TREAT AS
SENSITIVE
INFORMATION

ATTACHMENT A

Request for License Amendment Regarding Reactor Vessel Specimen Removal Schedule

EVALUATION OF PROPOSED CHANGES

1.0 INTRODUCTION

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company (EGC), LLC and AmerGen Energy Company (AmerGen), LLC are requesting changes to the Updated Final Safety Analysis Reports (UFSARs) for the Clinton Power Station (CPS), Unit 1, the Dresden Nuclear Power Station (DNPS), Units 2 and 3, the LaSalle County Station (LSCS), Units 1 and 2, the Limerick Generating Station (LGS), Units 1 and 2, the Oyster Creek Generating Station (Oyster Creek), the Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3, and the Quad Cities Nuclear Power Station (QCNPS), Units 1 and 2. The requested changes will remove the current facility reactor material surveillance capsule removal schedules from the facility UFSARs and specify that these facilities will participate in an integrated surveillance program (ISP) developed by the Boiling Water Reactor Vessel and Internals Project (BWRVIP). For LGS, the proposed changes will also remove the current facility reactor material specimen surveillance schedule from the Technical Specifications (TS).

EGC and AmerGen request approval of this license amendment request by October 6, 2003, to support revision to the facility UFSARs prior to October 2003, which is the next scheduled reactor surveillance capsule removal at DNPS.

All EGC and AmerGen submittals currently under review by the NRC were evaluated to determine the impact of these proposed changes. No submittals currently under review by the NRC are affected by the information presented in these license amendment requests. DNPS, PBAPS, and QCNPS are in the process of seeking license renewal. The BWRVIP may determine that future adjustments to the ISP are needed as a result of license renewal for these sites.

2.0 DESCRIPTION OF THE PROPOSED AMENDMENT

With the exception of Oyster Creek, the UFSARs of each of the facilities listed in this amendment request contain a withdrawal schedule for the reactor pressure vessel (RPV) material specimens. For facilities which are not scheduled to remove a material specimen as part of the ISP, the proposed amendment would remove these plant-specific schedules from the facility UFSARs and substitute a description of the facility's participation in the ISP. For facilities which are scheduled to remove a capsule as part of the ISP, the proposed amendment would revise the material specimen withdrawal schedule in accordance with the ISP. For Oyster Creek, which is not scheduled to remove any further material specimens, the proposed amendment would revise the UFSAR to state that Oyster Creek will participate in the ISP.

The revised UFSAR section for each facility will state that the facility will participate in the NRC-approved version of the ISP, which is currently described in References 1 and 2. EGC and AmerGen will evaluate any subsequent NRC-approved changes to the ISP, and will either incorporate these changes into each facility UFSAR or seek NRC

approval of a proposed alternative material specimen surveillance program in accordance with 10 CFR 50 Appendix H, "Reactor Vessel Material Surveillance Program Requirements."

The current NRC-approved revision of the BWRVIP ISP states that the following EGC/AmerGen units will remove material surveillance specimens as part of the ISP. All other EGC/AmerGen units will use material specimen results from other units in accordance with the ISP.

Unit	BWRVIP ISP Specimen Withdrawal (EFPY)
Dresden Nuclear Power Station, Unit 3	30
LaSalle County Station, Unit 1	17
Peach Bottom Atomic Power Station, Unit 2	30

For LGS, in addition to the UFSAR change, the proposed changes revise TS Section 3/4.4.6, "Pressure/Temperature Limits." Specifically, the proposed changes delete Surveillance Requirements (SRs) 4.4.6.1.3 and 4.4.6.1.4, and TS Table 4.4.6.1.3-1, which require removal and examination of the reactor vessel material surveillance specimens and reactor flux wire specimens in accordance with the schedule in Table 4.4.6.1.3-1. The revised LGS UFSAR description will replace the deleted TS requirements. The TS table of contents is also revised to reflect the deletion of Table 4.4.6.1.3-1. Also for LGS, the proposed changes revise the LGS TS Bases. A copy of the proposed marked-up TS and TS Bases changes is provided in Attachment B-4. The revised LGS TS pages are provided in Attachment C.

In addition, for CPS, marked-up TS Bases changes are provided in Attachment B-1. Upon approval of the proposed UFSAR changes, CPS will implement the indicated TS Bases changes in accordance with the CPS Bases Control Program.

3.0 BACKGROUND

In References 1 and 2, supplemented by References 3 and 4, the BWRVIP described the technical basis for the development and implementation of an ISP intended to support operation of all U. S. BWR RPVs through the completion of each facility's current 40-year operating license.

The BWR ISP was developed in response to an issue raised by the NRC regarding the potential lack of adequate unirradiated baseline Charpy V-notch (CVN) data for one or more materials in plant-specific RPV surveillance programs at several BWRs. The lack

of baseline properties would inhibit a licensee's ability to effectively monitor changes in the fracture toughness properties of RPV materials in accordance with 10 CFR 50 Appendix H. The BWR ISP, as approved by the NRC, resolves this issue.

Implementation of the ISP will provide additional benefits. When the original surveillance materials were selected for plant-specific surveillance programs, the state of knowledge concerning RPV material response to irradiation and post-irradiation fracture toughness was not the same as it is today. As a result, many facilities did not include what would be identified today as the plant's limiting RPV materials in their surveillance programs. Hence, this effort to identify and evaluate materials from other BWRs, which may better represent a facility's limiting materials, should improve the overall evaluation of BWR RPV embrittlement. Second, the inclusion of data from the testing of BWR Owners' Group (BWROG) Supplemental Surveillance Program (SSP) capsules will improve overall quality of the data being used to evaluate BWR RPV embrittlement. Further, occupational exposure will be reduced due to elimination of the need for some units to remove material specimens. Overall, the combined benefits of the ISP are substantial. Finally, implementation of the ISP is also expected to reduce the cost of surveillance testing and analysis since surveillance materials that are of little or no value will no longer be tested (either because they lack adequate unirradiated baseline CVN data or because they are not the best representative materials).

In Reference 5, the NRC determined that the ISP proposed by the BWRVIP is an acceptable alternative to all existing BWR plant-specific RPV surveillance programs for the purpose of maintaining compliance with requirements of 10 CFR 50 Appendix H. Reference 5 stated that licensees electing to participate in the ISP should provide information regarding what specific neutron fluence methodology will be implemented as part of participation in the ISP and to address the compatibility of comparison of neutron fluences calculated for each RPV with neutron fluences calculated for the surveillance capsules in the ISP.

In Reference 6, the NRC stated that licensees who elect to participate in the ISP should submit a license amendment request to incorporate this program into their licensing basis.

4.0 REGULATORY REQUIREMENTS AND GUIDANCE

10 CFR Part 50 Appendix G, "Fracture Toughness Requirements," which is invoked by 10 CFR 50.60, "Acceptance criteria for fracture prevention measures for light water nuclear power reactors for normal operation," specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary, including RPVs. In order to support evaluations to demonstrate that compliance with these requirements will be maintained, information regarding irradiated RPV material properties and the neutron fluence level of a licensee's RPV is necessary. Therefore, 10 CFR 50.60 also invokes 10 CFR 50 Appendix H, which requires licensees to implement an RPV material surveillance program.

However, an alternative to individual plant-specific RPV surveillance programs is addressed in Paragraph III.C. of 10 CFR 50 Appendix H. In accordance with Paragraph III.C. of Appendix H, an RPV ISP may be implemented, with approval of the Director of the Office of Nuclear Reactor Regulation, by two or more facilities with similar design and operating features. Paragraph III.C. also sets forth specific criteria upon which approval of an ISP shall be based. The specified criteria include:

a. The reactor in which the materials will be irradiated and the reactor for which the materials are being irradiated must have sufficiently similar design and operating features to permit accurate comparisons of the predicted amount of radiation damage;

b. Each reactor must have an adequate dosimetry program;

c. There must be adequate arrangement for data sharing between plants;

d. There must be a contingency plan to assure that the surveillance program for each reactor will not be jeopardized by operation at reduced power level or by an extended outage of another reactor from which data are expected; and,

e. There must be substantial advantages to be gained, such as reduced power outages or reduced personnel exposure to radiation, as a direct result of not requiring surveillance capsules in all reactors in the set.

In addition, no reduction in the requirements for the number of materials to be irradiated, specimen types, or number of specimens per reactor is permitted. Finally, no reduction in the amount of testing is permitted unless authorized by the Director of the Office of Nuclear Reactor Regulation.

In Reference 5, the NRC documented the BWRVIP ISP met the above criteria.

The proposed deletion of the LGS TS SRs regarding the material surveillance program meets the requirements of 10 CFR 50.36, "Technical specifications," paragraph (c)(3), which requires that the TS contain SRs relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that the facility operation will be within safety limits, and that the limiting conditions for operation will be met. The SRs related to the material specimen surveillance program support the RPV pressure-temperature (P-T) limits, which are contained in the LGS TS. The P-T limits restrict pressure and temperature changes during heatup and cooldown to maintain the RPV within the design assumptions and stress limits for cyclic operation.

In addition, the proposed deletion of TS SRs regarding the material surveillance program is consistent with the current NRC guidance regarding TS content. The deletion of the material specimen program from TS was allowed by NRC Generic Letter (GL) 91-01, "Removal of the Schedule for the Withdrawal of Reactor Vessel Material Specimens

from Technical Specifications." Thus, the SRs related to the material surveillance program are not required to be in the TS.

5.0 TECHNICAL ANALYSIS

BWRVIP-78 (Reference 1) described the technical basis related to material selection and testing on which the BWRVIP ISP was constructed. The report principally addressed the methodology established to identify existing plant-specific surveillance capsules and surveillance capsules from the SSP initiated by the BWROG in the late 1980s which contain important surveillance materials for inclusion within the ISP. In this case, "important" surveillance materials may be understood to be those which best represent the actual limiting (in terms of predicted fracture behavior) plate and weld materials from which BWR RPVs were constructed. The report also established the connection between the identified surveillance materials and the specific BWR RPV plate or weld materials which they represent and provided a proposed test matrix for the ISP.

BWRVIP-86 (Reference 2) addressed determination of ISP surveillance capsule withdrawal and testing dates, information on ISP project administration, additional information on neutron fluence determination issues, additional information on data utilization and sharing, and information on licensing aspects of ISP implementation.

The NRC's approval of the technical basis for the ISP (Reference 5) stated that licensees electing to participate in the ISP should provide information regarding what specific neutron fluence methodology will be implemented as part of participation in the ISP. With the exception of Oyster Creek, the EGC and AmerGen facilities listed in this amendment request have recently completed an evaluation of the RPV fluence using the methodology described in General Electric (GE) Company Licensing Topical Report (LTR) NEDC-32983P, "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation," (Reference 7). This methodology is in accordance with the recommendations of Regulatory Guide (RG) 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," dated March 1, 2001, and was approved by the NRC in Reference 8. The results from these evaluations will be used to update the currently-approved pressure-temperature (P-T) limits, if necessary, for these facilities. Future evaluations of RPV fluence at these facilities will be completed using a method in accordance with the recommendations of RG 1.190.

The Oyster Creek P-T limits are currently approved through end of plant life. If Oyster Creek determines there is a need to revise the current P-T limits, Oyster Creek will submit a license amendment request for the P-T limits based on an updated RPV fluence evaluation performed in accordance with RG 1.190.

Reference 5 also stated that licensees should address the compatibility of comparison of neutron fluences calculated for each RPV with neutron fluences calculated for the surveillance capsules in the ISP. The BWRVIP will evaluate the neutron fluence of the surveillance capsules withdrawn as part of the ISP using a method consistent with

RG 1.190. This will ensure compatibility of the methods used to calculate RPV and capsule neutron fluence.

6.0 REGULATORY ANALYSIS

The regulatory requirements for an ISP were discussed in Section 4.0 above. In Reference 5, the NRC concluded that the BWRVIP ISP met the regulatory criteria for approval of an ISP.

7.0 NO SIGNIFICANT HAZARDS CONSIDERATION

According to 10 CFR 50.92, "Issuance of amendment," paragraph (c), a proposed amendment to an operating license involves a no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

In support of this determination, an evaluation of each of the three criteria set forth in 10 CFR 50.92 is provided below regarding the proposed license amendment.

Overview

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company (EGC), LLC and AmerGen Energy Company (AmerGen), LLC are requesting changes to the Updated Final Safety Analysis Reports (UFSARs) for the Clinton Power Station (CPS), Unit 1, the Dresden Nuclear Power Station (DNPS), Units 2 and 3, the LaSalle County Station (LSCS), Units 1 and 2, the Limerick Generating Station (LGS), Units 1 and 2, the Oyster Creek Generating Station (Oyster Creek), the Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3, and the Quad Cities Nuclear Power Station (QCNPS), Units 1 and 2. The requested change will remove the current facility reactor material specimen surveillance schedules from the UFSARs and specify that these facilities will participate in an integrated surveillance program (ISP) developed by the Boiling Water Reactor Vessel and Internals Project (BWRVIP). In addition, for LGS, the proposed changes will also remove the current facility reactor material specimen surveillance schedule from the Technical Specifications.

ATTACHMENT A

Request for License Amendment Regarding Reactor Vessel Specimen Removal Schedule

The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change adopts an integrated surveillance program (ISP) for reactor material specimen surveillances. The ISP ensures that the reactor pressure vessel (RPV) will continue to meet all applicable fracture toughness requirements. No physical changes to the facilities will result from the proposed change. The initial conditions and methodologies used in accident analyses remain unchanged. The proposed change does not revise or alter the design assumptions for systems or components used to mitigate the consequences of accidents. Thus, accident analyses results are not affected by this proposed change.

Therefore, this proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change adopts an ISP for reactor material specimen surveillances. The ISP ensures that the RPV will continue to meet all applicable fracture toughness requirements. No physical changes to the facilities will result from the proposed change.

The proposed change does not affect the design or operation of any system, structure, or component (SSC) in the plant. The safety functions of the related SSCs are not changed in any manner, nor is the reliability of any SSC reduced. The change does not affect the manner by which the facility is operated and does not change any facility, structure, system, or component. No new or different type of equipment will be installed by this proposed change.

Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any previously evaluated.

The proposed change does not involve a significant reduction in a margin of safety.

The proposed change has no impact on the margin of safety of any Technical Specification. There is no impact on safety limits or limiting safety system settings. The change does not affect any plant safety parameters or setpoints. No physical or operational changes to the facility will result from the proposed changes. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

Conclusion

Based upon the above evaluation, EGC and AmerGen have concluded that the criteria of 10 CFR 50.92(c) are satisfied and that the proposed change involves no significant hazards consideration.

8.0 ENVIRONMENTAL CONSIDERATION

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company (EGC), LLC and AmerGen Energy Company (AmerGen), LLC are requesting changes to the Updated Final Safety Analysis Reports (UFSARs) for the Clinton Power Station (CPS), Unit 1, the Dresden Nuclear Power Station (DNPS), Units 2 and 3, the LaSalle County Station (LSCS), Units 1 and 2, the Limerick Generating Station (LGS), Units 1 and 2, the Oyster Creek Generating Station (Oyster Creek), the Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3, and the Quad Cities Nuclear Power Station (QCNPS), Units 1 and 2. The requested change will remove the current facility reactor material specimen surveillance schedules from the UFSARs and specify that these facilities will participate in an integrated surveillance program (ISP) developed by the Boiling Water Reactor Vessel and Internals Project (BWRVIP). In addition, for LGS, the proposed changes will remove the current facility reactor material specimen surveillance schedule from the Technical Specifications.

EGC and AmerGen have evaluated this proposed change against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21, "Criteria for and identification of licensing and regulatory actions requiring environmental assessments." EGC and AmerGen have determined that this proposed change meets the criteria for a categorical exclusion set forth in 10 CFR 51.22, "Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review," paragraph (c)(9), and as such, has determined that no irreversible consequences exist in accordance with 10 CFR 50.92, "Issuance of amendment," paragraph (b). This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities," which changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, "Standards for Protection Against Radiation," or that changes an inspection or a surveillance requirement, and the amendment meets the following specific criteria:

(i) The amendment involves no significant hazards consideration.

As demonstrated in Section 7.0, this proposed change does not involve any significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite.

The proposed change revises the reactor vessel material specimen surveillance program to adopt an integrated surveillance program (ISP). It does not allow for an increase in the unit power level, does not increase the production, nor alter the flow path or method of disposal of radioactive waste or byproducts. Therefore, the proposed change does not affect actual unit effluents.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

The proposed change will not result in changes in the operation or configuration of the facility. There will be no change in the level of controls or methodology used for processing of radioactive effluents or handling of solid radioactive waste, nor will the proposal result in any change in the normal radiation levels within the plant. Therefore, there will be no increase in individual or cumulative occupational radiation exposure resulting from this change.

9.0 PRECEDENT

The NRC approved the BWRVIP ISP in February 2002. Currently, no other licensee has received NRC approval to participate in the ISP.

10.0 REFERENCES

1. Letter from C. Terry (BWRVIP) to U. S. NRC, "Project No. 704 - BWR Vessel and Internals Project, BWR Integrated Surveillance Program Plan (BWRVIP-78)," dated December 22, 1999

2. Letter from C. Terry (BWRVIP) to U. S. NRC, "Project No. 704 - BWRVIP-86: BWR Vessel and Internals Project, BWR Integrated Surveillance Program Implementation Plan," EPRI Technical Report 1000888, dated December 22, 2000

3. Letter from C. Terry (BWRVIP) to U. S. NRC, "Project No. 704 – BWRVIP Response to NRC Request for Additional Information Regarding BWRVIP-78," dated December 15, 2000

4. Letter from C. Terry (BWRVIP) to U. S. NRC, "Project No. 704 – BWRVIP Response to Second NRC Request for Additional Information Regarding BWRVIP-78," dated May 30, 2001

5. Letter from U. S. NRC to C. Terry (BWRVIP), " Safety Evaluation Regarding EPRI Proprietary Reports 'BWR Vessel And Internals Project, BWR Integrated Surveillance Program Plan (BWRVIP-78)' and 'BWRVIP-86: BWR Vessel And Internals Project, BWR Integrated Surveillance Program Implementation Plan,'" dated February 1, 2002

6. NRC Regulatory Issue Summary 2002-05, "NRC Approval of Boiling Water Reactor Pressure Vessel Integrated Surveillance Program," dated April 8, 2002

7. Letter from J. F. Klapproth (GE) to U. S. NRC, "Submittal of GE Proprietary Document NEDC-32983P, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation," dated September 1, 2000

8. Letter from U. S. NRC to J. F. Klapproth (GE), "Safety Evaluation for NEDC-32983P, 'General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation,'" dated September 14, 2001

ATTACHMENT B-1

Request for License Amendment Regarding Reactor Vessel Specimen Removal Schedule

Marked-up UFSAR and TS Bases Pages
Clinton Power Station

UFSAR Pages

5.3-7
5.3-8
5.3-25
5.3-26

TS Bases Pages

B 3.4-53a
B 3.4-61a
B 3.4-61b

5.3.1.6 Material Surveillance

5.3.1.6.1 Compliance with "Reactor Vessel Material Surveillance Program Requirements"

The materials surveillance program monitors changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region resulting from their exposure to neutron irradiation and thermal environment.

Reactor vessel materials surveillance specimens ~~are~~ were provided in accordance with requirements of ASTM E 185-73 and 10 CFR 50 Appendix H. Materials for the program ~~are~~ selected to represent materials used in the reactor beltline region. Specimens ~~are~~ manufactured from a plate actually used in the beltline region and a weld typical of those in the beltline region and thus represent base metal, weld material, and the weld heat affected zone material. The plate and weld ~~are~~ heat treated in a manner which simulates the actual heat treatment performed on the core region shell plates of the completed vessel.

Each in-reactor surveillance capsule contains 36 Charpy-V-Notch specimens. The capsule loading consists of 12 specimens each of base metal, weld metal, and heat affected zone material. A set of out-of-reactor baseline Charpy-V-Notch specimens and archive material ~~are~~ was provided with the surveillance test specimens.

In accordance with the requirements of the edition of 10 CFR 50, Appendix H that was current at the time of vessel manufacture, three surveillance capsules ~~are~~ were provided since the predicted end of life adjusted reference temperature of the reactor vessel steel, as predicted at the time of design, was less than 100° F. The proposed withdrawal schedule is as follows:

First Capsule - 10 effective full power years.

Second Capsule - 20 effective full power years.

Insert A

Third Capsule -EOL, or at the time when the accumulated neutron fluence of the capsule corresponds to the approximate EOL fluence at the reactor vessel 1/4 T location, whichever comes first.

This is in accordance with the intent of ASTM E-185-82 as required by 10 CFR 50, Appendix H. The times in Table 1 of ASTM E-185-82 have been adjusted to account for the fact the BWR-6 lead factors are less than one (0.89 with respect to the vessel 1/4 T location and 0.67 with respect to the vessel inner wall), and to provide more meaningful results since the total Delta RT_{NDT} values (without Margin) for the surveillance samples, calculated in accordance with Regulatory Guide 1.99, Revision 2, are only about 20 degrees after 10 effective full power years.

In accordance with 10 CFR 50 Appendix H, NRC Staff review and approval is required for any changes to this specimen capsule withdrawal schedule. Specifically, in accordance with NRC Administrative Letter 97-04 "NRC Staff Approval for Changes to 10 CFR Part 50, Appendix H, Reactor Vessel Surveillance Specimen Withdrawal Schedules" schedule changes that conform to ASTM E-185 require NRC approval to verify conformance with the ASTM standard prior to implementation, schedule changes that do not conform to ASTM E-185 require submittal of a license amendment prior to implementation.

Fracture toughness testing of irradiated capsule specimens will be in accordance with requirements of 10 CFR 50, Appendix H.

5.3.1.6.2 Neutron Flux and Fluence Calculations

A description of the methods of analysis is contained in Subsections 4.1.4.5 and 4.3.3.

The peak fluence at the inside surface of the vessel beltline shell is 6.9×10^{18} n/cm^2 after 32 EFPY of operation. The peak fluence is used for calculating the effect of irradiation on USE. The fluence for the reactor vessel pressure/temperature calculations was conservatively determined by increasing the fluence by 25 percent to account for flux wire measurement uncertainty. Therefore, the beltline plate inside surface peak fluence is 8.7×10^{18} n/cm^2. The girth weld inside surface fluence is 8.1×10^{17} n/cm^2, which is adjusted for the axial flux distribution.

limit in further detail by assuring that the vessel bottom head region will not be warmed at an excessive rate caused by rapid sweep out of cold coolant in the vessel lower head region by recirculating pump operation or natural circulation (cold coolant can accumulate as a result of control drive inleakage and/or low recirculation flow rate during startup or hot standby). The Item 3 limit further restricts operation of the recirculating pumps to avoid high thermal stress effects in the pumps and piping, while also minimizing thermal stresses on the vessel nozzles.

The above operational limits when maintained insure that the stress limits within the reactor vessel and its components are within the thermal limits to which the vessel was designed for normal operating conditions. To maintain the integrity of the vessel in the event that these operational limits are exceeded the reactor vessel has also been designed to withstand a limited number of transients caused by operator error. Also, for abnormal operating conditions where safety systems or controls provide an automatic temperature and pressure response in the reactor vessel, the reactor vessel integrity is maintained since the severest anticipated transients have been included in the design conditions. Therefore, it is concluded that the vessel integrity will be maintained during the most severe postulated transients, since all such transients are evaluated in the design of the reactor vessel. The postulated transient for which the vessel has been designed is shown on Figure 5.2-5 and discussed in Section 5.2.2.

5.3.3.7 Inservice Surveillance

Inservice inspection of the pressure vessel will be performed in accordance with the requirements of Section XI of the ASME Boiler and Pressure and Vessel Code as described in Subsection 5.2.4.

The materials surveillance program will monitor changes in the fracture toughness properties of ferritic materials in the belt line region resulting from their exposure to neutron irradiation and thermal environment. Specimens of actual belt line material will be exposed in the reactor vessel and periodically withdrawn for impact testing. Operating procedures will accommodate or be changed to reflect the test results to assure adequate brittle fracture control.

Materials surveillance and inservice inspection programs are in accordance with the applicable requirements of 10 CFR 50, including Appendices G and H, and ASME Code requirements, and provide assurance that brittle fracture control and pressure vessel integrity will be maintained throughout the service lifetime of the reactor pressure vessel.

5.3.4 References

(1) Deleted.

(2) Watanabe, H., "Boiling Water Reactor Feedwater Nozzle/ Sparger Final Report", NEDE-21821-A (proprietary version), February 1980.

(3) Branlund, B.J., and Frew, B.D., "Pressure-Temperature Curves for AmerGen, Clinton Power Station Using the K_{IC} Methodology," GE-NE-B13-02084-00-01, Rev. 0 (Proprietary Version), August 2000.

Insert B

Clinton UFSAR Inserts

Insert A

In 2003, the NRC approved Clinton Power Station's participation in the BWR Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) as described in BWRVIP-78 and BWRVIP-86 (References 4 and 5). The NRC approved the ISP for the industry in Reference 6 and approved Clinton Power Station's participation in Reference 7. The ISP meets the requirements of 10 CFR 50 Appendix H and provides several advantages over the original program. The surveillance materials in many plant-specific programs do not represent the best match with the limiting vessel beltline materials since some were established prior to 10 CFR 50 Appendix H requirements. Also, the ISP allows for better comparison to unirradiated material data to determine actual shifts in toughness. Finally, for many plants, ISP data will be available sooner to factor into plant operations since there are more sources of data.

The current withdrawal schedule is based on the latest NRC-approved revision of BWRVIP-86 (Reference 5). Based on this schedule, Clinton Power Station is not scheduled to withdraw any additional material specimens.

Insert B

4. Letter from C. Terry (BWRVIP) to U. S. NRC, "Project No. 704 - BWR Vessel and Internals Project, BWR Integrated Surveillance Program Plan (BWRVIP-78)," dated December 22, 1999

5. Letter from C. Terry (BWRVIP) to U. S. NRC, "Project No. 704 - BWRVIP-86: BWR Vessel and Internals Project, BWR Integrated Surveillance Program Implementation Plan," EPRI Technical Report 1000888, dated December 22, 2000

6. Letter from U. S. NRC to C. Terry (BWRVIP), " Safety Evaluation Regarding EPRI Proprietary Reports 'BWR Vessel And Internals Project, BWR Integrated Surveillance Program Plan (BWRVIP-78)' and 'BWRVIP-86: BWR Vessel And Internals Project, BWR Integrated Surveillance Program Implementation Plan,'" dated February 1, 2002

7. [Insert letter approving Clinton participation in ISP]

BASES (continued)

BACKGROUND (continued)

10 CFR 50, Appendix G (Ref. 1), requires the establishment of P/T limits for material fracture toughness requirements of the RCPB materials. Reference 1 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME) Code, Section III, Appendix G (Ref. 2).

The actual shift in the RT_{NDT} of the vessel material will be established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185 (Ref. 3) and 10 CFR 50, Appendix H (Ref. 4). The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Reference 5.

Withdrawal of the first surveillance capsule has been deferred from a vessel exposure of 10 Effective Full Power Years (EPFY) to 10.4 EFPY (Ref. 20).

With regard to the reactor vessel material specimen capsule withdrawal schedule, NRC staff review and approval of any change to this schedule is required prior to implementation. Furthermore, changes to the capsule removal schedule that do not conform with ASTM E-185 (Ref. 3) require NRC approval in the form of a license amendment as described in NRC Administrative Letter 97-04 (Ref. 10).

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.11.8 and SR 3.4.11.9 (continued)

With regard to temperature difference values obtained pursuant to this SR, as read from plant indication instrumentation, the specified limit is considered to be a nominal value and therefore does not require compensation for instrument indication uncertainties (Refs. 16, 17).

REFERENCES

1. 10 CFR 50, Appendix G.
2. ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.
3. ASTM E 185-82, "Standard Practice for Conducting Surveillance Tests For Light-Water Cooled Nuclear Power Reactor Vessels."
4. 10 CFR 50, Appendix H.
5. Regulatory Guide 1.99, Revision 2, May 1988.
6. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.
7. NEDO-21778-A, "Transient Pressure Rises Affecting Fracture Toughness Requirements for BWRs," December 1978.
8. USAR, Section 15.4.4.
9. USAR, Section 5.3. Deleted
10. NRC Administrative Letter 97-04, "NRC Staff Approval for Changes to 10 CFR Part 50, Appendix H, Reactor Vessel Surveillance Specimen Withdrawal Schedules."
11. Calculation IP-0-0036.
12. Calculation IP-0-0037.
13. Calculation IP-0-0038.
14. Calculation IP-0-0039.

(continued)

BASES

REFERENCES (continued)

15. Calculation IP-0-0040.

16. Calculation IP-0-0041.

17. Calculation IP-0-0042.

18. GE-NE-B13-02084-00-01, Rev. 0, "Pressure-Temperature Curves for AmerGen, Clinton Power Station Using the K_{Ic} Methodology," August 2000.

19. NRC Letter from Jon B. Hopkins to Mike Reandeau, "Clinton Power Station, Unit 1 - Issuance of Amendment (TAC No. MA9862)," dated October 31, 2000.

20. License Amendment 143 dated March 8, 2002.

ATTACHMENT B-2

Request for License Amendment Regarding Reactor Vessel Specimen Removal Schedule

Marked-up UFSAR Pages
Dresden Nuclear Power Station, Units 2 and 3

Pages
5.3-4
5.3-15
Table 5.3-1

controlled. The applicable industry codes and standards for the station, at that time, did not contain these rules for brittle fracture.

A report describing the ductile yielding analysis of the reactor vessel, including a discussion of the assumptions, methods of analysis, and conclusions, has been prepared. The report also addresses thermal shock and brittle fracture.[1]

A comprehensive tabulation of fracture toughness results on the reactor vessel plate material and welds is contained in Appendix 5A. Details of the qualification results for the electroslag welds are given in Appendix 5B. Section 5.3.2 presents additional details on the provisions for fracture toughness evaluations during the operating life of the reactor vessels. Section 5.3.1.6 addresses the reactor vessel material surveillance program.

5.3.1.6 Material Surveillance

Vessel material surveillance samples are located within the reactor vessel to enable periodic monitoring of material properties with exposure. The program includes specimens of the base metal, weld metal, and heat-affected zone metal. These specimens receive higher neutron fluxes than the vessel wall ¼ T location and, therefore, lead it in integrated neutron flux. About 400 samples were initially inserted in the vessel; samples are periodically removed for Charpy V-notch and tensile strength tests.

The reactor vessel is a primary barrier against the release of fission products to the environment. In order to provide assurance that this barrier is maintained with a high degree of integrity, a materials surveillance program was developed and initiated at the beginning of nuclear operation of the reactors. This surveillance program ~~conforms to~~ was designed to be in conformance with the requirements of ASTM E185-62 with one exception. The base metal specimens of the vessels were made with their longitudinal axes parallel to the principal rolling direction of the vessel plate. Insert A

~~Neutron flux monitors and samples are installed in the reactor vessels adjacent to the vessel wall at the core midplane level. The monitor and sample programs, where possible, conform to ASTM E185-62. The flux monitors and metal samples are removed. The flux monitors are tested to experimentally verify the calculated values of integrated neutron flux that are used to determine the nil ductility transition (NDT) temperature using test data from the metal samples (see Section 5.3.2).~~

The withdrawal schedules for both units are shown in Table 5.3-1. The original withdrawal schedule ~~is~~ was based on the three capsule surveillance programs as defined in Section 11.C.3.a of 10 CFR 50, Appendix H. The accelerated capsule (near core top guide) is not required by Appendix H but was tested to provide additional information on the vessel material. The results of the tests and examinations of these samples are used to generate the information addressed in Section 5.3.2.

In the SEP Topic V-6, reactor vessel integrity was evaluated and the material surveillance program was found acceptable.

Insert B

5.3.4 References

1. L.C. Hsu, A Comprehensive Analysis of the Structural Integrity of GE-BWR Vessels Subject to the Design Basis Accident, November 1968.

2. Letter from M.H. Richter (CECo) to T.E. Murley (NRC), dated July 3, 1991, Reactor Vessel Head Closure Studs.

3. H.G. Mehta (GE), Fracture Mechanics Based Structural Margin Evaluation for Commonwealth Edison BWR Reactor Pressure Vessel Head Studs, GE-NE-523-93-0991, DRF 137-0010, September 1991.

4. H.H. Klepfer, et al., Investigation of Cause of Cracking in Austenitic Stainless Steel Piping, Volume 1, NEDO-21000, General Electric, July 1975, p. 8-1.

5. T.A. Caine (GE), Pressure-Temperature Curves per Regulatory Guide 1.99, Revision 2 for the Dresden and Quad Cities Nuclear Power Stations, SASR 89-54, DRF 137-0010, Revision 1, August 1989.

6. NRC Regulatory Guide 1.99, Revision 2, May 1988, Radiation Embrittlement of Reactor Vessel Materials.

7. Letter from R. Stols (CECo) to T.E. Murley (NRC), dated July 2, 1990, Reactor Vessel Fabrication History Summary (Transmitting Document 508-9006, Dresden II Upper Vessel Contract Variation Review by General Electric Company, June 29, 1990).

8. Letter from M.H. Richter (CECo) to T.E. Murley (NRC), dated September 4, 1990, Summary of Fabrication History for the Unit 3 Upper Reactor Vessel.

9. Letter from R. Stols (CECo) to T.E. Murley (NRC), dated January 3, 1991, Reactor Vessel Fabrication History Summary.

10. S.P. Selby and W.E. Brooks, "CRDM Nozzle Inspection," Nuclear Plant Journal, November/December 1992, pp. 56ff.

11. S. Ranganath and T.L. Chapman, "Inservice Inspection Experience in Boiling Water Reactors," Nuclear Plant Journal, November/December 1992, pp. 77ff.

12. T.A. Caine (GE), Tabulation of Thermal Cycles for Dresden Nuclear Power Station Units 2 and 3, SASR 89-111, Revision 2, November 1990.

13. Letter from P.L. Piet (CECo) to T.E. Murley (NRC), dated February 18, 1992, Dresden Nuclear Power Station Units 2 and 3, Inservice Inspection Plan (ISI), Revision 0 of the Third Ten-Year Inspection Interval ISI Plan.

INSERT C

Table 5.3-1

NEUTRON FLUX MONITOR AND BASE METAL SAMPLE WITHDRAWAL SCHEDULE

	Withdrawal ~~Year~~[1] Schedule (1)	Part Number	Location	Comments
Unit 2	1977	6	Near core top guide — 180°	Accelerated sample
	1980	8	Wall — 215°	
	2000	7	Wall — 95°	Standby
		9	Wall — 245°	Standby
		10	Wall — 275°	Standby
Unit 3	1978	16	Near core top guide — 180°	Accelerated sample
	1981	18	Wall — 215°	
	2001 30 EFPY	19	Wall — 245°	
		15	Wall — 65°	Standby
		20	Wall — 275°	Standby

Note:

1. ~~Adjustments to withdrawal year should be made due to unscheduled shutdowns and updated fuel exposure data.~~ Withdrawals completed are listed by year withdrawn. Future withdrawals are listed by the effective full power years (EFPY) anticipated at withdrawal.

Dresden UFSAR Inserts

Insert A

Two material specimens were removed and tested under this original program.

In 2003, the NRC approved Dresden's participation in the BWR Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) as described in BWRVIP-78 and BWRVIP-86 (References 14 and 15). The NRC approved the ISP for the industry in Reference 16 and approved Dresden's participation in Reference 17. The ISP meets the requirements of 10 CFR 50 Appendix H and provides several advantages over the original program. The surveillance materials in many plant-specific programs do not represent the best match with the limiting vessel beltline materials since some were established prior to 10 CFR 50 Appendix H requirements. Also, the ISP allows for better comparison to unirradiated material data to determine actual shifts in toughness. Finally, for many plants, ISP data will be available sooner to factor into plant operations since there are more sources of data.

Insert B

The current withdrawal schedule for both units is based on the NRC-approved revision of BWRVIP-86 (Reference 15).

Insert C

14. Letter from C. Terry (BWRVIP) to U. S. NRC, "Project No. 704 - BWR Vessel and Internals Project, BWR Integrated Surveillance Program Plan (BWRVIP-78)," dated December 22, 1999

15. Letter from C. Terry (BWRVIP) to U. S. NRC, "Project No. 704 - BWRVIP-86: BWR Vessel and Internals Project, BWR Integrated Surveillance Program Implementation Plan," EPRI Technical Report 1000888, dated December 22, 2000

16. Letter from U. S. NRC to C. Terry (BWRVIP), " Safety Evaluation Regarding EPRI Proprietary Reports 'BWR Vessel And Internals Project, BWR Integrated Surveillance Program Plan (BWRVIP-78)' and 'BWRVIP-86: BWR Vessel And Internals Project, BWR Integrated Surveillance Program Implementation Plan,'" dated February 1, 2002

17. [Insert letter approving Dresden participation in ISP]

ATTACHMENT B-3

Request for License Amendment Regarding Reactor Vessel Specimen Removal Schedule

Marked-up UFSAR Pages
LaSalle County Station , Units 1 and 2

Pages
5.2-33
5.2-34
Table 5.2-12
5.3-1
5.3-2
5.3-8

5.2.3.3.1.6 Temperature Limits for Preoperational System Hydrostatic Tests and ISI Hydrostatic or Leak Pressure Tests

Based on 10 CFR 50 Appendix G IV.A.2.d, which allows a reduced safety factor for tests prior to fuel loading, the preoperational system hydrostatic test at 1563 psig may be performed at a minimum temperature of 118°F (LSCS Unit 1) and 112°F (LSCS Unit 2) which is established by the RT_{NDT} of the non-beltline cylinder plate (LSCS Unit 1) and the beltline plate (LSCS Unit 2) plus 60°F. The fracture toughness analysis for system pressure tests resulted in the curves shown in the LaSalle County Station Units 1 & 2 Technical Specifications and Technical Requirements Manual. The curves are based on an initial beltline RT_{NDT} of 23° F (LSCS Unit 1) or 52°F (LSCS Unit 2). For LSCS Unit 1, the beltline weld material is expected to be more limiting at end-of-service fluence levels, and this weld material has an initial RT_{NDT} of -30°F.

5.2.3.3.1.7 Operating Limits During Heatup, Cooldown and Core Operation

The fracture toughness analysis was done for the normal heatup or cooldown rate of 100°F/hour. The temperature gradients and thermal stress effects corresponding to this rate were included. The results of the analyses are a set of operating limits for non-nuclear heatup or cooldown shown in the LaSalle County Station Units 1 & 2 Technical Specifications and Technical Requirements Manual. Curves labeled C on these figures apply whenever the core is critical. The basis for these Curves is described in References 15, 16, and 17.

5.2.3.3.1.8 Reactor Vessel Annealing

Inplace annealing of the reactor vessel because of radiation embrittlement is unnecessary because the predicted value in transition of adjusted reference temperature will not exceed 200° F (see 10 CFR 50, Appendix G, Paragraph IV.C)

5.2.3.3.1.9 Compliance with 10 CFR 50 Appendix H (Historical information)

Tables 5.2-11a and 5.2-11b for LaSalle 1 and 2 respectively in the FSAR indicated the level of compliance of the material surveillance program to the requirements of 10 CFR 50 Appendix H. The items of non-compliance indicated therein were reworked via an expanded program (1) to provide base-line Charpy V-notch data on unirradiated specimens, and (2) to provide base-line tensile data on unirradiated specimens. Additionally, an upgraded surveillance test program now includes additional test specimens as indicated below:

a. For Charpy V-Notch Testing

Specimen	Basket #1	Basket #2	Basket #3
HAZ	12	8 + (4)	8 + (4)

[No changes on this page. Provided for review only.]

Weld	12	8 + (4)	8 + (4)
Base Metal	12	8 + (4)	8 + (4)
TOTAL	36	36	36

where (4) means the upgraded base-line specimens which were added to constitute the expanded program for LaSalle.

b. For Tensile Testing

Specimen	Basket #1	Basket #2	Basket #3
HAZ	2	2	3
Weld	2	3	2
Base Metal	2	3	3
TOTAL	6	8	8

This expanded program enables the correlation of transverse data to longitudinal data with sufficient specimens in all baskets to more accurately define the upper shelf energy for irradiated conditions.

NRC acceptance of the LSCS material surveillance program pursuant to 10 CFR 50.12, was granted in NUREG 0519 Supplements 1 and 2 for Units 1 and 2 respectively.

5.2.3.3.2 Control of Welding

All welding performed on the RCPB is qualified in accordance with ASME B&PVC Section IX. All welds are inspected by the vendor and surveillance inspected by the owner in accordance with the procedures and the quality assurance program.

5.2.3.3.3 Nondestructive Examination

The LSCS primary pressure boundary equipment was examined nondestructively during manufacture, and following installation a Section XI baseline inspection was accomplished. The RPV's were purchased to Section III Code requirements. They were also subjected to in-process UT examination to Section XI standards. After the vessel was set, the RPV's were subjected to a Section XI ultrasonic baseline inspection. All Code Class 1 equipment also received this baseline inspection.

TABLE 5.2-12

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM-WITHDRAWL SCHEDULE

Unit 1

SPECIMEN HOLDER*	VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME (EFFECTIVE FULL POWER YEARS
Capsule 1	300°	0.98 ~~0.6~~	6
Capsule 2	120°	1.01 ~~0.6~~	~~15~~ 17
Capsule 3	30°	1.01 ~~0.6~~	Spare
Neutron Dosimeter	30°		1st Refueling Outage

Unit 2

SPECIMEN HOLDER*	VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME (EFFECTIVE FULL POWER YEARS
Capsule 1	300°	0.98 ~~0.6~~	6
Capsule 2	120°	0.99 ~~0.6~~	~~15~~ Spare
Capsule 3	30°	0.99 ~~0.6~~	Spare
Neutron Dosimeter	30°		1st Refueling Outage

* Each capsule includes an Fa, WI, and Cu flux wire. The neutron dosimeter contains three Cu and three ~~Fo~~ Fe flux wires.

three Cu, three Fe, and three Ni flux wires.

5.3 REACTOR VESSEL

5.3.1 Reactor Vessel Materials

5.3.1.1 Material Specifications

The materials used in the reactor pressure vessel and appurtenances are shown in Table 5.2-7 together with the applicable specifications.

5.3.1.2 Special Processes Used for Manufacturing and Fabrication

The basic manufacturing and fabrication controls for the LSCS RPV's were based on Section III, with radiography as the in-process and finished product inspection technique. During fabrication the advances in UT testing enabled the utilization of these techniques first as feasibility tests and then as confirmatory inspections on the RPV's.

5.3.1.3 Special Methods for Nondestructive Examination

The LSCS RPV's were fabricated to Section III requirements of the ASME Boiler and Pressure Vessel Code. They were also subjected to Section XI in-process ultrasonic inspection in addition to the standard NDE for Section III. Following completion of fabrication, each vessel was hydrostatically tested and then completely examined ultrasonically prior to shipping. A second hydro test and an ultrasonic baseline inspection per Section XI were performed following vessel set at LSCS.

5.3.1.4 Special Controls for Ferritic and Austenitic Stainless Steel

The specialized quality controls for cleaning, welding, slug control, and weld clad processes were applied for the austenitic stainless steel used in LSCS. Certifications of these controls were the responsibility of the NSSS vendor (GE).

5.3.1.5 Fracture Toughness

LSCS compliance with 10CFR50 Appendices G and H is discussed in Subsection 5.2.3.3.1.

5.3.1.6 Material Surveillance

5.3.1.6.1 Compliance with "Reactor Vessel Material Surveillance Program Requirements"

Insert A

installed original

Charpy impact specimens for the reactor vessel surveillance programs are of the longitudinal orientation consistent with the ASME requirements prior to the issue

of the Summer 1972 Addenda and ASTM E 185-73. Based on General Electric Company (GE) experience, the amount of shift measured by these irradiated longitudinal test specimens will be essentially the same as shift in an equivalent transverse specimen. The material surveillance program for LaSalle reactor pressure vessels includes three sets of specimens in each reactor. The specimens are manufactured from a plate actually used in the beltline region and a weld typical of those in the beltline region and thus represent base metal, weld metal, and the transition zone between base metal and weld. The expanded surveillance program includes: 12 additional samples in baskets No. 2 and 3 to give a total of 36 samples in each of the baskets. The 12 samples include 4 base metal, 4 weld metal, and 4 HAZ material. Additionally, 15 transverse samples are included in the baseline definition of both lower and upper shelf energies. These are to be examined prior to initial criticality. Sufficient tensile and Charpy V-notch specimens are provided in each of the three in-reactor sets and in the out-of- reactor set to measure strength, ductility, and toughness of each of the three materials (base, weld, HAZ), both in the unirradiated and irradiated conditions.

Insert B

5.3.1.6.2 Positioning of Surveillance Capsules

The sealed capsules are not attached to the vessel but are in welded capsule holders. The capsule holders are mechanically retained by capsule holder brackets welded to the vessel cladding. The capsule holder brackets allow the capsule holder to be removed at any desired time in the life of the plant for specimen testing. These brackets are designed, fabricated and analyzed to the requirements of Section III of the ASME Code. The capsule withdrawal schedule is given in Table 5.2-12.

5.3.1.6.3 Time and Number of Dosimetry Measurements

The NSSS vendor provided a separate neutron dosimeter which contains Cu flux wires and Fe flux wires. At the end of the first cycle the dosimeter is removed and a determination is made of the fluence of the vessel inside diameter during the first cycle, by measurements from these wires, to verify the predicted fluence at an early date in plant operation. This measurement is made over this short period to avoid saturation of the dosimeters now available. Once the fluence-to-thermal power output is verified, no further dosimetry is considered necessary because of the linear relationship between fluence and power output.

5.3.1.7 Reactor Vessel Fasteners

The boiling water reactor does not use borated water for reactivity control. Therefore, this subsection is not applicable.

The limit regarding the normal rate of heatup and cooldown (Item a) assures that the vessel closure, closure studs, vessel support skirt, and control rod drive housing and stub tube stresses and usage remain within acceptable limits. The limit regarding a vessel temperature limit on recirculating pump operation and power level increase restriction (Item b) augments the Item a limit in further detail by assuring that the vessel bottom head region will not be warmed at an excessive rate caused by rapid sweepout of cold coolant in the vessel lower head region by recirculating pump operation or natural circulation (cold coolant can accumulate as a result of control drive inleakage and/or low recirculation flow rate during startup or hot standby). The Item c limit further restricts operation of the recirculating pumps to avoid high thermal stress effects in the pumps and piping, while also minimizing thermal stresses on the vessel nozzles.

5.3.4 References

Insert C

LSCS UFSAR Inserts

Insert A

A materials surveillance program was developed and initiated at the beginning of nuclear operation of LSCS. This surveillance program was designed to be in conformance with the requirements of 10 CFR 50 Appendix H, as discussed in UFSAR Section 5.2.3.3.1.9.

Insert B

In 2003, the NRC approved LSCS participation in the BWR Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) as described in BWRVIP-78 and BWRVIP-86 (References 1 and 2). The NRC approved the ISP for the industry in Reference 3 and approved LSCS participation in Reference 4. The ISP meets the requirements of 10 CFR 50 Appendix H and provides several advantages over the original program. The surveillance materials in many plant-specific programs do not represent the best match with the limiting vessel beltline materials since some were established prior to 10 CFR 50 Appendix H requirements. Also, the ISP allows for better comparison to unirradiated material data to determine actual shifts in toughness. Finally, for many plants, ISP data will be available sooner to factor into plant operations since there are more sources of data.

The current withdrawal schedule for both units is given in UFSAR Table 5.2-12 and is based on the NRC-approved revision of BWRVIP-86 (Reference 2).

Insert C

1. Letter from C. Terry (BWRVIP) to U. S. NRC, "Project No. 704 - BWR Vessel and Internals Project, BWR Integrated Surveillance Program Plan (BWRVIP-78)," dated December 22, 1999

2. Letter from C. Terry (BWRVIP) to U. S. NRC, "Project No. 704 - BWRVIP-86: BWR Vessel and Internals Project, BWR Integrated Surveillance Program Implementation Plan," EPRI Technical Report 1000888, dated December 22, 2000

3. Letter from U. S. NRC to C. Terry (BWRVIP), " Safety Evaluation Regarding EPRI Proprietary Reports 'BWR Vessel And Internals Project, BWR Integrated Surveillance Program Plan (BWRVIP-78)' and 'BWRVIP-86: BWR Vessel And Internals Project, BWR Integrated Surveillance Program Implementation Plan,'" dated February 1, 2002

4. [Insert letter approving LSCS participation in ISP]

ATTACHMENT B-4

Request for License Amendment Regarding Reactor Vessel Specimen Removal Schedule

Marked-up UFSAR and TS/Bases Pages Limerick Generating Station, Units 1 and 2

<u>TS Pages</u>
xi (Unit 1)
3/4 4-19 (Unit 1)
3/4 4-21 (Unit 1)
xi (Unit 2)
3/4 4-19 (Unit 2)
3/4 4-21 (Unit 2)

<u>TS Bases Pages</u>
B 3/4 4-5 (Unit 1)
B 3/4 10-2 (Unit 1)
B 3/4 4-5 (Unit 2)
B 3/4 10-2 (Unit 2)

<u>UFSAR Pages</u>
5.3-7
5.3-8
5.3-9
5.3-25
Table 5.3-2 (Unit 1)
Table 5.3-2 (Unit 2)

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION		PAGE
REACTOR COOLANT SYSTEM (Continued)		
	Figure 3.4.1.1-1 Thermal Power versus Core Flow	3/4 4-3
	Jet Pumps	3/4 4-4
	Recirculation Pumps	3/4 4-5
	Idle Recirculation Loop Startup	3/4 4-6
3/4.4.2	SAFETY/RELIEF VALVES	3/4 4-7
3/4 4.3	REACTOR COOLANT SYSTEM LEAKAGE	
	Leakage Detection Systems	3/4 4-8
	Operational Leakage	3/4 4-9
	Table 3.4.3.2-1 Reactor Coolant System Pressure Isolation Valves	3/4 4-11
3/4.4.4	CHEMISTRY	3/4 4-12
	Table 3.4.4-1 Reactor Coolant System Chemistry Limits	3/4 4-14
3/4.4.5	SPECIFIC ACTIVITY	3/4 4-15
	Table 4.4.5-1 Primary Coolant Specific Activity Sample and Analysis Program	3/4 4-17
3/4.4.6	PRESSURE/TEMPERATURE LIMITS	
	Reactor Coolant System	3/4 4-18
	Figure 3.4.6.1-1 Minimum Reactor Pressure Vessel Metal Temperature Vs. Reactor Vessel Pressure	3/4 4-20
	Table 4.4.6.1.3-1 Reactor Vessel Material Surveillance Program - Withdrawal Schedule	3/4 4-21
	Reactor Steam Dome	3/4 4-22
3/4.4.7	MAIN STEAM LINE ISOLATION VALVES	3/4 4-23
3/4.4.8	STRUCTURAL INTEGRITY	3/4 4-24

Deleted

SURVEILLANCE REQUIREMENTS (Continued)

4.4.6.1.2 The reactor coolant system temperature and pressure shall be determined to be to the right of the criticality limit line of Figure 3.4.6.1-1 curve C within 15 minutes prior to the withdrawal of control rods to bring the reactor to criticality and at least once per 30 minutes during system heatup.

DELETED

4.4.6.1.3 The reactor vessel material surveillance specimens shall be removed and examined, to determine changes in reactor pressure vessel material properties, as required by 10 CFR Part 50, Appendix H in accordance with the schedule in Table 4.4.6.1.3-1. The results of these examinations shall be used to update the curves of Figure 3.4.6.1-1.

4.4.6.1.4 The reactor flux wire specimens located within the surveillance capsules shall be removed and examined to determine reactor pressure vessel fluence as a function of time and power level and used to modify Figure B 3/4 4.6-1 in accordance with the schedule in Table 4.4.6.1.3-1. The results of these fluence determinations shall be used to adjust the curves of Figure 3.4.6.1-1, as required.

4.4.6.1.5 The reactor vessel flange and head flange temperature shall be verified to be greater than or equal to 80°F:

DELETED

a. In OPERATIONAL CONDITION 4 when reactor coolant system temperature is:

   1. ≤ 100°F, at least once per 12 hours.

   2. ≤ 90°F, at least once per 30 minutes.

b. Within 30 minutes prior to and at least once per 30 minutes during tensioning of the reactor vessel head bolting studs.

MAY 1 1 1998

TABLE 4.4.6.1.3-1

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM-WITHDRAWAL SCHEDULE

CAPSULE NUMBER	VESSEL LOCATION	LEAD FACTOR*	WITHDRAWAL TIME** (EFPY)
117C 4944 G004	30°	1.20	15
117C 4944 G001	120°	1.20	30
117C 4944 G001	300°	1.20	Spare

*At 1/4 T.

**If the designated withdrawal time (EFPY) is reached during an operating cycle, withdrawal of the capsule may be deferred until the next scheduled refueling outage.

INFORMATION CONTAINED ON THIS PAGE HAS BEEN DELETED

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION		PAGE
REACTOR COOLANT SYSTEM (Continued)		
	Figure 3.4.1.1-1 Thermal Power versus Core Flow...	3/4 4-3
	Jet Pumps.....	3/4 4-4
	Recirculation Pumps.....	3/4 4-5
	Idle Recirculation Loop Startup.....	3/4 4-6
3/4.4.2	SAFETY/RELIEF VALVES.....	3/4 4-7
3/4.4.3	REACTOR COOLANT SYSTEM LEAKAGE	
	Leakage Detection Systems.....	3/4 4-8
	Operational Leakage.....	3/4 4-9
	Table 3.4.3.2-1 Reactor Coolant System Pressure Isolation Valves.....	3/4 4-11
3/4.4.4	CHEMISTRY.....	3/4 4-12
	Table 3.4.4-1 Reactor Coolant System Chemistry Limits.....	3/4 4-14
3/4.4.5	SPECIFIC ACTIVITY.....	3/4 4-15
	Table 4.4.5-1 Primary Coolant Specific Activity Sample and Analysis Program.....	3/4 4-17
3/4.4.6	PRESSURE/TEMPERATURE LIMITS	
	Reactor Coolant System.....	3/4 4-18
	Figure 3,4.6.1-1 Minimum Reactor Pressure Vessel Metal Temperature Vs. Reactor Vessel Pressure.....	3/4 4-20
	Table 4.4.6.1.3-1 Reactor Vessel Material Surveillance Program - Withdrawal Schedule.....	3/4 4-21
	Reactor Steam Dome.....	3/4 4-22
3/4.4.7	MAIN STEAM LINE ISOLATION VALVES.....	3/4 4-23
3/4.4.8	STRUCTURAL INTEGRITY.....	3/4 4-24

Deleted

AUG 2 5 1989

SURVEILLANCE REQUIREMENTS (Continued)

4.4.6.1.2 The reactor coolant system temperature and pressure shall be determined to be to the right of the criticality limit line of Figure 3.4.6.1-1 curve C within 15 minutes prior to the withdrawal of control rods to bring the reactor to criticality and at least once per 30 minutes during system heatup.

DELETED

4.4.6.1.3 The reactor vessel material surveillance specimens shall be removed and examined, to determine changes in reactor pressure vessel material properties, as required by 10 CFR Part 50, Appendix H in accordance with the schedule in Table 4.4.6.1.3-1. The results of these examinations shall be used to update the curves of Figure 3.4.6.1-1.

4.4.6.1.4 The reactor flux wire specimens located within the surveillance capsules shall be removed and examined to determine reactor pressure vessel fluence as a function of time and power level and used to modify Figure B 3/4 4.6-1 in accordance with the schedule in Table 4.4.6.1.3-1. The results of these fluence determinations shall be used to adjust the curves of Figure 3.4.6.1-1, as required.

4.4.6.1.5 The reactor vessel flange and head flange temperature shall be verified to be greater than or equal to 70°F:

a. In OPERATIONAL CONDITION 4 when reactor coolant system temperature is:

DELETED

1. ≤ 100°F, at least once per 12 hours.

2. ≤ 90°F, at least once per 30 minutes.

b. Within 30 minutes prior to and at least once per 30 minutes during tensioning of the reactor vessel head bolting studs.

TABLE 4.4.6.1.3-1

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM-WITHDRAWAL SCHEDULE

CAPSULE NUMBER	VESSEL LOCATION	LEAD FACTOR*	WITHDRAWAL TIME (EFPY)
131C 7717 G003	30°	1.20	15
131C 7717 G002	120°	1.20	30
131C 7717 G001	300°	1.20	Spare

*At 1/4 T.

INFORMATION CONTAINED ON THIS PAGE HAS BEEN DELETED

PRESSURE/TEMPERATURE LIMITS (Continued)

The operating limit curves of Figure 3.4.6.1-1 are derived from the fracture toughness requirements of 10 CFR 50 Appendix G and ASME Code Section XI, Appendix G. The curves are based on the RT_{NDT} and stress intensity factor information for the reactor vessel components. Fracture toughness limits and the basis for compliance are more fully discussed in FSAR Chapter 5, Paragraph 5.3.1.5, "Fracture Toughness."

The reactor vessel materials have been tested to determine their initial RT_{NDT}. The results of these tests are shown in Table B 3/4.4.6-1. Reactor operation and resultant fast neutron, E greater than 1 MeV, irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, nickel content and copper content of the material in question, can be predicted using Bases Figure B 3/4.4.6-1 and the recommendations of Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." The pressure/temperature limit curves, Figure 3.4.6.1-1, include a shift in RT_{NDT} for conditions at 32 EFPY. The A, B and C limit curves are predicted to be bounding for all areas of the RPV until 32 EFPY. In addition, an intermediate A curve has been provided for 22 EFPY.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with 10 CFR Part 50, Appendix H, irradiated reactor vessel flux wire and Charpy specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the flux wires, Charpy specimens and vessel inside radius are essentially identical, the irradiated Charpy specimens can be used with confidence in predicting reactor vessel material transition temperature shift. The operating limit curves of Figure 3.4.6.1-1 shall be adjusted, as required, on the basis of the flux wire and Charpy specimen data and recommendations of Regulatory Guide 1.99, Revision 2.

The pressure-temperature limit lines shown in Figures 3.4.6.1-1, curves C, and A, for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR Part 50 for reactor criticality and for inservice leak and hydrostatic testing.

The number of reactor vessel irradiation surveillance capsules and the frequencies for removing and testing the specimens in these capsules are provided in Table 4.4.6.1.3-1 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

SEP 15 2000

3/4.10.8 INSERVICE LEAK AND HYDROSTATIC TESTING

This special test exception permits certain reactor coolant pressure tests to be performed in OPERATIONAL CONDITION 4 when the metallurgical characteristics of the reactor pressure vessel (RPV) or plant temperature control capabilities during these tests require the pressure testing at temperatures greater than 200°F and less than or equal to 212°F (normally corresponding to OPERATIONAL CONDITION 3). The additionally imposed OPERATIONAL CONDITION 3 requirements for SECONDARY CONTAINMENT INTEGRITY provide conservatism in response to an operational event.

Invoking the requirement for Refueling Area Secondary Containment Integrity along with the requirement for Reactor Enclosure Secondary Containment Integrity applies the requirements for Reactor Enclosure Secondary Containment Integrity to an extended area encompassing Zones 1 and 3. Operations with the Potential for Draining the Vessel, Core alterations, and fuel handling are prohibited in this secondary containment configuration. Drawdown and inleakage testing performed for the combined zone system alignment shall be considered adequate to demonstrate integrity of the combined zones.

Inservice hydrostatic testing and inservice leak pressure tests required by Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code are performed prior to the reactor going critical after a refueling outage. The minimum temperatures (at the required pressures) allowed for these tests are determined from the RPV pressure and temperature (P/T) limits required by LCO 3.4.6, Reactor Coolant System Pressure/Temperature Limits. These limits are conservatively based on the fracture toughness of the reactor vessel, taking into account anticipated vessel neutron fluence. With increased reactor fluence over time, the minimum allowable vessel temperature increases at a given pressure. Periodic updates to the RCS P/T limit curves are performed as necessary, based upon the results of analysis of irradiated surveillance specimens removed from the vessel.

MAY 0 6 1999

PRESSURE/TEMPERATURE LIMITS (Continued)

The operating limit curves of Figure 3.4.6.1-1 are derived from the fracture toughness requirements of 10 CFR 50 Appendix G and ASME Code Section XI, Appendix G. The curves are based on the RT_{NDT} and stress intensity factor information for the reactor vessel components. Fracture toughness limits and the basis for compliance are more fully discussed in FSAR Chapter 5, Paragraph 5.3.1.5, "Fracture Toughness."

The reactor vessel materials have been tested to determine their initial RT_{NDT}. The results of these tests are shown in Table B 3/4.4.6-1. Reactor operation and resultant fast neutron, E greater than 1 MeV, irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, nickel content and copper content of the material in question, can be predicted using Bases Figure B 3/4.4.6-1 and the recommendations of Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." The pressure/temperature limit curve, Figure 3.4.6.1-1, curves A, B and C, includes an assumed shift in RT_{NDT} for the conditions at 32 EFPY. In addition, an intermediate A curve has been provided for 22 EFPY. The A, B and C limit curves are predicted to be bounding for all areas of the RPV until 32 EFPY.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with 10 CFR Part 50, Appendix H, irradiated reactor vessel flux wire and charpy specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the charpy specimens and vessel inside radius are essentially identical, the irradiated charpy specimens can be used with confidence in predicting reactor vessel material transition temperature shift. The operating limit curves of Figure 3.4.6.1-1 shall be adjusted, as required, on the basis of the flux wire and charpy specimen data and recommendations of Regulatory Guide 1.99, Revision 2.

The pressure-temperature limit lines shown in Figures 3.4.6.1-1, curves C, and A, for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR Part 50 for reactor criticality and for inservice leak and hydrostatic testing.

The number of reactor vessel irradiation surveillance capsules and the frequencies for removing and testing the specimens in these capsules are provided in Table 4.4.6.1.3-1 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

3/4.10.8 INSERVICE LEAK AND HYDROSTATIC TESTING

This special test exception permits certain reactor coolant pressure tests to be performed in OPERATIONAL CONDITION 4 when the metallurgical characteristics of the reactor pressure vessel (RPV) or plant temperature control capabilities during these tests require the pressure testing at temperatures greater than 200°F and less than or equal to 212°F (normally corresponding to OPERATIONAL CONDITION 3). The additionally imposed OPERATIONAL CONDITION 3 requirements for SECONDARY CONTAINMENT INTEGRITY provide conservatism in response to an operational event.

Invoking the requirement for Refueling Area Secondary Containment Integrity along with the requirement for Reactor Enclosure Secondary Containment Integrity applies the requirements for Reactor Enclosure Secondary Containment Integrity to an extended area encompassing Zones 2 and 3. Operations with the Potential for Draining the Vessel, Core alterations, and fuel handling are prohibited in this secondary containment configuration. Drawdown and inleakage testing performed for the combined zone system alignment shall be considered adequate to demonstrate integrity of the combined zones.

Inservice hydrostatic testing and inservice leak pressure tests required by Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code are performed prior to the reactor going critical after a refueling outage. The minimum temperatures (at the required pressures) allowed for these tests are determined from the RPV pressure and temperature (P/T) limits required by LCO 3.4.6, Reactor Coolant System Pressure/Temperature Limits. These limits are conservatively based on the fracture toughness of the reactor vessel, taking into account anticipated vessel neutron fluence. With increased reactor fluence over time, the minimum allowable vessel temperature increases at a given pressure. Periodic updates to the RCS P/T limit curves are performed as necessary, based upon the results of analysis of irradiated surveillance specimens removed from the vessel.

MAY 0 6 1999

~~ECR 99-00864~~

10CFR50, Appendix G includes the margin of safety implicit in the Appendix G requirement. The adjustment is made by increasing the minimum temperatures required by the difference between LGS and BWR/6 feedwater nozzle forging RT_{NDT}s. The adjustment is based on an RT_{NDT} of 40°F for Unit 1 and an RT_{NDT} of 40°F for Unit 2.

The reactor vessel closure studs have a minimum Charpy impact energy of 48 ft-lb and a 27 MLE at 10°F for LGS Unit 1. The studs for LGS Unit 2 have a minimum Charpy impact energy of 25 MLE and 46 ft-lb at 10°F. The lowest service temperature for bolt-up of LGS Unit 2 is 10°F. Charpy test results are discussed in Sections 5.3.1.7 and 5.3.1.8.

5.3.1.6 Material Surveillance

5.3.1.6.1 Compliance with "Reactor Vessel Material Surveillance Program Requirements"

The materials surveillance program monitors changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region resulting from their exposure to neutron irradiation and thermal environment.

Materials for the original surveillance program are selected were to represent materials used in the reactor beltline region. The specimens are manufactured from a plate actually used in the beltline region, and a weld typical of those in the beltline region, and thus represent base metal, weld metal, and the transition zone between base metal and weld. The plate and weld are heat treated in a manner that simulates the actual heat treatment performed on the core region shell plates of the completed vessel.

Details of the vessel surveillance program are provided in Section 5.3.1.10.

The original surveillance program includes three capsule holders per reactor vessel.

Information on the specimen arrangement is given in Table 5.3-12, referenced in Section 5.3.1.10.

A set of out-of-reactor baseline CVN specimens is provided with the surveillance test specimens.

Charpy impact specimens for the original reactor vessel surveillance programs are of the longitudinal orientation consistent with the ASME requirements prior to the issue of the 1972 Addenda and ASTM E185-73. Based on GE experience, the amount of shift measured by these irradiated longitudinal test specimens is essentially the same as the shift in an equivalent transverse specimen.

For LGS Units 1 and 2, each set of surveillance specimens is loaded in 6 small capsules rather than one large capsule. Therefore, each capsule holder which contains all 6 small capsules can be considered to be the same as one surveillance capsule as defined in 10CFR50, Appendix H. Three capsule holders are included in each reactor vessel. Since the predicted adjusted increase in reference temperature of the beltline region, estimated at the time of design, was less than 100°F at EOL and the calculated peak neutron fluence is less than $5x10^{18}$ n/cm^2, the use of three capsule holders meets the requirements of 10CFR50, Appendix H, and ASTM E185-73.

~~For each capsule withdrawal, the test procedures and reporting requirements must meet the requirements of ASTM E-185-82 to the extent practical for the configuration of the specimens in the capsule.~~

~~The withdrawal schedule of the three sets of specimens in the reactor is planned as follows:~~

~~a. 15 EFPY (Ref. 5.3-8 and 5.3-9)~~

~~b. 30 EFPY (Ref. 5.3-8 and 5.3-9)~~

~~c. The third set is a spare to be withdrawn based on previously developed data.~~

For the extent of compliance of the original surveillance program to 10CFR50, Appendix H, see Table 5.3-2.

Insert A

5.3.1.6.2 Neutron Flux and Fluence Calculations

A description of the methods of analysis is contained in Sections 4.1.4.5 and 4.3.2.8.

5.3.1.6.3 Predicted Irradiation Effects on Vessel Beltline Materials

Estimated maximum changes in RT_{NDT} (initial reference temperature) and upper shelf fracture energy as a function of the EOL fluence at the ¼ depth of the vessel beltline materials are provided in Section 5.3.1.7. The predicted peak EOL maximum neutron irradiation fluences for the 110% power rerate condition at the ¼ of the vessel beltline are $1.3x10^{18}$ n/cm^2 after 32 EFPY (where an EFPY is based on the rerated power level). For conservative flux calculations, 251 inches is used as the inside diameter of the beltline region with a wall thickness of 6-3/16 inches. Transition temperature changes and variations in upper-shelf energy were calculated in accordance with the rules of Regulatory Guide 1.99 (Rev 2). Reference temperatures were established in accordance with 10CFR50, Appendix G and NB-2330 of the ASME Code.

5.3.1.6.4 Positioning of Surveillance Capsules and Method of Attachment

Surveillance specimen capsules are located at three azimuths at a common elevation in the core beltline region. The sealed capsules are not attached to the vessel but are in welded capsule holders. The capsule holders are mechanically retained by capsule holder brackets welded to the vessel cladding as shown in Figure 5.3-3. The capsule holder brackets allow the capsule holder to be removed at any desired time in the life of the plant for specimen testing. These brackets are designed, fabricated, and analyzed to the requirements of ASME Section III. A positive spring-loaded locking device is provided to retain the capsules in position throughout any anticipated event during the lifetime of the vessel.

5.3.1.6.5 ~~Time and Number of~~ Dosimetry Measurements

Each surveillance capsule contains iron and copper flux wires. When the first capsule is removed, after about 15 EFPY of reactor operations, these wires can be used to determine the relationship between reactor power and neutron fluence.

5.3.1.7 Vessel Beltline Plates and Welds

This section supplements Section 5.3.1.5 in discussing the compliance to the intent of 10CFR50, Appendix G.

5.3.1.7.1 Test Data

Available Charpy V-notch and drop-weight impact test data are presented in Tables 5.3-3 and 5.3-4. The sample test welds are prepared in accordance with the ASME Code and do not include base material from the beltline. There are two categories of belt-line welds identified "shop" welds and "field" welds. The shop welds represent vessel vertical seams which were made prior to shipment of preassembled ring segments to the LGS Unit 1 and Unit 2 plant site. The flux material for the submerged arc weld is LINDE 124. The field welds (i.e., girth welds) were made at the plant site. However, exact identification of weld materials used in the beltline girth weld seam is not available. Therefore, a conservative assumption is made to consider all electrodes which were released for field welding the vessel shells.

Inservice inspection and testing of the RCPB is discussed in detail in Section 5.2.4.

5.3.4 REFERENCES

5.3-1 "Metal Progress", pp. 35-39, (July 1978).

5.3-2 "Radiation Effects in BWR Pressure Vessel Steels", GE Licensing Topical Report, NEDO-21708.

5.3-3 Letter MFN-414-77, G.G. Sherwood (GE) to Edson G. Case (NRC), (October 17, 1977).

5.3-4 Letter, Robert B. Minogue (NRC) to G.G. Sherwood (GE), (February 14, 1978).

5.3-5 NEDO-21778A, Transient Pressure Rise Affecting Fracture Toughness for BWR's, GE Licensing Topical Report (December 1978).

5.3-6 NEDO-32205-A, Revision 1, 10CFR50 Appendix G Equivalent Margin Analysis for Low Upper Shelf Energy in BWR/2 Through BWR/6 Vessels.

5.3-7 "Limerick Generating Station Unit 2 Intermediate Pressure Temperature Curves Considering Power Rerate", GENE-B11-00696, June 1996.

5.3-8 General Electric Co. Report GE-NE-B1100786-01R1, "Surveillance Specimen Program Evaluation for Limerick Generating Station Unit 1", dated December 1997.

5.3-9 General Electric Co. Report GE-NE-B1100786-02, "Surveillance Specimen Program Evaluation for Limerick Generating Station Unit 2", dated June 1998.

5.3-10 "Limerick Generating Station, Units 1 and 2, SRV Setpoint Tolerance Relaxation Licensing Report", NEDC-32645P (December 1998).

Insert B

Table 5.3-2

APPENDIX H MATRIX FOR LGS

(UNIT 1)

APPENDIX H PARA. NO.	TOPIC	COMPLY YES/NO OR N/A	ALTERNATE ACTIONS OR COMMENTS
I	Introduction	NA	
II.A	Fluence $<10^{17}$ n/cm^2 - surveillance program not required	NA	
II.B	Standards requirements (ASTM) for surveillance	No	Noncompliance with ASTM E185-73 in that the surveillance specimens are not necessarily from the limiting beltline material. Specimens are from representative beltline material, however, and can be used to predict behavior of the limiting material. Heat and heat/lot numbers for surveillance specimens are to be supplied.
II.C.1	Surveillance specimen is taken from locations alongside the fracture test specimens (Section III.B of Appendix G)	No	Noncompliance in that specimens are not necessarily taken from alongside specimens required by Section III of Appendix G and transverse CVNs are employed. However, representative materials are used, and RT_{NDT} shift appears to be independent of specimen orientation.
II.C.2	Locations of surveillance capsules in RPV	Yes	Code basis is used for the attachment of brackets to vessel cladding (Section 5.3.1.6.4).
II.C.3.a	Withdrawal schedule of capsules, $RT_{NDT} \leq 100°F$	Yes	~~Three capsules planned. Starting RT_{NDT} of limiting material is based on alternative action (see paragraph III.A of Appendix G).~~ (See Section 5.3.1.6.1)
II.C.3.b	Withdrawal schedule of capsules, $100°F < RT_{NDT} \leq 200°F$	NA	
II.C.3.c	Withdrawal schedule of capsules, $RT_{NDT} > 200°F$	NA	
III.A	Fracture toughness testing requirements of specimens	No	CVN tests only

Table 5.3-2 (Cont'd)

APPENDIX H MATRIX FOR LGS

(UNIT 2)

APPENDIX H PARA. NO.	TOPIC	COMPLY YES/NO OR N/A	ALTERNATE ACTIONS OR COMMENTS
I	Introduction	NA	
II.A	Fluence $<10^{17}$ N/cm^2 - surveillance program not required	NA	
II.B	Standards requirements (ASTM) for surveillance	No	Noncompliance with ASTM E185-73 in that the surveillance specimens are not necessarily from the limiting beltline material. Specimens are from representative beltline material, however, and can be used to predict behavior of the limiting material. Heat and heat/lot numbers for surveillance specimens are supplied (Section 5.3.1.10).
II.C.1	Surveillance specimens are taken from locations alongside the fracture test specimens (Section III.B of Appendix G)	No	Noncompliance in that specimens are not necessarily taken from alongside specimens required by Section III of Appendix G and transverse CVNs are not employed. However, representative materials are used, and RT_{NDT} shift appears to be independent of specimen orientation.
II.C.2	Locations of surveillance capsules in RPV	Yes	Code basis is used for the attachment of brackets to vessel cladding (Section 5.3.1.6.4).
II.C.3.a	Withdrawal schedule of capsules, $RT_{NDT} \leq 100°F$	Yes	Three capsules planned. Starting RT_{NDT} of limiting material is based on alternative action (see paragraph III.A of Appendix G). (See Section 5.3.1.6.1)
II.C.3.b	Withdrawal schedule of capsules, $100°F < RT_{NDT} \leq 200°F$	No	Material with RT_{NDT} shift >100°F is not limiting material; shift was predicted <100°F at time of surveillance program design.
II.C.3.c	Withdrawal schedule of capsules, $RT_{NDT} > 200°F$	NA	
III.A	Fracture toughness testing requirements of specimens	No	CVN tests only

LGS UFSAR Inserts

Insert A

In 2003, the NRC approved LGS participation in the BWR Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) as described in BWRVIP-78 and BWRVIP-86 (References 5.3-11 and 5.3-12). The NRC approved the ISP for the industry in Reference 5.3-13 and approved LGS participation in Reference 5.3-14. The ISP meets the requirements of 10 CFR 50 Appendix H and provides several advantages over the original program. The surveillance materials in many plant-specific programs do not represent the best match with the limiting vessel beltline materials since some were established prior to 10 CFR 50 Appendix H requirements. Also, the ISP allows for better comparison to unirradiated material data to determine actual shifts in toughness. Finally, for many plants, ISP data will be available sooner to factor into plant operations since there are more sources of data.

The current withdrawal schedule for both units is based on the latest NRC-approved revision of BWRVIP-86 (Reference 5.3-12). Based on this schedule, LGS is not scheduled to withdraw any additional material specimens.

Insert B

5.3-11 Letter from C. Terry (BWRVIP) to U. S. NRC, "Project No. 704 - BWR Vessel and Internals Project, BWR Integrated Surveillance Program Plan (BWRVIP-78)," dated December 22, 1999

5.3-12 Letter from C. Terry (BWRVIP) to U. S. NRC, "Project No. 704 - BWRVIP-86: BWR Vessel and Internals Project, BWR Integrated Surveillance Program Implementation Plan," EPRI Technical Report 1000888, dated December 22, 2000

5.3.13 Letter from U. S. NRC to C. Terry (BWRVIP), " Safety Evaluation Regarding EPRI Proprietary Reports 'BWR Vessel And Internals Project, BWR Integrated Surveillance Program Plan (BWRVIP-78)' and 'BWRVIP-86: BWR Vessel And Internals Project, BWR Integrated Surveillance Program Implementation Plan,'" dated February 1, 2002

5.3-14 [Insert letter approving LGS participation in ISP]

ATTACHMENT B-5

Request for License Amendment Regarding Reactor Vessel Specimen Removal Schedule

Marked-up UFSAR Pages
Oyster Creek Generating Station

Pages
5.3-2
5.3-3
5.3-6

Thermal stresses occur in a vessel when two segments or areas are at different temperatures. The thermal stresses and strains on the reactor vessel or reactor system which can result from system operation are limited in order to prevent fatigue or distortion of the vessel. The thermal strains are controlled by limitations on allowed heatup and cooldown rates, and limitations on cold inlet water temperatures (and thermal sleeves). The expected thermal strains on the vessel are analyzed and included in the fatigue analysis. It was concluded from the fatigue analysis that the vessel could withstand thermal strains beyond those anticipated during normal operation. In fact, conditions well beyond the design limits would be required to actually produce vessel failure. (Refer to Section 5.2 for anticipated operational cycles).

A discussion of conformance with pertinent Regulatory Guides is presented in Section 1.8.

5.3.1.2 Fracture Toughness

To ensure that ferritic components of pressure retaining components of the Reactor Coolant Pressure Boundary exhibit adequate fracture toughness under service hydrostatic tests and during heatup and cooldown, temperature and pressure limitations are established in the Technical Specifications (refer to Subsection 5.3.2). Beltline materials for the vessel were ordered with a reference nil ductility transition temperature (RT_{NDT}) of +10F or less.

5.3.1.3 Material Surveillance

The General Electric Company developed and provided for an irradiation surveillance program for the Oyster Creek reactor vessel. A series of mechanical test specimens from the base metal of the reactor vessel and from weld heat affected zone metal, and weld metal taken from a weld joint made from the reactor steel and simulating a welded joint in the reactor vessel were selected for the program at Oyster Creek. Specimens were placed in the reactor vessel close to the vessel wall to be exposed to conditions similar to that of the vessel wall. Both tensile specimens and Charpy specimens have been provided for all the above materials and locations. Various wires and materials are used to measure the integrated flux to which the specimens have been exposed. Selected groups of the specimens in the vessel are removed at recommended intervals over the life of the reactor and are tested to determine changes in mechanical properties, and effects on operational parameters (see Subsection 5.3.2).

The following reactor pressure vessel steel surveillance specimens from the actual vessel were placed in the Oyster Creek reactor vessel: Twelve impact specimens and one specimen each of iron, nickel and copper flux wire were placed in each impact capsule.

[No changes on this page. Included for information only.]

Two tensile specimens were placed in each tensile capsule. Groups of these capsules are held in baskets. Three of these baskets are hung on the wall of the vessel at the core midplane, spaced 120 degrees apart. The number of specimens is detailed in Table 5.3-1.

Material for base metal specimens has been taken from a plate used in vessel beltline regions or from a plate of the same heat of material. The same plate used for base metal specimens is used for production of heat affected zone specimens, and the weld specimens are produced by the identical weld practice and procedures used in the vessel fabrication. Thus, the surveillance specimens do represent the materials and processing of the vessel beltline region.

The steps taken during the production of BWR pressure vessel surveillance specimens assure reasonable representation of the vessel material. Any variations in irradiation behavior between the surveillance materials and additional heats of vessel materials is expected to be minimal.

Insert A

5.3.2 Pressure - Temperature Limits

The requirements of 10CFR50, Appendix G, "Fracture Toughness Requirements," establish that pressure temperature limits be established for Reactor Coolant System heatup and cooldown operations, inservice leak and hydrostatic tests, and reactor core operation. These limits are required to ensure that the stresses in the reactor vessel remain within acceptable limits, and to provide adequate margin against brittle fracture. They are intended to provide adequate margins of safety during any condition of normal operation, including anticipated operational transients.

The pressure temperature limits depend upon the metallurgical properties of the reactor vessel materials. The properties of materials in the vessel beltline region vary over the lifetime of the vessel because of the effects of neutron irradiation. One principal effect of the neutron irradiation is that it causes RT_{NDT} to increase with time. The pressure-temperature operating limits must be modified periodically to account for this radiation induced increase in RT_{NDT} by increasing the temperature required for a given pressure. The operating limits for a particular operating period are based on the material properties at the end of the operating period. By periodically revising the pressure temperature limits to account for radiation damage, the stresses and stress intensities in the reactor vessel can be held within acceptable limits. At the beginning of life, material other than that in the beltline region may be the limiting material because it is subjected to high stresses and stress intensities. However, since material outside the beltline region is not subjected to high level irradiation, its RT_{NDT} will not change as the beltline region will and at some period of life, the beltline materials will become limiting.

5.3.4 References

(1) Oyster Creek Nuclear Power Plant, FDSAR, Amendment 16, Reactor Pressure Vessel Design Report, September 1967.

(2) Oyster Creek Nuclear Power Plant, FDSAR, Amendment 29, Status Report on Reactor Vessel Repair Program, December 1967.

(3) Oyster Creek Nuclear Power Plant, FDSAR Amendment 35, Final Report on Reactor Vessel Repair Program, March 1968.

(4) Oyster Creek Nuclear Power Plant, FDSAR Amendment 36, Reactor Vessel Repair Program, March 1968.

(5) Oyster Creek Nuclear Power Plant, FDSAR Amendment 37, Reactor Vessel Repair Program Additional Information, April 1968.

(6) Oyster Creek Nuclear Power Plant, FDSAR Amendment 40, Report on Reactor Vessel Repair Program, August 1968.

(7) Oyster Creek Nuclear Power Plant, FDSAR Amendment 43, Supplemental Report Reactor Vessel Repair Program, October 1968.

(8) Oyster Creek Nuclear Power Plant, FDSAR Amendment 47, Post Hydro Examination, October 1968.

Insert B

Oyster Creek UFSAR Inserts

Insert A

In 2003, the NRC approved Oyster Creek's participation in the BWR Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) as described in BWRVIP-78 and BWRVIP-86 in References 9 and 10. The NRC approved the ISP for the industry in Reference 11 and approved Oyster Creek participation in Reference 12. The ISP meets the requirements of 10 CFR 50 Appendix H and provides several advantages over the original program. The surveillance materials in many plant-specific programs do not represent the best match with the limiting vessel beltline materials since some were established prior to 10 CFR 50 Appendix H requirements. Also, the ISP allows for better comparison to unirradiated material data to determine actual shifts in toughness. Finally, for many plants, ISP data will be available sooner to factor into plant operations since there are more sources of data.

The current withdrawal schedule is based on the latest NRC-approved revision of BWRVIP-86 (Reference 10). Based on this schedule, Oyster Creek is not scheduled to withdraw an additional material specimen.

Insert B

9. Letter from C. Terry (BWRVIP) to U. S. NRC, "Project No. 704 - BWR Vessel and Internals Project, BWR Integrated Surveillance Program Plan (BWRVIP-78)," dated December 22, 1999

10. Letter from C. Terry (BWRVIP) to U. S. NRC, "Project No. 704 - BWRVIP-86: BWR Vessel and Internals Project, BWR Integrated Surveillance Program Implementation Plan," EPRI Technical Report 1000888, dated December 22, 2000

11. Letter from U. S. NRC to C. Terry (BWRVIP), " Safety Evaluation Regarding EPRI Proprietary Reports 'BWR Vessel And Internals Project, BWR Integrated Surveillance Program Plan (BWRVIP-78)' and 'BWRVIP-86: BWR Vessel And Internals Project, BWR Integrated Surveillance Program Implementation Plan,'" dated February 1, 2002

12. [Insert letter approving Oyster Creek participation in ISP]

ATTACHMENT B-6

Request for License Amendment Regarding Reactor Vessel Specimen Removal Schedule

Marked-up UFSAR Pages
Peach Bottom Atomic Power Station, Units 2 and 3

Page
4.2-17

allow maximum accessibility for inspection. Some insulation panels or portions of panels outside the vessel support are removable to permit inspection of the vessel and vessel support surfaces. All nozzles (except those nozzles inside the vessel support, such as the CRD in-core instrument, and drain nozzles in the bottom head) have insulation designed so that it may be removed to expose the entire exterior of the nozzle and the adjacent vessel shell, and be readily replaced. Original components and locations to be examined were selected taking into account the probability of a defect occurring or enlarging at a certain location, the available accessibility; and Section XI of the ASME boiler and pressure vessel code. For inspection intervals subsequent to the original interval, examinations are performed to satisfy the ASME XI code requirements as specified in 10CFR50.55A. (Ref. Appendix I)

The surveillance test program provides for the preparation of a series of Charpy V-notch impact specimens and tensile specimens from the base metal of the reactor vessel, weld heat-affected zone metal, and weld metal from a reactor steel joint which simulates a welded joint in the reactor vessel. The specimens and neutron monitor wires ~~are~~ Were placed near core mid-height adjacent to the reactor vessel wall where the neutron exposure is similar to that of the vessel wall. The specimens ~~are~~ installed at startup or just prior to full-power operation. ~~Selected groups of specimens may be removed at intervals over the lifetime of the reactor and tested to compare mechanical properties with the properties of control specimens which are not irradiated.~~ The first of three available capsules were withdrawn from Units 2 and 3 and tested in 1988 and 1989, respectively. The results are documented in GE report SASR 88-24, part of DRF B13-01445, for Unit 2 and in GE report SASR 90-50, part of DRF B11-00494, for Unit 3.

Insert A

~~The specimen capsule removal schedule is as follows (refer to NRC Admin. Letter 97-04 for revisions to the removal schedule):~~

Unit No. 2	Unit No. 3
1. 7-9 EFPY (7.53 EFPY Actual)	1. 7-9 EFPY (7.57 EFPY Actual)
2. ~~15-18~~ 30 EFPY	2. ~~15-18 EFPY~~ Standby
3. Standby	3. Standby

~~The surveillance program is the standard surveillance program described in the GE-APED Topical Report, NEDO-10115, "Mechanical Property Surveillance of General Electric BWR Vessels." This surveillance program does not conform to ASTM E-185-66, "Recommended Practice for Surveillance Tests on Structural Materials in Nuclear Reactors," or its revision, ASTM E-185-70, "Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels." For a comparison of the surveillance program to the applicable portions of ASTM E-185-70 see Exhibit VII of Appendix K.~~

PBAPS UFSAR Inserts

Insert A

In 2003, the NRC approved PBAPS participation in the BWR Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) as described in BWRVIP-78 and BWRVIP-86 (Notes 1 and 2). The NRC approved the ISP for the industry (Note 3) and approved PBAPS participation (Note 4). The ISP meets the requirements of 10 CFR 50 Appendix H and provides several advantages over the original program. The surveillance materials in many plant-specific programs do not represent the best match with the limiting vessel beltline materials since some were established prior to 10 CFR 50 Appendix H requirements. Also, the ISP allows for better comparison to unirradiated material data to determine actual shifts in toughness. Finally, for many plants, ISP data will be available sooner to factor into plant operations since there are more sources of data.

The current withdrawal schedule for both units is as follows and is based on the NRC-approved revision of BWRVIP-86 (Note 2).

Notes for Page 4.2-17

1. Letter from C. Terry (BWRVIP) to U. S. NRC, "Project No. 704 - BWR Vessel and Internals Project, BWR Integrated Surveillance Program Plan (BWRVIP-78)," dated December 22, 1999

2. Letter from C. Terry (BWRVIP) to U. S. NRC, "Project No. 704 - BWRVIP-86: BWR Vessel and Internals Project, BWR Integrated Surveillance Program Implementation Plan," EPRI Technical Report 1000888, dated December 22, 2000

3. Letter from U. S. NRC to C. Terry (BWRVIP), " Safety Evaluation Regarding EPRI Proprietary Reports 'BWR Vessel And Internals Project, BWR Integrated Surveillance Program Plan (BWRVIP-78)' and 'BWRVIP-86: BWR Vessel And Internals Project, BWR Integrated Surveillance Program Implementation Plan,'" dated February 1, 2002

4. [Insert letter approving PBAPS participation in VIP]

ATTACHMENT C

Request for License Amendment Regarding Reactor Vessel Specimen Removal Schedule

Revised TS and TS Bases Pages
Limerick Generating Station, Units 1 and 2

<u>TS Pages</u>
xi (Unit 1)
3/4 4-19 (Unit 1)
3/4 4-21 (Unit 1)
xi (Unit 2)
3/4 4-19 (Unit 2)
3/4 4-21 (Unit 2)

<u>TS Bases Pages</u>
B 3/4 4-5 (Unit 1)
B 3/4 10-2 (Unit 1)
B 3/4 4-5 (Unit 2)
B 3/4 10-2 (Unit 2)

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION PAGE

REACTOR COOLANT SYSTEM (Continued)

Figure 3.4.1.1-1 Thermal Power versus Core Flow 3/4 4-3

Jet Pumps 3/4 4-4

Recirculation Pumps 3/4 4-5

Idle Recirculation Loop Startup 3/4 4-6

3/4.4.2 SAFETY/RELIEF VALVES 3/4 4-7

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

Leakage Detection Systems 3/4 4-8

Operational Leakage 3/4 4-9

Table 3.4.3.2-1 Reactor Coolant System Pressure Isolation Valves 3/4 4-11

3/4.4.4 CHEMISTRY 3/4 4-12

Table 3.4.4-1 Reactor Coolant System Chemistry Limits 3/4 4-14

3/4.4.5 SPECIFIC ACTIVITY 3/4 4-15

Table 4.4.5-1 Primary Coolant Specific Activity Sample and Analysis Program 3/4 4-17

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

Reactor Coolant System 3/4 4-18

Figure 3.4.6.1-1 Minimum Reactor Pressure Vessel Metal Temperature Vs. Reactor Vessel Pressure 3/4 4-20

Table 4.4.6.1.3-1 Deleted 3/4 4-21

Reactor Steam Dome 3/4 4-22

3/4.4.7 MAIN STEAM LINE ISOLATION VALVES 3/4 4-23

3/4.4.8 STRUCTURAL INTEGRITY 3/4 4-24

SURVEILLANCE REQUIREMENTS (Continued)

4.4.6.1.2 The reactor coolant system temperature and pressure shall be determined to be to the right of the criticality limit line of Figure 3.4.6.1-1 curve C within 15 minutes prior to the withdrawal of control rods to bring the reactor to criticality and at least once per 30 minutes during system heatup.

4.4.6.1.3 DELETED

4.4.6.1.4 DELETED

4.4.6.1.5 The reactor vessel flange and head flange temperature shall be verified to be greater than or equal to 80°F:

a. In OPERATIONAL CONDITION 4 when reactor coolant system temperature is:

   1. ≤ 100°F, at least once per 12 hours.

   2. ≤ 90°F, at least once per 30 minutes.

b. Within 30 minutes prior to and at least once per 30 minutes during tensioning of the reactor vessel head bolting studs.

INFORMATION CONTAINED ON THIS PAGE HAS BEEN DELETED

SECTION PAGE

REACTOR COOLANT SYSTEM (Continued)

Figure 3.4.1.1-1 Thermal Power versus Core Flow 3/4 4-3

Jet Pumps 3/4 4-4

Recirculation Pumps 3/4 4-5

Idle Recirculation Loop Startup 3/4 4-6

3/4.4.2 SAFETY/RELIEF VALVES 3/4 4-7

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

Leakage Detection Systems 3/4 4-8

Operational Leakage 3/4 4-9

Table 3.4.3.2-1 Reactor Coolant System Pressure Isolation Valves 3/4 4-11

3/4.4.4 CHEMISTRY 3/4 4-12

Table 3.4.4-1 Reactor Coolant System Chemistry Limits 3/4 4-14

3/4.4.5 SPECIFIC ACTIVITY 3/4 4-15

Table 4.4.5-1 Primary Coolant Specific Activity Sample and Analysis Program 3/4 4-17

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

Reactor Coolant System 3/4 4-18

Figure 3.4.6.1-1 Minimum Reactor Pressure Vessel Metal Temperature Vs. Reactor Vessel Pressure 3/4 4-20

Table 4.4.6.1.3-1 Deleted 3/4 4-21

Reactor Steam Dome 3/4 4-22

3/4.4.7 MAIN STEAM LINE ISOLATION VALVES 3/4 4-23

3/4.4.8 STRUCTURAL INTEGRITY 3/4 4-24

SURVEILLANCE REQUIREMENTS (Continued)

4.4.6.1.2 The reactor coolant system temperature and pressure shall be determined to be to the right of the criticality limit line of Figure 3.4.6.1-1 curve C within 15 minutes prior to the withdrawal of control rods to bring the reactor to criticality and at least once per 30 minutes during system heatup.

4.4.6.1.3 DELETED

4.4.6.1.4 DELETED

4.4.6.1.5 The reactor vessel flange and head flange temperature shall be verified to be greater than or equal to 70°F:

a. In OPERATIONAL CONDITION 4 when reactor coolant system temperature is:

   1. ≤ 100°F, at least once per 12 hours.

   2. ≤ 90°F, at least once per 30 minutes.

b. Within 30 minutes prior to and at least once per 30 minutes during tensioning of the reactor vessel head bolting studs.

INFORMATION CONTAINED ON THIS PAGE HAS BEEN DELETED

PRESSURE/TEMPERATURE LIMITS (Continued)

The operating limit curves of Figure 3.4.6.1-1 are derived from the fracture toughness requirements of 10 CFR 50 Appendix G and ASME Code Section XI, Appendix G. The curves are based on the RT_{NDT} and stress intensity factor information for the reactor vessel components. Fracture toughness limits and the basis for compliance are more fully discussed in FSAR Chapter 5, Paragraph 5.3.1.5, "Fracture Toughness."

The reactor vessel materials have been tested to determine their initial RT_{NDT}. The results of these tests are shown in Table B 3/4.4.6-1. Reactor operation and resultant fast neutron, E greater than 1 MeV, irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, nickel content and copper content of the material in question, can be predicted using Bases Figure B 3/4.4.6-1 and the recommendations of Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." The pressure/temperature limit curves, Figure 3.4.6.1-1, include a shift in RT_{NDT} for conditions at 32 EFPY. The A, B and C limit curves are predicted to be bounding for all areas of the RPV until 32 EFPY. In addition, an intermediate A curve has been provided for 22 EFPY.

The pressure-temperature limit lines shown in Figures 3.4.6.1-1, curves C, and A, for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR Part 50 for reactor criticality and for inservice leak and hydrostatic testing.

3/4.10.8 INSERVICE LEAK AND HYDROSTATIC TESTING

This special test exception permits certain reactor coolant pressure tests to be performed in OPERATIONAL CONDITION 4 when the metallurgical characteristics of the reactor pressure vessel (RPV) or plant temperature control capabilities during these tests require the pressure testing at temperatures greater than 200°F and less than or equal to 212°F (normally corresponding to OPERATIONAL CONDITION 3). The additionally imposed OPERATIONAL CONDITION 3 requirements for SECONDARY CONTAINMENT INTEGRITY provide conservatism in response to an operational event.

Invoking the requirement for Refueling Area Secondary Containment Integrity along with the requirement for Reactor Enclosure Secondary Containment Integrity applies the requirements for Reactor Enclosure Secondary Containment Integrity to an extended area encompassing Zones 1 and 3. Operations with the Potential for Draining the Vessel, Core alterations, and fuel handling are prohibited in this secondary containment configuration. Drawdown and inleakage testing performed for the combined zone system alignment shall be considered adequate to demonstrate integrity of the combined zones.

Inservice hydrostatic testing and inservice leak pressure tests required by Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code are performed prior to the reactor going critical after a refueling outage. The minimum temperatures (at the required pressures) allowed for these tests are determined from the RPV pressure and temperature (P/T) limits required by LCO 3.4.6, Reactor Coolant System Pressure/Temperature Limits. These limits are conservatively based on the fracture toughness of the reactor vessel, taking into account anticipated vessel neutron fluence. With increased reactor fluence over time, the minimum allowable vessel temperature increases at a given pressure.

PRESSURE/TEMPERATURE LIMITS (Continued)

The operating limit curves of Figure 3.4.6.1-1 are derived from the fracture toughness requirements of 10 CFR 50 Appendix G and ASME Code Section XI, Appendix G. The curves are based on the RT_{NDT} and stress intensity factor information for the reactor vessel components. Fracture toughness limits and the basis for compliance are more fully discussed in FSAR Chapter 5, Paragraph 5.3.1.5, "Fracture Toughness."

The reactor vessel materials have been tested to determine their initial RT_{NDT}. The results of these tests are shown in Table B 3/4.4.6-1. Reactor operation and resultant fast neutron, E greater than 1 MeV, irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, nickel content and copper content of the material in question, can be predicted using Bases Figure B 3/4.4.6-1 and the recommendations of Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." The pressure/temperature limit curve, Figure 3.4.6.1-1, curves A, B and C, includes an assumed shift in RT_{NDT} for the conditions at 32 EFPY. In addition, an intermediate A curve has been provided for 22 EFPY. The A, B and C limit curves are predicted to be bounding for all areas of the RPV until 32 EFPY.

The pressure-temperature limit lines shown in Figures 3.4.6.1-1, curves C, and A, for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR Part 50 for reactor criticality and for inservice leak and hydrostatic testing.

3/4.10.8 INSERVICE LEAK AND HYDROSTATIC TESTING

This special test exception permits certain reactor coolant pressure tests to be performed in OPERATIONAL CONDITION 4 when the metallurgical characteristics of the reactor pressure vessel (RPV) or plant temperature control capabilities during these tests require the pressure testing at temperatures greater than 200°F and less than or equal to 212°F (normally corresponding to OPERATIONAL CONDITION 3). The additionally imposed OPERATIONAL CONDITION 3 requirements for SECONDARY CONTAINMENT INTEGRITY provide conservatism in response to an operational event.

Invoking the requirement for Refueling Area Secondary Containment Integrity along with the requirement for Reactor Enclosure Secondary Containment Integrity applies the requirements for Reactor Enclosure Secondary Containment Integrity to an extended area encompassing Zones 2 and 3. Operations with the Potential for Draining the Vessel, Core alterations, and fuel handling are prohibited in this secondary containment configuration. Drawdown and inleakage testing performed for the combined zone system alignment shall be considered adequate to demonstrate integrity of the combined zones.

Inservice hydrostatic testing and inservice leak pressure tests required by Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code are performed prior to the reactor going critical after a refueling outage. The minimum temperatures (at the required pressures) allowed for these tests are determined from the RPV pressure and temperature (P/T) limits required by LCO 3.4.6, Reactor Coolant System Pressure/Temperature Limits. These limits are conservatively based on the fracture toughness of the reactor vessel, taking into account anticipated vessel neutron fluence. With increased reactor fluence over time, the minimum allowable vessel temperature increases at a given pressure.

ATTACHMENT B-7

Request for License Amendment Regarding Reactor Vessel Specimen Removal Schedule

Marked-up UFSAR Pages
Quad Cities Nuclear Power Station, Units 1 and 2

<u>Pages</u>
5.3-3
5.3-4
5.3-12
Table 5.3-1

Regulatory Guide 1.50

Preheat temperatures used when welding low alloy steel components (shells, flanges, plates) met applicable requirements or had contract variations approved by GE, the vendor responsible for supplying the RPV.

Regulatory Guide 1.99

Section 5.3.2.1 contains information on compliance with the methodology in Regulatory Guide 1.99, Rev. 2.

5.3.1.5 Fracture Toughness

Sections 5.2.3.3.1 and 5.3.2.1 describe fracture toughness provisions for the Quad Cities Units 1 and 2 RPVs.

5.3.1.6 Material Surveillance

5.3-8 Vessel material surveillance samples are located within the reactor vessel to enable periodic monitoring of changes in material properties with exposure. The samples include specimens of the base metal, weld zone metal, heat affected zone metal, and standard specimens. These specimens receive neutron exposures more rapidly than the vessel wall material of interest (i.e., the innermost 25% of vessel wall thickness) and therefore lead it in integrated neutron flux. The neutron exposure rate of the average specimen at the core midplane is approximately 1.2 times the exposure rate of the adjacent inside surface of the vessel wall.

Insert A

There were 401 samples initially inserted in the vessel, and these are removed in sample groups over the lifetime of the vessel for Charpy V-notch and tensile strength tests. Testing is in accordance with ASTM E208.

~~The Surveillance Specimen programs for the Quad Cities RPVs are the same as that~~ described in Fitzpatrick Supplement Docket No. 50-233, Comment 3.8, except for the following major differences:

1. Paragraph 3.3

   QC has a total of 401 tensile and impact specimens instead of the 195 for Fitzpatrick. This permits tests on both submerged arc and electroslag welds.

2. Paragraph 4.2

   Some of the extra specimens will be used for accelerated exposure determinations by placing them on the top guide instead of adjacent to the vessel wall.

A conservative estimate (based on past experience) of 160 equivalent operating years per year accelerated exposure has been confirmed.

The surveillance programs for the Unit 1 and Unit 2 RPVs are alike but independent.

5.3-9 The withdrawal schedule in Table 5.3-1 is based on the three capsule surveillance program as defined 10 CFR 50 Appendix H. The accelerated capsules (near core top guide) are not required by Appendix H.

This surveillance program conforms to ASTM E 185-82, "Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels," with one exception. The base metal specimens of the vessel were made with their longitudinal axes parallel to the principal rolling direction of the vessel plate.

5.3.1.7 Reactor Pressure Vessel Fasteners

5.3-10 The top head of the RPV is secured to the vessel with studs, nuts, and spherical washers. Nut torquing and detorquing is accomplished using a stud tensioner. Technical Specifications require that the RPV head bolting studs (or closure studs) not be under tension unless the metal temperature of the vessel shell immediately below the vessel flange is at or above 83°F. This value (83°F) comes from the reference temperature (RT_{NDT}) and ASME Code considerations as discussed in Section 5.3.2.

5.3-11 A fatigue usage analysis dated May 1999, demonstrated that the cumulative fatigue usage factor (CFUF) for the RPV closure studs would remain below 1.0 for current forty-year design lift. A previous fatigue usage analysis dated March 1990, using then current duty cycle values, originally predicted that the RPV closure studs would reach the CFUF limit of 1.0 in 1998. This prediction was recalculated using actual cycle data through November 1997, to demonstrate that the CFUF limit would be reached in 2002. The purpose of the May 1999 analysis was to reduce conservation used in the original vessel closure stud analysis and to qualify the studs for a forty-year design life using an updated fatigue evaluation. The primary means of reducing the fatigue usage was to use the actual number of operating cycles and perform new cycle pairing based on stress ranges and number of occurrences. Further reduction in fatigue usage was accomplished by using the appropriate ASME Code fatigue curve of 2.7Sm versus 3Sm. The results of this analysis show the CFUF for the vessel closure studs to be 0.73 for both Units 1 and 2 at the end of a forty-year design life. This value is well below the allowable CFUF limit of 1.0 established in the ASME Section III Code and as a result justifies at least forty years of operation.

5.3.2 Pressure - Temperature Limits

5.3-12 Fast (>1 MeV) neutron irradiation above 10^{17} nvt begins to affect the mechanical properties of ferritic steel. The most important consideration is that of the change in the temperature at which ferritic steel breaks in a brittle rather than a ductile mode (referred to as the Nil Ductility Transition Temperature or NDTT). The NDTT increases with increasing irradiation. ASME Section III, N-446 specifies the design conditions for determination of the NDTT. Extensive tests have established the magnitude of changes in the NDTT as a function of the integrated neutron dosage.

The SA 302B steel, with fabrication procedures specified by the ASME Code and by GE, is relatively insensitive to neutron irradiation. In fact, no change in the Adjusted Reference Temperature (ART) is expected to occur at neutron exposures less than 4.0×10^{17} nvt.

The flux levels were calculated using a modified Albert-Welton point kernel[1] which was originally developed as an approximate method of calculating the attenuation from a point fission source in water. The method represents fast neutron attenuation in water by a function which was experimentally fitted to data obtained for neutron attenuation by

5.3.4 References

1. "Reactor Handbook," 2nd Edition, Vol. III Part B, Shielding, pages 72 and 80.

Insert B

Table 5.3-1

REACTOR VESSEL MATERIAL SURVEILLANCE WITHDRAWAL SCHEDULE

UNIT 1

HOLD NUMBER	LOCATION	AZIMUTH	REMOVAL YEAR	STATUS
NEUTRON DOSIMETER	MOUNTED SIDE OF PART #7	95°	1974	REMOVED
2	TOP GUIDE	0°	1974	REMOVED
3	WALL	35°	1974	REMOVED
4	TOP GUIDE	90°	1979	REMOVED
5	WALL	65°	STANDBY	
6	TOP GUIDE	180°	1982	REMOVED
7	WALL	95°	2002	
8	WALL	215°	1982	REMOVED
9	WALL	245°	STANDBY	
10	WALL	275°	STANDBY	

STANDBY

UNIT 2

HOLD NUMBER	LOCATION	AZIMUTH	REMOVAL YEAR	STATUS
NEUTRON DOSIMETER	MOUNTED SIDE OF PART #7	95°	1975	------
12	TOP GUIDE	0°	1975	REMOVED
13	WALL	35°	1975	REMOVED
14	TOP GUIDE	90°	1979	REMOVED
15	WALL	65°	STANDBY	
16	TOP GUIDE	180°	1981	REMOVED
17	WALL	95°	2002	
18	WALL	215°	1981	REMOVED
19	WALL	245°	STANDBY	
20	WALL	275°	STANDBY	

NOTE: 0° IS DUE WEST.

Quad Cities UFSAR Inserts

Insert A

Table 5.3-1 provides the location and status of the material specimens.

In 2003, the NRC approved Quad Cities participation in the BWR Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) as described in BWRVIP-78 and BWRVIP-86 in References 2 and 3. The NRC approved the ISP for the industry in Reference 4 and approved Quad Cities participation in Reference 5. The ISP meets the requirements of 10 CFR 50 Appendix H and provides several advantages over the original program. The surveillance materials in many plant-specific programs do not represent the best match with the limiting vessel beltline materials since some were established prior to 10 CFR 50 Appendix H requirements. Also, the ISP allows for better comparison to unirradiated material data to determine actual shifts in toughness. Finally, for many plants, ISP data will be available sooner to factor into plant operations since there are more sources of data.

The current withdrawal schedule for both units is based on the latest NRC-approved revision of BWRVIP-86 (Reference 3). Based on this schedule, Quad Cities is not scheduled to withdraw an additional material specimen.

Insert B

2. Letter from C. Terry (BWRVIP) to U. S. NRC, "Project No. 704 - BWR Vessel and Internals Project, BWR Integrated Surveillance Program Plan (BWRVIP-78)," dated December 22, 1999

3. Letter from C. Terry (BWRVIP) to U. S. NRC, "Project No. 704 - BWRVIP-86: BWR Vessel and Internals Project, BWR Integrated Surveillance Program Implementation Plan," EPRI Technical Report 1000888, dated December 22, 2000

4. Letter from U. S. NRC to C. Terry (BWRVIP), " Safety Evaluation Regarding EPRI Proprietary Reports 'BWR Vessel And Internals Project, BWR Integrated Surveillance Program Plan (BWRVIP-78)' and 'BWRVIP-86: BWR Vessel And Internals Project, BWR Integrated Surveillance Program Implementation Plan,'" dated February 1, 2002

5. [Insert letter approving Quad Cities participation in ISP]