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Supplement 1

Reactor Pressure Vessel Status Report

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

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ABSTRACT

This report describes the issues raised as a result of the staff's review of Generic Letter (GL) 92-01, Revision 1, responses and plant-specific reactor pressure vessel (RPV) assessments and the actions taken or work in progress to address these issues. In addition, the report describes actions taken by the staff and the nuclear industry to develop a thermal annealing process for use at U.S. commercial nuclear power plants. This process is intended to be used as a means of mitigating the effects of neutron radiation on the fracture toughness of RPV materials.

The Nuclear Regulatory Commission (NRC) issued GL 92-01, Revision 1, Supplement 1, to obtain information needed to assess compliance with regulatory requirements and licensee commitments regarding RPV integrity. GL 92-01, Revision 1, Supplement 1, was issued as a result of generic issues that were raised in the NRC staff's reviews of licensee responses to GL 92-01, Revision 1, and plant-specific RPV evaluations. In particular, an integrated review of all data submitted in response to GL 92-01, Revision 1, indicated that licensees may not have considered all relevant data in their RPV assessments. This report is representative of submittals to and evaluations by the staff as of September 30, 1996. An update of this report will be issued at a later date.

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EXECUTIVE SUMMARY

This NUREG presents the actions taken by the U.S. Nuclear Regulatory Commission (NRC), as well as nuclear industry owners groups and individual licensees, regarding the ongoing assessment of reactor pressure vessel (RPV) integrity. Since the issuance of Generic Letter (GL) 92-01, Revision 1 (Ref. 1), in March 1992, and NUREG-1511 (Ref. 2), in December 1994, the staff has directed its actions toward determining the generic implications of the larger-than-expected variability observed in the chemical composition of RPV welds at the Palisades Nuclear Power Plant, and assuring that licensees take action to assure all relevant data are considered in their RPV assessments. The staff has also reviewed the Palisades thermal annealing plan and associated thermal annealing demonstration project activities. In addition, the staff has completed several plant-specific pressurized thermal shock (PTS) evaluations, and has developed and incorporated enhancements to the Reactor Vessel Integrity Database (RVID).

During the fall of 1994, the Consumers Power Company (CPCo, the licensee for the Palisades plant) performed material property tests and chemistry analyses of newly acquired samples of weld materials that were removed from the Palisades retired steam generators. When compared to the previous weld data, the copper and nickel measurements from the retired steam generator welds indicated that the variability in the weld chemistry was greater than previously anticipated during the development of the PTS rule, Section 50.61 to Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50 (Ref. 3).[1]

To address generic issues related to this larger-than-expected variability in weld chemical composition, the staff performed a generic PTS assessment and issued Supplement 1 to GL 92-01, Revision 1 (Ref. 4). The purpose of the staff's generic PTS assessment was to demonstrate that there is time to address the variability in weld chemistry. The generic PTS assessment used generic chemistry values and increased margin terms to account for the larger-than-expected variability in weld chemistry.

1 Henceforth, all sections to Title 10 of the *Code of Federal Regulations*, Part 50 will be abbreviated 10 CFR 50.XX or 10 CFR 50.XXX, as appropriate.

The results of applying generic values of chemistry and increased margin terms indicated that plants would be predicted to reach the PTS screening criteria at an earlier date than that given by the PTS assessment methodology in 10 CFR 50.61.

However, with the exception of six RPVs, the staff's generic assessment indicated that the RT_{PTS} values for the limiting beltline materials in all pressurized-water reactor (PWR) RPVs would still be below the PTS screening criteria at end-of-license (EOL) for the plants. The limiting RPV in this assessment was the RPV at the R.E. Ginna Nuclear Power Plant. The generic assessment did not consider plant-specific data, which could demonstrate that these six plants could have longer periods of time to reach the PTS screening criteria.

Subsequently, the Rochester Gas and Electric Company (RG&E, the licensee for Ginna) provided a plant-specific PTS assessment for the Ginna RPV. This assessment included RG&E's surveillance capsule data and all chemistry data representing the Ginna RPV beltline welds. This data indicated that the RT_{PTS} value for the limiting material in the Ginna RPV would be well below the PTS screening criteria at EOL. The staff reviewed RG&E's assessment and concurred with its conclusions. The plant-specific PTS assessment for the Ginna RPV demonstrates that the use of plant-specific data could extend the time for RPVs to reach the PTS screening criteria.

The staff compiled data from the responses to GL 92-01, Revision 1, in the RVID computerized database. Based on review of the data in the database and plant-specific reviews, the staff concluded that licensees may not have considered all the relevant data in their RPV assessments. Therefore, the staff, issued Supplement 1 to GL 92-01, Revision 1, in May 1995. The supplement required that all addressees identify, collect, and analyze the impact of any new data pertinent to the structural integrity of their RPVs relative to the requirements of 10 CFR 50.60 (Ref. 5), 10 CFR 50.61, and Appendices G and H to 10 CFR Part 50 (Refs. 6 and 7), as well as any potential impact on low temperature overpressure protection (LTOP) limits or pressure-temperature (P-T) limits.

All licensees have responded to GL 92-01, Revision 1, Supplement 1. Some licensees have provided additional data that were not provided in their initial response to the GL. However, in regard to GL 92-01, Revision 1, Supplement 1, no licensee has yet to identify any significant RPV integrity issue. Most licensees have indicated that they are participating in the owners group activities that will determine whether new information is available. The industry is coordinating the owners group activities through the Nuclear Energy Institute (NEI). The Boiling Water Reactor Vessel and Internals Project (BWRVIP) is coordinating activities for boiling-water reactors (BWRs). The Combustion Engineering Owners Group (CEOG) and Babcock and Wilcox Owners Group (B&WOG) have instituted programs to resolve the issue concerning weld chemistry variability.

The staff has also reviewed PTS assessments submitted by the licensees for Palisades and Calvert Cliffs Nuclear Power Plants (CCNPP). These licensees provided chemistry data that had not been included in previous assessments. The additional chemistry data for the CCNPP vessels indicated that the RT_{PTS} values for the limiting materials in the CCNPP RPVs would remain below the PTS screening criteria for up to 20 years after EOL. For the Palisades vessel, the additional chemistry data indicated that the embrittlement of the RPV could be greater than previously projected, but the RPV would still satisfy the requirements of the PTS rule until the end of the plant's fourteenth refueling outage, in late 1999.

Since the Palisades license expires in 2007, CPCo has submitted its preliminary thermal annealing plan for the Palisades RPV (See Section 4.8 for details and References). Thermal annealing is a process in which the RPV beltline is heated significantly above its operating temperature for an extended period. This process mitigates the effect of neutron radiation by recovering both the increase in transition temperature (TT) and the decrease in upper-shelf energy (USE). CPCo's annealing plan for the Palisades RPV addresses the critical engineering and metallurgical aspects of thermal annealing. The plan calls for the annealing to be performed using an indirect, gas-fired heating method that would heat the reactor vessel beltline region to the 850°F — 900°F temperature range for approximately 168 hours. The licensee projects that this annealing treatment should result in recovery of 80% to 90% of the fracture toughness lost as a result of radiation embrittlement.

CPCo has projected May 1998 for the anneal of the Palisades RPV. However, in a letter dated April 4, 1996 (Ref. 8), CPCo provided a revised PTS assessment indicating that the RT_{PTS} value for the limiting material in the Palisades RPV would not exceed the PTS screening criteria until after EOL. This revised assessment utilized the best-estimate chemistry that was previously reviewed by the staff, but utilized a lower projected neutron fluence at EOL. As a result of the reduced neutron fluence, the revised PTS assessment indicated that the Palisades RPV could satisfy the requirements of the PTS rule well after the plant's fourteenth refueling outage. The staff is currently reviewing CPCo's revised assessment.

To provide a regulatory framework for thermal annealing, the staff has issued a new regulation, 10 CFR 50.66 (Ref. 9), as well as Regulatory Guide (RG) 1.162 (Ref. 10). The Department of Energy awarded two contracts to demonstrate the engineering feasibility of the thermal annealing technology. The first demonstration project was performed at the Marble Hill facility and employed an indirect, gas-fired heating method. The second demonstration project has been tentatively scheduled to take place at the Midland facility and will employ an electric resistance heating approach. The staff has been closely following these projects in order to be prepared for the Palisades and other potential annealing applications. The staff is currently reviewing the Palisades thermal annealing plan and the two demonstration projects sponsored by the Department of Energy.

The staff has also developed and incorporated enhancements to the Reactor Vessel Integrity Database (RVID). The RVID was developed following the staff's review of licensee responses to Generic Letter (GL) 92-01, Revision 1. The database summarizes the properties of the reactor vessel beltline materials for each operating commercial nuclear power plant. The database has been issued to all U.S. licensees and some foreign regulatory authorities. The staff periodically enhances and updates the database based on feedback from the industry and revised data from the licensees. The RVID enables users to compare data from different licensees. In comparing the data, the staff observed that some licensees reported different data for welds that were fabricated from the same heats of weld wire. This led the staff to conclude that some licensees had not considered all relevant data when performing their RPV integrity assessments. The next updates to the RVID will incorporate any new information provided by licensees in response to the close-out letters to GL 92-01, Revision 1, and to GL 92-01, Revision 1, Supplement 1.

ABBREVIATIONS

10 CFR	Title 10 to the *Code of Federal Regulations*
ADP	Annealing Demonstration Project
AMES	Aging Materials and Evaluation Study
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
B&W	Babcock and Wilcox Nuclear Technologies
B&WOG	Babcock and Wilcox Owners Group
BG&E	Baltimore Gas and Electric Company
BNCS	Board of Nuclear Codes and Standards (ASME)
BWR	Boiling Water Reactor
BWRVIP	Boiling Water Reactor Vessel and Internals Project
CCNPP	Calvert Cliffs Nuclear Power Plant
CEOG	Combustion Engineering Owners Group
CPCo	Consumers Power Company
DOE	Department of Energy
EFPY	effective full power years
EOL	end of license
EPRI	Electric Power Research Institute
GL	Generic Letter
LTOP	low temperature overpressure protection
MOU	memorandum of understanding
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
OG	Owners Group
P-T	pressure-temperature
PTS	pressurized thermal shock
PWR	pressurized water reactor
RAI	request for additional information
RES	Office of Research (NRC)
RG	Regulatory Guide
RG&E	Rochester Gas and Electric Company
RPV	reactor pressure vessel
RVWG	Reactor Vessel Working Group
RVID	Reactor Vessel Integrity Database
SER	Safety Evaluation Report
SNSC	Southeast Nuclear Service Center
TAR	thermal annealing report
TT	transition temperature
USE	upper shelf energy

1 INTRODUCTION

The original version of the "Reactor Pressure Vessel Status Report," NUREG-1511 (December 1994), described the reactor pressure vessel (RPV), and discussed the effect of radiation embrittlement on RPV materials. NUREG-1511 also identified two indicators for measuring embrittlement: (1) an increase in the nil-ductility transition temperature; and (2) a decrease in upper-shelf energy (USE). Limits on radiation embrittlement to the RPV are defined in the pressurized thermal shock (PTS) rule, Section 50.61 to Title 10, *Code of Federal Regulations*, Part 50 (10 CFR 50.61), as well as Appendix G to 10 CFR Part 50. The PTS rule contains screening criteria that limit the increase in transition temperature (TT), and Appendix G contains screening criteria that limit the decrease in USE. NUREG-1511 also summarized the results of the staff's review of licensee responses to Generic Letter (GL) 92-01, Revision 1, as well as plant-specific RPV evaluations for all 37 boiling-water reactor (BWR) plants and 74 pressurized-water reactor (PWR) plants in the United States. The data resulting from the staff's review are stored in the NRC's computerized Reactor Vessel Integrity Database (RVID).

This updated "Reactor Pressure Vessel Status Report" discusses the staff's basis for issuing Supplement 1 to GL 92-01, Revision 1 (Section 2.1), the status of licensee responses to the supplement (Section 2.2), the current status of licensee compliance with the PTS rule (Section 3), thermal annealing (Section 4), and the staff's development of the RVID (Section 5).

The PTS rule adopted on July 23, 1985, and revised on May 15, 1991, and December 19, 1995, defines screening criteria for embrittlement of RPV materials and actions to be taken if these screening criteria are exceeded. These screening criteria are given in terms of reference temperature, or RT_{PTS}, at the end-of-license (EOL) for the plants. The RT_{PTS} is defined as follows:

$$RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS} + M$$

where:

- $RT_{NDT(U)}$ is the initial reference temperature of the unirradiated material
- ΔRT_{PTS} is the mean adjustment in reference temperature caused by irradiation
- M is the margin to be added to cover uncertainties in the initial reference temperature, copper and nickel contents, fluence, and calculational procedures.

The screening criteria are 270°F for plates, forgings, and axial welds and 300°F for circumferential welds. When these screening criteria are exceeded, the PTS rule requires that licensees perform further plant-specific evaluations of their reactor pressure vessels (RPVs) to justify continued operation of their reactors.

Based on the docketed information available at the time NUREG-1511 was issued, Beaver Valley Unit 1 and the Palisades Nuclear Power Plant were the only plants projected to exceed the PTS screening criteria prior to EOL. At that time, Beaver Valley Unit 1 (EOL 2016) and Palisades (EOL 2007) were projected to exceed the PTS screening criteria in 2012 and 2004, respectively. The Duquesne Light Company (the licensee for Beaver Valley Unit 1), and the Consumers Power Company (CPCo, the licensee for Palisades) indicated that the PTS results for their plants were based on the most current information and were subject to change. In a subsequent PTS assessment for the Palisades RPV, CPCo provided the staff with additional data indicating that Palisades would reach the PTS screening criteria as early as 1999. However, the licensee has recently revised the assessment based on a new neutron fluence projection. The staff is currently reviewing the neutron fluence projection and related PTS assessment. These data and related analyses are discussed in Section 3.3.1.

Appendix G to 10 CFR Part 50 specifies that the USE (as measured from the results of Charpy impact tests) must be greater than 102 joules (75 ft-lb) in the unirradiated condition. Furthermore , Appendix G specifies that the USE should remain above 68 joules (50 ft-lb) during the operating lifetime, unless analyses are performed to demonstrate that margins of safety exist for lower energies. Moreover, these safety margins must be equivalent to those specified in Appendix G to Section XI of the American Society of Mechanical Engineering (ASME) Boiler and Pressure Vessel Code (Ref. 11).

Through owners groups, the industry performed analyses to demonstrate the USE values necessary to satisfy the ASME Code for generic groupings of plants. In addition, some licensees performed plant-specific analyses. In NUREG/CR-6023, "Generic Analyses for Evaluation of Low Charpy USE Effects on Safety Margins Against Fracture of RPV Materials" (Ref. 12), the NRC staff concluded that PWR and BWR RPV materials can have USE values less than 68 joules (50 ft-lb) and still provide the required margins of safety against fracture. On the basis of the industry's equivalent margins analyses and the NRC's generic study, the staff concluded in NUREG-1511 that all RPVs will have adequate upper-shelf toughness throughout their current licensed operating life.

2 GENERIC LETTER (GL) 92-01, REVISION 1, SUPPLEMENT 1: REACTOR VESSEL STRUCTURAL INTEGRITY

2.1 Background

After evaluating licensee responses to Generic Letter (GL) 92-01, Revision 1, the staff entered the data from the responses into the Reactor Vessel Integrity Database (RVID). The staff then used the RVID to compare the data received from different licensees. As a result, the staff observed that some licensees reported different data for welds that were fabricated from the same heat of weld wire. In addition, the staff noted that the variability in the amount of copper in welds fabricated from copper-coated electrodes was greater than previously estimated. The staff therefore concluded that some licensees had not considered all relevant data when they performed their RPV integrity assessments and that welds fabricated from copper-coated electrodes had larger-than-expected variability in chemical composition. The staff's review of data from several plant-specific pressurized thermal shock (PTS) assessments confirmed these conclusions.

The variability in chemical composition highlighted the sensitivity of RPV embrittlement to small changes in the chemical composition of beltline materials. It also emphasized the need for licensees to use all relevant data in their RPV assessments and to adjust the surveillance data to the best-estimate chemistry, in accordance with the procedures in Regulatory Guide (RG) 1.99, Revision 2 (Ref. 13).

To obtain information needed to assess the significance of these issues, the NRC issued Supplement 1 to GL 92-01, Revision 1 (Ref. 4). This supplement, dated May 19, 1995, requested that all addressees identify, collect and analyze the impact of any new data pertinent to the structural integrity of their RPVs relative to the requirements of 10 CFR 50.60 (Ref. 5), 10 CFR 50.61 (Ref. 3), and Appendices G and H to 10 CFR Part 50 (Refs. 6 and 7), as well as any potential impact on low temperature overpressure protection (LTOP) limits or pressure-temperature (P-T) limits.

More specifically, in GL 92-01, Revision 1, Supplement 1, the staff requested that licensees take the following actions with respect to the RPV integrity assessments for their plants:

(1) describe the actions taken or planned to locate all data relevant to the evaluation of RPV integrity

(2) assess any change in best-estimate chemistry based on consideration of all relevant data

(3) determine the need for use of the ratio procedure in RG 1.99, Revision 2, when applying surveillance data to RPV integrity assessments

(4) assess the need for revision of the existing RPV integrity evaluations, including PTS, USE, P—T limit and LTOP limit evaluations.

The ratio procedure is defined in RG 1.99, Revision 2. When there is clear evidence that the copper and nickel content of the surveillance weld is different from that of the beltline weld, the surveillance weld data should be adjusted when determining the effect of neutron radiation on the beltline weld. The adjustment in the surveillance weld is dependent upon the amount of copper and nickel in the surveillance and beltline welds. According to the ratio procedure (as defined in RG 1.99, Revision 2), the measured increases in transition temperature (TT) from the surveillance data are to be adjusted by multiplying them by the ratio of the chemistry factor for the vessel weld to that of the surveillance weld. These chemistry factors are dependent upon the amount of copper and nickel, and are determined from a table in the RG.

2.2 Current Status

All licensees have responded to GL 92-01, Revision 1, Supplement 1, and some licensees have provided new data that were not considered in their responses to GL 92-01, Revision 1, in 1992. The staff is currently reviewing the data; however, the licensees have concluded that the data have no affect on previously submitted RPV integrity evaluations. Owners groups (OGs) have undertaken activities to identify, collect, and report any previously unreported data that may be relevant to the integrity of RPV materials. This data search requires the review of many years of welding records and is not scheduled to be completed until the summer of 1997. The schedules and associated activities of the OGs are described in the following sections.

On March 20, 1996, the staff met with representatives of Nuclear Energy Institute (NEI), the Electric Power Research Institute (EPRI), industry OGs, and licensees. A letter (Ref. 14) from B. Sheron (NRC) to A. Marion (NEI), dated April 18, 1996, summarizes the outcome of that meeting. The BWR Vessel and Internals Project (BWRVIP), Combustion Engineering Owners Group (CEOG) Reactor Vessel Working Group (RVWG), and EPRI informed the staff of their programs implemented in response to GL 92-01, Revision 1, Supplement 1. In turn, the staff informed the participants of its objectives related to GL 92-01, Revision 1, Supplement 1. As a result, the participants identified two topics where further discussions would prove to be beneficial, including:

(1) discussions regarding methods being used by industry to arrive at best-estimate chemistry values for families of welds; and

(2) discussions regarding the industry's development and maintenance of a database for information related to RPV materials.

2.2.1 Reactor Pressure Vessels Fabricated by Babcock and Wilcox

Licensees with pressurized-water RPVs fabricated by Babcock & Wilcox (B&W) are members of the B&W Owners Group's (B&WOG) Reactor Vessel Working Group (RVWG). In a letter from D.L. Howell to the NRC, dated August 1, 1995 (Ref. 15), and in the accompanying topical report (Ref. 16), the B&WOG RVWG provided its response to GL 92-01, Revision 1, Supplement 1, on behalf of the participating licensees in the owners group.

In that report (BAW-2257, Revision 1, dated October 1995), the B&WOG RVWG indicated that some additional data were available from domestic BWRs and foreign PWRs in regard to weld chemistries, and initial Charpy V-notch and Drop Weight impact toughness values. Nonetheless, the B&WOG RVWG contended that its members have appropriately considered the relevant data in regard to their reactor vessel integrity evaluations, and have previously reported the best-estimate weld metal chemistry values and valid RPV integrity evaluations. In addition, the B&WOG RVWG asserted that its members need not use the ratio procedure (as defined in RG 1.99, Revision 2), because the variability in chemistry for the surveillance welds was representative of the variability in chemistry for the beltline welds. However, the B&WOG RVWG provided a comparison of RT_{PTS} values when applying and not applying the ratio procedure. Regardless of the computational method, the RT_{PTS} values for all participating plants were shown to be below the PTS screening criteria at the EOL. The B&WOG RVWG listed 36 reports, previously submitted to the NRC, that form the basis for their conclusions. The staff is currently reviewing the information reported in BAW-2257, Revision 1.

2.2.2 Boiling-Water Reactor Pressure Vessels

On August 10, 1995, the BWRVIP submitted a report (Ref. 17) to address the RPV integrity of all BWR/2-6 plants. According to the report, the long term plan of the BWRVIP is to participate in database activities and cooperate with industry efforts to develop best-estimate chemistry values and methods to account for variability in weld chemistry. This report was reclassified to non-proprietary status on June 27, 1996 (Ref. 18).

The near-term response is discussed in EPRI Report No. TR-105908NP, "Bounding Assessment of BWR/2-6 Reactor Pressure Vessel Integrity Issues (BWRVIP-08NP)" (November 1995, Ref. 19). This report addresses the effects of chemistry variability on USE and pressure-temperature (P-T) limits.[2] The BWRVIP evaluations, which used bounding values of available weld chemistry data, were performed to determine the worst case impact of weld chemistry variability on USE and P-T limits. These evaluations indicate that all BWR licensees have satisfied the existing RPV regulatory requirements. The staff has reviewed and approved the P-T limits and USE evaluations for all plants discussed in the report.

2 EPRI Report No. TR-105908NP, November 1995, does not address the effects of chemistry variability on PTS and LTOP issues because BWR operating characteristics preclude PTS and LTOP events.

EPRI Report No. TR-105908NP indicated that weld chemistry variability had no impact on previously reviewed BWR USE evaluations. The report also indicated that, if bounding chemistry values are used (rather than best-estimate values as required in RPV integrity assessments), some P-T limits may not meet the safety margins of 10 CFR Part 50, Appendix G. However, for the limiting BWR operating condition (the leak rate test), the Appendix G safety factor of 1.5 could only potentially be reduced in the worst case to 1.3. This demonstrates that the safety factors required by Appendix G provide adequate margins even for a plant postulated to have an upper bound chemistry. The P-T limits may require revision if the best-estimate chemistries are revised as a result of new data being collected in response to GL 92-01, Revision 1, Supplement 1. The industry's effort is scheduled for completion in 1997.

2.2.3 Reactor Pressure Vessels Fabricated by Combustion Engineering

The CEOG RVWG has initiated a task related to better definition of weld chemistry. This task will compile heat specific information, including copper and nickel values, and will document the source(s) of the information. The compilation of the data requires the assembly and evaluation of fabrication records in the ABB/CE Southeast Nuclear Service Center (SNSC), located in Chattanooga, Tennessee. The records include over 450 boxes of original vessel fabrication records and drawings. The documents include material certifications, procedures, specifications, fabrication records, laboratory log book entries, and inspection and test records. The task group will also establish a model for determining the best-estimate copper and nickel values for each heat, based on the available data and weld process. The final report to the participating licensees is scheduled for December 31, 1996.

2.2.4 Generic Industry Activities

EPRI has developed an industry database entitled "RPVDATA". The database has the following objectives:

- Combine all available RPV materials data into an integrated, common material database.
- Develop special data search and retrieval capabilities for ease of use.
- Develop tools to assist utilities in performing vessel evaluations.
- Establish a convenient mechanism to incorporate new information into the database.

The initial version of RPVDATA was available through EPRI as of March 1996. RPVDATA included the RVID database that was assembled by the staff. However, it also included data that the staff has not reviewed. The staff plans to review all existing data, as well as any data resulting from OG activities related to GL 92-01, Revision 1, Supplement 1. The staff's goal is to include all RPV data in a RPV database that can be maintained and updated by the industry with oversight by the NRC (Refer to Section 5.3 for further discussion).

3 PRESSURIZED THERMAL SHOCK (PTS) EVALUATIONS

3.1 Revision of 10 CFR 50.61

On December 19, 1995, the NRC revised the PTS rule (10 CFR 50.61). The revisions to the rule incorporated the following changes:

- permission to use generic values of unirradiated reference temperatures different from those specified in the rule, if justification is provided;
- a requirement that the results from plant-specific surveillance programs be integrated into the RT_{PTS} estimate if plant specific surveillance data are deemed credible;
- incorporation into the rule of the credibility criteria specified in Regulatory Guide (RG) 1.99, Revision 2;
- a requirement that chemistry factors and margin values be calculated using the methodology and values specified in RG 1.99, Revision 2, if credible surveillance data are used to estimate the RT_{PTS}.

As a result of the revised rule and GL 92-01, Revision 1, Supplement 1, licensees may need to revise their best-estimate chemistries. In addition, licensees will need to review their surveillance data to determine whether the data satisfy the credibility criteria in the revised rule or RG 1.99, Revision 2, and whether application of the ratio procedure, as outlined in the revised rule or RG 1.99, Revision 2, is warranted.

3.2 Current Issues

3.2.1 Best-Estimate Chemistry

The PTS rule requires the use of the *best-estimate* chemical composition (percent copper and percent nickel) for evaluating embrittlement. The best-estimate for a weld is normally interpreted to be the mean of the measured values for weld deposits made from the same heat of weld wire as was used to fabricate the critical weld. However, this approach may not always yield the most accurate best-estimate chemistry compositions.

Several factors need to be considered for determination of the most accurate estimates of chemical composition, including:

- sources of variability (copper coating processes, separate nickel wire feeds, etc.);
- sample types (i.e. surveillance weld, nozzle dropout, etc.);
- quantity and pedigree of the data;
- weld wire sources;
- appropriate weighting techniques for the data.

These factors are the subjects of owners group (OG) research programs. The NRC staff and the OGs meet regularly to discuss progress on these issues.

A issue of particular regulatory concern has been the fact that a simple *average* of the data does not represent a *best-estimate* of the amount of copper in welds fabricated from copper-coated filler wire. The staff's reviews of RPVs with copper-coated filler wire indicate that there could be significant coil-to-coil variability in the amount of copper because of variability in the copper coating of the filler wire. The licensees for Calvert Cliffs and Palisades accounted for this variability by determining the best-estimate for copper content from a weighted average of the test results. In a weighted average, the average copper value from samples that represent more than one coil are weighted in accordance with the number of coils used to fabricate the weld.

3.2.2 Use of Surveillance Data

The revised PTS rule requires that licensees determine the RT_{PTS} values from surveillance data when the data meet the credibility criteria defined in RG 1.99, Revision 2, or in the revised rule. The use of plant-specific surveillance data may result in a RT_{PTS} value that is higher or lower than the RT_{PTS} value which would be determined by using the Tables in the revised rule or the RG. The revised rule also requires that the surveillance data must also be adjusted in accordance with the ratio procedure specified in the revised rule or RG 1.99, Revision 2, when there is clear evidence that the copper or nickel content of the surveillance weld is different from that of the beltline weld. The staff is continuing to evaluate the implications of variability in material properties and chemistry on the determination of credible surveillance data and integrated surveillance programs. This evaluation is being performed as part of the staff's overall RPV integrity program and will be addressed as warranted through revisions to RG 1.99, Revision 2, and Appendix H to 10 CFR Part 50.

3.3 Summary of Generic Assessment

Subsequent to the issuance of NUREG-1511, the licensee for Palisades submitted a revised PTS evaluation for staff review. As part of the revised evaluation, the licensee submitted chemistry data for welds in its retired steam generators. These welds were fabricated from the same procedure and weld wire heat lot as were used for fabrication of the limiting welds in the Palisades RPV. These data indicated that significant variability existed in the reported chemistries (i.e., copper and nickel contents) for welds fabricated from the same heat of weld wire.

The staff confirmed that the significant variability in weld chemistry was a generic issue by searching the RVID and comparing the chemistry data reported by different licensees. The results of the staff's efforts revealed that different licensees had reported different chemistry data for welds fabricated from the same heats of weld wire. This led the staff to conclude that the variability in the amount of copper in welds fabricated using copper-coated electrodes was greater than previously estimated.

Section 3.4.1 of this report discusses the staff's plant-specific assessment of the implications of the significant variability observed in the Palisades chemistry data. The staff recognized that this significant variability could impact other plant RPV evaluations. The staff is addressing this issue as part of its review of plant-specific PTS evaluations, and its ongoing reassessment of the PTS rule.

To ensure that all plants will maintain adequate protection against PTS events while the plant-specific assessments are in progress, the staff evaluated all PWR RPVs using generic chemistry values and increased margin terms in order to account for the potential variability in chemistries.[3] The results of applying generic values of chemistry and increased margin terms predict that plants would reach the PTS screening criteria at an earlier date than would be predicted by applying the PTS assessment methodology in 10 CFR 50.61 to the plant-specific data. The staff's generic assessment is documented in Commission Paper SECY-95-119 (Ref. 20)[4].

According to the staff's conservative generic assessment, no plant would be predicted to reach the PTS screening criteria in less than 7 effective full-power years (EFPY) from 1995, and most plants would reach the PTS screening criteria after the expiration of their current licenses.

It is important to emphasize that the staff's generic assessment was an extremely conservative analysis performed solely to demonstrate that there was sufficient time available to address the issues identified in GL 92-01, Revision 1, Supplement 1. The evaluation did not consider plant-specific information or surveillance data which the staff deems necessary to accurately assess the life of RPVs in the industry.

The staff's generic assessment was not intended to establish the operating life relative to the PTS screening criteria of 10 CFR 50.61, and the results of the assessment should not be interpreted in that way. This caution has been substantiated by a subsequent plant-specific PTS assessment for the Ginna RPV, which was predicted by the generic assessment to be the first plant to reach the screening criteria. The PTS assessment, which was based on plant-specific data, demonstrated that the Ginna RPV would not reach the PTS screening criteria prior to EOL. The plant-specific PTS assessment for the Ginna RPV is documented in a letter from the A. Johnson (NRC) to Dr. R. Mecredy, dated March 22, 1996 (Ref. 21).

3 These margin terms were increased using generic data for various classes of weldments.

4 The actual generic assessment is contained in a Memorandum from Jack R. Strosnider, Branch Chief, Materials and Chemical Engineering Branch, Division of Engineering, NRR, to Ashok C. Thadani, Associate Director for Technology, NRR, dated May 5, 1995. This memorandum is included as part of Commission Paper SECY-95-119.

This evaluation is summarized in Section 3.4.3. The industry will continue to perform plant-specific PTS assessments as required by 10 CFR 50.61. The NRC staff will review plant-specific assessments, and will perform a systematic reassessment of all the industry's RPV evaluations as part of its review of the industry's responses to GL 92-01, Revision 1, Supplement 1.

3.4 Plant-Specific PTS Assessments

3.4.1 Summary of the Palisades PTS Review

The staff issued an interim PTS safety evaluation for the Palisades plant in a letter dated July 12, 1994 (Ref. 22). In that evaluation, the staff concluded that the RT_{PTS} value for the limiting weld in the Palisades RPV would reach the PTS screening criterion in the year 2004, — prior to the expiration of the Palisades operating license in 2007. The staff based this conclusion on evaluation of the data available at that time. The staff indicated that this conclusion could change on the basis of test results from the retired steam generators, which contained weld metal fabricated from the same heats of weld wire (heats W5214 and 34B009) as were the limiting welds in the Palisades RPV beltline.

During the fall of 1994, the Consumer Powers Company (CPCo, licensee for the Palisades Nuclear Power Plant) performed material properties tests and chemistry analyses of newly acquired samples of weld material that had been removed from its retired steam generators. The copper and nickel measurements from the retired steam generators were added to the previous weld data to determine the best-estimate values of copper and nickel for the Palisades limiting welds.

To provide a common basis for comparing the copper measurements from different samples, and to determine a best-estimate weight percent copper, the licensee determined whether the measurements from a sample represented weld metal from one or more coils of weld wires. The number of coils of weld wire was determined by examining the weld record for the sample and the locations of the measurements from the sample. The licensee determined the best-estimate value for copper from a coil-weighted average of the samples.

The staff concluded that the coil-weighted average method is the preferable method of determining the best-estimate percent copper for welds fabricated from weld wire containing highly variable copper coatings. These tests and analyses indicated that the degree of embrittlement of the Palisades RPV could be higher than calculated in the July 1994 interim safety evaluation. With the new data included in the evaluation, analyses performed in accordance with the PTS rule indicate that the Palisades RPV will satisfy the requirements of the PTS rule until the end of the plant's fourteenth refueling outage, scheduled for late 1999.

As part of its review the staff noted that significant variability existed in the reported chemistry data (i.e., copper and nickel) for the limiting RPV weld. To assess this concern, the NRC staff employed the Palisades plant-specific chemistry and fluence data, and performed RPV failure frequency calculations similar to those in SECY-82-465 (Ref. 23), which established the basis for the PTS screening criteria. These analyses demonstrated that the safety margins intended by the PTS rule will be satisfied through the Palisades fourteenth refueling outage, even when considering the variability observed in the Palisades chemistry data.

As a result of its evaluations, the NRC determined that the Palisades RPV can be operated in compliance with the requirements of 10 CFR 50.61 through the plant's fourteenth refueling outage. The staff's safety evaluation is contained in a letter from E.G. Adensam to K.M. Haas, dated April 12, 1995 (Ref. 24). In October 1995 (Ref. 25), CPCo proposed a thermal anneal of the Palisades RPV for the plant's thirteenth refueling outage. Thermal annealing mitigates the effects of neutron radiation on the RPV materials, and would allow CPCo to operate the Palisades plant beyond the fourteenth refueling outage.

However, in a letter dated April 4, 1996 (Ref. 26), the licensee provided a revised PTS assessment indicating that the RT_{PTS} value for the limiting weld in the Palisades RPV would not exceed the PTS screening criteria until after the expiration of Palisades operating license. This revised assessment utilizes the best-estimate chemistry that was discussed in the safety assessment of April 12, 1995, and utilizes a lower projected neutron fluence at EOL. The staff is currently reviewing this revised assessment. If the revised neutron fluence calculation is found acceptable, the Palisades RPV could satisfy the requirements of the PTS rule well after the plant's fourteenth refueling outage. As a result, CPCo could defer the date for annealing the Palisades RPV beyond 1999.

3.4.2 Summary of the Calvert Cliffs PTS Review

In a letter dated January 2, 1996 (Ref. 27), the staff provided a PTS evaluation of the Calvert Cliffs Nuclear Power Plant (CCNPP), Units 1 and 2. Subsequently, the Baltimore Gas and Electric Company (BG&E, the licensee for CCNPP) updated the RT_{PTS} values for many of the CCNPP beltline materials as a result of information received from the following sources:

- Combustion Engineering fabrication records
- chemical analyses from samples of Shoreham RPV weldments and an archived surveillance block from the Pilgrim Nuclear Power Station,
- surveillance capsule data from McGuire Unit 1 and CCNPP Unit 2
- the most recent flux reduction measurements for CCNPP Units 1 and 2.

The chemical analyses from the Shoreham weldments and the Pilgrim surveillance block were used to determine the best-estimate chemistries for the CCNPP materials that were fabricated from the same heats of weld wire. To account for the variability in the amount of copper, the licensee used a weighted average based on the number of coils used to fabricate the test welds. A simple average was used to determine the best-estimate for nickel since the variability of this element was low.

BG&E also used surveillance weld data from the surveillance capsules in the McGuire, Unit 1 RPV. The McGuire surveillance weld was fabricated from the same heats of weld wire as were used to fabricate a weld in the CCNPP, Unit No. 1 beltline. The licensee compared the neutron and thermal environments in the CCNPP, Unit 1 vessel to those in the McGuire, Unit 1 vessel to demonstrate the equivalency of the environments, and to demonstrate that the McGuire surveillance data were applicable to CCNPP, Unit 1. In addition, to account for the difference in chemistry between the surveillance weld and the best-estimate chemistry of the beltline weld, BG&E adjusted the surveillance data by applying the ratio procedure of RG 1.99, Revision 2.

In its safety evaluation of January 2, 1996, the staff concluded that the RT_{PTS} values for the beltline materials in the CCNPP RPVs would remain below the PTS screening criteria 20 years after the expiration of the operating licenses for the plants. However, since this conclusion is dependent upon the available chemistry and surveillance data, it could be subject to change as new data become available.

3.4.3 Summary of the Ginna Review

In a letter dated March 22, 1996 (Ref. 21), the staff provided its safety evaluation of the PTS assessment for the Ginna RPV. The PTS assessment by the Rochester Gas and Electric Company (RG&E, the licensee for Ginna) is based on plant-specific RPV data. These data included the chemical composition data from two weld dropouts, its surveillance weld and a weld qualification test sample, which were all fabricated from the same heat of weld wire as were used to fabricate the limiting beltline weld in the Ginna RPV. In addition, the assessment included irradiated Charpy-V test data from the Ginna surveillance capsule welds.

Since the Ginna surveillance capsule weld was fabricated from the same heat of weld wire as was used for fabrication of the limiting beltline weld, and since the surveillance data met the credibility criteria in RG 1.99, Revision 2, RG&E utilized the surveillance data to determine the RT_{PTS} value of the limiting weld in the Ginna RPV. However, since the best-estimate chemical composition of the limiting beltline weld was different from the best-estimate chemical composition of the Ginna surveillance capsule weld, RG&E also applied the ratio procedure (as recommended in RG 1.99, Revision 2) in the Ginna PTS evaluation. On the basis of its evaluation of Ginna's plant-specific RPV data, RG&E concluded that the RT_{PTS} value of the limiting beltline weld in the Ginna RPV would be below the PTS screening criteria at EOL.

The staff evaluated the chemical composition data in a different manner than RG&E. The licensee used a simple average of the measured copper content values to determine the best-estimate for copper. As discussed in Section 3.2.1, the staff is concerned that a simple average of the data may not always represent a best-estimate for copper. The staff's review of other RPVs fabricated from copper-coated filler wire similar to that used in the Ginna RPV indicates that there could be significant coil-to-coil variability in the amount of copper because of variability in the copper coatings of the filler wires. The licensees for Calvert Cliffs and Palisades accounted for this variability by determining the best-estimate for copper from a weighted average of the test results. In a weighted average, the average copper value from samples that represent more than one coil are weighted according to the number of coils used to fabricate the weld. The staff discussed this issue with RG&E. RG&E indicated that it had insufficient information to accurately determine the number of coils used to fabricate the four welds that represent the Ginna data base.

Therefore, for the assessment of the Ginna RPV, the staff determined a best-estimate copper on the basis of the data source, and the number and location of the measurements. Based on the number and location of the measurements, the two weld dropouts contain data from many more weld coils than the data from the surveillance weld and weld qualification sample. Hence, to account for the coil-to-coil variability in the amount of copper, the staff gave a greater weight to the average copper values from the weld dropouts than to the average copper values from the surveillance weld and the weld qualification sample. Since the thickest weld dropout had the greater number of measurements and should have been fabricated with more coils, the staff concluded that the thicker weld dropout should provide a best-estimate copper value for the limiting Ginna beltline weld by conservatively accounting for the coil-to-coil variability in copper.

The weighted-average of the copper measurements from the thickest weld dropout resulted in a best-estimate for copper that was slightly greater than that calculated from a simple average of the measurements. As a result, the staff calculated a RT_{PTS} value at expiration of the Ginna operating license that was greater than that calculated by the licensee (268°F by the staff vs. 265°F by the licensee). However, the staff noted that both calculated RT_{PTS} values are significantly less than the 300°F screening criterion (as stated in 10 CFR 50.61) used for evaluation of the limiting beltline weld in the Ginna RPV. The staff therefore concluded that the Ginna RPV would satisfy the requirements of 10 CFR 50.61 until the EOL of the plant. This conclusion is predicated upon available chemistry and surveillance data, and could be subject to change as new data become available.

4 REACTOR PRESSURE VESSEL THERMAL ANNEALING

4.1 General Background

Reactor pressure vessels (RPVs) are fabricated from thick steel plates and/or forgings that are subject to embrittlement from neutron irradiation in the RPV beltline region. The embrittlement is manifested as a decrease in the fracture toughness of these materials. This decrease in fracture toughness is primarily a function of the following factors:

- total amount of neutron irradiation (fluence)
- chemical composition of the steels
- temperature of the irradiation

In order to limit the amount of neutron irradiation damage to the RPV beltline materials, many utilities have redesigned their fuel loading patterns to reduce the amount of neutron leakage from the core, or have used neutron poisons or shielding to protect the RPV in regions of high neutron flux. However, these techniques have only a limited effect if incorporated late in the life of the RPV.

As discussed in previous sections, the level of embrittlement is particularly sensitive to the chemical composition (specifically, the amounts of copper and nickel) of these steels. The NRC regulations (10 CFR 50.61, and Appendix G to 10 CFR Part 50) and Regulatory Guide (RG) 1.99, Revision 2, provide methodologies to conservatively estimate the increase in the transition temperature (TT) and decrease in the upper shelf energy (USE) of the beltline materials as a result of neutron irradiation.

An increased TT makes the RPV beltline materials more susceptible to rapid crack growth during startup or shutdown and under accident conditions such as pressurized thermal shock (PTS). The PTS rule (10 CFR 50.61) contains screening criteria to conservatively limit the amount of the shift in the TT.

The decrease in USE resulting from neutron irradiation can create the potential for ductile crack growth under normal operating and accident conditions. Appendix G to 10 CFR Part 50 contains screening criteria that conservatively limit the allowable decrease in USE.

A licensee can use a staff approved analysis to justify operation beyond the embrittlement screening criteria of 10 CFR 50.61 or Appendix G to 10 CFR Part 50, or else choose to thermally anneal the RPV. Publication of an NRC rule in 10 CFR 50.66 (Ref. 9) and RG 1.162 (Ref. 10) on thermal annealing was completed in February 1996, along with overall revisions to the RPV integrity regulations. However, the previous version of Appendix G to 10 CFR Part 50 recognized that "reactor vessels for which the predicted value of USE at end of life is below 50 ft-lbs or for which the predicted value of adjusted reference temperature at end of life exceeds 200°F must be designed to permit a thermal annealing treatment at a sufficiently high temperature to recover material toughness properties of ferritic materials of the reactor vessel beltline."

Annealing is also an option for extending the service lives of RPVs beyond the current end-of-license (EOL) or for establishing less restrictive plant operational pressure-temperature (P-T) limits for startup and shutdown.

4.2 Thermal Annealing Process and Technical Background

Thermal annealing is a process whereby the RPV beltline is heated to a temperature significantly above the operating temperature and held for an extended period. Thermal annealing can be performed either "dry" or "wet." Dry annealing is performed with the vessel drained and the fuel and internals removed. Wet annealing is typically performed with the full complement of primary coolant using the reactor coolant system pumps to provide the heat. The recovery in a wet anneal is usually limited, since it is practically difficult to achieve a large differential between the operating and annealing temperatures. The present discussion focuses on dry annealing.

The success of an annealing heat treatment in mitigating the effects of irradiation embrittlement is typically measured by the percent recovery in both TT and USE. The recovery of as-fabricated properties depends on the RPV steel chemistry, as well as the annealing time and temperature. For a specified steel composition, the lower the annealing temperature, the more time is required to achieve a given level of recovery. However, the Carbon-Manganese steels and weldments used in western RPVs are also potentially susceptible to metallurgical degradation (e.g., creep, temper embrittlement) when subjected to higher temperatures and longer heat treatment durations. Thus a "window" of potential temperatures and times exists for thermal annealing. This window is typically further constrained by plant-specific and heating-method-specific operational and economic considerations.

For a given annealing time and temperature, the amount of recovery primarily depends on the level of the irradiation embrittlement and the chemical composition of the steel. Server (Ref. 28) has shown that an annealing temperature 100°F above the RPV irradiation temperature is not high enough to obtain substantial mechanical property recovery. Therefore, to achieve a measurable amount of recovery in a relatively short time, a practical minimum for the annealing temperature would be on the order of at least 150°F above the RPV irradiation temperature.

For typical western RPVs with a nominal irradiation temperature of 550°F, this would imply a minimum annealing temperature of 700°F. A maximum temperature has not been defined; however, 940°F was agreed upon as the upper limit for ASME Code Case N-557 (Ref. 29; refer to Section 4.4). The 940°F limit was set to limit the potential for creep and other forms of metallurgical degradation that can result at elevated temperatures.

Durations of 168 hours have been typical for experimental annealing treatments that have been conducted in the 700°F — 900°F range. For western RPV steels and weldments, these treatments can restore the TTs and USEs to more than 90 percent of their initial values. Due to the relative scarcity of data from annealing treatments in the 700°F — 800°F range, proposed annealing treatments in the U.S. have been focused to occur in the 800°F — 900°F range.

Server (Ref. 28) also summarized the overall state of knowledge (as of 1985) for in-place thermal annealing of commercial RPVs. Server reviewed data on annealing recovery and reirradiation effects for high-copper welds, and concluded that thermal annealing at 850°F can cause significant recovery in both the TT shift and reduction in USE. Server also reviewed engineering studies on thermal annealing. Server concluded that annealing of U.S. reactors at 850°F is feasible using existing commercial heat treating methods, but also that plant-specific engineering problems would need to be resolved. Server also presented a thermal and structural analysis for a typical pressurized water RPV annealed at 850°F. This analysis predicted that vessel dimensional stability would be maintained and that post-anneal residual stresses would not be significant. However, Server's results also indicated that differential thermal expansion of the RPV during annealing can potentially lead to excessive bending of attached piping. Server has stated that careful temperature control is required during the annealing treatment in order to prevent this problem.

Mager and others (Refs. 30 and 31) reported on research to determine the extent of fracture toughness recovery as a function of annealing time and temperature for materials that are sensitive to neutron embrittlement. They concluded that excellent recovery of all properties could be achieved by annealing at 850°F for 168 hours, and that the reembrittlement after annealing would follow the same trend as the pre-annealing embrittlement rate. These reports also describe a thermal annealing procedure developed for field application.

The Department of Energy, Sandia National Labs, and EPRI conducted a "Reactor Pressure Vessel Thermal Annealing Workshop" in February 1994 (Ref. 32). The purpose of the workshop was to provide a forum for U.S. utilities and interested parties to discuss relevant experience and issues, and to identify potential solutions and approaches related to thermal annealing of RPVs. An extensive amount of information was presented ranging from mechanistic studies to an economic analysis of potential annealing benefits.

In 1995, Eason et. al. performed analyses of existing data on annealing of irradiated pressure vessel steels using both mechanistic and statistical considerations. These analyses led to the development of improved correlation models for estimating Charpy USE and TT after radiation and annealing. This work is reported in NUREG/CR-6327 (Ref. 33) and provides the basis for the equations (in RG 1.162) used to estimate the degree of recovery in fracture toughness properties following annealing.

Also in 1995, Pelli (Ref. 34) performed a "State of the Art Review on Thermal Annealing." This review led to the conclusion that, although annealing technology has been used successfully on Russian VVER-440 RPVs (See Section 4.3), application to Western-style reactors is more difficult because of the need to heat the entire beltline of plate-fabricated vessels. Furthermore, for the Russian materials, phosphorus was the limiting steel constituent for embrittlement as opposed to copper in U.S. steels.

4.3 Previous Experience

Although thermal annealing has not yet been applied to a U.S. commercial power reactor, it has been successfully applied to other reactors. Two Western-style RPVs that have been successfully annealed are the Army's SM-1A in 1967 (Ref. 35), and the BR-3 in Mol, Belgium, in 1984 (Ref. 36). Both of these reactors operated at temperatures low enough to permit "wet annealing" at a temperature of 650°F using the reactor coolant pumps as the heat source.

In addition, 14 Russian-designed VVER-440 PWRs in Russia, Finland, and Eastern Europe, which operate at temperature conditions similar to those at U.S. PWRs, have been annealed at temperatures of approximately 850°F, using dry air and radiant heaters as the heat source (Ref. 34). Details of the thermal annealing of the Novovoronezh Unit 3 RPV were reported by a U.S. delegation that witnessed the operation (Ref. 37). An NRC team also witnessed the annealing of the Lovissa Unit 1 RPV in Finland in August 1996.

4.4 Technical Codes and Standards for Thermal Annealing

4.4.1 ASTM Standard E 509

General guidance for in-service annealing is provided in American Society for Testing and Materials (ASTM) Standard E 509-86 (Ref. 38). Specifically, ASTM Standard E 509-86 prescribes general procedures for conducting an in-service thermal anneal of an RPV and for demonstrating the effectiveness and degree of recovery. ASTM Standard E 509-86 also provides direction for a post-annealing vessel radiation surveillance program. ASTM Standard E 509 is currently (1996) undergoing a major revision to provide updated guidance, particularly in the areas of technical references and verification of recovery and re-irradiation embrittlement.

4.4.2 ASME Code Case N-557 on Thermal Annealing

At the ASME Section XI meetings in Chicago in August 1995, the Task Group on Thermal Annealing undertook development of a Code Case on Thermal Annealing of Reactor Vessels on a high priority basis. The development of the Code Case was requested by the Consumers Power Company (CPCo, the licensee for the Palisades plant) and supported by the NRC. The Task Group appointed a special team to write the Code Case and technical basis document on an expedited basis. The Code Case (designated N-557) was passed by the ASME main committee on December 1, 1995 (Ref. 29).

Code Case N-557 addresses annealing conditions (temperature and duration), temperature monitoring, evaluation of loadings, and non-destructive examination techniques. Code Case N-557 received final approval by the ASME Board of Nuclear Codes and Standards (BNCS) on March 19, 1996. The supporting technical basis document for Code Case N-557 will be published in an appropriate technical journal in 1996.

4.5 NRC Annealing Rule and Regulatory Guide

The thermal annealing rule (10 CFR 50.66) was approved by the Commission and published in the *Federal Register* on December 19, 1995. The rule addresses the critical engineering and metallurgical aspects of thermal annealing. The regulatory process outlined in the proposed rule consists of several elements:

- a thermal annealing report (TAR, describing the licensee's plan for conducting the anneal) to be submitted to the NRC prior to annealing

- requirements for determining the percent recovery of RPV fracture toughness due to annealing and requirements for determining reembrittlement trends occurring during reactor operations after annealing

- confirmation that thermal annealing was performed in accordance with the TAR submitted in advance by the licensee

- public meetings to be held both before and after the anneal to allow the NRC to respond to questions from interested parties or individuals

The regulatory guide on thermal annealing (RG 1.162) was processed in parallel with the rule package and was published on February 15, 1996. NUREG/CR-6327 (Ref. 33), which provides the supporting technical basis for irradiation embrittlement recovery from thermal annealing, was issued in March 1995. The work in this report provides the basis for the computational embrittlement recovery models in RG 1.162.

4.6 Overview of Metallurgical and Engineering Issues

RG 1.162 contains a detailed listing of metallurgical and engineering issues that need to be addressed for thermal annealing of an RPV. (Background related to these issues was presented in Section 4.2.) Details regarding fracture toughness recovery and reembrittlement trends are covered in Section 3.0 of RG 1.162. Specifically, RG 1.162 presents three acceptable methods for estimating recovery:

(1) use of the vessel surveillance materials

(2) removal of specimens from the RPV beltline

(3) a generic computational method

The RG also provides methods for predicting post-annealing reembrittlement trends. The potential for elevated temperature degradation (e.g., creep, temper embrittlement) of western RPV steels was addressed in a technical basis document prepared for ASME Code Case N-557. Elevated temperature degradation of material properties for Western-style reactors was not considered to be an overriding concern for thermal annealing treatments in the range of 700°F — 900°F. The potential for creep, in particular, can be minimized by following the guidelines of ASME Code Case N-557.

Engineering issues that need to be addressed for thermal annealing include, but are not limited to the following:

- development of appropriate thermal and structural models for predicting limiting stress conditions and providing guidance for the placement and quantity of instrumentation

- control of thermal gradients during heatup and cooldown to minimize stresses and deformations in the vessel and attached piping

- adequate instrumentation (for temperature, strains and displacements) for monitoring the response of the RPV and piping to the anneal

- adequate onsite fire protection and proper adherence to National fire codes and standards (particularly with regard to gas-fired heating methods)

- protection of personnel from radiation hazards, including those associated with air-lifting internals within the containment and placement of instrumentation inside the bio-shield cavity

- protection of other equipment, components, and structures affected by the annealing (e.g., minimizing bio-shield wall temperatures)

Valuable insight regarding these and other engineering issues will be obtained from the Annealing Demonstration Project(s) (see Section 4.7).

4.7 Department of Energy Annealing Demonstration Project

The Department of Energy (DOE) is currently supporting thermal annealing for U.S. light water power reactors in three phases:

Phase 1:	Evaluate Engineering Feasibility and Material Property Recovery
Phase 2:	Assist in Establishing Regulatory Requirements Using Experimental and Analytical Data
Phase 3:	(a) Assist in Anneal of an Operating Plant and (b) Evaluate Post-Annealing Operability

The evaluation of engineering feasibility is referred to as the annealing demonstration project (ADP). Two contract awards for the ADP were announced on May 25, 1995. These contracts are with two separate consortia for demonstration of the feasibility of thermal annealing technology at two cancelled plant sites, Marble Hill, Indiana (CE RPV) and Midland, Michigan (B&W RPV). The first annealing demonstration was performed at the Marble Hill site and employed an indirect, gas-fired heating method. The Marble Hill annealing demonstration was completed in July 1996. The second demonstration at the Midland site will employ an electric resistance heating approach and is tentatively scheduled for December 1996. The NRC staff has been represented at meetings of both the Marble Hill and Midland Steering Committees and Design Reviews.

The NRC Office of Research (RES) has the lead in representing the NRC's interests in the ADPs. RES has prepared a memorandum of understanding (MOU) regarding NRC participation in the ADPs. This MOU was signed by NRC and DOE on August 4, 1995 (Ref. 39). The NRC's Office of Nuclear Reactor Regulation (NRR) staff has worked closely with RES as observers of the annealing demonstration projects in order to be prepared for the Palisades and other potential annealing applications.

4.8 Palisades Thermal Annealing Report

In the fall of 1994, the Consumers Power Company (CPCo), the licensee for the Palisades plant, developed chemical composition and mechanical property data for welds removed from their retired steam generators (Refs. 40 and 41). This new information changed the best estimate chemistry of the limiting RPV beltline weld. This information also indicated an increased variability in chemical composition of the weld when compared to that assumed for the development of the PTS rule. In combination, this information indicated that the plant would exceed the PTS screening criteria prior to EOL (2007). The staff issued a safety evaluation report (SER) regarding the variability of the Palisades RPV weld properties on April 12, 1995 (Ref. 23). The staff agreed with the licensee's best-estimate analysis of the chemical composition of the RPV welds and concluded that continued operation through late 1999 was acceptable. 10 CFR 50.61 requires submittal of a plant-specific analysis justifying operation beyond the screening criteria at least 3 years before exceeding the criteria. In the SER, the staff recognized that submission of information regarding thermal annealing of the Palisades RPV would obviate the need for the plant-specific analysis.

In October 1995, CPCo initiated submittal of a report describing the planned thermal annealing of the Palisades RPV (Ref. 25). CPCo's plan calls for the annealing to be performed using an indirect, gas-fired heating method, which would heat the reactor vessel beltline region to the 850°F — 900°F temperature range for approximately 168 hours. The licensee projects that this annealing treatment should result in recovery of 80 percent to 90 percent of the fracture toughness lost as a result of radiation embrittlement.

During the summer outage (May—August, 1995) at Palisades, the licensee obtained baseline information on the condition of the vessel insulation and the temperatures of the RPV supports and the cavity between the vessel and the bio-shield wall. The final sections of the preliminary TAR (Appendices A and B to Section 1.7 of the preliminary TAR) were submitted to the NRC on April 29, 1996 (Ref. 42). These appendices completed CPCo's submittal of the preliminary TAR for the Palisades RPV. The report is currently being reviewed by the staff. CPCo will be relying heavily on the results of the Marble Hill demonstration anneal (described previously) for completion and verification of the Palisades submittal. The submittal process is expected to be completed by end of 1996, when the results from the Marble Hill demonstration anneal are expected to be published. The licensee is currently projecting that the anneal of the Palisades RPV will commence in May 1998.

In addition, on April 4, 1995, CPCo submitted to the NRC a revised neutron fluence analysis for the Palisades RPV (Ref. 26). The analysis projects a significantly reduced neutron fluence at EOL for the RPV. If approved by the NRC, this analysis could enable operation of the plant well beyond 1999 without annealing.

4.9 Summary

The future is difficult to predict regarding thermal annealing of U.S. commercial nuclear power reactors. The commitment to anneal an RPV involves significant engineering and regulatory analyses and the assignment of substantial resources. However, the approach can reverse neutron irradiation embrittlement, thereby decreasing constraints on plant operation. This approach can enable operation to EOL for plants potentially challenged by the PTS screening criteria, and extend operation beyond EOL for others. In Sandia National Laboratories Report SAND94-1515/1 (Ref. 32), Griesbach examined the value of annealing, and concluded that each plant-specific case must be evaluated and compared to other alternatives. Important in this regard is the fact that uncertainties in RPV material properties can result in significant differences in the value of annealing for a plant.

Successful demonstration of the engineering feasibility of annealing technology in the DOE programs (see Section 4.7) will greatly facilitate future considerations for thermal annealing in the United States.

5 REACTOR VESSEL INTEGRITY DATABASE

5.1 Summary of Database Features

The Reactor Vessel Integrity Database (RVID) was developed following the NRC staff review of licensee responses to GL 92-01, Revision 1. The database summarizes the properties of the reactor vessel beltline materials for each operating commercial nuclear power plant.

In addition to the licensee responses to GL 92-01, Revision 1, various documents were included in the review process and development of the RVID. These documents include surveillance capsule reports; documents referenced in licensee responses to GL 92-01, Revision 1, submittals; PTS submittals, P—T limits reports; and responses to staff's requests for additional information (RAIs). The data from these source documents were reviewed and documented in the RVID tables.

The RVID was designed and developed to reflect the current status of RPV integrity, and the data and information is consolidated in a convenient and accessible manner. Some of the data categories are inputs of docketed information; others are computed values, which are not necessarily docketed. The programming logic used for calculations in the RVID follows the methodology in Regulatory Guide 1.99, Revision 2.

The RVID includes four tables (summary files) for each plant:

(1) background information table,

(2) chemistry data table

(3) upper-shelf energy table,

(4) P—T limits or PTS table

References and notes following each table document the source(s) of data and provide supplemental information. Additionally, the RVID includes sort and data search capabilities. Users can select a desired grouping of plants and then specify information categories to search and list. Tables 5.1, 5.2, and 5.3 provide examples of the Chemistry Data, USE, and PTS/P-T Limit Summary Files that may be accessed by the user in the RVID.

The RVID will be updated periodically to reflect the latest available information. Responses to GL 92-01, Revision 1, Supplement 1, and to the close-out letters to GL 92-01, Revision 1, are not necessarily reflected in the current version, but will be included in a future version of the RVID. Revisions that were made to make the original version of the RVID (Version 1.1) more user friendly are described in the following section.

5.2 Revisions Included in the RVID Version 1.1, Revision 1

The RVID was revised and the user's manual was expanded as a result of the staff's assessment of Version 1.1 and comments from database users. The database with the current changes was issued on the world-wide web as Revision 1 in June 1996. The staff's revisions of the RVID included the following changes:

- The table headings for the PTS summary files were revised to read "Summary File for PTS" for PWRs and "Summary File for Pressure-Temperature" for BWRs in order to reflect that BWRs do not have PTS evaluations. Information regarding P—T limits for BWR beltline materials (not PTS evaluations) are contained in the RVID.

- A means of identifying surveillance data as credible or non-credible was added to the database.[5]

5 At present, with the exception of the surveillance data for eight units, each surveillance data file is defaulted to Credible "Y", implying that, for the rest of the units with surveillance data, all data are credible. The eight units were assessed by the staff because the RVID reflected that surveillance data were being utilized for calculation of the chemistry factor for the limiting material in the RPV. The licensee's for Kewaunee, Indian Point 2, Indian Point 3, Maine Yankee, Robinson 2, and North Anna 1 had two or more sets of credible surveillance capsule data for their limiting material (reflected as credible "Y" in the database). The licensee's for Haddam Neck and Diablo Canyon 1 were determined to have less than two credible sets of surveillance capsule data for the limiting material in their vessels (reflected as credible "N" in the database). The staff will continue to review surveillance data and the appropriate changes for non-credible surveillance data will be made accordingly in future versions of the RVID.

- The code for determining the percent drop in USE was modified to correctly calculate USE values for beltline materials with very high copper values.

The staff's revisions to the user's manual included the following changes:

- The appropriate configuration of the "config.sys" file was outlined, and an explanation included to indicate why the file and buffer configurations in the computer file are important. This "config.sys" file should be stored in memory for the user to be capable of successfully running the RVID.

- The user's manual was modified to include a description of the types of problems that could arise while running the RVID in a networked environment, and specifically during simultaneous, multi-user access of the system.[6]

- Instructions were added regarding how to create a batch file that is needed for successful running of the RVID.

- The user's manual was revised to clarify that P—T limits reports, and not just PTS reports, are available in the RVID.

- Expanded information was provided regarding how to find a specific plant record while in the plant information screen.

- A paragraph was added stating that the database could contain information that has not been submitted to the NRC's Document Control Desk ("docketed" information).

These changes to the RVID and to the user's manual are outlined in NRC Administrative Letter 95-03, Revision 1, dated July 10, 1996 (Ref. 43).

6 The RVID was not designed to work in a networked environment. It can function under this scenario; however, error messages will appear when multiple users are logged into the system.

5.3 Future Revisions to the RVID

There are short-term as well as long-term goals for improving the RVID. Revision 1 was issued on the world wide web in June 1996, incorporating the changes described above. However, for many units the information provided in licensee responses to the staff's close-out letters to GL 92-01, Revision 1, was not included in Revision 1 to the RVID. The information provided in licensee responses to GL 92-01, Revision 1, Supplement 1 was also not included in Revision 1 of the RVID. Licensees have joined with owner's groups in order to address the issues raised in the GL. The new information is expected to be available at the end of 1997, and the RVID will be updated accordingly.

The data environment will eventually be switched to a more user-friendly format. Microsoft Access™ [7] software will be used to bring the database into the Windows environment (as opposed to the current DOS-based FoxPro™ software)[7]. The original version of the RVID is now contained in the industry database known as "RPVDATA," which is maintained by the EPRI. RPVDATA runs under Microsoft Access™ and is a flat file (no computations are performed within the database).

RPVDATA also contains several "tiers" of information that are not in the RVID. Examples include details of weld property sampling and comparisons of licensee "best-estimate" chemical composition values with docketed information. The NRC staff and industry are currently working to resolve inconsistencies between the two databases. Subsequent to the completion of the GL 92-01, Revision 1, Supplement 1 effort, the NRC will make revisions to the RVID. At that time, it will be possible to compare the NRC and industry databases, and establish an NRC approved database. Subsequently, it may be possible to have maintenance of the database performed by the industry with NRC oversight. The NRC will be able to verify the updates by comparisons with docketed plant information.

7 Microsoft Access™ and FoxPro™ are trademarks of the Microsoft Corporation.

Plant Name	Beltline Ident.	Heat No Ident.	% Cu	Data Source for Cu	Method of Determin. Cu	Range of Cu Values	Average Value of Cu	% Ni	Data Source for Ni	Method of Determin. Ni	Range of Ni Values	Average Value of Ni	% P	% S
Arkansas Nuclear 1 EOL: 05/20/14 Docket No.: 50-313														
	LOWER NOZZLE BELT FORGING	AYN 131	0.03					0.70					0.009	0.015
	LOWER SHELL COURSE	C-5114-1	0.15					0.52					0.010	0.016
	UPPER SHELL COURSE	C-5114-2	0.15					0.52					0.010	0.016
	LOWER SHELL COURSE	C-5120-1	0.17					0.55					0.014	0.013
	UPPER SHELL COURSE	C-5120-2	0.17					0.55					0.014	0.013
	UPPER/LOWER SHELL CIRC. WELD WF-112	406L44	0.31					0.59					0.016	0.015
	NOZZLE BELT/UPPER SHELL CIRC. WELD WF-182-1	821T44	0.24					0.63					0.014	0.013
	UPPER SHELL AXIAL WELD WF-18	8T1762	0.20					0.55					0.004	0.017
	LOWER SHELL AXIAL WELDS WF-18	8T1762	0.20					0.55					0.004	0.017

References for Arkansas Nuclear 1

UUSE for Weld WF-182-1 is 80
Fluence, IRTndt, and chemical composition data 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"

Chemistry Factor for plate C5114-1 was calculated using ANO1 surveillance data reported in BAW-2075, Rev. 1

Chemistry Factor for WF-182-1 weld was calculated from Davis-Besse surveillance data that was reported in BAW-1803, Rev. 1. The Davis-Besse surveillance weld was fabricated with the same heat number as WF-182-1.

Table 5.1 — Sample RVID Summary File for Chemistry Data

07/24/96
08:32:06

REACTOR VESSEL INTEGRITY DATABASE
Chemistry Data File Summary

Page: 2

Plant Name	Beltline Ident.	Heat No Ident.	% Cu	Data Source for Cu	Method of Determin. Cu	Range of Cu Values	Average Value of Cu	% Ni	Data Source for Ni	Method of Determin. Ni	Range of Ni Values	Average Value of Ni	% P	% S

References for Arkansas Nuclear 1 (continued)

Chemistry Factor for WF-112 weld was calculated from Oconee 1, Point Beach 2, B&WOG, ANO-1, and Rancho Seco surveillance data that was reported in BAW-1803, Rev. 1. These surveillance welds were fabricated with the same heat number as WF-112.

IRTndt, for nozzle belt forging is a mean value from 24 forings similar to AYN 131. The data is reported in BAW-10046P and has a standard devaiation of 31oF.

Fluence data report in BAW-2222.

UUSE for plates C5120-2, C5114-2, C5120-1 and C 5114-1 reported in BAW-2222.

UUSE for forging AYN 131 was 95/95 tolerance limit that was reported in BAW-2222.

Table 5.1 (Continued) — Sample RVID Summary File for Chemistry Data

07/24/96
08:30:14

REACTOR VESSEL INTEGRITY DATABASE
Summary File for Upper Shelf Energy

Page: 1

Plant Name	Beltline Ident.	Heat No Ident.	Material Type	USE @ EOL @ 1/4T	1/4T Neut. Flu @ EOL	Unirr USE	Method Determ Unirr USE	% Drop USE @ EOL @ 1/4T	Method Determ % Drop	Cu	
Arkansas Nuclear 1 EOL: 05/20/14 Docket No.: 50-313											
	LOWER NOZZLE BELT FORGING	AYN 131	A 508-2	91	0.520	109	GENERIC	16.3%	Position 1 of RG 1.99, Rev. 2	0.03	
	LOWER SHELL COURSE	C-5114-1	A 533B	60	0.567	96	DIRECT	37.5%	Surveillance data	0.15	
	UPPER SHELL COURSE	C-5114-2	A 533B	82	1.984	107	DIRECT	23.3%	Surveillance data	0.15	
	LOWER SHELL COURSE	C-5120-1	A 533B	62	0.567	80	65%	22.6%	Position 1 of RG 1.99, Rev. 2	0.17	
	UPPER SHELL COURSE	C-5120-2	A 533B	60	1.984	86	65%	30.2%	Position 1 of RG 1.99, Rev. 2	0.17	
	UPPER/LOWER SHELL CIRC. WELD WF-112	406L44	LINDE 80	EMA	0.567	EMA	EMA	EMA	EMA	0.31	
	NOZZLE BELT/UPPER SHELL CIRC. WELD WF-182-1	821T44	LINDE 80	EMA	0.520	EMA	SISTER PLANT	EMA	EMA	0.24	
	UPPER SHELL AXIAL WELD WF-18	8T1762	LINDE 80	EMA	0.425	EMA	EMA	EMA	EMA	0.20	
	LOWER SHELL AXIAL WELDS WF-18	8T1762	LINDE 80	EMA	0.419	EMA	EMA	EMA	EMA	0.20	

References for Arkansas Nuclear 1

UUSE for Weld WF-182-1 is 80
Fluence, IRTndt, and chemical composition data 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"

Chemistry Factor for plate C5114-1 was calculated using ANO1 surveillance data reported in BAW-2075, Rev. 1

Table 5.2 — Sample RVID Summary File for Upper Shelf Energy (USE)

Plant Name	Beltline Ident.	Heat No Ident.	Material Type	USE @ EOL @ 1/4T	1/4T Neut. Flu @ EOL	Unirr USE	Method Determ Unirr USE	% Drop USE @ EOL @ 1/4T	Method Determ % Drop	Cu		

References for Arkansas Nuclear 1 (continued)

Chemistry Factor for WF-182-1 weld was calculated from Davis-Besse surveillance data that was reported in BAW-1803, Rev. 1. The Davis-Besse surveillance weld was fabricated with the same heat number as WF-182-1.

Chemistry Factor for WF-112 weld was calculated from Oconee 1, Point Beach 2, B&WOG, ANO-1, and Rancho Seco surveillance data that was reported in BAW-1803, Rev. 1. These surveillance welds were fabricated with the same heat number as WF-112.

IRTndt, for nozzle belt forging is a mean value from 24 forings similar to AYN 131. The data is reported in BAW-10046P and has a standard devaiation of 31oF.

Fluence data report in BAW-2222.

UUSE for plates C5120-2, C5114-2, C5120-1 and C 5114-1 reported in BAW-2222.

UUSE for forging AYN 131 was 95/95 tolerance limit that was reported in BAW-2222.

Table 5.2 (Continued) — Sample RVID Summary File for Upper Shelf Energy (USE)

Plant Name	Beltline Ident.	Heat No Ident.	RTpts @ EOL	ID Neut. Fluence @ EOL	IRTndt	Method of Determin. IRTndt	ΔRTndt at EOL	Fluence Factor @ EOL	Chemistry Factor	Method of Determin. CF	Margin	Method of Determin. Margin	Cu%	Ni%
Arkansas Nuclear 1 EOL: 05/20/14 Docket No.: 50-313														
	LOWER NOZZLE BELT FORGING	AYN 131	93	0.86200	3	B&W GENERI	19.2	0.958	20.00	Table	70.71	TABLE	0.030	0.700
	LOWER SHELL COURSE	C-5114-1	88	0.94000	0	PLANT SPEC	54.0	0.983	54.90	Override	34.00	Surv. Capsule	0.150	0.520
	UPPER SHELL COURSE	C-5114-2	129	3.29000	-10	PLANT SPEC	138.5	1.312	105.60	Table	34.00	TABLE	0.150	0.520
	LOWER SHELL COURSE	C-5120-1	145	0.94000	-10	PLANT SPEC	120.7	0.983	122.75	Table	34.00	TABLE	0.170	0.550
	UPPER SHELL COURSE	C-5120-2	146	3.29000	-10	PLANT SPEC	161.0	1.312	122.75	Table	34.00	TABLE	0.170	0.550
	UPPER/LOWER SHELL CIRC. WELD WF-112	406L44	207	0.94000	-5	B&W GENERI	173.3	0.983	176.27	Override	48.33	Surv. Capsule	0.310	0.590
	NOZZLE BELT/UPPER SHELL CIRC. WELD WF-182-1	821T44	214	0.86200	-5	B&W GENERI	155.3	0.958	162.09	Override	48.33	TABLE	0.240	0.630
	UPPER SHELL AXIAL WELD WF-18	8T1762	201	0.70500	-5	B&W GENERI	137.3	0.902	152.25	Table	68.47	TABLE	0.200	0.550
	LOWER SHELL AXIAL WELDS WF-18	8T1762	200	0.69500	-5	B&W GENERI	136.7	0.898	152.25	Table	68.47	TABLE	0.200	0.550

<u>References for Arkansas Nuclear 1</u>

UUSE for Weld WF-182-1 is 80
Fluence, IRTndt, and chemical composition data 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"

Chemistry Factor for plate C5114-1 was calculated using ANO1 surveillance data reported in BAW-2075, Rev. 1

Chemistry Factor for WF-182-1 weld was calculated from Davis-Besse surveillance data that was reported in BAW-1803, Rev. 1. The Davis-Besse surveillance weld was fabricated with the same heat number as WF-182-1.

Table 5.3 — Sample RVID Summary File for Pressurized Thermal Shock (PTS)

07/24/96
08:30:58

REACTOR VESSEL INTEGRITY DATABASE
Summary File for PTS

Plant Name	Beltline Ident.	Heat No Ident.	RTpts @ EOL	ID Neut. Fluence @ EOL	IRTndt	Method of Determin. IRTndt	ΔRTndt at EOL	Fluence Factor @ EOL	Chemistry Factor	Method of Determin. CF	Margin	Method of Determin. Margin	Cu%	Ni%

References for Arkansas Nuclear 1 (continued)

Chemistry Factor for WF-112 weld was calculated from Oconee 1, Point Beach 2, B&WOG, ANO-1, and Rancho Seco surveillance data that was reported in BAW-1803, Rev. 1. These surveillance welds were fabricated with the same heat number as WF-112.

IRTndt, for nozzle belt forging is a mean value from 24 forings similar to AYN 131. The data is reported in BAW-10046P and has a standard devaiation of 31oF.

Fluence data report in BAW-2222.

UUSE for plates C5120-2, C5114-2, C5120-1 and C 5114-1 reported in BAW-2222.

UUSE for forging AYN 131 was 95/95 tolerance limit that was reported in BAW-2222.

Table 5.3 (Continued) — Sample RVID Summary File for Pressurized Thermal Shock (PTS)

6 SUMMARY AND CONCLUSIONS

The NRC staff has reviewed licensee responses to GL 92-01, Revision 1, Supplement 1. The staff has also performed a generic PTS assessment of all pressurized-water RPVs. The generic assessment was evaluated by the staff using plant-specific surveillance data. On the basis of these reviews, the NRC staff has confirmed that the RT_{PTS} values for most of the domestic RPVs are not projected to exceed the pressurized thermal shock (PTS) screening criteria prior to the end of their current operating licenses.

As discussed in the original "Reactor Pressure Vessel Status Report" (NUREG-1511), the RT_{PTS} values for the limiting materials in the Beaver Valley Unit 1 and Palisades vessels were the only plants projected to exceed their PTS screening criteria prior to expiration of the plant operating licenses.[8] However, based on the results of subsequent chemical composition and mechanical properties tests of weld materials from Palisades' retired steam generators, the Consumers Power Company (CPCo, the licensee for the Palisades plant) projected that the degree of embrittlement of the Palisades RPV could be greater than previously predicted. As a result, CPCo concluded that the RT_{PTS} value for the limiting material in the Palisades RPV would exceed the PTS screening criteria in 1999.

Recently, however, in a letter dated April 4, 1996, CPCo provided a revised PTS assessment that projected a lower neutron fluence at the expiration of the Palisades operating license. As a result of the reduction in neutron fluence, CPCo concluded that the RT_{PTS} value for the limiting material in the Palisade RPV would not reach the PTS screening criteria until many years after 1999. This revised PTS assessment is being reviewed by the staff.

It is important to note that the staff and licensee assessments are based on currently available information reported by the licensees and are subject to change. The dates at which the RT_{PTS} values for the limiting materials in the vessels are projected to exceed 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 recover a large percentage of the vessel's fracture toughness lost to neutron irradiation.

In their responses to GL 92-01, Revision 1, Supplement 1, the industry's Owners Groups (OGs) informed the NRC that additional data and information will be submitted for review by the staff. The OGs' programs include extensive searches for relevant data and the development of methodologies for determining the best-estimate chemistries for welds fabricated using copper-coated electrodes. The last of the OG's programs is not expected to be completed until the summer of 1997. The staff's review of the OGs' data will include a reassessment of each licensee's RPV. After completing this review, the staff will incorporate any new data into the RVID and prepare another update of this report.

It is difficult to predict what the status will be regarding thermal annealing of commercial RPVs in the United States. The commitment to anneal an RPV involves significant engineering and regulatory analyses and the assignment of substantial resources. However, the approach can recover a large percentage of the fracture toughness lost to neutron irradiation, thereby decreasing constraints on plant operation. This approach will enable plant operation to EOL for RPVs potentially challenged by the PTS screening criteria, and extend operation beyond EOL for others.

Successful demonstration of the engineering feasibility of annealing technology in the DOE programs will greatly facilitate future considerations for thermal annealing in the United States.

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8 Based on information available in 1994, the RT_{PTS} values for the limiting materials in the Beaver Valley 1 and Palisades RPVs were projected to exceed the PTS screening limit in 2012 and 2004, prior to EOL in 2016 and 2007, respectively.

7 REFERENCES

1. Generic Letter 92-01, Revision 1, "Reactor Vessel Structural Integrity, 10 CFR 50.54(f)," March 6, 1992.

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

3. Title 10, *Code of Federal Regulations*, Section 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events."

4. Generic Letter 92-01, Revision 1, Supplement 1, "Reactor Vessel Structural Integrity," May 19, 1995.

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

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

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

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

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

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

11. Appendix G to Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code.

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

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

14. Letter (untitled) from B.W. Sheron, Director — Division of Engineering, Office of Nuclear Reactor Regulation, to A. Marion, Nuclear Energy Institute, enclosing internal NRC Memorandum of April 15, 1996, "Summary of Meeting Held with the Nuclear Energy Institute (NEI) and Industry Representatives to Discuss Reactor Vessel Integrity," April 18, 1996.

15. Letter JHT/95-77 from D.L. Howell, Acting Manager — Licensing Services, the Babcock and Wilcox Owners Group, to the U.S. Nuclear Regulatory Commission, "B&W Owners Group Reactor Vessel Integrity Program," August 1, 1995.

16. Babcock and Wilcox Topical Report BAW-2257, Revision 1, "B&W Owners Group Reactor Vessel Working Group Response to Generic Letter 92-01, Revision 1, Supplement 1," October 1995.

17. Letter 95-404 from the J.T. Beckham, Chairman — BWR Vessel and Internals Project, to the U.S. Nuclear Regulatory Commission, forwarding "BWRVIP Response to NRC Generic Letter 92-01, Revision 1, Supplement 1, Reactor Vessel Structural Integrity," August 10, 1995.

18. Letter 96-335 from Vaughn Wagner, Technical Chairman, BWRVIP Integration Committee, to the U.S. Nuclear Regulatory Commission, "Reclassification of a BWRVIP Report Submitted in Response to Generic Letter 92-01, Revision 1, Supplement 1," June 26, 1997.

19. EPRI TR-105908NP, "BWR Vessel and Internals Project, Bounding Assessment of BWR/2-6 Reactor Pressure Vessel Integrity Issues (BWRVIP-08NP)," November 1995.

20. SECY-95-117, "Status of Reactor Pressure Vessel Issues," May 8, 1995.

21. Letter from A.R. Johnson, Project Manager, Project Directorate I-1, Division of Reactor Projects I/II, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, to Dr. R.C. Mecredy, Vice President — Nuclear Operations, Rochester Gas & Electric Company, "R.E. Ginna Nuclear Power Plant, Pressurized Thermal Shock Evaluation (TAC No. M93827)," March 22, 1996.

22. Letter from A.H. Hsia, Project Manager, Project Directorate III-1, Division of Reactor Projects, Office of Nuclear Reactor Regulation, to R.A. Fenech, Consumers Power Company, "Palisades Plant - Interim Safety Evaluation of Pressurized Thermal Shock (PTS) and Upper-Shelf Energy (USE) Analyses," July 12, 1994

23. SECY-82-465, "Pressurized Thermal Shock," November 23, 1982.

24. Letter from E.G. Adensam, Acting Director — Division of Reactor Projects III/IV, Office of Nuclear Reactor Regulation, to K.M. Haas, Plant Safety and Licensing Director, Palisades Plant, "Palisades Plant — Pressurized Thermal Shock Safety Evaluation (TAC No. M83227)," April 12, 1995.

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

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

27. Letter from D.G. McDonald, Jr., Senior Project Manager, Project Directorate I-1, Division of Reactor Projects I/II, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, to R.E. Denton, Vice President — Nuclear Energy, Baltimore Gas and Electric Company, "Updated Values for Pressurized Thermal Shock Reference Temperatures — Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 (TAC Nos. M93230 and M93231)," January 2, 1996.

28. NUREG/CR-4212, "In-Place Thermal Annealing of Nuclear Reactor Pressure Vessels," W.L. Server, April 1985.

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

30. T.R. Mager, "Feasibility of and Methodology for Thermal Annealing of an Embrittled Reactor Vessel," EPRI NP-2712, Vol. 2, Electric Power Research Institute, Palo Alto, CA, November 1982.

31. T.R. Mager et al., "Thermal Annealing of an Embrittled Reactor Vessel, Feasibility and Methodology," EPRI NP-6113-SD, Electric Power Research Institute, Palo Alto, CA, January 1989.

32. SAND94-1515/1, "Proceedings of the DOE/SNL/EPRI Sponsored Reactor Pressure Vessel Thermal Annealing Workshop," September 1994.

33. NUREG/CR-6327, "Models for Embrittlement Recovery Due to Annealing of Reactor Pressure Vessel Steels," E. Eason, J. Wright, R. Odette, E. Mader, February 1995.

34. R. Pelli and K. Torronnen, "State of the Art Review on Thermal Annealing," European Commission, DG XII - Joint Research Centre, Institute for Advanced Materials, DG XI - Safety of Nuclear Installations, Aging Materials and Evaluation Studies (AMES) Report No. 2, December, 1994.

35. U. Potapovs, J.R. Hawthorne, and C.Z. Serpan, Jr., "Notch Ductility Properties of SM-1A Reactor Pressure Vessel Following the In-Place Annealing Operation," *Nuclear Applications*, Vol. 5, No. 6, pp. 389—409, 1968.

36. A. Fabry et al., "Annealing of the BR-3 Reactor Pressure Vessel," in *Proceedings of the Twelfth Water Reactor Safety Research Information Meeting*, NUREG/CP-0058, Vol. 4, pp. 144—175, NRC, January 1985.

37. N. M. Cole and T. Friderichs, "Report on Annealing of the Novovoronezh Unit 3 Reactor Vessel in the USSR," NUREG/CR-5760 (MPR Associates, Inc., MPR-1230), NRC, July 1991.

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

39. Letter from J.M. Taylor, Executive Director for Operations, U.S. Nuclear Regulatory Commission, to Dr. T.R. Lash, Director — Office of Nuclear Energy, Science and Technology, U.S. Department of Energy, enclosing 2 copies of the Memorandum of Understanding for the DOE Annealing Demonstration Project, August 4, 1995.

40. Consumers Power Company submittal to the U.S. Nuclear Regulatory Commission, Revision 1 to Calculation No. EA-RDS-94-02, "Evaluation of Palisades Current PTS Screening Criteria Margin," dated November 8, 1994.

41. Letter from R.A. Fenech, Vice President — Nuclear Operations, Consumers Power Company, to the U.S. Nuclear Regulatory Commission, "Docket 50-255 — License DPR-20 — Palisades Plant 10CFR50.61 — Pressurized Thermal Shock — Additional Preliminary Information," November 10, 1995

42. Letter from R.W. Smedley, Manager — Licensing, Consumers Power Company, to the U.S. Nuclear Regulatory Commission, "Docket 50-255 — License DPR-20 — Palisades Plant 10CFR50.61 — Preliminary Thermal Annealing Report, Thermal Annealing Operating Plan, Section 1.7, Thermal and Stress Analysis, Appendix A and Appendix B," April 29,1996.

43. NRC Administrative Letter 95-03, Revision 1, "Availability of Reactor Vessel Integrity Database," July 10, 1996.

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11. ABSTRACT *(200 words or less)*

This report describes the issues raised as a result of the staff's review of Generic Letter (GL) 92-01, Revision 1, responses and plant-specific reactor pressure vessel (RPV) assessments and the actions taken or work in progress to address these issues. In addition, the report describes actions taken by the staff and the nuclear industry to develop a thermal annealing process for possible use at U.S. commercial nuclear plants to mitigate the effects of neutron radiation on the fracture toughness of RPV materials.

The Nuclear Regulatory Commission (NRC) issued GL 92-01, Revision 1, Supplement 1, to obtain information needed to assess compliance with regulatory requirements and licensee commitments regarding RPV integrity. GL 92-01, Revision 1, Supplement 1, was issued as a result of generic issues raised in the NRC staff's review of licensee responses to GL 92-01, Revision 1, and plant-specific RPV evaluations. In particular, an integrated review of all data submitted in response to GL 92-01, Revision 1, indicated that licensees may not have considered all relevant data in their RPV assessments.

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