Omaha Public Power District

444 South 16th Street Mall
Omaha NE 68102-2247

November 8, 2001
LIC-01-0107

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

References:
1. Docket No. 50-285
2. Letter from Nuclear Regulatory Commission (A. B. Wang) to Omaha Public Power District (S. K. Gambhir), dated June 6, 2001, "Fort Calhoun Station, Unit No. 1 - Issuance of Amendment - Deletion of Section 3.D, License Term," (TAC No. MA9690) (NRC-01-058)

SUBJECT: Reactor Vessel Surveillance Capsule Removal Schedule Change Request

Pursuant to 10 CFR 50 Appendix H Section II.B.3, Omaha Public Power District (OPPD) requests NRC staff authorization to change the schedule for removal of reactor vessel surveillance capsules associated with the Fort Calhoun Station (FCS) Reactor Vessel Integrity (RVI) Program.

OPPD is currently preparing an application for renewal of the FCS operating license, which when approved will extend the license term to August 2033. The current surveillance capsule removal schedule, the proposed schedule modifications to reflect the extended license term, and the technical basis for the schedule modification are contained in the attached Reactor Vessel Surveillance Program Withdrawal Schedule Modifications (WCAP-15741, Rev. 00)

The optimized capsule removal schedule is presented to yield data assuring that the vessel meets the requirements of Regulatory Guide 1.99, Revision 02, position 2.1 analysis. Consistent with Reference 2, OPPD will continue to evaluate applicable surveillance data from other reactor vessels to ensure that the conclusions of Reference 2 remain valid.

The specific objectives of the FCS RVI Program are stated on page 4 of the Attachment. It should be noted that implementation of a Pressure and Temperature Limits Report (PTLR), objective 5, is not being requested for approval at this time. OPPD currently plans to make an additional PTLR submittal after the spring 2002 refueling outage.

OPPD requests approval of the proposed schedule modifications before April 1, 2002, in order to allow OPPD to use the modified schedule during the spring 2002 refueling outage, scheduled to begin May 4, 2002.

A008

If you have any questions, please call me at (402) 533-7210.

R. L. Phelps 11-8-01

R. L. Phelps
Division Manager
Nuclear Engineering

RLP/RLJ/rlj

Attachment

c: E. W. Merschoff, NRC Regional Administrator, Region IV
A. B. Wang, NRC Project Manager
W. C. Walker, NRC Senior Resident Inspector
Winston & Strawn

Attachment

to

LIC-00-0107

Reactor Vessel Surveillance Program Withdrawal Schedule Modifications

WCAP-15741 Rev. 00

WCAP-15741
Rev. 00

Reactor Vessel Surveillance Program Withdrawal Schedule Modifications

Fort Calhoun Station
Omaha Public Power District

September 2001

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WCAP-15741, Rev. 00

Reactor Vessel Surveillance Program Withdrawal Schedule Modifications

September 2001

VERIFICATION STATUS: COMPLETE

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1. Introduction and Objective

Neutron irradiation of the reactor vessel (RV) results in embrittlement of the plate and weld materials, as manifested by a loss of ductility and fracture toughness. The reactor vessel may be operated in accordance with 10CFR50.61 (Ref. 6.2) and 10CFR50.60/Appendix G, Fracture Toughness Requirements (Refs. 6.1 and 6.3), provided that the margin of safety defined by regulatory requirements is maintained. The major concern with embrittlement is for extreme accidents, such as a large main steamline break or a small-break loss of coolant accident (LOCA), followed by cold repressurization. Hypothetically, as a result of stresses induced in the vessel during the associated rapid cool-down and cold repressurization of the RCS, the reactor vessel could initiate a crack that might extend through-wall. The worst case consequence is the loss of integrity, specifically the inability to maintain any core cooling inventory in the RV which could result in severe core damage.

The Fort Calhoun Station (FCS) reactor vessel was fabricated in 1967-1969 by Combustion Engineering using state-of-the-art methods and materials. The weld wire heats used in the fabrication of the vessel contain high nickel and copper content. These elements are now known to result in an increased rate of embrittlement of the reactor vessel materials. The FCS plate material has a low copper content. Thus the reactor vessel materials subject to the highest projected embrittlement are the welds. The most limiting reactor vessel materials are the 3-410 A, B and C axial welds located at 60° (A), 180° (B), and 300° (C). Due to the fuel management used, the two welds at 60° and 300°, are the most limiting of the three lower shell course axial welds. These 3-410A-C welds cover approximately the lower 40% of the active core height.

The primary reactor vessel integrity (RVI) issues include the following aspects:

1. Pressurized Thermal Shock (PTS) screening criteria per 10CFR50.61 (Ref. 6.2)
2. The RV Surveillance Program per 10CFR50, Appendix H (Ref. 6.4)
3. Heat-up and cool-down limits per 10CFR50, Appendix G (Ref. 6.3)
4. Low Temperature Overpressure Protection (LTOP) per Standard Review Plan (NUREG-0800) 5.2.2 (Ref. 6.5) and Branch Technical Position RSB 5-2 (Ref. 6.6)
5. Upper Shelf Energy screening criteria per 10CFR50, Appendix G (Ref. 6.3)

Significant efforts have been expended in order to prevent FCS from reaching regulatory limits associated with reactor vessel operations. This includes efforts to minimize the fast neutron flux to the limiting welds, to research fabrication records and determine the most representative data associated with the FCS reactor vessel materials, and to apply surveillance data from FCS and other plants. The principle goal has been to safely operate the FCS reactor vessel to the end of licensed life of the plant, which includes both the current license date of August 9, 2013 and the planned license renewal date of August 9, 2033. This entails the demonstration that the reactor vessel will not exceed the regulatory screening criteria limits as defined in 10CFR50.61 and 10CFR50.60/Appendix G.

The specific objectives of the FCS RVI Program are to:

1. Ensure that the FCS RV may be operated until at least August 9, 2033 (License Renewal) without exceeding the regulatory limits associated with reactor vessel operation. Presently these limits are defined in 10CFR50.60/Appendix G and 10CFR50.61.

2. Maintain a reactor vessel surveillance program (10CFR50 Appendix H) that provides adequate monitoring of the fluence and level of embrittlement of the FCS reactor vessel. This includes use of data from other surveillance programs containing the limiting weld material for FCS and application of Regulatory Guide 1.99, Rev. 02, Position 2.1 (Ref. 6.7) to satisfy the requirements of 10CFR50.60/Appendix G and 10CFR50.61. It also includes developing and maintaining an optimized 10CFR50, Appendix H Surveillance Program that complies with ASTM E 185-82 (Ref. 6.10) for removal and testing of surveillance capsules. (Note that the FCS Surveillance Program was designed to ASTM E 185-66 and contains RV materials consistent with this version of the Standard.)

3. Ensure that sufficient operating margins exist to adequately maneuver the plant during heat-up and cool-down cycles (10CFR50 Appendix G), as reflected in the P-T limits and LTOP set points.

4. Maintain a fluence analysis necessary to support the above objectives. The methodology used shall be consistent with regulatory expectations (e.g., Regulatory Guide 1.190, Ref. 6.8 using the ENDF/B-VI cross-section library).

5. Develop, implement, and maintain a NRC-approved Pressure-Temperature Limits Report (PTLR), which is similar to the Core Operating Limits Report (COLR). The PTLR will remove the P-T limits (see #3 above) from the Technical Specifications and relocate them in the Technical Data Book, similar to the COLR. This will be done in accordance with Reference 6.9. (CEOG Task 1174 provides the topical report for implementation.)

The goal of this report is to define an optimized reactor vessel surveillance program that, through its maintenance, will ensure surveillance coverage through the end of License Renewal. This also means that the optimum number of surveillance capsules will be removed. For example, only those capsules necessary to comply with Appendix H will be removed. Conversely, the W-275 supplemental capsule removal schedule will be optimized to yield data that will best support the FCS RVI initiatives (e.g., Regulatory Guide 1.99, Revision 02, Position 2.1 analysis).

2.0 Background and Regulatory Requirements

The FCS RV was fabricated by Combustion Engineering between 1967 and 1969. For the RV welds that are considered with respect to PTS and P-T/LTOP, the following table summarizes some of the fabrication history information:

Table 2-1 FCS Reactor Vessel Weld Fabrication Information

Weld Number	Type	Heat Number(s)	Flux Lot	Fabrication Dates
1-410[1]	Axial	27204	3714	10/67 to 11/67
2-410	Axial	51989	3687	9/67 to 10/67
3-410	Axial	12008,13253, and 27204	3774	2/68 to 3/68
8-410	Circum.	13253	3774	3/68
9-410	Circum.	20291	3833	1/69

Note 1. Nozzle dropout from weld 1-410B (Heat 27204) used in W-275S and other plant's supplemental capsules.

The Reactor Vessel Surveillance Program consists of two elements for FCS. The first is the FCS Surveillance Program (Ref. 6.21 and 6.22) as required by 10CFR50, Appendix H, including the supplemental capsules. The second is the so-called "Integrated Surveillance Program" used in CEN-636, Revision 2 (Ref. 6.13, Attachment 5) to apply data originating from the surveillance programs at other plants. This was done in accordance with 10CFR50.61 and Regulatory Guide 1.99, Position 2.1, and it was approved by the NRC in Reference 6.27. Use of these other data was beneficial because the FCS Surveillance Program did not originally include material from any of the welds described in Table 2-1. (The FCS Surveillance Program was designed in accordance with ASTM E 185-66. At that time, the surveillance weld was only required to be representative of the beltline welds, and specimens were obtained from weld wire heat 305414. The present criterion, established first in ASTM E185-73, is to include in the surveillance program a weld made from the same heat as the limiting reactor vessel weld material.)

Consideration of reactor vessel integrity issues has expanded since the FCS reactor vessel was designed and fabricated. In 1980 vessel integrity issues were heightened (Ref. 6.11) with the advent of Pressurized Thermal Shock (PTS) concerns. The benefit to FCS of an integrated surveillance program came with the issue of Regulatory Guide 1.99, Revision 2 (Ref. 6.7) and the CEOG approach for using surveillance data from other vessels fabricated by Combustion Engineering (Ref. 6.12). Consideration of such data for the evaluation of PTS risk became a requirement in the 1995 amendment to 10CFR50.61 (Ref. 6.2).

3.0 Current Surveillance Program and Capsule Withdrawal Schedule

All of the FCS Reactor Vessel Integrity Program objectives relate to the current surveillance program. Those objectives are 1) to ensure that the FCS RV may be operated at least through the end of the current license period, August 9, 2013, as well as through August 9, 2033 for license renewal, 2) to maintain a reactor vessel surveillance program in accordance with 10CFR50 Appendix H (including use of data from other surveillance programs), 3) to ensure that sufficient operating margins exist in accordance with 10CFR50 Appendix G, 4) to maintain a fluence analysis necessary to support the above objectives, and 5) to develop, implement, and maintain a NRC-approved Pressure-Temperature Limits Report (PTLR). This report defines an optimized reactor vessel surveillance program in Section 3.1 that provides for monitoring vessel embrittlement and neutron fluence accumulation in accordance with the second and the fourth objectives. Furthermore it provides for the license renewal period in accordance with the second objective. Through assessment of the W-275S (supplemental) capsule and surveillance capsules from other plants, additional data will become available to address these objectives. The integrated surveillance program is described in Section 3.2.

3.1 10CFR50 Appendix H, Reactor Vessel Surveillance Program

The FCS Surveillance Program is described in Section 4.5 of the FCS USAR. The original program was designed and fabricated based on ASTM E185-66 (Ref. 6.10) as documented in References 6.21 and 6.22. Six surveillance capsules were fabricated and installed in holders attached to the vessel wall at the 45, 85, 95, 225, 265 and 275-degree azimuthal locations. Each capsule contained test specimens from one of the FCS reactor vessel plates and from a weld fabricated using heat 305414 with Linde 1092 flux.

Three replacement surveillance capsules were fabricated and installed into vacated holders in the FCS reactor vessel. Capsules W-225S and W-265S were fabricated in 1983 as documented in Reference 6.23 and installed during the EOC 7 Refueling Outage. Each capsule contained Charpy impact, fracture toughness and tensile specimens from the FCS surveillance program weld (i.e., fabricated using heat 305414). Capsule W-275S was fabricated in 1993 in accordance with Reference 6.24 and installed during the EOC 14 Refueling Outage (approximately October 1993). It contained Charpy impact specimens from the FCS nozzle dropout weld seam 1-410B (fabricated using heat 27204) and from the Maine Yankee nozzle dropout weld (fabricated using heats 12008 and 13253).

The surveillance capsule withdrawal schedule is summarized below in Table 3-1. This is the schedule as currently described in the FCS USAR. The exposure to the vessel is given in terms of effective full power years assuming an 80% load factor. The capsule location is given as the azimuthal location of the capsule. "W" refers to the vessel wall position, and "S" denotes those supplemental capsules that replaced the ones already evaluated. W-265S is designated as a Standby capsule indicating there was no plan to remove it for evaluation. The future plans to use W-275S are discussed in Section 4.

Table 3-1 FCS Surveillance Capsule Withdrawal Schedule

Removal Sequence	Exposure[1] of Vessel	End of Fuel Cycle	Capsule Location
1	2.5	3	W-225[2]
2	5.9	7	W-265[2]
3	13.6	14	W-275[2]
4	20	20	W-45
5	21	20	W-85
6	27	25	W-95
7	32	29	W-225S
8	Standby	-	W-265S
9	Standby	-	W-275S

1) Exposure in effective-full-power years.
2) Capsule removed and tested.

Capsules W-225, W-265, and W-275 were removed and evaluated as reported in References 6.15, 6.16 and 6.17. Capsules W-225S, W-265S, and W-275S were installed to replace them. The surveillance capsule removal schedule in Table 3-1 was intended to meet the requirements of ASTM E 185-82 and 10CFR50, Appendix H. The 1995 version of Appendix H requires that the withdrawal schedule (and the design of the program) comply with the edition of ASTM E 185 in effect for the ASME Code year to which the vessel was built, up through ASTM E 185-82. That was the 1966 edition of the standard for Fort Calhoun Station. ASTM E 185-66 provides that: "It is recommended that sets of specimens be withdrawn at three or more separate times. One of the data points obtained shall correspond to the neutron exposure of the component near the end of its design life." ASTM E 185-82 provides that five capsules be scheduled when the predicted transition temperature shift is 200° F or greater. The third and fourth withdrawals are to coincide with the end-of-life fluence at the 1/4 t and vessel inside surface, respectively. The fifth capsule, if tested, is to be withdrawn after receiving a neutron fluence equal to but not more than twice the end-of-life fluence at the vessel inside surface. Therefore, the schedule in Table 3-1 exceeds current requirements for surveillance through the year 2013. Recommended modifications to meet the FCS reactor vessel integrity objectives are discussed in Section 4.

3.2 Integrated Surveillance Program

3.2.1 Integrated Surveillance Requirements

An integrated surveillance program is intended to provide reactor pressure vessel irradiation data from another reactor vessel surveillance program. 10CFR50, Appendix H establishes specific rules for an integrated surveillance program. Those rules encompass features such as the similarity of the irradiation environment and provisions for data sharing. (The latter is not relevant to Fort Calhoun.) The issue of similarity of the irradiation environment and relevance to plant specific materials of the data obtained has been addressed in CEN-405-P (Ref. 6.12). An application of the integrated surveillance approach to FCS is given in CEN-636, Revision 2 (Ref. 6.13, Attachment 5). Use of this report was approved by the NRC in Reference 6.27. CEN-636, Revision 2 utilized data originating from the surveillance programs at Mihama 1, Diablo Canyon 1, Palisades, D.C. Cook 1, and Salem 2 to refine the estimates of RT_{PTS} for the FCS reactor vessel. It provided the basis for an on-going integrated surveillance program for FCS.

The FCS integrated surveillance program is intended to fulfil the requirement of 10CFR50.61, (c)(2) to:

> "...consider plant-specific information that could affect the level of embrittlement. This information includes but is not limited to the reactor vessel operating temperature and any related surveillance program results."

It was necessary to utilize the surveillance data from the plants cited above because the FCS Surveillance Program did not originally include weld material from any of the materials described in Table 2-1. (The original surveillance capsules and two of the replacement capsules contain specimens from weld wire heat 305414.) The other surveillance programs, including the critical weld materials and the number of capsules, are described in the following paragraphs.

3.2.2 Other Surveillance Program Descriptions

Mihama Unit 1- The surveillance program for Mihama 1 includes a weldment made using heats 12008 and 27204. (The materials and post-irradiation test results are described in CE NPSD-1204-P, Ref. 6.14.) The weldment corresponds to two of the three heats used in the FCS lower shell course axial weld seams. Three capsules have been removed and evaluated. The evaluation as applied to FCS is reported in CEN-636, Revision 2.

The fourth capsule is scheduled for removal in 2001 with results expected in 2002. The fifth capsule is scheduled for removal in 2010 with results expected in 2011. The sixth capsule is designated for Standby with no scheduled removal date.

Diablo Canyon 1- The surveillance program for Diablo Canyon 1 includes a weldment made using heat 27204. The materials and post-irradiation test results are described in WCAP-11567 (Capsule DC1-S, December 1987, Ref. 6.18) and in WCAP-13750 (Capsule DC1-Y, July 1993, 6.19).

Supplemental capsules for Diablo Canyon 1 were assembled using reconstituted specimens from capsule DC1-Y and weld specimens from the FCS nozzle drop-out weld 1-410 B, heat 27204. Those capsules are described in Reference 6.20 (WCAP-13440, December 1992). Two capsules containing weld specimens have been removed and evaluated. The DC1-V capsule is scheduled for removal in May 2002 with results expected in 2003. The DC1-C (supplemental) capsule is scheduled for removal in October 2005 with results expected in 2006. Three supplemental capsules (A, B and C) assembled using weld specimens from the FCS nozzle dropout were also installed, but there is no designated removal date.

Palisades Supplemental Capsules- The Palisades supplemental surveillance capsules include a weldment made using heat 27204. (See Ref. 6.13, Attachment 5.) The weld metal specimens were obtained from the FCS nozzle dropout weld 1-410 B. The status of the Palisades capsules is:

a) Capsule SA-60-1 was pulled and evaluation data are found in a preliminary report ATI-99-006-002 (8/4/99). The formal capsule report is expected to be issued in 2001.

b) Capsule SA-240-1 was pulled and has been evaluated. The capsule report is expected to be issued in 2002.

FCS W-275S Capsule- The FCS supplemental capsule W-275S includes one weldment made using heat 27204, and a second made using heats 12008 and 13253. (Heat 27204 is from the FCS nozzle drop-out including weld 1-410B. Heat 12008 and 13253 is from the Maine Yankee nozzle drop-out including weld 1-203.) This capsule was fabricated in 1993 (see Ref. 6.24) and installed in October 1993. The capsule contains 24 Charpy V-notch specimens from each weldment and nine uniaxial tension test specimens (a total of nine split between the two heats).

Other Capsules- Data originating from the surveillance programs at D.C. Cook 1 and Salem 2 were examined as part of projections of RT_{PTS} for the FCS reactor vessel (see Ref. 6.13, Attachment 5). The weldments in those surveillance programs were fabricated using wire heat 13253. This heat is not one of the limiting materials in the FCS vessel and the assessment of those surveillance data confirmed that fact. As part of the Integrated Surveillance Program for FCS, the weld results from future Mihama Unit 1, Diablo Canyon Unit 1 and Palisades capsules will be monitored to ascertain that the assessment remains valid (Ref. 6.27).

4. Schedule Modifications

4.1 Needs for Surveillance Capsules and Associated Bases

10CFR50, Appendix H Requirements for Current Design Life- The surveillance capsule removal schedule in Table 3-1 is from the FCS USAR. It was prepared to meet the requirements of 10CFR50, Appendix H for operation through the end of the current design life. The current design life is for forty years of operation (approximately 32 effective full power years at an 85% capacity factor) and goes through 2013. Earlier versions of Appendix H provided requirements for capsule withdrawal. The 1995 version of Appendix H requires that the withdrawal schedule comply with the edition of ASTM E 185 in effect for the ASME Code year to which the vessel was built. For FCS, that was the 1966 version of the standard, ASTM E 185-66. Appendix H also allows compliance with the editions of ASTM E 185 up through ASTM E 185-82. The schedule in Table 3-1 reflects the requirements from a combination of the preceding standards and regulations.

The surveillance capsule withdrawal requirements of ASTM E 185-66 are as follows:

"It is recommended that sets of specimens be withdrawn at three or more separate times. One of the data points obtained shall correspond to the neutron exposure of the component near the end of its design life."

According to these requirements, three or more capsules should be scheduled for withdrawal. The original surveillance program for FCS scheduled all six surveillance capsules for withdrawal such that the withdrawal requirements of ASTM E 185-66 were exceeded. The version of 10CFR50, Appendix H issued in July 1973 required that five capsules be provided for withdrawal, including one for standby, in cases where the adjusted reference temperature was expected to exceed 200° F during the service life. The first capsule was to be withdrawn when the predicted shift was approximately 50° F or at one-fourth service life, whichever is earlier. The fourth capsule was to be removed at three-fourths service life. The second and third capsules were to be removed at approximately one-third and two-thirds the time interval between the first and fourth capsules.

As indicated above, later versions of Appendix H were revised to cite the withdrawal schedule in ASTM 185 that is current on the issue date of the ASME Code to which the reactor vessel was purchased, but including only those editions through 1982. The ASTM 185-82 edition of the standard practice specifies that five capsules be provided in cases where the predicted transition temperature was expected to exceed 200° F at the vessel inside surface during the service life. Four or more capsules are to be tested with the third and fourth withdrawals to coincide with the end-of-life fluence at the 1/4 t and vessel inside surface, respectively. The fifth capsule, if tested, is to be withdrawn after receiving a neutron fluence equal to but not more than twice the end-of-life fluence at the vessel inside surface. Therefore, five capsules should be provided, and at least four should be withdrawn and tested. The sixth capsule could be designated for standby.

The actual schedule followed for the three FCS capsules removed to date was to withdraw the capsules after 2.5, 5.9 and 13.6 effective full power years (EFPY). The W-225 capsule provided data at a fluence that is consistent with the ASTM E 185-82 requirement; that was for a fluence of approximately 5E18 n/cm^2. The timing of the W-275 capsule withdrawal was also consistent with ASTM E 185-82. It was withdrawn at a time corresponding approximately to the end-of-life fluence at the quarter-thickness location in the vessel. The W-265 capsule was withdrawn at a time when the fluence was mid-way between that of the first and the third capsule, which is also consistent with the ASTM E 185-82 requirement. Two more capsules are to be scheduled for withdrawal per ASTM E 185-82. The fifth capsule should have attained at least once but not greater than twice the end-of-life fluence at the inner wall of the vessel, which is approximately 1.7E19 n/cm^2 or greater. That would correspond to approximately 27 to 32 EFPY for the lower lead factor capsules and 20 to 32 EFPY for the higher lead factor capsules. The fourth capsule should have attained the end-of-life fluence at the inner wall of the vessel. That corresponds to approximately 27 EFPY for the lower lead factor capsules and 20 EFPY for the higher lead factor capsules. The remaining capsules could be designated for standby.

When the integrated surveillance data are taken into consideration, the ASTM E 185-82 requirements for all five capsule have already been met. The three tested surveillance capsules from Mihama Unit 1, the one tested supplemental capsule from Palisades, and the two tested surveillance capsules from Diablo Canyon Unit 1 provide data on three of the limiting welds from Fort Calhoun. The data encompass the range of neutron fluences that is specified in ASTM E 185-82. On that basis, the FCS surveillance program and the integrated surveillance program in combination have satisfied the requirements even if no additional capsule were tested. (As noted in 3.2.2, the weld results from future Mihama Unit 1, Diablo Canyon Unit 1 and Palisades capsules will be monitored as part of the integrated surveillance program for FCS. Furthermore, as will be noted in Table 4.1, additional capsule withdrawals are planned for FCS that will provide for continued vessel surveillance.)

<u>10CFR50, Appendix H Requirements for License Renewal Period</u>- The surveillance capsule removal schedule for FCS for sixty years of operation (i.e., operation through 2033) would be based on criteria similar to that of the current design life. Those criteria would be the requirements of ASTM E 185-82 and 10CFR50, Appendix H. In accordance with those criteria, the schedule would change as described below.

For the first three FCS capsules, the schedule followed was in accordance with the requirements of ASTM E 185-82 and 10CFR50, Appendix H for the sixty year operating period. The capsules were withdrawn after 2.5, 5.9 and 13.6 effective full power years (EFPY). The W-225 capsule (Ref. 6.15) provided data at a time that corresponds with the ASTM E 185-82 requirement, a fluence of approximately 5E18 n/cm^2. The W-275 capsule was withdrawn at a time corresponding to the approximate end-of-life fluence at the quarter-thickness location in the vessel (Ref. 6.17). The W-265 capsule was withdrawn at a time when the fluence was between that of the first and the third capsule (Ref. 6.16). Two more capsules are to be scheduled for withdrawal per ASTM E 185-82. The fifth capsule should have attained at least but not greater than twice the end-of-life fluence at the inner wall of the vessel. For the sixty-year life, the estimated end-of-life fluence is approximately 2.4E19 n/cm^2 (Ref. 6.25). The withdrawal times corresponding to one to two times the end-of-life fluence are approximately 40 to 48 EFPY for the lower lead factor capsules and 30 to 48 EFPY for the higher lead factor capsules. The fourth capsule should be withdrawn when it has attained the end-of-life fluence at the inner wall of the vessel. The withdrawal time corresponding to the end-of-life fluence is approximately 40 EFPY for the lower lead factor capsules and 30 EFPY for the higher lead factor capsules. The remaining capsules would not have to be removed and should be designated for standby.

4.2 Modifications to Meet 10CFR50 Appendix H Requirements for Extended License Life

In addition to meeting the ASTM E 185-82 requirements, the NRC identified in Ref. 6.26 three considerations for the license renewal period that might necessitate modifications to the Calvert Cliffs reactor vessel surveillance program. (The reference is based on statements from NUREG-1705, Appendix E, NRC Safety Evaluation Report for Calvert Cliffs License Renewal.) Those considerations are paraphrased as follows:

1. Modify the schedule as necessary to obtain data at neutron fluences equal to or greater than the projected peak neutron fluence at the end of the period of extended operation.

2. If the last capsule is withdrawn before the 55th (calendar) year, the licensee is to establish reactor vessel neutron environment conditions (fluence, spectrum, temperature, and neutron flux) applicable to the surveillance data and the reactor vessel pressure temperature limit curves. If the vessel subsequently operates outside the limits established by these conditions, the licensee shall inform the NRC and determine the impact of that condition on reactor vessel integrity.

3. If the last capsule is withdrawn before the 55th year, the licensee shall install neutron dosimetry to permit tracking of the reactor vessel fluence.

The three NRC considerations for the license renewal period as stated above could be met by the surveillance capsule removal schedule for FCS. The availability of standby capsules and the ability to expose the fifth capsule for the full length of the 60 calendar year service lifetime provides this capability.

In addition to meeting the general considerations for the license renewal period, the FCS reactor vessel integrated surveillance program will also entail the use of the W-275S capsule and the integrated surveillance program (i.e., use of surveillance data from other plants). A primary consideration for the withdrawal is the minimum fluence exposure desired. The W-275S capsule was fabricated in 1993 and installed prior to the start of Cycle 15 (approximately October 1993). The capsule contains 24 Charpy V-notch specimens from two weldments and nine uniaxial tension test specimens (a total of nine split between the two weldments). One weldment had been made using heat 27204, and a second weldment had been made using heats 12008 and 13253. (Heat 27204 is from the FCS nozzle dropout including weld 1-410B.

Heat 12008 and 13253 is from a Maine Yankee nozzle dropout that includes weld 1-203.) The average fluence to the W-275S capsule is 8.63E17 n/cm^2 per EFPY. After 20 EFPY (approximately 2017) the accumulated fluence on W-275S is 1.719E+19 n/cm^2. That fluence corresponds approximately to the fluence in 2033 on the FCS vessel weld 3-410 A/C.

A schedule that would meet ASTM E 185-82 requirements and the additional considerations in NUREG-1705, Appendix E is provided in Table 4-1.

Table 4-1 Recommended Modifications to FCS Surveillance Capsule Withdrawal Schedule and Integrated Program

Removal Sequence	Exposure of Vessel	End of Fuel Cycle	Capsule Location
1	2.5[1]	3	W-225[2]
1	6E18[4]	-	Mihama 1[2]
2	5.9[1]	7	W-265[2]
3	13.6[1]	14	W-275[2]
3	1.2E19[4]	-	Mihama 1[2]
4	2.1E19[4]	-	Mihama 1[2]
5	~3.0E19[4]	-	Pal. SA-240[2]
6	33.6[1]	30	W-275S
7	48[1,3]	41	W-95
8	Standby	-	W-45
9	Standby	-	W-85
10	Standby	-	W-225S
11	Standby	-	W-265S

Notes:
1. Exposure in effective-full-power years (EFPY)
2. Capsule removed and tested.
3. 48 EFPY or EOL
4. Exposure in neutron fluence (n/cm^2)

4.3 Technical Bases

The changes from the USAR schedule were made to establish a program that will satisfy both the current requirements for a reactor surveillance program and for the anticipated requirements for the license renewal period. The current requirements as defined in 10CFR50, Appendix H and ASTM E 185-82 are met by the combined FCS surveillance program and the integrated surveillance program. The fourth and fifth capsules are from the integrated program. The second and third Mihama 1 capsules were exposed to a fluence comparable to the one-quarter thickness fluence for the FCS vessel weld 3-410 A/C in 2013 and 2033, respectively. The second supplemental capsule from Palisades will be exposed to a fluence of approximately 3E+19 n/cm^2. That capsule will provide data for a fluence well in excess of the projected fluence for the FCS vessel weld 3-410 A/C in the year 2033. Two more capsules are scheduled for withdrawal per ASTM E 185-82 to meet anticipated requirements for the license renewal period. The last capsule should have attained at least but not greater than twice the end-of-life fluence at the inner wall of the vessel. Capsule W-95 with a nominal lead factor of 1.2 shall have achieved a fluence between those limits for the peak location on the vessel inside surface. Another capsule should be withdrawn when it has attained the end-of-life fluence at the inner wall of the vessel. The accumulated fluence on W-275S is projected to be 1.719E+19 n/cm^2 after 20 EFPY from the 1993 insertion (approximately 2017). That fluence corresponds approximately to the fluence on the FCS vessel weld 3-410 A/C in the year 2013, which is 1.728E+19 n/cm^2. The remaining capsules would not have to be removed and have been designated for standby.

The integrated surveillance program for the FCS is comprised of the analysis as presented in Ref. 6.13 (CEN-636, Revision 2). The data from future analyses of surveillance welds from Mihama 1, the second supplemental capsule from Palisades, and Diablo Canyon Unit 1 will be reviewed against the CEN-636, Revision 2 analysis. The purpose of the review is to verify that the future measurements do not exhibit any anomalous behavior. If significant anomalies are discovered, the effect of the anomalies on the conservatism of the CEN-636, Revision 2 analysis will be assessed. Appropriate adjustments will be made if the CEN-636, Revision 2 analysis has clearly become non-conservative.

5. Summary

The FCS Reactor Vessel Surveillance Program consists of two elements. The first is the FCS Surveillance Program required by 10CFR50, Appendix H, including the supplemental capsule, W-275S. The second is the so-called "Integrated Surveillance Program" to utilize data originating from the surveillance programs at Mihama 1, Diablo Canyon 1, and Palisades. Use of CEN-636, Revision 2 has been approved by the NRC. This additional program is a result of utilizing 10CFR50.61 and Regulatory Guide 1.99, Position 2.1.

The FCS Reactor Vessel Integrity Program objectives rely on both elements of the surveillance program. Those objectives are 1) to ensure that the FCS RV may be operated until at least through the end of the current license period, August 9, 2013, as well as through August 9, 2033 for license renewal, 2) to maintain a reactor vessel surveillance program in accordance with 10CFR50 Appendix H (including use of data from other surveillance programs), 3) to ensure that sufficient operating margins exist in accordance with 10CFR50 Appendix G, 4) to maintain a fluence analysis necessary to support the above objectives, and 5) to develop, implement, and maintain a NRC-approved Pressure-Temperature Limits Report (PTLR). This report defines an optimized reactor vessel surveillance program that provides for monitoring vessel embrittlement and neutron fluence accumulation. Furthermore it provides for the license renewal period through assessment of the W-275S (supplemental) capsule and surveillance capsules from other plants.

The current surveillance capsule removal schedule was intended to meet the requirements of ASTM E 185-82 and 10CFR50, Appendix H. This schedule exceeds current requirements for surveillance through the year 2013. Modifications were made to the removal schedule and an integrated surveillance program was defined to better utilize the surveillance capsule resources, to meet the FCS reactor vessel integrity objectives. The combination of the FCS reactor vessel surveillance program and the integrated surveillance program, including the surveillance capsule results from Mihama 1 and Palisades in Table 4-1, have satisfied the requirements for surveillance through the year 2013. Modifications were also made to establish a program that will satisfy anticipated surveillance program requirements for the license renewal period. The current requirements are defined in 10CFR50, Appendix H and ASTM E 185-82.

Two more capsules are scheduled for withdrawal in accordance with ASTM E 185-82. The last capsule should have attained at least but not greater than twice the end-of-life fluence at the inner wall of the vessel. Capsule W-95 with a nominal lead factor of 1.2 shall have achieved a fluence between those limits for the peak location on the vessel inside surface. Another capsule is to be withdrawn when it has attained the end-of-life fluence at the inner wall of the vessel. This will be met by the second supplemental capsule from Palisades, SA-240. That capsule will have been exposed to a fluence equal to or greater than that projected for the FCS vessel weld 3-410 A/C in the year 2033. After 20 EFPY (after insertion in 1993) the accumulated fluence on W-275S is projected to correspond approximately to the inside surface fluence on the FCS vessel weld 3-410 A/C in the year 2013. Data from W-275S will be used to obtain surveillance measurements early in the license renewal period and for comparison with results from the integrated surveillance program. The remaining capsules would not be removed and are scheduled for standby.

The integrated surveillance program for the FCS is comprised of the analysis of data from current and future analyses of the surveillance welds from Mihama 1, Palisades supplemental capsules, and Diablo Canyon Unit 1. Future data will be reviewed against previous analyses to verify that those future measurements do not exhibit any anomalous behavior. If significant anomalies are discovered, the effect of the anomalies on the conservatism of the previous analyses will be assessed. Appropriate adjustments will be made if the analysis has clearly become non-conservative.

6. References

6.1 10CFR50.60: Acceptance Criteria for Fracture Prevention Measures for Light Water Nuclear Power Reactors for Normal Operation

6.2 10CFR50.61: Fracture Toughness Requirements for Protection against Pressurized Thermal Shock (PTS Rule)

6.3 10CFR50, Appendix G: Fracture Toughness Requirements

6.4 10CFR50, Appendix H: Reactor Vessel Material Surveillance Program Requirements

6.5 SRP 5.2.2, Revision 2: "Overpressure Protection", November 1988

6.6 RSB 5-2, Revision 1: "Overpressurization Protection of Pressurized Water Reactors While Operating at Low Temperatures", November 1988.

6.7 RG 1.99, Rev. 02: Radiation Embrittlement of Reactor Vessel Materials (issued May 1988)

6.8 RG 1.190: Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence (Draft DG-1053 published 9/1999 and DG-1025)

6.9 GL 96-03: Relocation of the Pressure-Temperature Limit Curves and Low Temperature Overpressure Protection System Limits, dated January 31, 1996

6.10 ASTM E 185-66 (-73, -79, and –82), "Recommended Practice for Surveillance Tests on Structural Materials in Nuclear Reactors"

6.11 CEN-189, "Evaluation of Pressurized Thermal Shock Effects due to Small Break LOCAs with Loss of Feedwater for the Combustion Engineering NSSS", December 1981

6.12 CEN-405-P, Revision 2, "Application of Reactor Vessel Surveillance Data for Embrittlement Management", July 1993

6.13 Letter LIC-00-0096 from OPPD (W.G. Gates) to NRC (Document Control Desk), "Supplemental Information to Support an Application for Amendment of Operating License", dated November 17, 2000

6.14 Letter LIC-01-0018 from OPPD (S.K. Gambhir) to NRC (Document Control Desk), "Supplemental Information to Support an Application for Amendment of Operating License", dated February 14, 2001

6.15 TR-O-MCM-001, Revision 1, "Fort Calhoun Station Unit No. 1 Evaluation of Irradiated Capsule W-225", dated August 28, 1980. [Contained in Letter LIC-81-0011 from OPPD (W.C. Jones) to NRC (H.R. Denton), dated January 23, 1981]

6.16 TR-O-MCM-002, "Fort Calhoun Station Unit No. 1 Evaluation of Irradiated Capsule W-265", dated March 7, 1984. [Contained in Letter LIC-84-124 from OPPD (W.C. Jones) to NRC (D.G. Eisenhut), dated April 25, 1984]

6.17 BAW-2226, "Fort Calhoun Station Unit No. 1 Evaluation of Irradiated Capsule W-275", dated November 1994. [Contained in Letter LIC-94-0250 from OPPD (T.L. Patterson) to NRC (Document Control Desk), dated December 9, 1994]

6.18 WCAP-11567, "Analysis of Capsule S from the Pacific Gas and Electric Company Diablo Canyon Unit 11 Reactor Vessel Radiation Surveillance Program", December 1987

6.19 WCAP-13750, "Analysis of Capsule Y from the Pacific Gas and Electric Company Diablo Canyon Unit 11 Reactor Vessel Radiation Surveillance Program", July 1993

6.20 WCAP-13440, "Supplemental Reactor Vessel Radiation Surveillance Program for the Pacific Gas and Electric Company Diablo Canyon Unit No. 1, December 1992

6.21 "Recommended Program for Irradiation Surveillance of the Fort Calhoun Reactor Vessel Materials", dated February 21, 1969.

6.22 CENPD-33, "Summary Report on Manufacture of Test Specimens and Assembly of Capsules for Irradiation Surveillance of the Fort Calhoun Reactor Vessel Materials", dated November 15, 1971.

6.23 TR-O-MCM-001, "Summary Report on Manufacture of Test Specimens and Assembly of Capsules for Replacement Assemblies in the Fort Calhoun Reactor Vessel", dated August 15, 1983.

6.24 MCC-92-QP-014, "Manufacture of the Fort Calhoun W-275 Replacement Surveillance Capsule Assembly", dated June 23, 1993.

6.25 "Fast Neutron Fluence Evaluation for the Fort Calhoun Reactor Pressure Vessel", dated July 2000, WCAP-15443, Rev. 0.

6.26 "Safety Evaluation of Request to Revise Reactor Pressure Vessel Surveillance Program Schedule for Calvert Cliffs Nuclear Power Plant, Units 1 and 2", USNRC letter dated November 8, 2000.

6.27 Letter from NRC (A.B. Wang) to OPPD (S.K. Gambhir), "Fort Calhoun Station, Unit No. 1 - Issuance of Amendment - Deletion of Section 3.D "License Term" (TAC No. MA9690)", (Amendment No. 199).

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