculation MC-Q1B13-91145, Rev. 0."

1/4T fluence is from GL 92-01 submittal dated July 6, 1992.

Ni contents for welds 5P6214B and 627260 are from an attachment to GL 88-11 submittal dated February 28, 1989.

The rest of the data are from FSAR.

Plant Name: Haddam Neck

Docket Number: 50-213

NSSS Vendor: Westinghouse

Vessel Manufacturer: Combustion Engineering

Edition of ASME Code for Design: Section VIII of the 1959 ASME Code Utilizing Code Cases 1270N and 1273N

Date of Commercial Operation: January 1, 1968

Date of License Expiration: June 29, 2007

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate axial welds 6-373A, B, C, heat 86054B
ID Fluence at EOL: 8.35E19 n/cm^2
Initial RT_{NDT}: -56°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 156°F
Margin: 66°F
RT_{pts} at EOL: 166°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Nozzle shell plate W9807-6, heat A5897
1/4T Fluence at EOL: 2.22E19 n/cm^2
Initial USE: 65 ft-lb
Percent Drop at EOL: 20%
USE at EOL: 52 ft-lb
Date at which USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Surveillance data per Paragraph C.2.2 of RG 1.99, Rev. 2

References:

Fluence, chemical composition, initial RT_{NDT}, and initial USE data are from July 6, 1992, letter from J. F. Opeka (NNECo) to USNRC Document Control Desk, Subject: Haddam Neck Plant; Millstone Power Station, Units 1, 2, and 3: Reactor Vessel Structural Integrity, 10CFR50.54(f), Generic Letter 92-01, Revision 1.

Plant Name: Hatch 1

Docket Number: 50-321

NSSS Vendor: General Electric

Vessel Manufacturer: Combustion Engineering

Edition of ASME Code for Design: Winter 1966 Addenda to the 1965 ASME Code

Date of Commercial Operation: December 31, 1975

Date of License Expiration: August 6, 2014

ART for the Limiting Beltline Material:

Limiting Beltline Material: Circumferential weld 1-313, heat 90099
ID Fluence at EOL: 2.5E18 n/cm^2
Initial RT_{ndt}[1]: -10°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase RT_{ndt} at EOL: 147°F
Margin: 56°F
ART at EOL: 193°F

USE for the Limiting Beltline Material:

Limiting Beltline Material: Lower intermediate axial welds, heats IP2815 and IP2809
1/4T Fluence at EOL: 1.8E18 n/cm^2
Initial USE: Heat specific value not reported
Percent Drop at EOL: 28.5%
USE at EOL: Not applicable since heat specific value not reported
Date at which USE Screening Limit will be Exceeded: Not applicable since heat specific value not reported
Bases for Accepting the USE at EOL: Equivalent margins analysis was performed in Topical Report NEDO-32205, Revision 1[2]

References:

Chemical composition and initial RT data are from July 2, 1992, letter from J. T. Beckham, Jr. to USNRC Document Control Desk, subject: Response to NRC Generic Letter 92-01, Revision 1, Reactor Vessel Structural Integrity

Fluence data are from July 1, 1994 letter from J. T. Beckham, Jr to USNRC

---

[1]Staff will review the forthcoming GE Topical Report on initial RT_{NDT} determination

[2]Staff will review applicability of Topical Report NEDO-32205, Rev. 1

Plant Name: Hatch 2

Docket Number: 50-366

NSSS Vendor: General Electric

Vessel Manufacturer: Combustion Engineering

Edition of ASME Code for Design: Summer 1970 Addenda to the 1968 ASME Code

Date of Commercial Operation: September 5, 1979

Date of License Expiration: June 13, 2018

ART for the Limiting Beltline Material:

Limiting Beltline Material: Lower shell axial welds 101-842, heat 10137
ID Fluence at EOL: 1.4E18 n/cm^2
Initial RT_{ndt}[1]: -50°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase RT_{ndt} at EOL: 75°F
Margin: 56°F
ART at EOL: 81°F

USE for the Limiting Beltline Material:

Limiting Beltline Material: Lower intermediate shell plate G6601-4, heat C8579-2
1/4T Fluence at EOL: 1.0E18 n/cm^2
Initial USE: 70 ft-lb
Percent Drop at EOL: 12%
USE at EOL: 62 ft-lb
Date at which USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Chemistry data per paragraph C.1.2 of RG 1.99, Rev. 2

References:

Initial RT, initial USE, and chemical composition data are from July 2, 1992, letter from J. T. Beckham, Jr. to USNRC Document Control Desk, subject: Response to NRC Generic Letter 92-01, Revision 1, Reactor Vessel Structural Integrity

Fluence data are from July 1, 1994 letter from J. T. Beckham, Jr. to USNRC

[1]Staff will review the forthcoming GE Topical Report on initial RT_{NDT} determination

Plant Name: Hope Creek

Docket Number: 50-354

NSSS Vendor: General Electric

Vessel Manufacturer: Hitachi

Edition of ASME Code for Design: Winter 1969 Addenda to the 1968 ASME Code

Date of Commercial Operation: December 20, 1986

Date of License Expiration: April 11, 2026

ART for the Limiting Beltline Material:
- Limiting Beltline Material: Lower shell axial welds, heat D53040
- ID Fluence at EOL: 1.70E18 n/cm^2
- Initial RT_{NDT}[1]: -30°F
- Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
- Increase in RT_{NDT} at EOL: 55°F
- Margin: 55°F
- ART at EOL: 80°F

USE for the Limiting Beltline Material:
- Limiting Beltline Material: Intermediate shell plate, heat 5K3025-1
- 1/4T Fluence at EOL: 2.6E17 n/cm^2
- Initial USE: Heat-specific value not reported
- Percent Drop at EOL: 11%
- USE at EOL: Not applicable since heat-specific value not reported
- Date at which USE Screening Limit will be Exceeded: Not applicable since heat-specific value not reported
- Bases for Accepting the USE at EOL: Equivalent margins analysis was performed in Topical Report NEDO-32205, Revision 1[2]

References:

June 6, 1994, letter to the NRC, response to Generic Letter 92-01.

Fluence data are from the licensee's response to GL 88-11, dated March 8, 1990.

Chemical composition and initial USE data are from June 30, 1992, letter from S. Miltenberger (PSEG) to USNRC Document Control Desk, Subject: Response to Generic Letter 92-01, Revision 1, Reactor Vessel Structural Integrity, 10 CFR 50.54(f).

---

[1]Staff will review the forthcoming GE Topical Report on initial RT_{NDT} determination

[2]Staff will review applicability of Topical Report NEDO-32205, Rev. 1

Plant Name: Indian Point 2

Docket Number: 50-247

NSSS Vendor: Westinghouse

Vessel Manufacturer: Combustion Engineering

Edition of ASME Code for Design: Summer 1965 Addenda to the 1965 ASME Code

Date of Commercial Operation: August 1, 1974

Date of License Expiration: September 28, 2013

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate Shell Plate B 2002-3, heat B4922-1
ID Fluence at EOL: 1.21E19 n/cm^2
Initial RT_{NDT}: 21°F
Method of Determining Chemistry Factor: Surveillance data per Paragraph C.2.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 192°F
Margin: 17°F
RT_{pts} at EOL: 230°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate shell plate B2002-3, heat B4922-1
1/4T Fluence at EOL: 7.22E18 n/cm^2
Initial USE: 74 ft-lb
Percent Drop at EOL: 32%
USE at EOL: 50 ft-lb
Date USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Surveillance data per Paragraph C.2.2 of RG 1.99, Rev. 2

REFERENCES:

July 6, 1992 and October 12, 1993, letters from S. B. Bram (ConEd) to USNRC

Plant Name: Indian Point 3

Docket Number: 50-286

NSSS Vendor: Westinghouse

Vessel Manufacturer: Combustion Engineering

Edition of ASME Code for Design: Summer 1965 Addenda to the 1965 ASME Code

Date of Commercial Operation: August 30, 1976

Date of License Expiration: December 15, 2015

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Lower shell course plate B2803-3, heat A-0512-2
ID Fluence at EOL: 1.04E19 n/cm^2
Initial RT_{NDT}: 74°F
Method of Determining Chemistry Factor: Surveillance data per Paragraph C.2.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 169°F
Margin: 17°F
RT_{pts} at EOL: 260°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material[1]:

Limiting Beltline Material: Lower shell course plate B2803-3, heat A-0512-2
1/4T Fluence at EOL: 6.20E18 n/cm^2
Initial USE: 68 ft-lb
Percent Drop at EOL: 19%
USE at EOL: 55 ft-lb
Date USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Surveillance data per Paragraph C.2.2 of RG 1.99, Rev. 2,

REFERENCES:

July 9, 1992 and November 24, 1993, letters from R. E. Beedle (NYPA) to USNRC

[1]Additional information will be provided by the CEOG in 1995 to further demonstrate welds 2-042A, B, & C and 3-042A, B, & C have irradiated USE of greater than 50 ft-lb at EOL.

Plant Name: Kewaunee

Docket Number: 50-305

NSSS Vendor: Westinghouse

Vessel Manufacturer: Combustion Engineering

Edition of ASME Code for Design: Winter 1968 Addenda to 1968 ASME Code

Date of Commercial Operation: June 16, 1974

Date of License Expiration: December 21, 2013

RT_{PTS} for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate to lower shell circumferential weld, heat IP3571
ID Fluence at EOL: 3.21E19 n/cm^2
Initial RT_{NDT}: -56°F
Method of Determining Chemistry Factor: Surveillance data per Paragraph C.2.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 250°F
Margin: 44°F
RT_{PTS} at EOL: 238°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate to lower shell circumferential weld, heat IP3571
1/4T Fluence at EOL: 2.17E19 n/cm^2
Initial USE: 126 ft-lb
Percent Drop at EOL: 44%
USE at EOL: 71 ft-lb
Date at which USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Surveillance data per Paragraph C.2.2 of RG 1.99, Rev. 2

References:

Fluence, initial USE, and initial RT_{NDT} are from July 2, 1992, letter from C. R. Steinhardt (WPSCo) to USNRC Document Control Desk, Subject: Kewaunee Nuclear Power Plant, Reactor Vessel Structural Integrity.

Plant Name: LaSalle 1

Docket Number: 50-373

NSSS Vendor: General Electric

Vessel Manufacturer: Combustion Engineering

Edition of ASME Code for Design: Winter 1969 Addenda to the 1968 ASME Code

Date of Commercial Operation: January 1, 1984

Date of License Expiration: May 17, 2022

RT_{pts} for the Limiting Beltline Material:
- Limiting Beltline Material: Middle shell axial welds 3-308A, B, C, heat 1P3571
- ID Fluence at EOL: 6.1E17 n/cm^2
- Initial RT_{NDT}[1]: -30°F
- Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
- Increase in RT_{NDT} at EOL: 78°F
- Margin: 56°F
- ART at EOL: 104°F

USE for the Limiting Beltline Material:
- Limiting Beltline Material: Middle shell axial welds 3-308A, B, C, heat 1P3571
- 1/4T Fluence at EOL: 3.9E17 n/cm^2
- Initial USE: Heat-specific value not reported
- Percent Drop at EOL: 25.0%
- USE at EOL: Not applicable since heat-specific value not reported
- Date at which USE Screening Limit will be Exceeded: Not applicable since heat-specific value not reported
- Bases for Accepting the USE at EOL: Equivalent margins analysis was performed in Topical Report NEDO-32205, Revision 1[2]

References:

The August 11, 1994 response to GL 92-01 closeout letter indicated that the fluence data were from GE Report MDE-89-0786, DRF A00-02764, "Flux Wire Dosimeter Evaluation for LaSalle Nuclear Power Station, Unit 1," July 1986

Chemical composition and initial RT are from July 1, 1992, letter from M. A. Jackson (CECo) to T. E. Murley (USNRC), Subject: Dresden Station Units 2 and 3; Quad Cities Station Units 1 and 2; LaSalle County Station Units 1 and 2

---

[1]Staff will review the forthcoming GE Topical Report on initial RT_{NDT} determination

[2]Staff will review applicability of Topical Report NEDO-32205, Rev. 1

Plant Name: LaSalle 2

Docket Number: 50-374

NSSS Vendor: General Electric

Vessel Manufacturer: Chicago Bridge and Iron

Edition of ASME Code for Design: Winter 1970 Addenda to the 1968 ASME Code

Date of Commercial Operation: October 19, 1984

Date of License Expiration: December 16, 2023

ART for the Limiting Beltline Material:
Limiting Beltline Material: Lower intermediate shell plate, heat C9404-2
ID Fluence at EOL: 6.4E17 n/cm^2
Initial RT_{NDT} [1]: 52°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 16°F
Margin: 16°F
ART at EOL: 84°F

USE for the Limiting Beltline Material:
Limiting Beltline Material: Lower shell plate, heats C9425-1 and C9425-2 and Lower intermediate shell plate, heat C9601-2
1/4T Fluence at EOL: 4.2E17 n/cm^2
Initial USE: Heat-specific values not reported
Percent Drop at EOL: 9.0%
USE at EOL: Not applicable since heat-specific values not reported
Date at which USE Screening Limit will be Exceeded: Not applicable since heat-specific values not reported
Bases for Accepting the USE at EOL: Equivalent margins analysis was performed in Topical Report NEDO-32205, Revision 1[2]

References:

The August 11, 1994 response to GL 92-01 closeout letter indicated that the fluence data was from GE Report SASR 87-59, DRF A00-02764, "Flux Wire Dosimeter Evaluation for LaSalle Nuclear Power Station, Unit 2," October 1987

Chemical composition and initial RT are from July 1, 1992, letter from M. A. Jackson (CECo) to T. E. Murley (USNRC), Subject: Dresden Station Units 2 and 3; Quad Cities Station Units 1 and 2; LaSalle County Station Units 1 and 2

---

[1]Staff will review the forthcoming GE Topical Report on initial RT_{NDT} determination

[2]Staff will review applicability of Topical Report NEDO-32205, Rev. 1

Plant Name: Limerick 1

Docket Number: 50-352

NSSS Vendor: General Electric

Vessel Manufacturer: Chicago Bridge and Iron

Edition of ASME Code for Design: Summer 1969 Addenda to the 1968 ASME Code

Date of Commercial Operation: February 1, 1986

Date of License Expiration: October 26, 2024

ART for the Limiting Beltline Material:

Limiting Beltline Material: Plate, heat C7677-1
ID Fluence at EOL: 1.73E18 n/cm^2
Initial RT_{NDT} [1]: 20°F
Method of Determining Chemistry Factor: Chemistry data per paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 39°F
Margin: 34°F
ART at EOL: 93°F

USE for the Limiting Beltline Material:

Limiting Beltline Material: Plates, heats C7688-1 and C7688-2
1/4T Fluence at EOL: 1.2E18 n/cm^2
Initial USE: Heat specific value not reported
Percent Drop at EOL: 12.5%
USE at EOL: Not applicable since heat value not reported
Date at which USE Screening Limit will be Exceeded: Not applicable since heat value not reported
Bases for Accepting the USE at EOL: Equivalent margins analysis was performed in Topical Report NEDO-32205, Rev. 2 [2]

References:

Chemical composition, initial RT, and fluence data are from November 23, 1992 letter to USNRC

[1] Staff will review the forthcoming GE Topical Report on initial RT_{ndt} determination

[2] Staff will review applicability of Topical Report NEDO-32205, Rev. 1

Plant Name: Limerick 2

Docket Number: 50-353

NSSS Vendor: General Electric

Vessel Manufacturer: Chicago Bridge and Iron

Edition of ASME Code for Design: Winter 1969 Addenda to the 1968 ASME Code

Date of Commercial Operation: February 1, 1986

Date of License Expiration: October 26, 2024

ART for the Limiting Beltline Material:

Limiting Beltline Material: Plate, heat B4316-1
ID Fluence at EOL: 1.73E18 n/cm^2
Initial RT_{ndt}[1]: 40°F
Method of Determining Chemistry Factor: Chemistry data per paragraph C.1.1 of RG 1.99, Rev. 2
Increase RT_{ndt} at EOL: 54°F
Margin: 34°F
ART at EOL: 128°F

USE for the Limiting Beltline Material:

Limiting Beltline Material: Plate, heat C9621-2
1/4T Fluence at EOL: 1.2E18 n/cm^2
Initial USE: Heat specific value not reported
Percent Drop at EOL: 14.8%
USE at EOL: Not applicable since heat specific value not reported
Date at which USE Screening Limit will be Exceeded: Not applicable since heat specific value not reported
Bases for Accepting the USE at EOL: Equivalent margins analysis was performed in Topical Report NEDO-32205, Rev. 2[2]

References:

Chemical composition, initial RT, and fluence data are from November 23, 1992 letter to USNRC

[1]Staff will review the forthcoming GE Topical Report on initial RT_{NDT} determination

[2]Staff will review applicability of Topical Report NEDO-32205, Rev. 1

Plant Name: Maine Yankee

Docket Number: 50-309

NSSS Vendor: Combustion Engineering

Vessel Manufacturer: Combustion Engineering

Edition of ASME Code for Design: Winter 1966 Addenda to the 1965 ASME Code

Date of Commercial Operation: December 28, 1972

Date of License Expiration: October 21, 2008

RT_{PTS} for the Limiting Beltline Material:

Limiting Beltline Material: Axial Welds 3-203, heats 13253/12008
ID Fluence at EOL: 1.48E19 n/cm^2
Initial RT_{NDT}: -56°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 229°F
Margin: 66°F
RT_{PTS} at EOL: 239°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material[1]:

Limiting Beltline Material: Axial Welds 2-203 A, B, C, heat 51989
1/4T Fluence at EOL: 8.82E18 n/cm^2
Initial USE: 75 ft-lb
Percent Drop at EOL: 29%
USE at EOL: 53 ft-lb
Date at which USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Chemistry data per Paragraph C.1.2 of RG 1.99, Rev. 2

References:

Initial USE data are from December 3, 1993, letter to USRNC, subject: Response to Generic Letter 92-01.

Fluence data are from October 28, 1991, letter to USNRC, Subject: Update of PTS Assessment to Address the Revised PTS Rule.

Initial RT_{NDT} data are from Attachment F of December 2, 1988 MY letter to USNRC, Subject: Proposed Change No. 145--Combined Heatup, Cooldown and Pressure-Temperature Limitations.

---

[1]Additional information will be provided by the CEOG in 1995 to further demonstrate welds 2-203 and 3-203 A, B, & C have irradiated USE of greater than 50 ft-lb at EOL.

Plant Name: McGuire 1

Docket Number: 50-369

NSSS Vendor: Westinghouse

Vessel Manufacturer: Combustion Engineering

Edition of ASME Code for Design: Summer 1971 Addenda to the 1971 ASME Code

Date of Commercial Operation: December 1, 1981

Date of License Expiration: June 12, 2021

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Lower shell axial weld M1.32, heats 21935/12008
ID Fluence at EOL: 1.46E19 n/cm^2
Initial RT_{ndt}: -56°F
Method of Determining the Chemistry Factor: Chemistry data per paragraph C.1.1 of RG 1.99, Rev. 2
Adjusted RT_{ndt} at EOL: 232°F
Margin: 66°F
RT_{pts} at EOL: 242°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Lower shell axial welds M1.32, heats 21935/12008
1/4T Fluence at EOL: 8.7E18 n/cm^2
Initial USE: 90 ft-lb
Percent Drop at EOL: 34.8%
USE at EOL: 59 ft-lb
Date at which USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Chemistry data per Paragraph C.1.2 of RG 1.99, Rev. 2

References:

Chemical composition and initial USE data are from July 3, 1992, letter from H. B. Tucker (DPCo) to USNRC Document Control Desk, subject: McGuire Nuclear Station, Units 1 and 2; Catawba Nuclear Station Units 1 and 2; Oconee Nuclear Station, Units 1, 2, and 3: Response to Generic Letter 92-01, Revision 1, Reactor Vessel Structural Integrity.

Fluence data are from Table 6-13 of WCAP-12354.

Plant Name: McGuire 2

Docket Number: 50-370

NSSS Vendor: Westinghouse

Vessel Manufacturer: Rotterdam Dockyard

Edition of ASME Code for Design: Winter 1971 Addenda to the 1971 ASME Code

Date of Commercial Operation: March 1, 1984

Date of License Expiration: March 3, 2023

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Lower shell forging 04, heat 411337-11
ID Fluence at EOL: 2.04E19 n/cm^2
Initial RT_{ndt}: -30°F
Method of Determining Chemistry Factor: Chemistry Table per Paragraph C.1.1 of RG 1.99, Rev. 2
Adjusted RT_{ndt} at EOL: 138°F
Margin: 34°F
RT_{pts} at EOL: 142°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Lower shell forging 04, heat 411337-11
1/4T Fluence at EOL: 1.225E19 n/cm^2
Initial USE: 97 ft-lb
Percent Drop at EOL: 25.2%
USE at EOL: 71 ft-lb
Date at which USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Chemistry data per paragraph C.1.2 of RG 1.99, Rev. 2

References:

Chemical composition and initial USE data are from July 3, 1992, letter from H. B. Tucker (DPCo) to USNRC Document Control Desk, subject: McGuire Nuclear Station, Units 1 and 2; Catawba Nuclear Station Units 1 and 2; Oconee Nuclear Station, Units 1, 2, and 3: Response to Generic Letter 92-01, Revision 1, Reactor Vessel Structural Integrity.

Fluence data are from Table 6-14 of WCAP-12556.

Plant Name: Millstone 1

Docket Number: 50-245

NSSS Vendor: General Electric

Vessel Manufacturer: Combustion Engineering

Edition of ASME Code for Design: Winter 1966 Addenda to the 1965 ASME Code

Date of Commercial Operation: March 1, 1971

Date of License Expiration: October 6, 2010

ART for the Limiting Beltline Material:

Limiting Beltline Material: Lower intermediated shell plate G2002-5, heat C1079-1
ID Fluence at EOL: 1.29E18 n/cm^2
Initial RT_{NDT}: 26°F
Method of Determining Chemistry Factor: Surveillance data per Paragraph C.2.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 110°F
Margin: 17°F
ART at EOL: 153°F

USE for the Limiting Beltline Material:

Limiting Beltline Material: Lower intermediate shell plate G2002-5, heat C1079-1
1/4T Fluence at EOL: 9.0E17 n/cm^2
Initial USE: 65 ft-lb
Percent Drop at EOL: 21.5%
USE at EOL: 51 ft-lb
Date at which USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Surveillance data per Paragraph C.2.2 of RG 1.99, Rev. 2

References:

Fluence, chemical composition, initial RT_{NDT}, and initial USE data are from July 6, 1992, letter from J. F. Opeka (NNECo) to USNRC Document Control Desk, Subject: Haddam Neck Plant; Millstone Power Station, Units 1, 2, and 3: Reactor Vessel Structural Integrity, 10CFR50.54(f), Generic Letter 92-01, Revision 1

Plant Name: Millstone 2

Docket Number: 50-336

NSSS Vendor: Combustion Engineering

Vessel Manufacturer: Combustion Engineering

Edition of ASME Code for Design: Summer 1969 Addenda to the 1968 ASME Code

Date of Commercial Operation: December 26, 1975

Date of License Expiration: July 31, 2015

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Lower shell plate, C-506-3, heat C-5518-1
ID Fluence at EOL: 2.29E19 n/cm^2
Method of Determining the Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Initial RT_{ndt}: 37°F
Increase in RT_{ndt} at EOL: 116°F
Margin: 34°F
RT_{pts} at EOL: 187°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material[1]:

Limiting Beltline Material: Intermediate shell plate, heat C-5843-1
1/4T Fluence at EOL: 1.365E19 n/cm^2
Initial USE: 73 ft-lb
Percent Drop at EOL: 26%
USE at EOL: 54 ft-lb
Date at which USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Chemistry data per paragraph C.1.2 of RG 1.99, Rev. 2

References:

Fluence, chemical composition, and initial RT_{ndt} data are from July 6, 1992, letter from J. F. Opeka (NNECo) to USNRC Document Control Desk.

May 23, 1994 letter from J. F. Opeka (Northeast Utilities) to USNRC, subject: Millstone Nuclear Power Station, Unit No. 2 Response to NRC Request for Additional Information, Generic Letter 92-01, Revision 1 "Reactor Vessel Structural Integrity".

---

[1]Additional information will be provided by the CEOG in 1995 to further demonstrate welds 2-203 and 3-203 A, B, & C have irradiated USE of greater than 50 ft-lb at EOL.

Plant Name: Millstone 3

Docket Number: 50-423

NSSS Vendor: Westinghouse

Vessel Manufacturer: Combustion Engineering

Edition of ASME Code for Design: Summer 1973 Addenda to the 1971 ASME Code

Date of Commercial Operation: April 23, 1986

Date of License Expiration: November 25, 2025

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate shell plate B9805-1, heat C-4039-2
ID Fluence at EOL: 3.03E19 n/cm^2
Initial RT_{ndt}: 60°F
Method of Determining the Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{ndt} at EOL: 40°F
Margin: 34°F
RT_{pts} at EOL: 134°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Lower shell plate B9820-2, heat D1242-2
1/4T Fluence at EOL: 1.82E19 n/cm^2
Initial USE: 76 ft-lb
Percent Drop at EOL: 22.5%
USE at EOL: 59 ft-lb
Date at which USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Chemistry data per paragraph C.1.2 of RG 1.99, Rev. 2

References:

Chemical composition and initial RT_{ndt} data are from July 6, 1992, letter from J. F. Opeka (NNECo) to USNRC Document Control Desk, subject: Haddam Neck Plant; Millstone Power Station, Units 1, 2, and 3: Reactor Vessel Structural Integrity, 10CFR50.54(f), Generic Letter 92-01, Revision 1

Fluence data are from May 23, 1994 letter from J. F. Opeka (Northeast Utilities) to USNRC, subject: Millstone Nuclear Power Station, Unit 3, Generic Letter 92-01 Reactor Vessel Structural Integrity."

Plant Name: Monticello

Docket Number: 50-263

NSSS Vendor: General Electric

Vessel Manufacturer: Chicago Bridge and Iron

Edition of ASME Code for Design: Winter 1966 Addenda to the 1965 ASME Code

Date of Commercial Operation: June 3, 1971

Date of License Expiration: September 8, 2010

ART for the Limiting Beltline Material:

Limiting Beltline Material: Lower intermediate shell plate, heat C2220-2
ID Fluence at EOL: 5.15E18 n/cm^2
Initial RT_{ndt}[1]: 14°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase RT_{ndt} at EOL: 102°F
Margin: 34°F
ART at EOL: 150°F

USE for the Limiting Beltline Material:

Limiting Beltline Material: Lower shell plates I-14, I-16, and I-17, heat C2220-1, A0946-1, and C2193-1; and welds (no heat numbers)
1/4T Fluence at EOL: 3.8E18 n/cm^2
Initial USE: Heat specific value not reported
Percent Drop at EOL: 20.6% for plates and 15% for welds
USE at EOL: Not applicable since heat specific value not reported
Date at which USE Screening Limit will be Exceeded: Not applicable since heat specific value not report
Bases for Accepting the USE at EOL: Equivalent margins analysis was performed in Topical Report NEDO-32205, Revision 1[2]

References:

Fluence, chemical composition, initial RT, and initial USE data are from July 6, 1992, letter from T. M. Parker (NSP) to USNRC Document Control Desk, subject: Response to Generic Letter 92-01, Reactor Vessel Structural Integrity

---

[1]Staff will review the forthcoming GE Topical Report on initial RT_{NDT} determination

[2]Staff will review applicability of Topical Report NEDO-32205, Rev. 1

Plant Name: Nine Mile Point 1

Docket Number: 50-220

NSSS Vendor: General Electric

Vessel Manufacturer: Combustion Engineering

Edition of ASME Code for Design: Section I of the 1962 ASME Code Utilizing Code Cases 1270N and 1273N

Date of Commercial Operation: December 1, 1969

Date of License Expiration: August 22, 2009

ART for the Limiting Beltline Material:

Limiting Beltline Material: Upper shell course plate G-307-4, heat P2076
ID Fluence at EOL: 2.21E18 n/cm^2
Initial RT_{NDT}: 40°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99 Rev. 2
Increase in RT_{NDT} at EOL: 103°F
Margin: 34°F
ART at EOL: 177°F

USE for the Limiting Beltline Material:

Limiting Beltline Material: Upper shell course plate G-307-4, heat P2076
1/4T Fluence at EOL: 1.44E18 n/cm^2
Initial USE: 52 ft-lb
Percent Drop at EOL: 23.0%
USE at EOL: 40 ft-lb
Date at which USE Screening Limit will be Exceeded: Before EOL
Bases for Accepting the USE at EOL: Plant specific equivalent margins analysis was approved by the NRC in a April 20, 1994 letter to the licensee

References:

Fluence, chemical composition, and initial USE data are from July 2, 1992, letter from C. D. Terry (NMPCo) to USNRC Document Control Desk, Subject: Generic Letter 92-01, Revision 1, Reactor Vessel Structural Integrity, 10 CFR 50.54(f).

April 26, 1994 letter to the USNRC, subject: Response to Generic Letter 92-01.

Plant Name: Nine Mile Point 2

Docket Number: 50-410

NSSS Vendor: General Electric

Vessel Manufacturer: CB&I Nuclear

Edition of ASME Code for Design: Winter 1972 Addenda to the 1971 ASME Code

Date of Commercial Operation: March 11, 1988

Date of License Expiration: October 31, 2026

ART for the Limiting Beltline Material:

Limiting Beltline Material: Number 2 shell ring, heat C3147-1 and Number 1 shell ring, heat C3147-2
ID Fluence at EOL: 1.72E18 n/cm^2
Initial RT_{NDT}: 0°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 40°F
Margin: 40°F
ART at EOL: 80°F

USE for the Limiting Beltline Material:

Limiting Beltline Material: Number 2 shell ring, heat C3147-1
1/4T Fluence at EOL: 1.18E18 n/cm^2
Initial USE: 70 ft-lb
Percent Drop at EOL: 14.3%
USE at EOL: 60 ft-lb
Date at which USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Chemistry data per paragraph C.1.2 of RG 1.99, Rev. 2

References:

Fluence, chemical composition, and initial USE data are from July 2, 1992, letter from C. D. Terry (NMPCo) to USNRC Document Control Desk, Subject: Generic Letter 92-01, Revision 1, Reactor Vessel Structural Integrity, 10 CFR 50.54(f)

April 26, 1994 letter to the USNRC, subject: Response to Generic Letter 92-01.

Plant Name: North Anna 1

Docket Number: 50-338

NSSS Vendor: Westinghouse

Vessel Manufacturer: Rotterdam Dockyard

Edition of ASME Code for Design: Winter 1968 Addenda to the 1968 ASME Code

Date of Commercial Operation: June 6, 1978

Date of License Expiration: April 1, 2018

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Weld 04, heat 25531
ID Fluence at EOL: 3.95E19 n/cm^2
Initial RT_{NDT}: 19°F
Method of Determining Chemistry Factor: Surveillance data per Paragraph C.2.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 126°F
Margin: 28°F
RT_{pts} at EOL: 173°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material[1]:

Limiting Beltline Material: Lower shell forging 03, heat 990400/292332
1/4T Fluence at EOL: 2.49E19 n/cm^2
Initial USE: 85 ft-lb
Percent Drop at EOL: 31.5%
USE at EOL: 58 ft-lb
Date at which USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Surveillance data per Paragraph C.2.2 of RG 1.99, Rev. 2

References:

Fluence data for weld 04 are from June 29, 1992, letter to the NRC; fluence data for other materials are from September 23, 1993, letter to the NRC.

Chemical composition, initial RT, and initial USE data for forging 03 are from BAW-2168 (attached to the GL 92-01 response).

Initial USE data for forging 04 and weld 04 are from BAW-1911, Rev. 1.

[1]Additional information will be provided by the CEOG in 1995 to further demonstrate welds 04 and 05 have irradiated USE of greater than 50 ft-lb at EOL.

Plant Name: North Anna 2

Docket Number: 50-339

NSSS Vendor: Westinghouse

Vessel Manufacturer: Rotterdam Dockyard

Edition of ASME Code for Design: Winter 1968 Addenda to the 1968 ASME Code

Date of Commercial Operation: December 14, 1980

Date of License Expiration: August 21, 2020

RT_{pts} for the Limiting Beltline Material:
Limiting Beltline Material: Lower shell forging 03, heat 990533/207355
ID Fluence at EOL: 4.47E19 n/cm^2
Initial RT_{NDT}: 56°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 133°F
Margin: 34°F
RT_{pts} at EOL: 223°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:
Limiting Beltline Material: Intermediate shell forging 04, heat 990496/292424
1/4T Fluence at EOL: 2.82E19 n/cm^2
Initial USE: 74 ft-lb
Percent Drop at EOL: 30.6%
USE at EOL: 51 ft-lb
Date at which USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Surveillance data per Paragraph C.2.2 of RG 1.99, Rev. 2.

References:

BAW-2168, which is attached to June 29, 1992, letter W.L. Stewart (VPCo) to USNRC Document Control Desk, Subject: Response to Generic Letter 92-01, Reactor Vessel Structural Integrity, contains chemical composition and the initial RT_{NDT} data for all the beltline materials.

Fluence data are from September 23, 1993, letter to the NRC.

Initial USEs for forgings 05 and 03 are from December 29, 1992, letter to the NRC.

Plant Name: Oconee 1

Docket Number: 50-269

NSSS Vendor: Babcock and Wilcox

Vessel Manufacturer: Babcock and Wilcox

Edition of ASME Code for Design: Summer 1967 Addenda to the 1965 ASME Code

Date of Commercial Operation: July 15, 1973

Date of License Expiration: February 6, 2013

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate to lower shell circumferential weld SA-1229, heat 71249
ID Fluence at EOL: 7.98E18 n/cm^2
Initial RT_{ndt}: -5°F
Method of Determining Chemistry Factor: Chemistry data per paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{ndt} at EOL: 162°F
Margin: 68°F
RT_{pts} at EOL: 225°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate to lower shell circumferential weld SA-1229, heat 71249
1/4T Fluence at EOL: 4.46E18 n/cm^2
Initial USE: Heat specific value not reported
Percent Drop at EOL: 31.6%
USE at EOL: Not applicable since heat specific value not reported
Date at which USE Screening Limit will be Exceeded: Not applicable since heat specific value was not reported
Bases for Accepting the USE at EOL: Equivalent margin analyses were performed in Topical Reports BAW-2178P and BAW-2192P[1].

References:

Fluence, chemical composition, and initial USE are from reports BAW-2166 and BAW-2222.

---

[1]Staff will review applicability of Topical Reports BAW-2178P and BAW-2192P.

Plant Name: Oconee 2

Docket Number: 50-270

NSSS Vendor: Babcock and Wilcox

Vessel Manufacturer: Babcock and Wilcox

Edition of ASME Code for Design: Summer 1967 Addenda to the 1965 ASME Code

Date of Commercial Operation: September 9, 1974

Date of License Expiration: October 6, 2013

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Middle circumferential weld WF-25, heat 299L44
ID Fluence at EOL: 9.19E18 n/cm^2
Initial RT_{ndt}: -7°F
Method of Determining Chemistry Factor: Chemistry data per paragraph C.2.1 of RG 1.99, Rev. 2
Increase in RT_{ndt} at EOL: 216°F
Margin: 49°F
RT_{pts} at EOL: 258°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Middle circumferential weld WF-25, heat 299L44
1/4T Fluence at EOL: 5.15E18 n/cm^2
Initial USE: Heat specific value not reported
Percent Drop at EOL: 38%
USE at EOL: Not applicable since heat specific value not reported
Date at which USE Screening Limit will be Exceeded: Not applicable since specific value was not reported
Bases for Accepting the USE at EOL: Equivalent margin analyses were performed in Topical Reports BAW-2178P and BAW-2192P[1]

References:

Fluence, chemical composition, initial USE, and Initial RT_{ndt}s are from reports BAW-2166 and BAW-2222.

---

[1]Staff will review applicability of Topical Reports BAW-2178P and BAW-2192P.

Plant Name: Oconee 3

Docket Number: 50-287

NSSS Vendor: Babcock and Wilcox

Vessel Manufacturer: Babcock and Wilcox

Edition of ASME Code for Design: Summer 1967 Addenda to the 1965 ASME Code

Date of Commercial Operation: December 16, 1974

Date of License Expiration: July 19, 2014

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Upper to lower circumferential weld WF-67, heat 72442
ID Fluence at EOL: 9.01E18 n/cm^2
Initial RT_{ndt}: -5°F
Method of Determining Chemistry Factor: Chemistry data per paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{ndt} at EOL: 168°F
Margin: 68°F
RT_{pts} at EOL: 231°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Upper to lower circumferential weld, WF-67
1/4T Fluence at EOL: 5.05E18 n/cm^2
Initial USE: Heat specific value was not reported
Percent Drop at EOL: 32.9%
USE at EOL: Not applicable since heat specific value was not reported
Date at which USE Screening Limit will be Exceeded: Not applicable since heat specific value was not reported.
Bases for Accepting the USE at EOL: Equivalent margin analysis were performed in Topical Reports BAW-2178P and BAW-2192P[1]

References:

Fluence, chemical composition, and initial RT_{ndt} are from BAW-2166 and BAW-2222.

[1]Staff will review applicability of Topical Reports BAW-2178P and BAW-2192P.

Plant Name: Oyster Creek

Docket Number: 50-219

NSSS Vendor: General Electric

Vessel Manufacturer: Combustion Engineering

Edition of ASME Code for Design: Section I of the 1962 ASME Code

Date of Commercial Operation: December 1, 1969

Date of License Expiration: December 15, 2004

ART for the Limiting Beltline Material:
- Limiting Beltline Material: Lower intermediate shell axial welds 2-564D, E, F, heat 86054B
- ID Fluence at EOL: 3.62E18 n/cm^2
- Initial RT_{NDT}[1]: -8°F
- Method of Determining the Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
- Increase in RT_{NDT} at EOL: 121°F
- Margin: 56°F
- ART at EOL: 169°F

USE for the Limiting Beltline Material:
- Limiting Beltline Material: Lower intermediate shell plate, heat P2161-1 and axial weld, heat 86054B
- 1/4T Fluence at EOL: 2.36E18 n/cm^2
- Initial USE: 51 ft-lb for plate P2161-1 and value not reported for weld
- Percent Drop at EOL: 21.4% for plate and 34% for weld
- USE at EOL: 40 ft-lb for plate and not applicable for weld since heat specific value not reported
- Date at which Screening Limit will be Exceeded: Not applicable since heat-specific value not reported
- Bases for Accepting the USE at EOL: Equivalent margins analysis was performed in Topical Report NEDO-32205, Rev. 1[2]

References:

Initial USE and fluence data for plates are from June 16, 1992, letter from A. W. Dromerick (USNRC) to Distribution (USNRC).

Initial RT_{NDT} data are from January 11, 1991, letter USNRC

Initial USE for welds are from July 17, 1991, letter from J.C. Devin (GPUN) to USNRC

---

[1] Staff will review the forthcoming GE Topical Report on initial RT_{NDT} determination

[2] Staff will review applicability of Topical Report NEDO-32205, Rev. 1

Plant Name: Palisades

Docket Number: 50-255

NSSS Vendor: Combustion Engineering

Vessel Manufacturer: Combustion Engineering

Edition of ASME Code for Design: Winter 1965 Addenda to 1965 ASME Code

Date of Commercial Operation: December 31, 1971

Date of License Expiration: March 14, 2007

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Axial welds, heat W5214
ID Fluence at EOL: 1.91E19 n/cm^2
Initial RT_{NDT}: -56°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 265°F
Margin: 66°F
RT_{pts} at EOL: 275°F
Date at which PTS Screening Limit will be exceeded: 2004

USE for the Limiting Beltline Material:

Limiting Beltline Material: Plate D-3804-1, heat C-1308
1/4T Fluence at EOL: 1.615E19 n/cm^2
Initial USE: 72 ft-lb
Percent Drop at EOL: 31%
USE at EOL: 50 ft-lb
Date USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Chemistry data per Paragraph C.1.2 of RG 1.99, Rev. 2

REFERENCES:

July 3, 1992, letter from G. B. Slade (CPCo) to USNRC Document Control Desk, Subject: Palisades Plant--Reactor Vessel Structural Integrity--Response to Generic Letter 92-01, Revision 1

February 23, 1994, letter from D.W. Rogers (CPCo) to USNRC

August 31, 1990, letter from G.B. Slade (CPCo) to USNRC

July 12, 1994, letter from A. Hsia (NRC) to CPCo

Plant Name: Palo Verde 1

Docket Number: 50-528

NSSS Vendor: Combustion Engineering

Vessel Manufacturer: Combustion Engineering

Edition of ASME Code for Design: Winter 1973 Addenda to the 1971 ASME Code

Date of Commercial Operation: January 28, 1986

Date of License Expiration: December 31, 2024

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate shell plates M-6701-2 and -3 (no heat numbers)
ID Fluence at EOL: 3.29E19 n/cm^2
Initial RT_{NDT}: 40°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 49°F
Margin: 34°F
RT_{pts} at EOL: 123°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate shell plate M-6701-1 (no heat numbers)
1/4T Fluence at EOL: 1.681E19 n/cm^2
Initial USE: 83 ft-lb
Percent Drop at EOL: 21.5%
USE at EOL: 65 ft-lb
Date at which USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Chemistry data per Paragraph C.1.2 of RG 1.99, Rev. 2

References:

Fluence, chemical compositions, and initial RT_{NDT} data are from January 31, 1989, letter from D.B. Karner (APS) to USNRC Document Control Desk, Subject: Palo Verde Nuclear Generating Station (PVNGS) Units 1,2, and 3, Generic Letter 88-11, Radiation Embrittlement of Reactor Vessel Materials.

Information on plate and weld material types is from FSAR; plate initial USE data are from Table 5.2-5A of FSAR, and weld initial USE values are from Charpy curves of FSAR.

Plant Name: Palo Verde 2

Docket Number: 50-529

NSSS Vendor: Combustion Engineering

Vessel Manufacturer: Combustion Engineering

Edition of ASME Code for Design: Winter 1973 Addenda to the 1971 ASME Code

Date of Commercial Operation: September 19, 1986

Date of License Expiration: December 9, 2025

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate shell plate F-765-6 (no heat numbers)
ID Fluence at EOL: 3.29E19 n/cm^2
Initial RT_{NDT}: 10°F
Method of Determining Chemical Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 34°F
Margin: 34°F
RT_{pts} at EOL: 78°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Lower shell axial welds 101-142 A, B, C, (no heat numbers)
1/4T Fluence at EOL: 1.91E19 n/cm^2
Initial USE: 100 ft-lb
Percent Drop at EOL: 25.5%
USE at EOL: 75 ft-lb
Date at which USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Chemistry data per Paragraph C.1.2 of RG 1.99, Rev. 2

References:

Fluence, chemical compositions, and initial RT_{NDT} data are from January 31, 1989, letter from D.B. Karner (APS) to USNRC Document Control Desk, Subject: Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2 and 3, Generic Letter 88-11, Radiation Embrittlement of Reactor Vessel Materials.

Information on plate and weld material types is from FSAR; plate initial USE data are from Table 5.2-5A of FSAR, and weld initial USE values are from Charpy curves of FSAR.

Plant Name: Palo Verde 3

Docket Number: 50-530

NSSS Vendor: Combustion Engineering

Vessel Manufacturer: Combustion Engineering

Edition of ASME Code for Design: Winter 1973 Addenda to the 1971 ASME Code

Date of Commercial Operation: January 8, 1988

Date of License Expiration: March 25, 2027

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Lower shell plate F-6411-2 (no heat numbers)
ID Fluence at EOL: 3.29E19 n/cm^2
Initial RT_{NDT}: 0°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 34°F
Margin: 34°F
RT_{pts} at EOL: 68°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Circumferential weld 101-171 and Lower shell axial welds 101-142A, B, C, (no heat numbers)
1/4T Fluence at EOL: 1.681E19 n/cm^2 for weld 101-171 and 1.91E19 n/cm^2 for welds 101-142 A, B, C
Initial USE: 90 ft-lb (101-171) and 100 ft-lb (101-142 A, B, C)
Percent Drop at EOL: 21.5% (101-171) and 22.1% (101-142 A, B, C)
USE at EOL: 71 ft-lb (101-171) and 78 ft-lb (101-142 A, B, C)
Date at which USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Chemistry data per Paragraph C.1.2 of RG 1.99, Rev. 2

References:

Fluence, chemical compositions, and initial RT_{NDT} data are from January 31, 1989, letter from D.B. Karner (APS) to USNRC

Licensee indicated in a letter dated July 25, 1994 that weld chemical composition values are higher than those in FSAR and will be used by APS in the future.

Information on plate and weld material types is from FSAR; plate initial USE data are from Table 5.2-5A of FSAR, and weld initial USE values are from Charpy curves of FSAR.

Plant Name: Peach Bottom 2

Docket Number: 50-277

NSSS Vendor: General Electric

Vessel Manufacturer: Combustion Engineering, and Chicago Bridge and Iron

Edition of ASME Code for Design: Winter 1965 Addenda to the 1965 ASME Code

Date of Commercial Operation: July 5, 1974

Date of License Expiration: January 31, 2008

ART for the Limiting Beltline Material:

Limiting Beltline Material: Lower intermediate shell plate, heat C2873-1
ID Fluence at EOL: 8.0E17 n/cm^2
Initial RT_{NDT}[1]: -6°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of Rev. 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 31°F
Margin: 31°F
ART at EOL: 56°F

USE for the Limiting Beltline Material:

Limiting Beltline Material: Lower intermediate shell plate, heat C2894-2 and circumferential weld, heat S-3986
1/4T Fluence at EOL: 5.5E17 n/cm^2
Initial USE: Heat-specific values not reported
Percent Drop at EOL: 10.0%
USE at EOL: Not applicable since heat-specific values not reported
Date at which Screening Limit will be Exceeded: Not applicable since heat-specific values not reported
Bases for Accepting the USE at EOL: Equivalent margins analysis was performed in Topical Report NEDO-32205, Rev. 1[2]

References:

Initial RT_{NDT}, Fluence, chemical composition, and initial USE data are from July 10, 1992, letter from G. J. Beck (PECo) to USNRC Document Control Desk, Subject: Response to Generic Letter 92-01, Revision 1, "Reactor Vessel Structural Integrity, 10 CFR 50.54(f)"

---

[1]Staff will review the forthcoming GE Topical Report on initial RT_{NDT} determination

[2]Staff will review applicability of Topical Report NEDO-32205, Rev. 1

Plant Name: Peach Bottom 3

Docket Number: 50-278

NSSS Vendor: General Electric

Vessel Manufacturer: Babcock and Wilcox, and Chicago Bridge and Iron

Edition of ASME Code for Design: Winter 1965 Addenda to the 1965 ASME Code

Date of Commercial Operation: December 31, 1974

Date of License Expiration: January 31, 2008

ART for the Limiting Beltline Material:

Limiting Beltline Material: Lower intermediate shell plate, heat C2773-2
ID Fluence at EOL: 7.2E17 n/cm^2
Initial RT_{NDT}: 10°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 37°F
Margin: 34°F
ART at EOL: 81°F

USE for the Limiting Beltline Material:

Limiting Beltline Material: Lower intermediate shell plate, heat C2773-2
1/4T Fluence at EOL: 5.0E17 n/cm^2
Initial USE: Heat-specific value not reported
Percent Drop at EOL: 12%
USE at EOL: Not applicable since heat-specific value not reported
Date at which Screening Limit will be Exceeded: Not applicable since heat-specific value not reported
Bases for Accepting the USE at EOL: Equivalent margins analyses was performed in Topical Report NEDO-32205, Rev. 1[1]

References:

Initial RT_{NDT}, Fluence, chemical composition, and initial USE data are from July 10, 1992, letter from G. J. Beck (PECo) to USNRC Document Control Desk, Subject: Response to Generic Letter 92-01, Revision 1, "Reactor Vessel Structural Integrity, 10 CFR 50.54(f)"

[1] Staff will review applicability of Topical Report NEDO-32205, Rev. 1

Plant Name: Perry

Docket Number: 50-441

NSSS Vendor: General Electric

Vessel Manufacturer: Chicago Bridge and Iron

Edition of ASME Code for Design: Winter 1972 Addenda to the 1971 ASME Code

Date of Commercial Operation: November 18, 1987

Date of License Expiration: March 18, 2026

ART for the Limiting Beltline Material:

Limiting Beltline Material: Axial weld, heat 627260/B322A27AE
ID Fluence at EOL: 5.4E18 n/cm^2
Initial RT_{NDT}: -30°F
Method of Determining the Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 68°F
Margin: 56°F
ART at EOL: 94°F

USE for the Limiting Beltline Material:

Limiting Beltline Material: Plate, heat C2557-1
1/4T Fluence at EOL: 3.7E18 n/cm^2
Initial USE: 84 ft-lb
Percent Drop at EOL: 24%
USE at EOL: 64 ft-lb
Date at which Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Chemistry data per Paragraph C.1.2 of RG 1.99, Rev. 2

References:

Initial RT_{NDT}, chemical composition, and initial USE are from the Perry 1 FSAR.

Fluence data are from September 14, 1990, letter from M. D. Lyster (CE) to USNRC Document Control Desk, Subject: Technical Specification Change Request--Reactor Pressure-Temperature Limits

Plant Name: Pilgrim

Docket Number: 50-293

NSSS Vendor: General Electric

Vessel Manufacturer: Combustion Engineering

Edition of ASME Code for Design: 1966 Addenda to the 1965 ASME Code

Date of Commercial Operation: December 1, 1972

Date of License Expiration: June 8, 2012

ART for the Limiting Beltline Material:

Limiting Beltline Material: All axial welds, heats 27204/12008
ID Fluence at EOL: 1.39E18 n/cm^2
Initial RT_{NDT}: 0°F
Method of Determining the Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 82°F
Margin: 56°F
ART at EOL: 138°F

USE for the Limiting Beltline Material:

Limiting Beltline Material: All axial Welds, heats 27204/12008
1/4T Fluence at EOL: 9.9E17 n/cm^2
Initial USE: 75 ft-lb
Percent Drop at EOL: 16%
USE at EOL: 63 ft-lb
Date at which Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Chemistry data per Paragraph C.1.2 of RG 1.99, Rev. 2

References:

Chemical composition and initial USE are from "Pilgrim Nuclear Power Station Reactor Pressure Vessel Pressure Temperature Limits," Technical Report TR-6052B-1, Rev. 1, Boston Edison, June 26, 1986

Initial RT data are from June 11, 1991, letter from G. W. Davis (BE) to USNRC Document Control Desk, Subject: Proposed Changes to the Reactor Pressure Vessel Thermal and Pressurization Technical Specification Limits

Fluence data are from April 16, 1991, letter from J. C. Tsacoyeanes (BE) to USNRC Document Control Desk, Subject: Pilgrim RPV Pressure Temperature Limits

Plant Name: Point Beach 1

Docket Number: 50-266

NSSS Vendor: Westinghouse

Vessel Manufacturer: Babcock and Wilcox

Edition of ASME Code for Design: 1965 ASME Code

Date of Commercial Operation: December 21, 1970

Date of License Expiration: October 5, 2010

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Circumferential weld SA-1101, heat 71249
ID Fluence at EOL: 2.43E19 n/cm^2
Initial RT_{NDT}: 10°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 for RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 223°F
Margin: 56°F
RT_{pts} at EOL: 289°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Lower shell axial welds SA-847, heat 61782 and circ. weld SA-1101, heat 71249
1/4T Fluence at EOL: 1.21E19 n/cm^2 for SA-847 and 1.78E19 n/cm^2 for SA-1101
Initial USE: Heat-specific value not reported
Percent Drop at EOL: 43% for SA-847 and SA-1101
USE at EOL: Not applicable since heat-specific value not reported
Date USE Screening Limit will be Exceeded: Not applicable since heat specific value not reported
Bases for Accepting the USE at EOL: Equivalent margin analyses was performed in Topical Reports BAW-2178P and BAW-2192P[1]

REFERENCES:

June 25, 1992, letter from B. Link (WEPCo) to USNRC Document Control Desk, Subject: Response to Generic Letter 92-01, Revision 1, Reactor Vessel Structural Integrity, 10 CFR 50.54(f)

Fluence data are from May 2, 1994, letter from B. Link (WEPCo) and BAW-2222.

---

[1]Staff will review applicability of Topical Reports BAW-2178P and BAW-2192P.

Plant Name: Point Beach 2

Docket Number: 50-301

NSSS Vendor: Westinghouse

Vessel Manufacturer: Combustion Engineering

Edition of ASME Code for Design: 1965 ASME Code

Date of Commercial Operation: October 1, 1972

Date of License Expiration: March 8, 2013

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Circumferential weld SA-1484, heat 72442
ID Fluence at EOL: 2.52E19 n/cm^2
Initial RT_{NDT}: -5°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 217°F
Margin: 68°F
RT_{pts} at EOL: 280°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Circumferential weld SA-1484, heat 72442
1/4T Fluence at EOL: 1.85E19 n/cm^2
Initial USE: Heat-specific value not reported
Percent Drop at EOL: 43%
USE at EOL: Not applicable since heat-specific value not reported
Date USE Screening Limit will be Exceeded: Not applicable since heat specific value not reported
Bases for Accepting the USE at EOL: Equivalent Margin Analyses were performed in Topical Reports BAW-2178P and BAW-2192[1]

REFERENCES:

June 25, 1992, letter from B. Link (WEPCo) to USNRC Document Control Desk, Subject: Response to Generic Letter 92-01, Revision 1, Reactor Vessel Structural Integrity, 10 CFR 50.54(f)

Fluence data are from May 2, 1994, letter from B. Link (WEPCo) and BAW-2222

[1]Staff will review applicability of Topical Reports BAW-2178P and BAW-2192P.

Plant Name: Prairie Island 1

Docket Number: 50-282

NSSS Vendor: Westinghouse

Vessel Manufacturer: Creusot-Loire (formerly SFAC)

Edition of ASME Code for Design: Winter 1972 addenda to the 1971 ASME Code

Date of Commercial Operation: December 16, 1973

Date of License Expiration: August 9, 2013

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate shell forging C, heat 21918/38566
ID Fluence at EOL: 4.5E19 n/cm^2
Initial RT_{NDT}: 14°F
Method of Determining Chemistry Factor: Surveillance data per Paragraph C.2.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 63°F
Margin: 17°F
RT_{pts} at EOL: 94°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Circumferential Weld, heat 1752
1/4T Fluence at EOL: 3.41E19 n/cm^2
Initial USE: 78 ft-lb
Percent Drop at EOL: 36%
USE at EOL: 50 ft-lb
Date at which USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Chemistry data per Paragraph C.1.2 of RG 1.99, Rev. 2

References:

Fluence and initial RT_{NDT} for all materials, initial USE data for Forging and the circumferential weld, and chemical composition data for Forgings C are from July 6, 1992, letter from T.M. Parker (NSP) to USNRC Document Control Desk, Subject: Response to GL 92-01, Rev. 1.

Chemical composition values for circumferential weld are from November 26, 1993, letter to the NRC.

The 1/4T fluence and the USE at EOL for the circumferential weld are from March 31, 1994, letter to the NRC.

Plant Name: Prairie Island 2

Docket Number: 50-306

NSSS Vendor: Westinghouse

Vessel Manufacturer: Creusot-Loire (formerly SFAC)

Edition of ASME Code for Design: Winter 1972 addenda to the 1971 ASME Code

Date of Commercial Operation: December 21, 1974

Date of License Expiration: October 29, 2014

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate shell forging C, heat 22829
ID Fluence at EOL: 4.5E19 n/cm^2
Initial RT_{NDT}: 14°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 69°F
Margin: 34°F
RT_{pts} at EOL: 117°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Circumferential weld, heat 2721
1/4T Fluence at EOL: 3.01E19 n/cm^2
Initial USE: 103 ft-lb
Percent Drop at EOL: 34%
USE at EOL: 68 ft-lb
Date at which USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Chemistry data per Paragraph C.1.2 of RG 1.99, Rev. 2

References:

The initial USE for the circumferential weld are from July 6, 1992, letter from T.M. Parker (NSP) to USNRC Document Control Desk, Subject: Response in GL 92-01, Reactor Vessel Structural Integrity.

Chemical composition values and initial RT_{NDT} data for forgings C and D are from November 24, 1993, letter to the NRC.

The 1/4T fluence and USE at EOL for the circumferential weld are from March 31, 1994, letter to the NRC.

Plant Name: Quad Cities 1

Docket Number: 50-254

NSSS Vendor: General Electric

Vessel Manufacturer: Babcock and Wilcox

Edition of ASME Code for Design: Summer 1965 Addenda to the 1965 ASME Code

Date of Commercial Operation: February 18, 1973

Date of License Expiration: December 14, 2012

ART for the Limiting Beltline Material:

Limiting Beltline Material: Axial welds, heat PQ2563
ID Fluence at EOL: 3.5E17 n/cm^2
Initial RT_{NDT}[1]: 40°F
Method of Determining the Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 65°F
Margin: 56°F
ART at EOL: 161°F

USE for the Limiting Beltline Material:

Limiting Beltline Material: Lower shell plate, heat B5524-1 and Axial Welds, heat PQ2563
1/4T Fluence at EOL: 2.4E17 n/cm^2
Initial USE: Heat-specific values not reported
Percent Drop at EOL: 17% for plate and 26% for weld
USE at EOL: Not applicable because heat-specific values not reported
Date at which Screening Limit will be Exceeded: Not applicable because heat-specific values not reported
Bases for Accepting the USE at EOL: Equivalent margins analysis was performed in Topical Report NEDO-32205, Rev. 1[2]

References:

Initial RT_{NDT}, chemical composition, initial USE, and fluence data are from July 1, 1992, letter from M. A. Jackson (CECo) to T. E. Murley (USNRC), Subject: Dresden Station Units 2 and 3; Quad Cities Station Units 1 and 2; LaSalle County Station Units 1 and 2

---

[1]Staff will review the forthcoming GE Topical Report on initial RT_{NDT} determination

[2]Staff will review applicability of Topical Report NEDO-32205, Rev. 1

Plant Name: Quad Cities 2

Docket Number: 50-265

NSSS Vendor: General Electric

Vessel Manufacturer: CB&I, B&W, and Rotterdam Dockyard Co.

Edition of ASME Code for Design: Summer 1965 Addenda to the 1965 ASME Code

Date of Commercial Operation: March 10, 1973

Date of License Expiration: December 14, 2012

ART for the Limiting Beltline Material:

Limiting Beltline Material: Axial welds, heat PQ-1300
ID Fluence at EOL: 4.9E17 n/cm^2
Initial RT_{NDT}[1]: 40°F
Method of Determining the Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 46°F
Margin: 46°F
ART at EOL: 132°F

USE for the Limiting Beltline Material:

Limiting Beltline Material: Lower shell plate, heat C-1501-2 and axial welds, heat PQ-1300
1/4T Fluence at EOL: 3.4E17 n/cm^2
Initial USE: Heat-specific values not reported
Percent Drop at EOL: 12% for plate and 20% for weld
USE at EOL: Not applicable because heat-specific values not reported
Date at which Screening Limit will be Exceeded: Not applicable because heat-specific values not reported
Bases for Accepting the USE at EOL: Equivalent margins analysis was performed in Topical Report NEDO-32205, Rev. 1[2]

References:

Initial RT_{NDT}, chemical composition, initial USE, and fluence data are from July 1, 1992, letter from M. A. Jackson (CECo) to T. E. Murley (USNRC), Subject: Dresden Station Units 2 and 3; Quad Cities Station Units 1 and 2; LaSalle County Station Units 1 and 2

---

[1]Staff will review the forthcoming GE Topical Report on initial RT_{NDT} determination

[2]Staff will review applicability of Topical Report NEDO-32205, Rev. 1

Plant Name: River Bend

Docket Number: 50-458

NSSS Vendor: General Electric

Vessel Manufacturer: Chicago Bridge and Iron

Edition of ASME Code for Design: Summer 1973 Addenda to the 1971 ASME Code

Date of Commercial Operation: June 16, 1986

Date of License Expiration: August 29, 2025

ART for the Limiting Beltline Material:

Limiting Beltline Material: Axial welds, heat 5P6756
ID Fluence at EOL: 6.64E18 n/cm^2
Initial RT_{NDT}: -50°F
Method of Determining the Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 108°F
Margin: 56°F
ART at EOL: 114°F

USE for the Limiting Beltline Material:

Limiting Beltline Material: Plate, heat C3138-2
1/4T Fluence at EOL: 4.8E18 n/cm^2
Initial USE: 79 ft-lb
Percent Drop at EOL: 15.9%
USE at EOL: 67 ft-lb
Date at which Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Chemistry data per Paragraph C.1.2 of RG 1.99, Rev. 2

References:

Initial RT_{NDT}, chemical composition, fluence, and initial USE data are from July 2, 1992, letter from W. H. Odell (GSUCo) to USNRC Document Control Desk, Subject: River Bend--Unit 1, Docket No. 50-458

Plant Name: Robinson 2

Docket Number: 50-261

NSSS Vendor: Westinghouse

Vessel Manufacturer: Combustion Engineering

Edition of ASME Code for Design: 1965 ASME Code

Date of Commercial Operation: March 7, 1971

Date of License Expiration: July 31, 2010

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Lower circumferential weld 11-273, heat 34B009
ID Fluence at EOL: 2.0E19 n/cm^2
Initial RT_{ndt}: -56°F
Method of Determining the Chemistry Factor: Chemistry data per paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{ndt} at EOL: 235°F
Margin: 66°F
RT_{pts} at EOL: 245°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Upper shell plate W10201-1, heat A6623-1
1/4T Fluence at EOL: 1.03E19 n/cm^2
Initial USE: 54 ft-lb
Percent Drop at EOL: 22%
USE at EOL: 42 ft-lb
Date at which USE Screening Limit will be Exceeded: Before EOL
Bases for Accepting the USE at EOL: Equivalent margins analysis was performed in Westinghouse Topical Report WCAP-13587, Revision 1 and approved by the staff

References:

Initial RT_{ndt} data are from February 4, 1986, letter from S. R. Zimmerman (CP&L) to L. S. Rubinstein (USNRC)

Fluence and chemistry data are from July 6, 1992, letter from R. B. Starkey (CP&L) to USNRC Document Control Desk.

June 13, 1994, letter from R.M. Krich (CP&L) to USNRC Document Control Desk.

Safety Evaluation for equivalent margin analysis is in an USNRC letter to the licensee dated September 23, 1994

Plant Name: Salem 1

Docket Number: 50-272

NSSS Vendor: Westinghouse

Vessel Manufacturer: Combustion Engineering

Edition of ASME Code for Design: Winter 1965 Addenda to the 1965 ASME Code

Date of Commercial Operation: June 30, 1977

Date of License Expiration: August 13, 2016

RT_{pts} for the Limiting Beltline Material:
Limiting Beltline Material: Lower shell axial welds 3-042A, B, C, heat 34B009
ID Fluence at EOL: 1.37E19 n/cm^2
Initial RT_{NDT}: -56°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 240°F
Margin: 66°F
RT_{pts} at EOL: 250°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:
Limiting Beltline Material: Intermediate shell axial welds 2-042, heat 39B196/34B009
1/4T Fluence at EOL: 6.7E18 n/cm^2
Initial USE: 75 ft-lb[1]
Percent Drop at EOL: 29.3%
USE at EOL: 53 ft-lb
Date at which USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Chemistry data per Paragraph C.1.2 of RG 1.99, Rev. 2

References:

Fluence data are from June 20, 1994, letter from S. LaBruna (PSE&G) to the NRC.

Chemical composition, initial RT_{NDT}, and initial USE data are from June 30, 1992, letter from S. E. Miltenberger (PSE&G) to USNRC Document Control Desk, Subject: Response to Generic Letter 92-01, Revision 1, Reactor Vessel Structure Integrity, 10CFR50.54(f), Salem Generating Station 1 and 2.

---

[1]Initial USE is an NRC generic value

Plant Name: Salem 2

Docket Number: 50-311

NSSS Vendor: Westinghouse

Vessel Manufacturer: Combustion Engineering

Edition of ASME Code for Design: Winter 1966 Addenda to the 1965 ASME Code

Date of Commercial Operation: October 13, 1981

Date of License Expiration: April 18, 2020

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Lower shell axial welds 3-442, heats 21935/12008
ID Fluence at EOL: 1.18E19 n/cm^2
Initial RT_{NDT}: -56°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 209°F
Margin: 66°F
RT_{pts} at EOL: 219°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material[1]:

Limiting Beltline Material: Intermediate to lower shell circumferential weld 9-442, heat 90099
1/4T Fluence at EOL: 8.6E18 n/cm^2
Initial USE: 75 ft-lb
Percent Drop at EOL: 31%
USE at EOL: 52 ft-lb
Date at which USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Chemistry data per Paragraph C.1.2 of RG 1.99, Rev. 2

References:

Fluence data are from June 20, 1994, letter from S. LaBruna (PSE&G) to the NRC.

Chemical composition, initial RT_{NDT}, and initial USE data are from June 30, 1992, letter from S. E. Miltenberger (PSE&G) to USNRC Document Control Desk, Subject: Response to Generic Letter 92-01, Revision 1, Reactor Vessel Structure Integrity, 10CFR50.54(f), Salem Generating Station 1 and 2.

---

[1]Additional information will be provided by the CEOG in 1995 to further demonstrate welds 2-442, 3-442, and 9-442 have irradiated USE of greater than 50 ft-lb at EOL.

Plant Name: San Onofre 2

Docket Number: 50-361

NSSS Vendor: Combustion Engineering

Vessel Manufacturer: Combustion Engineering

Edition of ASME Code for Design: Summer 1971 Addenda to the 1971 ASME Code

Date of Commercial Operation: August 8, 1983

Date of License Expiration: October 18, 2013

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Lower shell plate C-6404-5, heat C-7585-1
ID Fluence at EOL: 4.2E19 n/cm^2
Initial RT_{NDT}: 10°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 102°F
Margin: 34°F
RT_{pts} at EOL: 146°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate axial welds 2-203 A, B, C, heat BOLA
1/4T Fluence at EOL: 2.503E19 n/cm^2
Initial USE: 93 ft-lb
Percent Drop at EOL: 23.6%
USE at EOL: 71 ft-lb
Date at which USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Chemistry data per Paragraph C.1.2 of RG 1.99, Rev. 2

References:

Chemical composition, fluence, initial RT_{NDT}, and initial USE data are from July 22, 1994, letter from W. C. Marsh (SCE) to USNRC Document Control Desk, Subject: Revision to supplemental response to Generic Letter 92-01, Revision 1, "Reactor Vessel Structural Integrity, 10 CFR 50.54(f)" San Onofre Nuclear Generating Station Units 2 and 3

Plant Name: San Onofre 3

Docket Number: 50-362

NSSS Vendor: Combustion Engineering

Vessel Manufacturer: Combustion Engineering

Edition of ASME Code for Design: Summer 1971 Addenda to 1971 ASME Code

Date of Commercial Operation: April 1, 1984

Date of License Expiration: October 18, 2013

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate shell plate C6802-1, heat C9195-2
ID Fluence at EOL: 4.2E19 n/cm^2
Initial RT_{NDT}: 40°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 51°F
Margin: 34°F
RT_{pts} at EOL: 125°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Lower shell plate C6802-6, heat B-3388-1 and intermediate to lower shell circumferential welds 9-203, heat 90144
1/4T Fluence at EOL: 2.503E19 n/cm^2
Initial USE: 92 ft-lb (C6802-6); 91 ft-lb (9-203)
Percent Drop at EOL: 24% (C6802-6); 23% (9-203)
USE at EOL: 70 ft-lb
Date at which USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Chemistry data per Paragraph C.1.2 of RG 1.99, Rev. 2

References:

Chemical composition, fluence, initial RT_{NDT}, and initial USE data are from July 22, 1994, letter from W. C. Marsh (SCE) to USNRC Document Control Desk, Subject: Revision to supplemental response to Generic Letter 92-01, Revision 1, "Reactor Vessel Structural Integrity, 10 CFR 50.54(f)" San Onofre Nuclear Generating Station Units 2 and 3

Plant Name: Seabrook

Docket Number: 50-443

NSSS Vendor: Westinghouse

Vessel Manufacturer: Combustion Engineering

Edition of ASME Code for Design: Winter 1972 Addenda to the 1971 ASME Code

Date of Commercial Operation: August 19, 1990

Date of License Expiration: October 17, 2026

RT_{PTS} for the Limiting Beltline Material:

Limiting Beltline Material: Lower shell plate R1808-3 (no heat number)
ID Fluence at EOL: 3.1E19 n/cm^2
Initial RT_{NDT}: 40°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 48°F
Margin: 34°F
RT_{PTS} at EOL: 122°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Lower shell plate R1808-2 (no heat number)
1/4T Fluence at EOL: 1.85E19 n/cm^2
Initial USE: 77 ft-lb
Percent Drop at EOL: 22%
USE at EOL: 60 ft-lb
Date at which USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Chemistry data per Paragraph C.1.2 of RG 1.99, Rev. 2

References:

Initial USE and initial RT_{NDT} data are from August 17, 1992, letter from T. C. Feigenbaum (PSNH) to USNRC Document Control Desk, Subject: Reactor Vessel Surveillance Capsule Report.

Fluence data are from pressurized thermal shock submittal, dated August 17, 1992.

Plant Name: Sequoyah 1

Docket Number: 50-327

NSSS Vendor: Westinghouse

Vessel Manufacturer: Rotterdam Dockyard

Edition of ASME Code for Design: 1968 ASME Code

Date of Commercial Operation: July 1, 1981

Date of License Expiration: September 17, 2020

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate shell forging 05, heat 980807/281489
ID Fluence at EOL: 3.838E19 n/cm^2
Initial RT_{NDT}: 40°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 156°F
Margin: 34°F
RT_{pts} at EOL: 230°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate shell forging 05, heat 980807/281489
1/4T Fluence at EOL: 2.29E19 n/cm^2
Initial USE: 74 ft-lb
Percent Drop at EOL: 28.9%
USE at EOL: 53 ft-lb
Date at which USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Chemistry data per Paragraph C.1.2 of RG 1.99, Rev. 2

References:

Initial RT_{NDT} value for forgings 05 has been revised to 40°F as shown in response to GL 92-01 dated September 27, 1993.

Chemical composition, initial USE, initial RT_{NDT}, and fluence data are from July 7, 1992, letter from R. M. Shell (TVA) to USNRC Document Control Desk, Subject: Browns Ferry Nuclear Plant, Sequoyah Nuclear Plant, and Watts Bar Nuclear Plant--Response to Generic Letter 92-01 (Reactor Vessel Structural Integrity)

Plant Name: Sequoyah 2

Docket Number: 50-328

NSSS Vendor: Westinghouse

Vessel Manufacturer: Rotterdam Dockyard

Edition of ASME Code for Design: 1968 ASME Code

Date of Commercial Operation: June 1, 1982

Date of License Expiration: September 15, 2021

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate shell Forging 05, heat 288757/981057
ID Fluence at EOL: 1.669E19 n/cm^2
Initial RT_{NDT}: 10°F
Method of Determining Chemistry Factor: Surveillance data per Paragraph C.2.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 105°F
Margin: 17°F
RT_{pts} at EOL: 132°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Circumferential weld, heat 4278
Initial USE: 105 ft-lb
Percent Drop at EOL: 34%
USE at EOL: 70 ft-lb
Date at which Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Surveillance data per Paragraph C.2.2 of RG 1.99, Rev. 2

References:

Chemical composition, initial RT_{NDT}, initial USE, and fluence data are from July 7, 1992, letter from R. M. Shell (TVA) to USNRC Document Control Desk, Subject: Browns Ferry Nuclear Plant, Sequoyah Nuclear Plant, and Watts Bar Nuclear Plant--Response to Generic Letter 92-01 (Reactor Vessel Structural Integrity)

Plant Name: Shearon Harris

Docket Number: 50-400

NSSS Vendor: Westinghouse

Vessel Manufacturer: Chicago Bridge and Iron

Edition of ASME Code for Design: Winter 1971 Addenda to the 1971 ASME Code

Date of Commercial Operation: May 2, 1987

Date of License Expiration: October 24, 2026

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate shell plate, heat A9153-1
ID Fluence at EOL: 3.42E19 n/cm^2
Initial RT_{NDT}: 60°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 77°F
Margin: 34°F
RT_{pts} at EOL: 171°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate shell plates, heats A9153-1 and B4197-2
1/4T Fluence at EOL: 2.158E19
Initial USE: 83 ft-lb for A9153-1 and 71 ft-lb for B4197-2
Percent Drop at EOL: 22.7% for A9153-1 and 10% for B4197-2
USE at EOL: 64 ft-lb
Date at which USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Chemistry data per Paragraph C.1.2 of RG 1.99, Rev. 2 for plate A9153-1 and surveillance data per Paragraph C.2.2 of RG 1.99, Rev. 2 for plate B4197-2.

References:

Fluence, Chemistry, initial RT_{NDT}, and initial USE data are from April 2, 1992 surveillance capsule report, "Analysis of Capsule V"

Plant Name: South Texas 1

Docket Number: 50-493

NSSS Vendor: Westinghouse

Vessel Manufacturer: Combustion Engineering

Edition of ASME Code for Design: Summer 1973 Addenda to the 1971 ASME Code

Date of Commercial Operation: August 25, 1988

Date of License Expiration: August 20, 2027

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate shell plate R1606-3, heat C4326-2
ID Fluence at EOL: 2.90E19 n/cm^2
Initial RT_{NDT}: 10°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 39°F
Margin: 34°F
RT_{pts} at EOL: 83°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate shell plate R1606-2, heat B8120-1
1/4T Fluence at EOL: 1.73E19 n/cm^2
Initial USE: 94 ft-lb
Percent Drop at EOL: 21.6%
USE at EOL: 74 ft-lb
Date at which USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Chemistry data per Paragraph C.1.2 of RG 1.99, Rev. 2

References:

Fluence and plate heat numbers are from June 21, 1994, letter to the NRC.

All the remaining data are from the FSAR.

Plant Name: South Texas 2

Docket Number: 50-499

NSSS Vendor: Westinghouse

Vessel Manufacturer: Combustion Engineering

Edition of ASME Code for Design: Summer 1973 Addenda to the 1971 ASME Code

Date of Commercial Operation: June 19, 1989

Date of License Expiration: December 15, 2028

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate shell plate R2507-2, heat NR62248-1
ID Fluence at EOL: 2.89E19 n/cm^2
Initial RT_{NDT}: -10°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 39°F
Margin: 34°F
RT_{pts} at EOL: 63
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Circumferential weld, heat 90209
1/4T Fluence at EOL: 1.72E19 n/cm^2
Initial USE: 101 ft-lb
Percent Drop at EOL: 21.5%
USE at EOL: 79 ft-lb
Date at which USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Chemistry data per Paragraph C.1.2 of RG 1.99, Rev. 2

References:

Fluence and plate heat numbers are from June 21, 1994, letter to the NRC.

All the remaining data are from the FSAR.

Plant Name: St. Lucie 1

Docket Number: 50-335

NSSS Vendor: Combustion Engineering

Vessel Manufacturer: Combustion Engineering

Edition of ASME Code for Design: Winter 1967 Addenda to the 1965 ASME Code

Date of Commercial Operation: December 21, 1976

Date of License Expiration: March 1, 2016

RT_{pts} for the Limiting Beltline Material:
Limiting Beltline Material: Lower shell axial welds 3-203, heat 305424
ID Fluence at EOL: 2.13E19 n/cm^2
Initial RT_{NDT}: -60°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 230°F
Margin: 56°F
RT_{pts} at EOL: 226°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material[1]:
Limiting Beltline Material: Lower shell plates, heats C-5935-1 and A-4567-2
1/4T Fluence at EOL: 2.01E19 n/cm^2
Initial USE: 82 ft-lb (C-5935-1); 76 ft-lb (A-4567-2)
Percent Drop at EOL: 28% (C-5935-1); 29% (A-4567-2)
USE at EOL: 59 ft-lb
Date at which USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Chemistry data per Paragraph C.1.2 of RG 1.99, Rev. 2

References:

Chemical composition and initial RT_{ndt} for weld 3-203are from the November 15, 1993, letter to the NRC (response to GL 92-01 RAI).

Fluence, chemical composition, and initial USE data are from July 1, 1992, letter from W. H. Bohlke (FPL) to USNRC Document Control Desk, Subject: Generic Letter 92-01, Revision 1, Response

[1]Additional information will be provided by the CEOG in 1995 to further demonstrate that weld heats A-8746/34B009 have irradiated USE of greater than 50 ft-lb at EOL

Plant Name: St. Lucie 2

Docket Number: 50-389

NSSS Vendor: Combustion Engineering

Vessel Manufacturer: Combustion Engineering

Edition of ASME Code for Design: Summer 1972 Addenda to the 1971 ASME Code

Date of Commercial Operation: August 8, 1983

Date of License Expiration: April 6, 2023

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate shell plate M-605-2, heat B-3416-2
ID Fluence at EOL: 3.07E19 n/cm^2
Initial RT_{NDT}: 10°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 119°F
Margin: 34°F
RT_{pts} at EOL: 163°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Lower shell plate M-4116-1, heat B-8307-2
1/4T Fluence at EOL: 1.83E19 n/cm^2
Initial USE: 91 ft-lb
Percent Drop at EOL: 21.9%
USE at EOL: 71 ft-lb
Date at which USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Chemistry data per Paragraph C.1.2 of RG 1.99, Rev. 2

References:

All data are from the November 15, 1993, letter from D. A. Sager to the NRC, "St. Lucie Units 1 and 2, Response to Request for Additional Information GL 92-01, Revision 1."

Plant Name: Summer

Docket Number: 50-395

NSSS Vendor: Westinghouse

Vessel Manufacturer: Chicago Bridge and Iron

Edition of ASME Code for Design: 1971 ASME Code

Date of Commercial Operation: January 1, 1984

Date of License Expiration: August 6, 2022

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Lower shell plates, heats C9923-2 and C9923-1
ID Fluence at EOL: 3.87E19 n/cm^2
Initial RT_{NDT}: 10°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 69°F
Margin: 34°F
RT_{pts} at EOL: 113°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate shell plate, heat A-9154-1
1/4T Fluence at EOL: 2.43E19 n/cm^2
Initial USE: 81 ft-lb
Percent Drop at EOL: 9%
USE at EOL: 74 ft-lb
Date USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Surveillance data per Paragraph C.2.2 of RG 1.99, Rev. 2

REFERENCES:

June 30, 1992, letter from J. F. Skolds (SCE&G) to USNRC Document Control Desk, Subject: Response to Generic Letter 92-01, Revision 1, Reactor Vessel Structural Integrity

Plant Name: Surry 1

Docket Number: 50-280

NSSS Vendor: Westinghouse

Vessel Manufacturer: Rotterdam Dockyard, B&W

Edition of ASME Code for Design: Winter 1968 Addenda to the 1968 ASME Code

Date of Commercial Operation: December 22, 1972

Date of License Expiration: May 25, 2012

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Lower shell axial weld SA-1526, heat 299L44
ID Fluence at EOL: 6.39E18 n/cm^2
Initial RT_{NDT}: -7°F
Method of Determining Chemistry Factor: Surveillance data per Paragraph C.2.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 192°F
Margin: 50°F
RT_{pts} at EOL: 235°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate to lower shell circumferential welds SA-1585, heat 72445 and J726, heat 25017; axial weld SA-1494, heat 8T1554
1/4T Fluence at EOL: 2.20E19 n/cm^2 for SA-1585, 3.54E18 n/cm^2 for SA-1494 and 2.64E18 for J726.
Initial USE: 77 ft-lb for SA-1585 and heat specific value not reported for SA-1494 and J726
Percent Drop at EOL: 42% for SA-1585, 25% for SA-1494, and 34% for J726
USE at EOL: 45 ft-lb for SA-1585; not applicable for heats SA-1494 and J726 since heat specific values not reported.
Date USE Screening Limit will be Exceeded: Before EOL
Bases for Accepting the USE at EOL: Equivalent margin analyses were performed in Topical Reports BAW-2178P and BAW 2192P[1]

REFERENCES:

Fluence, chemical composition, and initial RT_{ndt} are from reports BAW-2166, BAW-1803, Rev. 1, and BAW-2222.

---

[1]Staff will review applicability of Topical Reports BAW-2178P and BAW-2192P.

Plant Name: Surry 2

Docket Number: 50-281

NSSS Vendor: Westinghouse

Vessel Manufacturer: Rotterdam Dockyard, B&W

Edition of ASME Code for Design: Winter 1968 Addenda to the 1968 ASME Code

Date of Commercial Operation: May 1, 1973

Date of License Expiration: January 29, 2013

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Lower shell axial welds WF-4 and WF-8, heat 8T1762
ID Fluence at EOL: 7.14E18 n/cm^2
Initial RT_{NDT}: -5°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 138°F
Margin: 68°F
RT_{pts} at EOL: 201°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Lower shell axial welds WF-4 and WF-8, heat 8T1762; Weld L737, heat 4275
1/4T Fluence at EOL: 3.88E18 n/cm^2 for WF-4 and WF-8; and 2.23E18 n/cm^2 for L737
Initial USE: Heat-specific value not reported
Percent Drop at EOL: 27.5% for WF-4 and WF-8; 34% for L737
USE at EOL: Not applicable since heat-specific value not reported
Date USE Screening Limit will be Exceeded: Not applicable since heat specific value not reported
Bases for Accepting the USE at EOL: Equivalent margin analyses were performed in Topical Reports BAW-2178P and BAW 2192P[1]

REFERENCES:

Fluence, chemical composition, and initial RT_{ndt} are from reports BAW-2166, BAW-1803, Rev. 1, and BAW-2222.

---

[1]Staff will review applicability of Topical Reports BAW-2178P and BAW-2192P.

Plant Name: Susquehanna 1

Docket Number: 50-387

NSSS Vendor: General Electric

Vessel Manufacturer: Chicago Bridge and Iron

Edition of ASME Code for Design: Summer 1970 Addenda to the 1968 ASME Code

Date of Commercial Operation: June 8, 1983

Date of License Expiration: July 17, 2022

ART for the Limiting Beltline Material:

Limiting Beltline Material: Lower intermediate shell plate, heat C2433-1
ID Fluence at EOL: 7.1E17 n/cm^2
Initial RT_{NDT}[1]: 18°F
Method of Determining the Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 23°F
Margin: 23°F
ART at EOL: 64°F

USE for the Limiting Beltline Material:

Limiting Beltline Material: Shell plate 22-2, heat C0776-1
1/4T Fluence at EOL: 4.9E17 n/cm^2
Initial USE: Heat-specific values not reported
Percent Drop at EOL: less than 15%
USE at EOL: Not applicable because heat-specific values not reported
Date at which Screening Limit will be Exceeded: Not applicable because heat-specific values not reported
Bases for accepting USE at EOL: Equivalent margins analysis was performed in Topical Report NEDO-32205, Rev. 1[2]

References:

Initial RT_{NDT}, Fluence and chemical composition data are from July 10, 1992, letter from H. W. Keiser (PP&L) to C. L. Miller (USNRC), Subject: Response to Generic Letter 92-01 and from June 23, 1994, letter from R. G. Byram (PP&L) to C. L. Miller

---

[1]Staff will review the forthcoming GE Topical Report on initial RT_{NDT} determination

[2]Staff will review applicability of Topical Report NEDO-32205, Rev. 1

Plant Name: Susquehanna 2

Docket Number: 50-388

NSSS Vendor: General Electric

Vessel Manufacturer: Chicago Bridge and Iron

Edition of ASME Code for Design: Summer 1970 Addenda to the 1968 ASME Code

Date of Commercial Operation: December 12, 1985

Date of License Expiration: March 23, 2025

ART for the Limiting Beltline Material:

Limiting Beltline Material: Lower intermediate shell plate, heat 6C1053-1-1
ID Fluence at EOL: 6.4E17 n/cm^2
Initial RT_{NDT}[1]: 10°F
Method of Determining the Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 22°F
Margin: 22°F
ART at EOL: 54°F

USE for the Limiting Beltline Material:

Limiting Beltline Material: Weld, heat 09M057/C109A27A
1/4T Fluence at EOL: 5.4E17 n/cm^2
Initial USE: Heat-specific values not reported
Percent Drop at EOL: less than 15%
USE at EOL: Not applicable because heat-specific values not reported
Date at which Screening Limit will be Exceeded: Not applicable because heat-specific values not reported
Bases for Accepting the USE at EOL: Equivalent margins analysis was performed in Topical Report NEDO-32205, Rev. 1[2]

References:

Initial RT_{NDT}, Fluence and chemical composition data are from July 10, 1992, letter from H. W. Keiser (PP&L) to C. L. Miller (USNRC), Subject: Response to Generic Letter 92-01 and from June 23, 1994, letter from R. G. Byram (PP&L) to C. L. Miller

[1] Staff will review the forthcoming GE Topical Report on initial RT_{NDT} determination

[2] Staff will review applicability of Topical Report NEDO-32205, Rev. 1

Plant Name: TMI-1

Docket Number: 50-289

NSSS Vendor: Babcock and Wilcox

Vessel Manufacturer: Babcock and Wilcox

Edition of ASME Code for Design: Summer 1967 Addenda to the 1965 ASME Code

Date of Commercial Operation: September 2, 1974

Date of License Expiration: April 19, 2014

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Upper to lower circumferential weld WF-25 heat 299L44
ID Fluence at EOL: 8.6E18 n/cm^2
Initial RT_{NDT}: -7°F
Method of Determining Chemistry Factor: Surveillance data per Paragraph C.2.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 208°F
Margin: 50°F
RT_{pts} at EOL: 251°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Axial weld SA-1526, heat 299L44
1/4T Fluence at EOL: 4.35E18 n/cm^2.
Initial USE: Heat-specific value not reported
Percent Drop at EOL: 37.5%
USE at EOL: Not applicable since heat-specific value not reported
Date at which USE Screening Limit will be Exceeded: Not applicable since heat-specific value not reported
Bases for Accepting the USE at EOL: Equivalent margins analyses were performed in Topical Reports BAW-2178P and BAW-2192P[1]

References:

Chemical composition, fluence, and initial RT_{ndt} are from reports BAW-2222, BAW-2166, and BAW-1820.

---

[1]Staff will review applicability of Topical Reports BAW-2178P and BAW-2192P.

Plant Name: Turkey Point 3

Docket Number: 50-250

NSSS Vendor: Westinghouse

Vessel Manufacturer: Babcock and Wilcox

Edition of ASME Code for Design: Summer 1966 Addenda to the 1965 ASME Code

Date of Commercial Operation: December 14, 1972

Date of License Expiration: April 27, 2007

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate to lower shell circumferential weld SA-1101, heat 71249
ID Fluence at EOL: 2.64E19 n/cm^2
Initial RT_{NDT}: 10°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 227°F
Margin: 56°F
RT_{pts} at EOL: 293°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate to lower shell circumferential weld SA-1101, heat 71249
1/4T Fluence at EOL: 1.58E19 n/cm^2
Initial USE: 65 ft-lb
Percent Drop at EOL: 41%
USE at EOL: 38 ft-lb
Date at which USE Screening Limit will be Exceeded: Before EOL
Bases for Accepting the USE at EOL: Equivalent margins analyses were performed in the plant-specific report BAW-2118P and Topical Report BAW-2178P[1]

References:

The chemical composition and ID fluence data for all beltline materials, and the initial USE data for weld SA-1101 are from BAW-2166.

[1]Staff will review applicability of Topical Report BAW-2178P

Plant Name: Turkey Point 4

Docket Number: 50-251

NSSS Vendor: Westinghouse

Vessel Manufacturer: Babcock and Wilcox

Edition of ASME Code for Design: Summer 1966 Addenda to the 1965 ASME Code

Date of Commercial Operation: September 7, 1973

Date of License Expiration: April 27, 2007

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate to lower shell circumferential weld SA-1101, heat 71249
ID Fluence at EOL: 2.53E19 n/cm^2
Initial RT_{NDT}: 10°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 225°F
Margin: 56°F
RT_{pts} at EOL: 291°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate to lower shell circumferential weld SA-1101, heat 71249
1/4T Fluence at EOL: 1.59E19 n/cm^2
Initial USE: 65 ft-lb
Percent Drop at EOL: 41%
USE at EOL: 38 ft-lb
Date at which USE Screening Limit will be Exceeded: Before EOL
Bases for Accepting the USE at EOL: Equivalent margins analyses were performed in the plant-specific report BAW-2118P and Topical Report BAW-2178P[1]

References:

The chemical composition and ID fluence data for all beltline materials were from BAW-2166.

Initial USE data for weld SA-1101 are from the FSAR.

[1]Staff will review applicability of Topical Report BAW-2178P

Plant Name: Vermont Yankee

Docket Number: 50-271

NSSS Vendor: General Electric

Vessel Manufacturer: Chicago Bridge and Iron

Edition of ASME Code for Design: 1966 and 1967 Addenda to the 1965 ASME Code

Date of Commercial Operation: November 30, 1972

Date of License Expiration: March 21, 2012

ART for the Limiting Beltline Material:

Limiting Beltline Material: Plate, heat C3017-2 and C3116-2
ID Fluence at EOL: 2.3E17 n/cm^2
Initial RT_{NDT}: 40°F (C3017-2) and 30°F (C3116-2)
Method of Determining the Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 14°F (C2017-2) and 19°F (C3116-2)
Margin: 14°F (C3017-2) and 19°F (C3116-2)
ART at EOL: 68°F

USE for the Limiting Beltline Material:

Limiting Beltline Material: Plate, heat C2640-1 and shielded metal arc welds (no heat numbers)
1/4T Fluence at EOL: 1.7E17 n/cm^2
Initial USE: Heat-specific values not reported
Percent Drop at EOL: less than 10%
USE at EOL: Not applicable since heat-specific value not reported
Date at which Screening Limit will be Exceeded: Not applicable since heat-specific value not reported
Bases for Accepting the USE at EOL: Equivalent margin analysis was performed in Topical Report NEDO-32205, Rev. 1 and approved in an NRC letter to the licensee dated July 20, 1994

References:

Fluence data are from July 3, 1992, letter from J. P. Pelletier (VYNPC) to USNRC Document Control Desk, Subject: Vermont Yankee Response to Generic Letter 92-01 Regarding Reactor Vessel Structural Integrity

Chemical composition and initial RT_{NDT} data are from licensee's response to GL 92-01 dated September 24, 1993

Equivalent margin analysis submitted in a December 21, 1993, letter from licensee

Plant Name: Vogtle 1

Docket Number: 50-424

NSSS Vendor: Westinghouse

Vessel Manufacturer: Combustion Engineering

Edition of ASME Code for Design: Summer 1972 Addenda to the 1971 ASME Code

Date of Commercial Operation: June 1, 1987

Date of License Expiration: January 16, 2027

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate shell plate B8805-2 (no heat number)
ID Fluence at EOL: 3.16E19 n/cm^2
Initial RT_{ndt}: 20°F
Method of Determining the Chemistry Factor: Chemistry data per paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{ndt} at EOL: 67°F
Margin: 34°F
RT_{pts} at EOL: 121°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate shell plate B8805-1 (no heat number)
1/4T Fluence at EOL: 1.883E19 n/cm^2
Initial USE: 90 ft-lb
Percent Drop at EOL: 22%
USE at EOL: 70 ft-lb
Date at which USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Chemistry data per paragraph C.1.2 of RG 1.99, Rev. 2

References:

June 25, 1992, letter from C. K. McCoy (GPCo) to USNRC Document Control Desk, subject: Vogtle Electric Generating Plant, Reactor Vessel Structural Integrity, Generic Letter 92-01, Revision 1

November 5, 1993 and January 24, 1994 letters to the USNRC, subject: Response to the staff request for additional information.

Plant Name: Vogtle 2

Docket Number: 50-425

NSSS Vendor: Westinghouse

Vessel Manufacturer: Combustion Engineering

Edition of ASME Code for Design: Summer 1972 Addenda to the 1971 ASME Code

Date of Commercial Operation: May 20, 1989

Date of License Expiration: February, 9, 2029

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Lower shell plate B8628-1 (no heat number)
ID Fluence at EOL: 3.17E19 n/cm^2
Initial RT_{ndt}: 50°F
Method of Determining the Chemistry Factor: Chemistry data per paragraph C.1.1 of RG 1.99
Increase in RT_{ndt} at EOL: 40°F
Margin: 34°F
RT_{pts} at EOL: 124°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Lower shell plate B8825-1 (no heat number)
1/4T Fluence at EOL: 1.889E19 n/cm^2
Initial USE: 83 ft-lb
Percent Drop at EOL: 22.1%
USE at EOL: 65 ft-lb
Date at which USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Chemistry data per Paragraph C.1.2 of RG 1.99, Rev. 2

References:

June 25, 1992, letter from C. K. McCoy (GPCo) to USNRC Document Control Desk, subject: Vogtle Electric Generating Plant, Reactor Vessel Structural Integrity, Generic Letter 92-01, Revision 1

November 5, 1993 and January 24, 1994 letters to the USNRC, subject: Response to the staff request for additional information.

Plant Name: WNP-2

Docket Number: 50-397

NSSS Vendor: General Electric

Vessel Manufacturer: Chicago Bridge and Iron

Edition of ASME Code for Design: Summer 1971 Addenda to the 1971 ASME Code

Date of Commercial Operation: December 13, 1984

Date of License Expiration: December 20, 2023

ART for the Limiting Beltline Material:

Limiting Beltline Material: Ring #1 plate, heat C1272-1
ID Fluence at EOL: 1.1E18 n/cm^2
Initial RT_{NDT}[1]: 28°F
Method of Determining the Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 48°F
Margin: 34°F
ART at EOL: 110°F

USE for the Limiting Beltline Material:

Limiting Beltline Material: Ring #1 plate, heat C1272-1, Ring #2 plates, heats C1337-1 and C1337-2 and axial welds, heat 3P4966
1/4T Fluence at EOL: 7.3E17 n/cm^2 for Ring #2 plates and 6.8E17n/cm^2 for Ring #1 plate and axial welds
Initial USE: Heat-specific values not reported
Percent Drop at EOL: 13.8% for plates and 11% for axial welds
USE at EOL: Not applicable because heat-specific values not reported
Date at which Screening Limit will be Exceeded: Not applicable because heat-specific values not reported
Bases for Accepting the USE at EOL: Equivalent margins analysis was performed in Topical Report NEDO-32205, Rev. 1[2]

References:

Plate chemical composition data are from Table 5.3-4 of FSAR.

Plate initial USE data are from page B 3/4.4.6-1 of the WNP-2 Technical Specifications.

Fluence data are based on power update request reported in July 28, 1994, letter from J. V. Parrish to USNRC

---

[1]Staff will review the forthcoming GE Topical Report on initial RT_{NDT} determination

[2]Staff will review applicability of Topical Report NEDO-32205, Rev. 1

Plant Name: Waterford 3

Docket Number: 50-382

NSSS Vendor: Combustion Engineering

Vessel Manufacturer: Combustion Engineering

Edition of ASME Code for Design: Summer 1971 Addenda to the 1971 ASME Code

Date of Commercial Operation: September 24, 1985

Date of License Expiration: December 18, 2024

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Lower shell plate M-1004-2, heat 57286-1
ID Fluence at EOL: 3.68E19 n/cm^2
Initial RT_{NDT}: 22°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 27°F
Margin: 27°F
RT_{pts} at EOL: 76°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate shell plate M-1003-3, heat 56484-1
1/4T Fluence at EOL: 2.193E19 n/cm^2
Initial USE: 90 ft-lb
Percent Drop at EOL: 22.9%
USE at EOL: 69 ft-lb
Date at which USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Chemistry data per Paragraph C.1.2 of RG 1.99, Rev. 2

References:

Chemical composition and initial RT_{NDT} data are from July 6, 1992, letter from R. F. Burski (EO) to USNRC Document Control Desk, Subject: Waterford 3 SES, Generic Letter 92-01, Revision 1, Response

Plate initial USEs and fluence data are from BAW-2177

Plant Name: Watts Bar 1

Docket Number: 50-390

NSSS Vendor: Westinghouse

Vessel Manufacturer: Rotterdam Dockyard

Edition of ASME Code for Design: Summer 1971 Addenda to the 1971 ASME Code

Date of Commercial Operation: License has not been issued

Date of License Expiration: License has not been issued

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate shell forging 05, heat 527536
ID Fluence at EOL: 3.18E19 n/cm^2
Initial RT_{NDT}: 47°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 172°F
Margin: 34°F
RT_{pts} at EOL: 253°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate shell forging 05, heat 527536
1/4T Fluence at EOL: 1.913E19 n/cm^2
Initial USE: 62 ft-lb
Percent Drop at EOL: 29.9%
USE at EOL: 43 ft-lb
Date USE Screening Limit will be Exceeded: Before EOL
Bases for Accepting the USE at EOL: Equivalent margin analyses were performed in accordance with WOG Topical Report.[1]

REFERENCES:

July 7, 1992, letter from R. M. Shell (TVA) to USNRC Document Control Desk, Subject: Browns Ferry Nuclear Plant, Sequoyah Nuclear Plant, and Watts Bar Nuclear Plant--Response to Generic Letter 92-01 (Reactor Vessel Structural Integrity)

---

[1]Staff has reviewed the licensee's equivalent margins analysis and plans to publish its review in a future SER.

Plant Name: Watts Bar 2

Docket Number: 50-391

NSSS Vendor: Westinghouse

Vessel Manufacturer: Rotterdam Dockyard

Edition of ASME Code for Design: Summer 1971 Addenda to the 1971 ASME Code

Date of Commercial Operation: License has not been issued

Date of License Expiration: License has not been issued

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Circumferential weld, heat 895075
ID Fluence at EOL: 3.18E19 n/cm^2
Initial RT_{NDT}: -50°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 89°F
Margin: 56°F
RT_{pts} at EOL: 95°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Lower shell forging 04, heat 528658
1/4T Fluence at EOL: 1.90E19 n/cm^2
Initial USE: 105 ft-lb
Percent Drop at EOL: 22.1%
USE at EOL: 82 ft-lb
Date USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Chemistry data per Paragraph C.1.2 of RG 1.99, Rev. 2

REFERENCES:

July 7, 1992, letter from R. M. Shell (TVA) to USNRC Document Control Desk, Subject: Browns Ferry Nuclear Plant, Sequoyah Nuclear Plant, and Watts Bar Nuclear Plant--Response to Generic Letter 92-01 (Reactor Vessel Structural Integrity)

Plant Name: Wolf Creek

Docket Number: 50-482

NSSS Vendor: Westinghouse

Vessel Manufacturer: Combustion Engineering

Edition of ASME Code for Design: Winter 1972 Addenda to the 1971 ASME Code

Date of Commercial Operation: September 3, 1985

Date of License Expiration: March 11, 2025

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Lower shell plate R2508-1 (no heat number)
ID Fluence at EOL: 2.5E19 n/cm^2
Initial RT_{NDT}: 0°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 72°F
Margin: 34°F
RT_{pts} at EOL: 106°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Lower shell plate R2508-1 (no heat numbers)
1/4T Fluence at EOL: 1.36E19 n/cm^2
Initial USE: 87 ft-lb
Percent Drop at EOL: 20%
USE at EOL: 70 ft-lb
Date at which USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Chemistry data per Paragraph C.1.2 of RG 1.99, Rev. 2

References:

Initial USE data are from Table A-3 of WCAP-11553. The rest of the information on beltline materials is from WCAP-13365.

Plant Name: Zion 1

Docket Number: 50-295

NSSS Vendor: Westinghouse

Vessel Manufacturer: Babcock and Wilcox

Edition of ASME Code for Design: Summer 1966 Addenda to the 1965 ASME Code

Date of Commercial Operation: December 31, 1973

Date of License Expiration: April 6, 2013

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Middle circumferential weld WF-70, heat 72105
ID Fluence at EOL: 1.73E19 n/cm^2
Initial RT_{NDT}: -26°F
Method of Determining the Chemistry Factor: Surveillance data per Paragraph C.2.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 230°F
Margin: 28°F
RT_{pts} at EOL: 232°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Middle circumferential weld WF-70, heat 72105 and axial welds WF-4 and WF-8, heat 8T1762
1/4T Fluence at EOL: 9.34E18 n/cm^2 for WF-70 and 3.04E18 n/cm^2 for WF-4 and WF-8
Initial USE: Heat-specific value not reported
Percent Drop at EOL: 40% for WF-70 and 26% for WF-4 and WF-8
USE at EOL: Not calculated since heat-specific value was not reported
Date USE Screening Limit will be Exceeded: After EOL.
Bases for Accepting the USE at EOL: Equivalent margin analyses were performed in Topical Reports BAW-2178P and BAW-2192P.[1]

REFERENCES:

July 2, 1992, letter from M. A. Jackson (CECo) to T. E. Murley (USNRC)

Fluences are from reports BAW-2166 and BAW-2222

The initial RT_{NDT} for the middle circumferential weld, WF-70, was approved in a February 22, 1994, letter from C.Y. Shiraki (USNRC) to D.L. Farrar (CECo)

---

[1]Staff will review applicability of Topical Reports BAW-2178P and BAW-2192P.

Plant Name: Zion 2

Docket Number: 50-304

NSSS Vendor: Westinghouse

Vessel Manufacturer: Babcock and Wilcox

Edition of ASME Code for Design: Summer 1966 Addenda to the 1965 ASME Code

Date of Commercial Operation: September 17, 1974

Date of License Expiration: November 14, 2013

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate to lower shell circumferential weld, SA-1769, heat 71249
ID Fluence at EOL: 1.69E19 n/cm^2
Initial RT_{NDT}: -5°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 209°F
Margin: 68°F
RT_{pts} at EOL: 272°F
Date at which PTS Screening Limit will be exceeded: After EOL

USE for the Limiting Beltline Material:

Limiting Beltline Material: Intermediate Shell axial welds, WF-70, heat 72105.
1/4T Fluence at EOL: 3.26E18 n/cm^2
Initial USE: Heat-specific value not reported
Percent Drop at EOL: 32%
USE at EOL: Not calculated since heat-specific value not reported
Date USE screening limit will be exceeded: After EOL
Bases for Accepting the USE at EOL: Equivalent margin analyses were performed in Topical Reports BAW-2178P and BAW-2192P[1]

References:

July 2, 1992, letter from M. A. Jackson (CECo) to T. E. Murley (USNRC), Subject: Braidwood Station, Units 1 and 2; Byron Station, Units 1 and 2; Zion Station, Units 1 and 2

Fluences are from reports BAW-2166 and BAW-2222

[1]Staff will review applicability of Topical Reports BAW-2178P and BAW-2192P.

NRC FORM 335
(2-89)
NRCM 1102,
3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG-1511

2. TITLE AND SUBTITLE

Reactor Pressure Vessel Status Report

3. DATE REPORT PUBLISHED

MONTH	YEAR
December	1994

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

J. Strosnider, K. Wichman, B. Elliot, C. Fairbanks, E. Hackett,
S. Sheng, J. Tsao, L. Lois, M. Mayfield, M. Mitchell

6. TYPE OF REPORT

Technical

7. PERIOD COVERED (Inclusive Dates)

8. PERFORMING ORGANIZATION - NAME AND ADDRESS (If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)

Division of Engineering
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS (If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)

Same as above

10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

The Nuclear Regulatory Commission (NRC) issued Generic Letter 92-01 (GL 92-01) to obtain information needed to assess compliance with requirements and commitments regarding reactor vessel integrity in view of certain concerns raised in the NRC staff's review of reactor vessel integrity for the Yankee Nuclear Power Station.

This report gives a brief description of the reactor pressure vessel (RPV), followed by a discussion of the radiation embrittlement of RPV beltline materials and the two indicators for measuring embrittlement, the end-of-license (EOL) reference temperature and the EOL upper-shelf energy. It also summarizes the GL 92-01 effort and presents, for all 37 boiling water reactor plants and 74 pressurized water reactor plants in the United States, the current status of compliance with regulatory requirements related to ensuring RPV integrity. The staff has evaluated the material data needed to predict neutron embrittlement of the reactor vessel beltline materials. These data will be stored in a computer database entitled the reactor vessel integrity database (RVID). This database will be updated annually to reflect the changes made by the licensees in future submittals and will be used by the NRC staff to assess the issues related to vessel structural integrity.

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

Equivalent Margins Analysis, Fracture Mechanics, Generic Letter 92-01, Pressurized Thermal Shock, Radiation Embrittlement, Reactor Pressure Vessel, Upper-Shelf Energy

13. AVAILABILITY STATEMENT

Unlimited

14. SECURITY CLASSIFICATION

(This Page)

Unclassified

(This Report)

Unclassified

15. NUMBER OF PAGES

16. PRICE

Printed on recycled paper

Federal Recycling Program

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

OFFICIAL BUSINESS
PENALTY FOR PRIVATE USE, $300

SPECIAL FOURTH-CLASS RATE
POSTAGE AND FEES PAID
USNRC
PERMIT NO. G-67

120555111020 1 1AN19L
US NRC-RES
DIV SAFETY ISSUE RESOLUTION
BRANCH CHIEF
SEVERE ACCIDENT ISSUES BR
2WFN-10F13
WASHINGTON DC 20555