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U. S. Nuclear Regulatory Commission
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Dresden Nuclear Power Station, Units 2 and 3
Facility Operating License Nos. DPR-19 and DPR-25
NRC Docket No. 50-237 and 50-249

Quad Cities Nuclear Power Station, Units 1 and 2
Facility Operating License Nos. DPR-29 and DPR-30
NRC Docket Nos. 50-254 and 50-265

Subject: Additional Information for the Review of the License Renewal Applications for Dresden Nuclear Power Station, Units 2 and 3 and Quad Cities Nuclear Power Station, Units 1 and 2

Reference: (1) Letter from J. A. Benjamin (Exelon Generation Company, LLC) to U. S. NRC, "Application for Renewed Operating Licenses," dated January 3, 2003

Exelon Generation Company, LLC (EGC) is submitting the additional information requested in email requests sent by Tae Kim (NRC) to EGC on October 23, 2003. This additional information provides a response to questions regarding the Section 3.1 and associated Aging Management Programs sections of Reference 1 to support the NRC review. EGC responses to requests for additional information for RAI 2.3.4.2-3, 3.1-1 and B.1.4 will be submitted in a later correspondence.

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A098

Should you have any questions, please contact Al Fulvio at 610-765-5936.

I declare under penalty of perjury that the foregoing is true and correct.

Respectfully,

11/21/03
Executed on

Patrick R. Simpson
Patrick R. Simpson
Manager – Licensing

Attachment : Response to Request for Additional Information – Section 3.1 and Associated Aging Management Programs

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Quad Cities Nuclear Power Station
NRC Senior Resident Inspector – Dresden Nuclear Power Station
Illinois Emergency Management Agency

Attachment

Response to Request for Additional Information – Section 3.1 and Associated Aging Management Programs

RAI 3.1-4 Supplemental Information Request

Clarify whether the fluence calculations for the reactor vessel included the effects of power uprates when determining which components are susceptible to loss of fracture toughness.

Response:

The fluence calculations for the reactor vessel did include the effects of power uprates when determining which components are susceptible to loss of fracture toughness.

RAI 3.1-11 Supplemental Information Request

The applicant's response to RAI 3.1-11 states that the Aging Management program B.2.6 manages loss of material and crack initiation and growth in the Dresden isolation condensers. This program applies to the tubing, tubesheet, channel heads and shells, and consists of performing eddy current testing of the tubes as well as temperature and radiation monitoring of the shell-side water (which are consistent with NUREG-1801). However, NUREG-1801, items IV.C.1.4-a and b requires the following two enhancements in addition to the AMPs in Chapter XI.M1 and XI.M2: (1) augmented inspections to detect cracking due to SCC and cyclic loading or loss of material due to pitting and crevice corrosion in isolation condenser components (tubing, tubesheet, channel head, and shell, and (2) verification of the effectiveness of the program. NUREG-1801 identifies temperature and radioactivity monitoring of the shell-side water, and eddy current testing of the tubes as a verification program. This verification program does not verify that no cracking or loss of material is occurring in the isolation condenser shell, channel head and tubesheet. Therefore, provide augmented inspection of the Dresden isolation condenser (i.e. VT or UT) to manage loss of material, and crack initiation and growth in the isolation condenser tubesheet, channel head, and shell, as required by NUREG-1801.

Response:

The response is divided into two parts. For clarity of understanding, background information on the RAI is first provided. Afterwards, the response to the supplemental information request is provided.

Background information for RAI 3.1-11 Supplemental Information Request

Aging analysis of the isolation condenser can be found in Section IV C1 in NUREG 1801. This section was intended for systems connected to the pressure vessel extending outward to the second containment isolation valves. The Dresden isolation condensers are installed outboard of the second containment isolation valve. As a result, the isolation condenser are not classified as ISI Class 1, but are classified as ISI Class 2 on the tube side, and ISI Class 3 on the shell side. As described in the License Renewal Application (LRA) Section 2.3.2.5, the isolation condenser is a heat exchanger which consists of two tube bundles immersed in a large storage tank, with the tank vented to the atmosphere.

NUREG 1801, items IV.C.1.4-a and b, require that the aging management program (AMP) in Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD" for Class 1 components be augmented to detect cracking due to stress corrosion cracking and cyclic loading or loss of material due to pitting and crevice corrosion. It also requires that the effectiveness of AMP XI.M1 be verified. It states that an acceptable verification program is to include temperature and radiation monitoring of the shell side water, and eddy current testing of the tubes.

LRA Table 3.1-1, items 3.1.1.2 (loss of material due to general, pitting, and crevice corrosion) and 3.1.1.7 (crack initiation and growth due to stress corrosion cracking or cyclic loading) credited the inservice inspection and water chemistry AMPs as described in the LRA Appendix B, Sections B.1.1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," and B.1.2, "Water Chemistry." Items 3.1.1.2 and 3.1.1.7 included discussions of further evaluations in LRA Sections 3.1.1.1.2 and 3.1.1.1.7 respectively. Both Sections 3.1.1.1.2 and 3.1.1.1.7 stated that AMP B.1.1 will be augmented by AMP B.2.6, and that B.2.6 activities include temperature and radiation monitoring of the shell side water, and eddy current testing of the tubes, to ensure intended function is maintained.

The initial RAI 3.1-11 stated that "LRA Appendix B.1.1 requires VT-2 examinations of the reactor coolant pressure boundary during system pressure testing. This is not adequate for detecting crack initiation and growth in the isolation condenser components before their intended function (pressure boundary) is compromised. Identify the augmented inspection program for detecting loss of material, and crack initiation and growth in the Dresden isolation condenser tubesheet, channel head, and shell as recommended by Items C1.4-a and C1.4-b, Chapter IV.C1 of NUREG-1801."

The response to RAI 3.1-11 stated that "Aging management program B.2.6, "Heat Exchanger Test and Inspection Program," manages loss of material and crack initiation and growth in the Dresden isolation condensers. This program, as it applies to the isolation condensers, which include the tubing, tube sheets, channel heads and shells, consists of performing eddy current testing of the tubes as well as temperature and radiation monitoring of the shell-side (cooling) water."

This supplemental information request to RAI 3.1-11 stated that "NUREG-1801 identifies temperature and radioactivity monitoring of the shell-side water, and eddy current testing of the tubes as a verification program. This verification program does not verify that no cracking or loss of material is occurring in the isolation condenser shell, channel head and tubesheet. Therefore, provide augmented inspection of the Dresden isolation condenser (i.e. VT or UT) to manage loss of material, and crack initiation and growth in the isolation condenser tubesheet, channel head, and shell, as required by NUREG-1801."

RAI 3.1-11 Supplemental Information Request Response

Aging management program (AMP) B.2.6, "Heat Exchanger Test and Inspection Program," is a ten-element program that was developed to address heat exchangers in scope of license renewal that are not inspected under other AMPs. The intent of the AMP B.2.6, as originally developed and described In the LRA, is to require a visual inspect of the isolation condenser channel head, tube sheet, and shell, in addition to performing eddy current testing of the tubes, and temperature and radiation monitoring

of the shell-side water. These are all new activities that will be implemented prior to the period of extended operation.

License Renewal Application, Appendix B, Section B.2.6, summarized Aging management program B.2.6, "Heat Exchanger Test and Inspection Program," but did not clearly describe the visual inspection of the isolation condenser tubesheet, channel head, and shell in the description of the isolation condenser augmented activities.

In addition to identifying the augmented isolation condenser inspection activities of temperature and radiation monitoring of the shell-side water, and eddy current testing of the tubes, AMP B.2.6 provides for condition monitoring, inspection, and performance testing of heat exchangers in scope of license renewal that are not inspected under other AMPs, including the isolation condensers.

Aging management program AMP B.2.6 requires that the isolation condenser be inspected in according with the station heat exchanger inspection program, and eddy current tested in accordance with corporate procedures. These inspections and testing are performed by qualified inspectors.

(1) In conjunction with the periodic eddy current testing of the tubes, a visual inspection to detect cracking and loss of material of the channel head and tube sheets will be performed on the tube-side of the isolation condenser in accordance with the station heat exchanger inspection program as an augmented inspection to manage loss of material and crack initiation and growth in the isolation condenser tube sheet and channel head.

(2) Shell-side visual inspections are presently periodically performed to verify the integrity of shell-side internal structural components. These inspections will be expanded in accordance with the station heat exchanger inspection program to visually inspect the shell to detect cracking and loss of material of the shell as an augmented inspection to manage loss of material and crack initiation and growth in the isolation condenser shell.

RAI 3.1-13 Supplemental Information Request

LRA Section 3.1.1.2.1 did not address loss of preload as an aging effect, but this section did state that loss of preload mechanisms are typically addressed during installation and subsequent maintenance of closure bolting. However, the applicant's response to RAI 3.1-13 stated that loss of preload will be managed, and that the enhanced Bolting Integrity aging management program will be comprised of periodic In-Service Inspections and piping and components Preventative Maintenance inspections. Describe the maintenance program activities that are performed on the bolts so that loss of preload is significantly reduced or eliminated, and identify whether re-torquing of the bolts to the design preload values is performed after the component is reassembled.

Response:

Exelon procedure governing "Torquing and Tightening of Bolted Connections" requires the inspection and documentation of closure bolting connections prior to disassembly for sign of leakage; missing, cracked, degraded or loose bolts or nuts; misalignment of connection; gasket condition; and corrosion. When closure bolting connections are disassembled, bolts are loosened in 2 increments using a crossing sequence to prevent distortion of the mating surfaces. Bolts, studs, and washers are cleaned of corrosion, grit and dried lubricant. They are also inspected for galling of threads or nut facing, dings and dents in threads, cracks, pits, erosion, corrosion, bent bolts, dished or galled washers, and elongation. If damaged bolts, studs, nuts or washers are found, they are replaced or repaired. Prior to reassembly, the correct torque value is determined from history that has been validated by successful performance, design drawings, vendor prints, plant design specification or procedures. When required by design drawings, vendor prints, plant design specifications or procedures, approved lubricant is applied to the bolts, studs and nuts. Bolts or nuts are first hand tightened using a crossing sequence then torque tightened using a diametrically opposed pattern in 3 passes (ALARA areas) or 4 passes (Non-ALARA areas). Conditions that involve leakage, insufficient gasket compression or other situations that require torque values to be increased require Engineering review. All torque values are documented.

The In-Service Inspection procedure requires corrective action if leakage occurs at a bolted connection. The bolts are removed, VT-3 examined for corrosion, and evaluated in accordance with IWA-3100. A corrective action report is written to document relevant/recordable indications that require corrective actions. A work order is then initiated to correct the identified leakage.

System Engineering walkdowns will be performed per the "License Renewal Bolting Integrity Program Walkdowns" procedure. This procedure requires the system engineer to inspect for leakage and visual indications of loose bolts. Leakage may be evidenced by the existence of steam or vapor; hissing; surface wetness or ice buildup; dripping; puddles; abnormal or unlikely locations of corrosion; or wet, damp, or sagging non-metallic insulation. Loose bolts may be evidenced by the signs of leakage; missing, cracked, bent, or degraded bolts; missing, cracked, or degraded nuts; missing or cracked washers; lack of full thread engagement; or visible gaps between the bolt head and the mating surface. If degradation due to leakage or loose bolting is detected a condition report will be written and corrective action taken. Results will be trended in accordance with "System Performance Monitoring and Analysis" procedure.

<u>RAI 3.1-21(b) Supplemental Information Request</u>

The applicant's response to RAI 3.1-21(b) states that water chemistry controls that are sufficiently effective to prevent loss of material at stagnant flow locations are also expected at preventing stress corrosion cracking. Therefore, one-time inspections for loss of material would cover the need for a one-time inspection for cracking due to SCC. The staff does not agree with the applicant, because the locations susceptible to loss of material are generally different than the locations susceptible to cracking due to SCC. Therefore, provide a one-time inspection for SCC at the susceptible locations.

Response:

The process fluid temperature in the control rod drive (CRD) hydraulic system is less than 100 deg-F, and the typical flow conditions are either low flow (in the cooling water line) or stagnant flow (in the charging water and drive water lines). Under these conditions, it is expected that the stagnant flow locations that are susceptible to loss of material due to pitting and crevice corrosion are also the locations that may be susceptible to cracking due to SCC. In the piping system for the control rod drive (CRD) hydraulic system, where the same water chemistry applies to all parts of the system, Exelon believes that water chemistry controls sufficient to prevent loss of material due to pitting and crevice corrosion are also sufficient to prevent stress corrosion cracking. In addition, with process temperatures below 140 deg-F, EPRI TR 1003056 Mechanical Tools Appendix A states that cracking due to SSC is very unlikely to occur. Nonetheless, Exelon will include inspection for stress corrosion cracking as part of its one-time inspection to validate the effectiveness of the Water Chemistry Program (LRA Appendix B.1.2) in managing the aging of stainless steel components in the CRD hydraulic system.

<u>4.2-BWRVIP Supplemental Information Request</u>

RAI 4.2-BWRVIP asked that the applicant referencing a particular report for license renewal should identify and evaluate any potential TLAA issues and/or commitments to perform future inspections when inspection tooling is made available. The applicant's response is incomplete because it does not address the request for commitments to perform future inspections when inspection tooling is made available. The applicant should commit to perform future inspections when inspection tooling is made available.

Response:

Dresden and Quad Cities will perform the additional inspections when new inspection techniques and tooling are developed, incorporated into the applicable BWRVIP document(s), and approved by the NRC.

<u>RAI B.1.3 (a) Supplemental Information Request</u>

The applicant's response to RAI B.1.3 (a) states that they have updated their ISI Programs to be consistent with the 1995 Edition, 1996 Addenda of ASME Section XI, but that the requirements of ASME Section XI, Subsection IWB, Table IWB 2500-1, will be augmented by Code Case N-307-2. Since the use of Code Case N-307-2 provides information that augments the requirements of the ASME code, confirm that the UFSAR supplement is revised to reference this code case.

Response:

It is not known at this time what other ASME Section XI Editions, Addenda, Code Cases or Relief Requests that Exelon may incorporate during the period of extended operation. However, Exelon will update the UFSAR Supplement to indicate that the requirements of ASME Section XI will be implemented in accordance with 10 CFR 50.55(a).

RAI B.1.5 Supplemental Information Request

The BWR Feed Water Nozzle AMP (LRA Appendix B.1.5) references GE report GE-NE-523-A71-0594, which is not the NRC approved version of the report. Confirm that the applicant will implement the recommendations of Revision 1, Version A of the report (GE -NE-523-A71-0594-A, Revision 1) which is approved by the staff.

Response:

Exelon will implement the recommendations of Revision 1, Version A of the report (GE - NE-523-A71-0594-A, Revision 1) which was approved by the NRC staff.

B.1.9 (a) Supplemental Information Request

The applicant's response to RAI B.1-9(a) states that Dresden and Quad Cities are following BWRVIP-26 using EVT-1 of the top guide for evaluating IASCC. The examination extent and frequency is a 10% sample of the total population within 12 years, etc. However, this response never addressed the total population. Therefore, identify the locations of the top guide, equivalent to Peach Bottom, and confirm that their fluences exceed the IASCC threshold limit of BWRVIP-26.

Response:

BWRVIP-26 figures 2-1 and 2-3 identify 94 potential locations (total population) of failure. The top guide grid beam locations selected for examination will be contingent on the CRDH guide tube locations selected for inspection per BWRVIP-47. This is not necessarily limited to the center or near the center of the core locations, as was stated by Peach Bottom.

However, the locations selected for examination will be in areas that have surpassed the fluence threshold for IASCC noted in BWRVIP-26. As stated in BWRVIP-26, section 2.1.1, "based on estimates made in the late 1980's all BWR/2 through BWR/5's have reached or surpassed the fluence threshold for IASCC at the top guide grid beam locations". Dresden and Quad Cities (BWR/3 plants) are consistent with this evaluation.

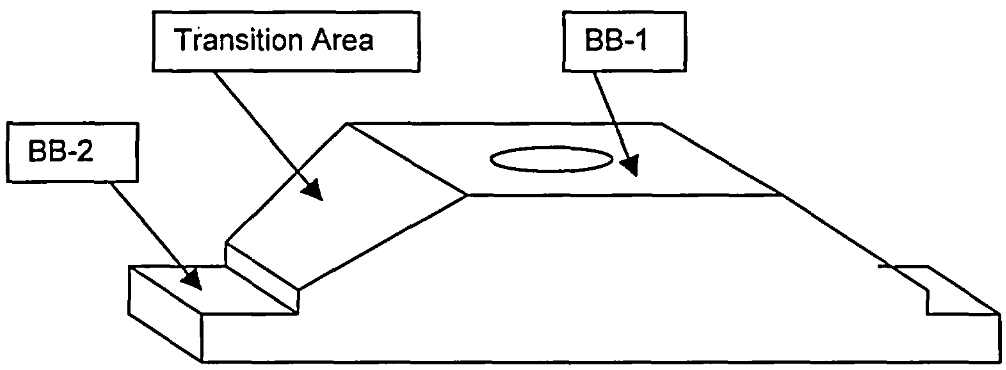
RAI B.1.9 (d) Supplemental Information Request

As specified by the Response to RAI B.1.9 (d), the applicant provides experience of the jet pump failure by referencing Quad Cities Licensee Event Report LER 1-02-001. The LER states that although Quad Cities Nuclear Power Station implemented the requirements of BWRVIP-41, including periodic inspections of the required areas of the jet pump hold down beams, the transitions areas of the beam was not required to be inspected and the crack was not identified prior to failure. However, Table 3.3-1 of BWRVIP-41 provides inspection guidelines for location BB-2 (Beam Transition Arm Region) of the jet pump. Explain this contradiction and confirm that the inspections required by BWRVIP-41 will be performed, including the beam transition area of the jet pump.

Response:

The statement is not a contradiction. Inspections recommended in BWRVIP-41 are divided into both baseline and re-inspection recommendations based on an inspection cycle of six years. Baseline inspection required UT or other NDE technique of all beams during the next inspection cycle. Fifty percent of the inspections were to be performed in the next refueling outage. Re-inspection requirements are 50% per inspection cycle for the first 20 years of service and 100% per inspection cycle beyond 20 years of service. Both Dresden and Quad Cities have performed 100 percent UT inspections of the BB-1 and BB-2 each refueling outage since 1995, which exceeds BWRVIP-41 recommendations.

However, the cracking occurred in the transition area, located between BB-1 (the area surrounding the bolt hole) and BB-2 (the machined radius or ear). The transition area is a low stress location, outside the area of interest specified by the BWRVIP-41. (See below sketch.)



As part of the corrective actions from the LER root cause investigation, Exelon has requested the BWRVIP committee to provide further evaluations of the adequacy of the examinations performed on the hold down beams. Exelon will continue to perform the inspections required by BWRVIP-41 (BB-1 and BB-2) and will incorporate the additional changes in inspection methodology when incorporated into BWRVIP-41.

RAI B.1.9 (d1) Supplemental Information Request

In its response to RAI B.1.9-c, the applicant states that the aging management activities associated with detecting these cracks in the core spray piping were based on BWRVIP-18 "BWR Core Spray Internals Inspection and Evaluation Guidelines". The examination methods included EVT-1. The staff finds the applicant's use of the BWRVIP-18 recommendations acceptable for detecting cracks in the core spray piping because BWRVIP-18 has been reviewed and approved by the staff. With respect to the access hole covers, the applicant states that the covers were inspected by VT-1 and VT-3 based on the recommendations of GE SIL 462 and Supplement 1 "Shroud Access Hole Cover Cracks". As a result of these inspections, cracks were found in the welded access hole covers for Dresden 2 and Quad Cities 1 & 2 and all access hole covers at these three units were subsequently replaced with bolted mechanical covers. The applicant further states that future inspections of the Dresden 3 welded access hole covers will continue to be based on the SIL guidance. The staff finds the use of visual inspection for detecting cracking in the welded access cover unacceptable because it is not consistent with Item B1.1-d, Chapter IV, NUREG-1801. Item B1.1-d recommends

ultrasonic inspection or other demonstrated acceptable inspection of the access hole cover welds instead of visual inspection because visual inspection are not suited for detecting cracks initiated deep in the crevice regions that are likely to be located in the welds. In addition, GE Nuclear Energy has issued SIL 462, Revision 1 (published March 21, 2001), which supersedes and voids SIL 462 and its three supplements. Therefore, the applicant cannot refer to SIL 462 and Supplement 1. The staff notes that SIL 462, Revision 1, recommends inspection methods that are capable of detecting both circumferential and radial crack indications in the AHC plate/assembly, connecting weld and adjacent shroud support plate. Provide an inspection program that meets Item B1.1-d, Chapter IV, NUREG-1801, such as an ultrasonic inspection or other demonstrated acceptable inspection method of the access hole cover welds.

Response:

In the original response to RAI B.1.9-c Exelon stated what examination methods had been used to originally detect the cracking in the Dresden 2 and Quad Cities 1 & 2 access hole cover plate welds. These cracks were found in the 1991 – 1992 time frame. This was not meant to describe the program going forward.

Future examinations of the Dresden 3 welded access hole cover will be ultrasonic examination, consistent with the guidance provided in SIL 462, Revision 1.

RAI B.1.22 Supplemental Information Request

The response to RAI B.1.22 stated that the reactor vessel surveillance program will be enhanced to be consistent with BWRVIP-78 and 86, and that this commitment is already included in section A.1.22 of the UFSAR supplement. However, this section of the UFSAR supplement only states that the program will be consistent with BWRVIP 78 and 86, and does not specify the requirements for the license renewal period. Confirm that the applicant will implement, and include in their UFSAR supplement, the ISP for the LR period (Proposed BWRVIP-116) when approved by NRC, and if not approved, the applicant will provide a plant-specific surveillance plan for the LR period in accordance with 10CFR Part 50, Appendices G and H, which should include:

- Capsules must be removed periodically to determine the rate of embrittlement and a least one capsule with neutron fluence not less than once or greater than twice the peak beltline neutron fluence must be removed before the expiration of the license renewal period.
- Capsules must contain material to monitor the impact of irradiation on the limiting beltline materials and must contain dosimetry to monitor neutron fluence.
- If capsules are not being removed from the plant during the license renewal period, the applicant must supply operating restrictions (i.e., inlet temperature, neutron spectrum, and flux) to ensure that the RPV is operating within the environment of the surveillance capsules, and must supply ex-vessel dosimetry for monitoring neutron fluence.

This is similar to the Peach Bottom LR.

Response:

Exelon will implement, and include in the UFSAR supplements, the ISP for the LR period (Proposed BWRVIP-116) when approved by NRC, and if not approved, Exelon will provide a plant-specific surveillance plan for the LR period in accordance with 10CFR Part 50, Appendices G and H.

The program description in Section B.1.22 of the license renewal application should have read as follows (corrections underlined in bold type):

B.1.22 Reactor Vessel Surveillance

Description

The reactor vessel surveillance aging management program provides management of irradiation embrittlement of the reactor pressure vessel through testing that monitors reactor vessel beltline materials. The program is implemented through station procedures that conform to the requirements of 10 CFR 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements." Neutron embrittlement for the period of extended operation is predicted utilizing chemistry tables and Position 1.3 limitations as described in Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials."

The program will be consistent with the BWRVIP-78, "Integrated Surveillance Program," (ISP.) The program will provide for capsule testing consistent with BWRVIP-78, and saving withdrawn capsules for future reconstitution. Additionally the program will be consistent with BWRVIP-86, "BWR Integrated Surveillance Program Implementation Plan"**. The program will also be consistent with BWRVIP-116, "BWR Vessel and Internals Project Integrated Surveillance Program (ISP) Implementation for License Renewal" upon receipt of NRC approval.** The ISP will increase technical quality and make most effective use of existing BWR surveillance capsules and data.

The enhanced program provides for capsule testing consistent with BWRVIP-78 and BWRVIP-86. Existing capsules not included in the current ISP will be maintained in the vessels as a contingency for the future.

The BWRVIP-78 report described the technical basis related to material selection and testing on which the proposed BWRVIP ISP was constructed. The report principally addressed the methodology established to identify existing plant-specific surveillance capsules and surveillance capsules from the Supplemental Surveillance Program initiated by the Boiling Water Reactors Owners' Group in the late 1980s, which contain important surveillance materials for inclusion within the ISP.

The BWRVIP-86 report was submitted to follow up on the material presented in the BWRVIP-78 report by establishing specific guidelines for ISP implementation. The BWRVIP-86 report 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 **during the current operating period. BWRVIP-116 report addresses implementation during the extended period of operation.**

NUREG-1801 Consistency

With enhancements the aging management program for reactor vessel surveillance is consistent with the aging management program XI.M31, "Reactor Vessel Surveillance," described in NUREG-1801.

Enhancements

- The program will provide for performance of reactor beltline material surveillance consistent with BWR ISP, BWRVIP-78, BWRVIP-86**, BWRVIP-116 (upon approval by the NRC)**, the NRC SER for the ISP and associated RAIs.

Enhancements are scheduled for implementation prior to the period of extended operation.

Operating Experience

Material capsules are withdrawn and testing is performed in accordance with the program requirements. Program updating is performed as new industry standards are developed. No adverse conditions have been identified.

Conclusion

The reactor vessel surveillance aging management program provides reasonable assurance that the aging effects of irradiation embrittlement are adequately managed so that the intended functions of reactor vessel components within the scope of license renewal are maintained during the period of extended operation.

The UFSAR supplement description in Section A.1.22 of the license renewal application for Dresden and Quad Cities will be revised to read as follows (corrections underlined in bold type):

(Dresden) A.1.22 Reactor Vessel Surveillance

The reactor vessel surveillance aging management program includes periodic testing of metallurgical surveillance samples to monitor the progress of neutron embrittlement of the reactor pressure vessel as a function of neutron fluence, in accordance with Regulatory Guide (RG) 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2.
Prior to the period of extended operation the program will be consistent with BWRVIP-78, "Integrated Surveillance Program," ~~and~~ BWRVIP-86, "BWR Integrated Surveillance Program Implementation Plan", **and BWRVIP-116, "BWR Vessel and Internals Project Integrated Surveillance Program (ISP) Implementation for License Renewal" (upon approval by the NRC).** The program will ensure coupon availability during the period of extended operation, and provide for saving withdrawn coupons for future reconstitution.

(Quad Cities) A.1.22 Reactor Vessel Surveillance

The reactor vessel surveillance aging management program includes periodic testing of metallurgical surveillance samples to monitor the progress of neutron embrittlement of the reactor pressure vessel as a function of neutron fluence, in accordance with

Regulatory Guide (RG) 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2.
Prior to the period of extended operation the program will be consistent with BWRVIP-78, "Integrated Surveillance Program," ~~and~~ BWRVIP-86, "BWR Integrated Surveillance Program Implementation Plan", **and BWRVIP-116, "BWR Vessel and Internals Project Integrated Surveillance Program (ISP) Implementation for License Renewal" (upon approval by the NRC).** The program will ensure coupon availability during the period of extended operation, and provide for saving withdrawn coupons for future reconstitution.