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JUL 25 2002

U. S. Nuclear Regulatory Commission
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Washington, DC 20555

SUSQUEHANNA STEAM ELECTRIC STATION
PROPOSED AMENDMENT NO. 247 TO UNIT 1
LICENSE NO. NPF-14 AND PROPOSED
AMENDMENT NO. 212 TO UNIT 2
LICENSE NO. NPF-22: REVISION TO THE RPV
MATERIAL SURVEILLANCE PROGRAM
PLA-5498

Docket Nos. 50-387
and 50-388

Reference: 1. *Letter from W. H. Bateman (USNRC) to C. Terry (BWRVIP Chairman) titled, "Safety Evaluation Regarding EPRI Proprietary Report '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.*

2. *Regulatory Issue Summary No. 2002-05, "NRC Approval of Boiling Water Reactor Pressure Vessel Integrated Surveillance Program," dated April 8, 2002.*

The purpose of this letter is to propose changes to the Susquehanna Steam Electric Station Final Safety Analysis Report (Susquehanna SES FSAR) for Unit 1 and Unit 2. This proposed change revises the Reactor Pressure Vessel Material Surveillance Program in accordance with References 1 and 2.

Attachment 1 to this letter is the "Safety Assessment" supporting this change.

Attachment 2 is the No Significant Hazards Considerations evaluation performed in accordance with the criteria of 10 CFR 50.92 and the Environmental Assessment.

Attachment 3 to this letter contains the applicable pages of the Susquehanna SES FSAR for Unit 1 and Unit 2, marked to show the proposed change.

A008

The proposed change has been approved by the Susquehanna SES Plant Operations Review Committee and reviewed by the Susquehanna Review Committee.

Consistent with the process established between the NRC and the BWRVIP, this change is being processed as a license amendment to facilitate NRC review and approval.

PPL plans to implement the proposed changes in the Spring of 2003 to support deletion of work from the Unit 2 11th Refueling and Inspection Outage. Therefore, we request NRC complete its review of this change by December 1, 2002 with the changes effective 30 days after approval.

Any questions regarding this request should be directed to Mr. Cornelius T. Coddington at (610) 774-4019.

Sincerely,

R. L. Anderson

Attachments: (1) Safety Assessment - Revision to the Reactor Pressure Vessel Material Surveillance Program
(2) No Significant Hazards Considerations and Environmental Assessment
(3) Final Safety Analysis Report Mark-Ups

copy: NRC Region I
Mr. D. J. Allard, PA DEP
Mr. T. G. Colburn, NRC Sr. Project Manager
Mr. S. L. Hansell, NRC Sr. Resident Inspector
Mr. R. Janati, DEP/BRP
Mr. E. M. Thomas, NRC Project Manager

BEFORE THE
UNITED STATES NUCLEAR REGULATORY COMMISSION

In the Matter of :

PPL Susquehanna, LLC: Docket No. 50-388

PROPOSED AMENDMENT NO. 212 TO LICENSE NPF-22:
REVISION TO THE REACTOR PRESSURE VESSEL
MATERIAL SURVEILLANCE PROGRAM
UNIT NO. 2

Licensee, PPL Susquehanna, LLC, hereby files a revision to its Facility Operating License No. NPF-22 dated March 23, 1984.

This amendment involves a revision to the Susquehanna SES Final Safety Analysis Report.

PPL Susquehanna, LLC
By:

R. L. Anderson
Vice President - Nuclear Operations

Sworn to and subscribed before me
this 25th day of July, 2002.

Nancy L. Garcia
Notary Public

Notarial Seal
Nancy L. Garcia, Notary Public
Salem Twp., Luzerne County
My Commission Expires May 31, 2003
Member, Pennsylvania Association of Notaries

BEFORE THE
UNITED STATES NUCLEAR REGULATORY COMMISSION

In the Matter of :

PPL Susquehanna, LLC: Docket No. 50-387

PROPOSED AMENDMENT NO. 247 TO LICENSE NPF-14:
REVISION TO THE REACTOR PRESSURE VESSEL
MATERIAL SURVEILLANCE PROGRAM
UNIT NO. 1

Licensee, PPL Susquehanna, LLC, hereby files a revision to its Facility Operating License No. NPF-14 dated July 17, 1982.

This amendment involves a revision to the Susquehanna SES Final Safety Analysis Report Specifications.

PPL Susquehanna, LLC
By:

R. L. Anderson
Vice President - Nuclear Operations

Sworn to and subscribed before me
this 25th day of July, 2002.

Nancy L. Garcia
Notary Public

Notarial Seal
Nancy L. Garcia, Notary Public
Salem Twp., Luzerne County
My Commission Expires May 31, 2003
Member, Pennsylvania Association of Notaries

Attachment 1 to PLA-5498

Safety Assessment

Revision to the Reactor Pressure Vessel Material Surveillance Program

Safety Assessment

Revision to the Reactor Pressure Vessel Material Surveillance Program

The following provides the basis for the proposed revision to the reactor pressure vessel material surveillance program.

1.0 DESCRIPTION OF THE PROPOSED CHANGE

PPL Susquehanna, LLC (PPL) proposes to revise the licensing basis for Susquehanna Steam Electric Station Units 1 and 2 (SSES) by replacing the current plant-specific reactor pressure vessel (RPV) material surveillance program with the Boiling Water Reactor (BWR) Integrated Surveillance Program (ISP), which was approved by the NRC in its Safety Evaluation (SE) dated February 1, 2002 (Reference 1). The proposed revision to the SSES Final Safety Analysis Report reflecting this change is provided for information in Attachment 3.

2.0 REASON FOR THE PROPOSED CHANGE

The BWR ISP was developed in response to an issue raised by the NRC staff 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 Appendix H to 10 CFR 50. 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 the overall quality of the data being used to evaluate BWR RPV embrittlement. 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 (either because they lack adequate unirradiated baseline CVN data or because they are not the best representative materials) will no longer be tested.

3.0 TECHNICAL ANALYSIS

Reference 1 concludes that the proposed ISP, if implemented in accordance with the conditions in the SE, has been determined to be an acceptable alternative to all existing BWR plant-specific RPV surveillance programs for the purpose of maintaining compliance with the requirements of Appendix H to 10 CFR Part 50 through the end of current facility 40 year operating licenses. Reference 1 requires that each licensee (1) provide information regarding what specific neutron fluence methodology will be implemented as part of participation in the ISP and (2) address the neutron fluence methodology compatibility issue as it applies to the comparison of neutron fluences calculated for its RPV versus the neutron fluences calculated for surveillance capsules in the ISP which are designated to represent its RPV. This information is provided in the following discussion.

The SSES Technical Specifications, as discussed in Amendment No. 200 to SSES Unit 1 Operating License (NPF-14) and Amendment No. 174 to SSES Unit 2 Operating License (NPF-22) require that new P-T curves be implemented based on updated fluence calculations by May 1, 2005 and May 1, 2006 (Unit 2 and Unit 1 respectively). See Reference 2 for additional information.

PPL intends to use the BWRVIP RAMA code or other NRC approved methodology to revise the calculations for both Units 1 and 2. The RAMA code will perform a full 3D-neutron transport solution to determine fluence within the vessel. The analysis will use the BUGLE-96 data library as recommended by Regulatory Guide 1.190. It will perform a full uncertainty analysis to determine the accuracy of the calculation.

The current schedule for completion of the BWRVIP RAMA code is December 2002. The BWRVIP intends to submit a topical report on the RAMA code to the NRC for review, with the objective of receiving a safety evaluation in 2003 approving use of the methodology.

The first surveillance capsule to be tested under the ISP is the River Bend 183° capsule. The test report is scheduled to be submitted to the NRC by February 2003. Coincidentally, these capsules, according to the ISP, are the substitute capsules for SSES Unit 2. Thus in accordance with the ISP, the SSES Unit 2 capsule will not be removed and tested.

The ISP requires the Unit 1 surveillance capsules be removed in 2012 and tested in 2013. The Unit 1 fluence calculations will be reevaluated both in 2006 and after this ISP testing.

REFERENCES:

1. Letter from W. H. Bateman (USNRC) to C. Terry (BWRVIP Chairman) titled, "Safety Evaluation Regarding EPRI Proprietary Report '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.

2. Letter from D. S. Collins (USNRC to R. G. Byram (PPL) titled, "Susquehanna Steam Electric Station Units 1 and 2 – Issuance of Amendment RE: Reactor Pressure Vessel Pressure-Temperature Limit Curves," dated February 7, 2002.

Attachment 2 to PLA-5498

No Significant Hazards Considerations and Environmental Assessment

No Significant Hazards Considerations and Environmental Assessment

The Commission has provided standards in 10 CFR 50.92(c) for determining whether a significant hazards consideration exists. A proposed amendment to an operating license for a facility involves 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; (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.

PPL proposes to revise the licensing basis for SSES by replacing the plant-specific RPV material surveillance program with the BWR ISP. This change is acceptable because the BWR ISP has been approved by the NRC staff as meeting the requirements of paragraph III.C of Appendix H to 10 CFR 50 for an integrated surveillance program.

In accordance with the criteria set forth in 10 CFR 50.92, PPL has evaluated the proposed TS change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. ***Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?***

   No. The proposed change implements an integrated surveillance program that has been evaluated by the NRC staff as meeting the requirements of paragraph III.C of Appendix H to 10 CFR 50. Consequently, the proposed change does not significantly increase the probability of any accident previously evaluated. The proposed change provides the same assurance of RPV integrity. As a result, the consequences of any accident previously evaluated are not significantly increased. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. ***Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?***

   No. The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change maintains an equivalent level of RPV material surveillance and does not introduce any new accident initiators. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. ***Does the proposed change involve a significant reduction in a margin of safety?***

No. The proposed change has been evaluated as providing an acceptable alternative to the plant-specific RPV material surveillance program that meets the requirements of the regulations for RPV material surveillance. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

ENVIRONMENTAL CONSIDERATION

10 CFR 51.22(c)(9) identifies certain licensing and regulatory actions, which are eligible for categorical exclusion from the requirement to perform an environmental assessment. A proposed amendment to an operating license for a facility does not require an environmental assessment if operation of the facility in accordance with the proposed amendment would not (1) involve a significant hazards consideration; (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite; or (3) result in a significant increase in individual or cumulative occupational radiation exposure. PPL has evaluated the proposed change and has determined that the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Accordingly, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with issuance of the amendment. The basis for this determination, using the above criteria, follows:

Basis

1. As demonstrated in the No Significant Hazards Consideration Evaluation, the proposed amendment does not involve a significant hazards consideration.

2. There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite. The proposed change does not involve any physical alteration of the plant (no new or different type of equipment will be installed) or change in methods governing normal plant operation.

There is no significant increase in individual or cumulative occupational radiation exposure. The proposed change does not involve any physical alteration of the plant (no new or different type of equipment will be installed) or change in methods governing normal plant operation.

Attachment 3 to PLA-5498

Final Safety Analysis Report Mark-Ups

5.3.1.5.1.7 Reactor Vessel Annealing

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

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 program are selected 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 which simulates the actual heat treatment performed on the core region shell plates of the completed vessel.

The surveillance program includes three capsule holders per reactor vessel. Charpy impact specimens for the reactor vessel surveillance programs are of the longitudinal orientation consistent with the ASME requirements prior to the issuance of the Summer 1972 Addenda and ASTM-E-185-82. Based on GE experience, the amount of shift measured by these irradiated longitudinal test specimens will be essentially the same as the shift in an equivalent transverse specimen.

The program for implementation of the scheduling and testing of the surveillance specimens is governed and controlled by BWRVIP-86, BWR Integrated Surveillance Program (ISP) Implementation Plan. The Unit 1 second holder (131C7717G2) will be pulled in accordance with the schedule in BWRVIP-86. For Unit 2, all the information will come from other plants in BWRVIP-86 ISP Program. No capsules are scheduled to be withdrawn from Unit 2. Other plants will remove and test specimens in accordance with BWRVIP-86. The results from these tests will provide the necessary data to monitor embrittlement for Unit 2. Since the predicted adjusted reference temperature of the reactor vessel beltline steel is less than 100°F at end of life, the use of the capsules per BWRVIP-86 meets the requirements of 10 CFR 50, Appendix H, and ASTM-E-185-82. The withdrawal schedule and other requirements are provided in BWRVIP-86.

For the extent of compliance to 10 CFR 50, Appendix H, see Tables 5.3-1 b and 5.3-2b.

Each holder is loaded with capsules which contain the following surveillance specimens and dosimeter wires:

First holder (131 C7717G3):

36 Charpy impact specimens including 12 base metal, 12 weld metal, and 12 heat affected zone metal specimens; 10 tensile specimens including 3 base metal, 4 weld metal, and 3 weld heat affected zone metal specimens; 9 metal wire dosimeters including 3 iron, 3 nickel, and 3 copper.

After the first capsule holders (for both Units 1 and 2) were withdrawn and the specimens tested (see references 5.3-4 and 5.3-5), the broken specimens were remachined as miniature specimens and reloaded in the vessels during the next refueling outages. The contents of the new "reconstituted" capsules (for both Units 1 and 2) are as follows:

2 Charpy specimen packets each containing 12 Charpy specimens - 1 packet for base metal specimens and 1 for weld metal specimens. (EXCEPTION: The Unit 1 weld metal capsule only has 11 specimens).

Copper, Iron and Niobium flux wires are included in the capsules with the Charpy specimens.

2 tensile specimen tubes - 1 containing one tensile capsule with four 0.113 inch diameter miniature tensile specimens, the other containing 1 capsule with one 0.113 inch diameter miniature tensile specimen and one 0.250 inch diameter original weld metal tensile specimen.

The new holders have the same geometry as the original capsule holders.

Second holder (131 C7717G2):

24 Charpy impact specimens including 8 base metal, 8 weld metal, and 8 weld heat affected zone metal specimens; 8 tensile specimens including 3 base metal, 3 weld metal, and 2 weld heat affected zone metal specimens; 6 metal wire dosimeters including 2 iron, 2 nickel, and 2 copper.

Third holder (131 C7717G1):

24 Charpy impact specimens including 8 base metal, 8 weld metal and 8 weld heat affected zone metal specimens; 6 tensile specimens including 2 base metal, 2 weld metal, and 2 weld heat affected zone metal specimens; 6 metal wire dosimeters including 2 iron, 2 nickel, and 2 copper.

A set of out-of-reactor baseline Charpy V-notch specimens is provided with the surveillance test specimens.

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~~Charpy impact specimens for the reactor vessel surveillance programs are of the longitudinal orientation consistent with the ASME requirements prior to the issuance of the Summer 1972 Addenda and ASTM E 185-73. Based on GE experience, the amount of shift measured by these irradiated longitudinal test specimens will be essentially the same as the shift in an equivalent transverse specimen.~~

~~The program includes three capsules in the reactor. Since the predicted adjusted reference temperature of the reactor vessel beltline steel is less than 100°F at end of life, the use of three capsules meets the requirements of 10 CFR 50, Appendix H, and ASTM E 185-73. The withdrawal schedule is provided in Table 5.3-3.~~
~~For the extent of compliance to 10 CFR 50, Appendix H, see Tables 5.3-1b and 5.3-2b.~~

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 4.3.2.8.

5.3.1.6.3 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 Section III of the ASME Code. A positive spring-loaded locking device is provided to retain the capsules in position throughout any anticipated event during the lifetime of the vessel.

5.3.1.6.4 Time and Number of Dosimetry Measurements

GE has provided neutron dosimetry wires in each of the specimen holders. In addition, one holder in each vessel is designed with a separately removable dosimeter, to be removed after one fuel cycle. The first cycle dosimeter was removed from Unit 1 in 1986 and analyzed. A first cycle dosimeter was not available for removal from Unit 2. However, the first cycle dosimetry for Unit 1 provides a good estimate of flux for Unit 2, because vessel geometries and core power shapes are very similar.

The first cycle dosimetry provides a means of calibrating the flux distribution calculations to actual vessel conditions. Dosimetry will be updated as holders are removed and tested. The holder withdrawal schedule is listed in Table 5.3-3.

5.3.1.7 Reactor Vessel Fasteners

5.3.4 References

5.3-1 Faynshtein, K., and D. R. Pankratz, "Power Uprate Engineering Report for Susquehanna Steam Electric Station, Units 1 and 2," General Electric Report NEDC-32161P, as revised by PP&L Calculation EC-PUPC-1001, Revision 0, March, 1994.

5.3-2 Carey, R. G., "Susquehanna Steam Electric Station Unit 1 Vessel Surveillance Materials Testing and Fracture Toughness Analysis," General Electric Report GE-NE-523-169-1292, Revision 1, October, 1993. Attached to PP&L letter PLA-3953, R. G. Byram to C. L. Miller, NRC, "Susquehanna Steam Electric Station, Submittal of Reactor Vessel Material Surveillance Test Report per 10CFR50 Appendix H for Unit 1," April 8, 1993.

5.3-3 Contreras, G. W., "Susquehanna Steam Electric Station Unit 2 Vessel Surveillance Materials Testing and Fracture Toughness Analysis," General Electric Report GE-NE-523-107-0893, Revision 1, October, 1993. Attached to PP&L Letter PLA-4126, R. G. Byram to C. L. Miller, NRC, "Susquehanna Steam Electric Station, Submittal of Revision to Reactor Vessel Material Surveillance Test Report per 10CFR50 Appendix H for Unit 2," May 19, 1994.

5.3-4 DuBord, R.M., "Susquehanna Steam Electric Station Unit 1 Fabrication of New Surveillance Capsule with Reconstituted Charpy Specimens" General Electric Nuclear Energy report GE-NE-523-A054-0595, May 1995.

5.3-5 DuBord, R.M., "Susquehanna Steam Electric Station Unit 2 Fabrication of New Surveillance Capsule with Reconstituted Charpy Specimens" General Electric Nuclear Energy report GE-NE-523-A055-0595, May 1995.

5.3-6 Structural Integrity Associates Report No. SIR-00-167, Revision 0, "Revised Pressure-Temperature Curves for Susquehanna Units 1 and 2", January 2001.

5.3-7 BWRVIP-86: BWR Vessel Internals Project, BWR Integrated Surveillance Program Implementation Plan, February 2002, including the latest revisions.

TABLE 5.3-1b

APPENDIX H MATRIX FOR SUSQUEHANNA SES UNIT 1

APPENDIX H PARA. NO.	TOPIC	COMPLY YES/NO OR N.A.	ALTERNATE ACTIONS OR COMMENTS
I	Introduction	N/A	
II.A	Fluence $<10^{17}$ n/cm^2 – Surveillance Program Not Required	N/A	
II.B	Standards Requirements (ASTM) for Surveillance	No	Noncompliance with ASTM E185-82in 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 Shall be Taken from Locations Alongside the Fracture Test Specimens (Section III.B of Appendix G)	No	Noncompliance in that specimens may not have necessarily been taken from along-side specimens required by Section III of Appendix G and transverse CVNs may not be employed. However, representative materials have been 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 attachment of brackets to vessel cladding. See Section 5.3.1.6.4.
II.C.3.a	Withdrawal Schedule of Capsules, $RT_{NDT}<100°F$	Yes	~~Three capsules planned.~~ Starting RT_{NDT} of limiting material is based on alternative action (see Paragraph III.A of Appendix G). One capsule complete. Other capsules are scheduled and tested in accordance with Reference 5.3-7.
II.C.3.b	Withdrawal Schedule of Capsules, $RT_{NDT}<200°F$	N/A	
II.C.3.c.	Withdrawal Schedule of Capsules, $RT_{NDT}<200°F$	N/A	
III.A	Fracture Toughness Testing Requirements of Specimens	Yes	See Section 5.3.1.6
III.B	Method of Determining Adjusted Reference Temp. for Base Metal, HAZ and Weld Metal	Yes	See Section 5.3.1.5
IV.A	Reporting Requirements of Test Results	Yes	See Section 5.3.1.6
IV.B	Requirement for Dosimetry Measurement	Yes	See Section 5.3.1.6.2, 5.3.1.6.4
IV.C	Reporting Requirements of Press/Temp. Limits	Yes	See Section 5.3.2

TABLE 5.3-3

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM-WITHDRAWAL SCHEDULE

Specimen Holder	Vessel Location	Lead Factor *	Withdrawal Time (EFPY)#
UNIT 1			
131C7717G1	300°	1.20	Spare
131C7717G2	120°	1.20	~~15~~22
131C7717G3	30°	1.20	6 (Actual Date - Spring 1992)
G3 Reconstituted Specimens	30°	1.20	Spare
UNIT 2			
131C7717G1	300°	1.20	Spare
131C7717G2	120°	1.20	~~15~~ Spare
131C7717G3	30°	1.20	6 (Actual Date - Fall 1992)
G3 Reconstituted Specimens	30°	1.20	Spare

* At 1/4 T.

~~Note: The Unit 1 surveillance specimens at the 30° location were removed from the vessel for testing during the Spring 1992 inspection outage and these specimens were reconstituted and replaced back into the vessel 30° location during the Fall 1993 inspection outage (U1-7RIO). The Unit 2 surveillance specimens were removed from the vessel 30° location for testing during the Fall 1992 inspection outage and these specimens were reconstituted and replaced back into the vessel 30° location during the Spring 1994 inspection outage (U2-6RIO). Details of the reconstitution process and the capsule contents can be found in Reference 5.3-4 and 5.3-5.~~

Withdrawal Time is in accordance with Reference 5.3-7.

TABLE 5.3-2b

APPENDIX H MATRIX FOR SUSQUEHANNA SES UNIT 2

APPENDIX H PARA. NO.	TOPIC	COMPLY YES/NO OR N.A.	ALTERNATE ACTIONS OR COMMENTS
I	Introduction	N/A	
II.A	Fluence $<10^{17}$ n/cm^2 – Surveillance Program Not Required	N/A	
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 Shall be Taken from Locations Alongside the Fracture Test Specimens (Section III.B of Appendix G)	No	Noncompliance in that specimens may not have necessarily been taken from alongside specimens required by Section III of Appendix G and transverse CVN's may not be employed. However, representative materials have been 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 attachment of brackets to vessel cladding. See Subsections 5.3.1.6.4.
II.C.3.a	Withdrawal Schedule of Capsules, $RT_{NDT}<100°F$	Yes	~~Three capsules planned.~~ Starting RT_{NDT} of limiting material is based on alternative action (see Paragraph III.A of Appendix G). One capsule completed. Other capsules are scheduled and tested in accordance with Reference 5.3-7.
II.C.3.b	Withdrawal Schedule of Capsules, $RT_{NDT}<200°$	N/A	
II.C.3.c	Withdrawal Schedule of Capsules, $RT_{NDT}<200°$	N/A	
III.A	Fracture Toughness Testing Requirements of Specimens	Yes	See Section 5.3.1.6
III.B	Method of Determining Adjusted Reference Temp. for Base Metal, HAZ and Weld Metal	Yes	See Section 5.3.1.5
IV.A	Reporting Requirements of Test Results	Yes	See Section 5.3.1.6
IV.B	Requirement for Dosimetry Measurement	Yes	See Section 5.3.1.6.2, 5.3.1.6.4
IV.C	Reporting Requirements of Press/Temp. Limits	Yes	See Section 5.3.2