ENCLOSURE

NEDC 05-019 REV. 0
REACTOR PRESSURE VESSEL FLUENCE EVALUATION

COOPER NUCLEAR STATION
NRC DOCKET 50-298, LICENSE DPR-46

Title: Reactor Pressure Vessel Fluence Evaluation	Calculation Number: NEDC 05-019
	CED/EE Number: 04-015
System/Structure: NB	Setpoint Change/Part Eval Number: N/A
Component: RPV	Discipline: Nuclear
Classification: [X] Essential; [ ] Non-Essential	SQAP Requirements Met? [ ] Yes; [X] N/A

Proprietary Information Included? [ ] Yes; [X] No

Description: This calculation determines the reactor vessel neutron fluence at 21 and 32 EFPY using the BWRVIP RAMA methodology. Reference TransWare calculation number EPR-VIP-003-R-002, Revision 0.

Conclusions and Recommendations:

Accept the TransWare calculation. The calculated fluence at 32 EFPY is slightly higher than the fluence used in the generation of the current P-T curves. Therefore the P-T curves should be limited to 30 EFPY: [(1.57e18 n/cm2)/(1.67e18 n/cm2)* 32 EFPY= 30.1 EFPY]. A Licensing Basis Change Request per procedure 0.29.1 will be required.

0	1	TransWare Enterprises, Inc.	K. B. Thomas 5/09/05	N/A	6/23/05
Rev. Number	Status	Prepared By/Date	Reviewed By/Date	IDVed By/Date	Approved By/Date

Status Codes

1. Active
2. Information Only
3. Pending
4. Superseded or Deleted
5. OD/OE Support Only
6. Maintenance Activity Support Only
7. PRA/PSA

Nebraska Public Power District

DESIGN CALCULATION CROSS-REFERENCE INDEX

ITEM NO.	DESIGN INPUTS	REV. NO.	PENDING CHANGES TO DESIGN INPUTS
1	NEDC 04-018	0	None
2	NEDC 04-019	0	None
3	EE 03-092	0	None

Nebraska Public Power District

DESIGN CALCULATION CROSS-REFERENCE INDEX

ITEM NO.	AFFECTED DOCUMENTS	REV. NUMBER
1	Technical Specification Figures 3.4.9-1. 3.4.9-2, and 3.4.9-3	Amendment 204

The purpose of this form is to assist the Preparer in screening new and revised design calculations to determine potential impacts to procedures and plant operations.®[1]

	SCREENING QUESTIONS	YES	NO	UNCERTAIN
1.	Does it involve the addition, deletion, or manipulation of a component or components which could impact a system lineup and/or checklist for valves, power supplies (breakers), process control switches, HVAC dampers, or instruments?	[ ]	[X]	[ ]
2.	Could it impact system operating parameters (e.g., temperatures, flow rates, pressures, voltage, or fluid chemistry)?	[X]	[ ]	[ ]
3.	Does it impact equipment operation or response such as valve closure time?	[ ]	[X]	[ ]
4.	Does it involve assumptions or necessitate changes to the sequencing of operational steps?	[ ]	[X]	[ ]
5.	Does it transfer an electrical load to a different circuit, or impact when electrical loads are added to or removed from the system during an event?	[ ]	[X]	[ ]
6.	Does it influence fuse, breaker, or relay coordination?	[ ]	[X]	[ ]
7.	Does it have the potential to affect the analyzed conditions of the environment for any part of the Reactor Building, Containment, or Control Room?	[ ]	[X]	[ ]
8.	Does it affect TS/TS Bases, USAR, or other Licensing Basis documents?	[X]	[ ]	[ ]
9.	Does it affect DCDs?	[ ]	[X]	[ ]
10.	Does it have the potential to affect procedures in any way not already mentioned (refer to review checklists in Procedure EDP-06)? If so, identify:	[ ]	[X]	[ ]

If all answers are NO, then additional review or assistance is not required.

If any answers are YES or UNCERTAIN, then the Preparer shall obtain assistance from the System Engineer and other departments, as appropriate, to determine impacts to procedures and plant operations. Affected documents shall be listed on Attachment 2.

Nebraska Public Power District

DESIGN CALCULATIONS SHEET

PURPOSE:

This calculation determines the reactor vessel neutron fluence at 21 and 32 EFPY using the BWRVIP RAMA methodology. Reference TransWare calculation number EPR-VIP-003-R-002, Revision 0.

ASSUMPTIONS:

See section IV of the TransWare calculation for assumptions.

METHODOLOGY:

See section IV of the TransWare calculation for a description of the RAMA Fluence methodology.

CONCLUSION:

Implement the TransWare calculation. Projected neutron fluence values are presented for two points in time: The end of cycle 21 and the current end of design life, 32 EFPY. It was assumed in projecting the 32 EFPY fluence that cycle 21 was an equilibrium cycle and representative of future operating cycles. Thus, the incremental fluence determined for cycle 21 is assumed to be constant which allows the fluence between cycle 21 and 32 EFPY to be estimated with linear interpolation for the purpose of evaluating the P-T curves.

The fluence at 32 EFPY calculated by RAMA is 1.67e18 n/cm2 (NEDC 05-019, Table 7-3). The fluence used in the development of the current P-T curves at 32 EFPY is 1.57e18 n/cm2 (EE 03-92, Table 1). The revised fluence causes a slight increase in ART for the limiting material (EE 03-92, Table 1, Part #1-233, was 127.6°F). In order to adjust for this increase the P-T curves should be limited to 30 EFPY: [(1.57e18 n/cm2)/(1.67e18 n/cm2)* 32 EFPY= 30.1 EFPY]. A Licensing Basis Change Request per procedure 0.29.1 will be required.

REFERENCES:

1. EE03-092, Review of Structural Integrity Report, Revised Pressure Temperature Curves

ATTACHMENTS:

TransWare calculation number EPR-VIP-003-R-002, Revision 0.

TWE

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Certificate of Conformance

Date: April 28, 2005

Contract No.: EP-P13466/C6669QA

Product: TransWare Enterprises Inc. document number EPR-VIP-003-R-002, Revision 0, "Cooper Nuclear Station Reactor Pressure Vessel Fluence Evaluation"

Product Form: ■ Transmitted as printed copy

Revision: 0

Reference: None

TransWare Enterprises Inc. hereby certifies that the Product has been reviewed and approved in accordance with the TransWare Quality Assurance Program, and is in conformance with all requirements prescribed in the contract, 10CFR50 Appendix B, 10CFR21, including or in addition to, applicable codes and standards as specified in TransWare Enterprises Inc. Project Plan No. EPR-VIP-003-Q-001, Revision 1, dated March 17, 2004.

Kenneth E. Watkins
Manager, Quality Assurance

4/28/05
Date

COOPER NUCLEAR STATION REACTOR PRESSURE VESSEL FLUENCE EVALUATION

TransWare Document Number: EPR-VIP-003-R-002
Revision 0

TransWare Enterprises Inc.
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April 2005

COOPER NUCLEAR STATION REACTOR PRESSURE VESSEL FLUENCE EVALUATION

Document Number: EPR-VIP-003-R-002
Revision 0

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This document has been prepared in accordance with the requirement of 10CFR50 Appendix B, 10CFR21, and TransWare Enterprises Inc.'s 10CFR50 Appendix B quality assurance program.

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ACKNOWLEDGMENT

Mr. Ken Thomas and Mr. Steven Luther of Nebraska Public Power District are acknowledged for their support of the Cooper Nuclear Station reactor pressure vessel fluence evaluation. Mr. Bob Carter of the Electric Power Research Institute, Inc. and the Boiling Water Reactor Vessel Internals Project (BWRVIP) are acknowledged for their support of this project. Mr. Steven Baker of TransWare Enterprises Inc. is acknowledged for his support in the Cooper Nuclear Station data collection process.

Charles A. Schwalbach, TransWare Enterprises Inc.

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1 INTRODUCTION

This report presents the results of the reactor pressure vessel (RPV) fluence evaluation performed for the Cooper Nuclear Station (CNS) using the RAMA Fluence Methodology. Fluence values are calculated at the end of operating cycle 21 and projected fluence values are presented for the end of the reactor's design lifetime of 32 effective full power years (EFPY). Neutron fluence values are determined for the RPV shell and weld locations located in the RPV beltline region. This evaluation was performed in accordance with guidelines presented in U. S. Nuclear Regulatory Guide 1.190 [1].

This evaluation also includes the prediction of specific activities for flux wires that were irradiated in two CNS surveillance capsules. One of the capsules was irradiated for nine cycles (6.8 EFPY) and the other was irradiated for 14 cycles (11.2 EFPY). Activation measurements were conducted on the flux wires and impact testing was performed on the Charpy specimens extracted from the surveillance capsules [2,3]. In this evaluation, the specific activities predicted by the RAMA Fluence Methodology are compared to the activity measurements.

The RAMA Fluence Methodology (hereinafter referred to as the Methodology) has been developed for the Electric Power Research Institute, Inc. (EPRI) and the Boiling Water Reactor Vessel and Internals Project (BWRVIP) for the purpose of calculating neutron fluence in Boiling Water Reactor (BWR) components. The Methodology includes a transport code, model builder codes, a fluence calculator code, an uncertainty methodology, and a nuclear data library. The transport code, fluence calculator, and nuclear data library are the primary software components for calculating the neutron flux and fluence. The transport code uses a deterministic, three-dimensional, multigroup nuclear particle transport theory to perform the neutron flux calculations. The transport code couples the nuclear transport method with a general geometry modeling capability to provide a flexible and accurate tool for calculating fluxes in light water reactors. The fluence calculator uses reactor operating history information with isotopic production and decay data to estimate activation and fluence in the reactor components over the operating life of the reactor. The nuclear data library contains nuclear cross-section data and response functions that are needed in the flux, fluence, and reaction rate calculations. The cross sections and response functions are based on the BUGLE-96 nuclear data library [4]. The Methodology and procedures for its use are described in the following reports: Theory Manual [5] and Procedures Manual [6].

Previous analyses have been conducted to benchmark the Methodology against other benchmark problems as recommended in the U. S. Nuclear Regulatory Guide 1.190. The results of the Methodology benchmarks are presented in [7]. The Methodology has also been used to perform three additional surveillance capsule fluence evaluations [8-10].

The information and associated evaluations provided in this report have been performed in accordance with the requirements of 10CFR50 Appendix B.

2
SUMMARY AND CONCLUSIONS

This section provides a summary of the results of the reactor pressure vessel fluence evaluation for Cooper Nuclear Station at the end of operating cycle (EOC) 21 through the projected end of normal operating life (32 EFPY). Detailed tables of all fluence results are presented in Section 7 of this report. The primary purpose of this evaluation is to determine the reactor pressure vessel fluence for energy >1.0 MeV at selected welds and shells in the reactor pressure vessel beltline region. Fluence is calculated at the inner surface (0T), 1/4T and 3/4T locations on each RPV weld and shell.

Table 2-1 summarizes the peak fluence values generated from this evaluation for energy >1.0 MeV at EOC 21 and 32 EFPY. One value represents the peak fluence for the weld locations and the other represents the value at the shell locations. Note that the peak fluence for both the RPV welds and shells occurs at the inner diameter at the point closest to the edge of the core (0T). The peak fluence for the weld locations is in circumferential weld VCB-BA-2 with a value of 8.11E+17 n/cm^2 at EOC 21 and 1.22E+18 n/cm^2 at 32 EFPY. The peak fluence for the RPV shells is in shell ring #2 with a value of 1.13E+18 n/cm^2 at EOC 21 and 1.67E+18 n/cm^2 at 32 EFPY.

Another observation is that fluence has exceeded 1.00E+17 n/cm^2 in all welds located in shell rings 1 and 2 as of the end of cycle 21. The elevations at which the fluence value first exceeds 1.00E+17 n/cm^2 are 511.73 cm (201.47 inches) for EOC 21 and 502.85 cm (197.97 inches) for 32 EFPY.

Table 2-1
Peak >1.0 MeV Neutron Fluence for RPV Weld and Shell Locations at Inner Diameter (0T)

Weld/Shell Location	Elevation cm (inches)	Peak Fluence for EOC 21 (n/cm^2)	Peak Fluence for 32 EFPY (n/cm^2)
Weld VCB-BA-2	638.81 (251.50)	8.11E+17	1.22E+18
Shell Ring #2 (1)	772.12 (303.98)	1.13E+18	----
Shell Ring #2 (1)	787.35 (309.98)	----	1.67E+18

(1) The peak fluence value occurred at a different elevation for EOC 21 and 32 EFPY, however, both elevations are in Shell Ring 2.

In addition to the prediction of RPV fluence and flux values, specific activities were predicted for the copper, iron, and nickel flux wires in two CNS surveillance capsules identified in this report as Capsules 1 and 2. Capsule 1 is positioned at azimuth 30° and Capsule 2 is at azimuth 300°. The flux wires from Capsule 1 were irradiated from cycles 1 through 9 for 6.8 EFPY. The flux wires from Capsule 2 were irradiated from cycles 1 through 14 for 11.2 EFPY. The predicted activities were compared to measurements. The total average calculated-to-measured (C/M) result of specific activities for all flux wires was determined to be 1.05 with a standard deviation of ±15%. Tables containing more detailed capsule activation results are presented in Section 5 of this report. These C/M ratios are in good agreement indicating the RAMA Fluence Methodology is accurately predicting fluence and flux. Note that the Methodology provides a direct solution of the reactions, i.e., no multiplicative or other adjustment is made to the results.

Another result from this evaluation is the calculated RPV fluence combined uncertainty values. By combining the measurement uncertainty and analytic uncertainty, the combined RPV fluence uncertainty is determined to be 9.2% for energy >1.0 MeV.

In conclusion, the RAMA Fluence Methodology produces accurate results that compare very well with measured data. The Methodology for determining the neutron fluence for the RPV shell and weld locations has been performed in accordance with the guidelines presented in Regulatory Guide 1.190.

3
DESCRIPTION OF THE REACTOR SYSTEM

This section describes the CNS design inputs used in the RPV fluence evaluation. The basic design inputs include component mechanical designs, material compositions, and reactor operating history. Mechanical design drawings and structural material data were provided by Nebraska Public Power District and were used to build the Cooper Nuclear Station RAMA geometry model [11]. Detailed operating history data was provided for this project by Nebraska Public Power District [12] for cycles 15 through 21. Detailed operating data for cycles 1 through 14 of the CNS was not available for the fluence predictions so data for these cycles was approximated using power and exposure distributions derived from detailed operating cycles of reactors of comparable core design and energy production [13].

3.1 Reactor System Mechanical Design Inputs

The CNS employs a boiling water reactor (BWR) nuclear steam supply system. The reactor is a General Electric BWR/4 class reactor located in Brownville, Nebraska. The reactor core consists of 548 fuel assemblies with a rated thermal power of 2381 MWt.

The CNS is modeled with the RAMA Fluence Methodology. The Methodology employs a three-dimensional modeling technique to describe the reactor geometry for the neutron transport calculations. Detailed mechanical design information is used in order to build an accurate three-dimensional RAMA computer model of the reactor system. A summary of the important design inputs is presented in this subsection.

Figure 3-1 illustrates the basic planar geometry configuration of the reactor at the axial elevation corresponding to the core mid-plane elevation. All radial regions comprising the fluence model are illustrated. Beginning at the center of the reactor and projecting outwards, the regions include: the core region, including control rod locations and fuel assembly locations (fuel locations are shown only for the northeast quadrant); core reflector region (bypass water); central shroud wall; downcomer water region including the jet pumps; reactor pressure vessel (RPV) wall; mirror insulation; biological shield (concrete wall); and cavity regions between the RPV and biological shield. Also shown are the azimuthal positions of the surveillance capsules in the downcomer region at 30, 210, and 300 degrees. The surveillance capsules are positioned radially near the inner surface of the RPV wall. The azimuthal positions of the jet pump assemblies are illustrated as well at 30, 60, 90, 120, 150, 210, 240, 270, 300, and 330 degrees.

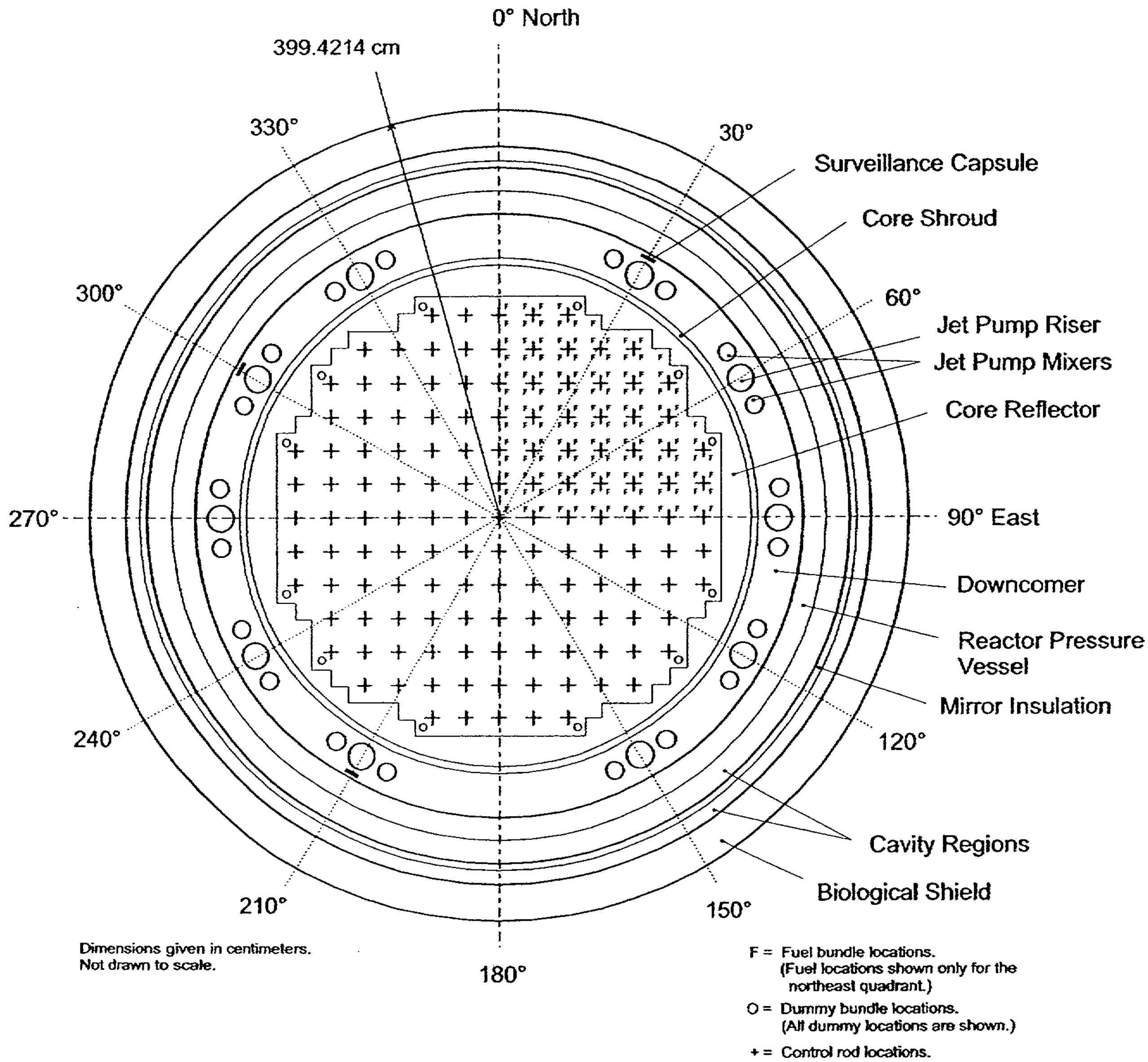

Figure 3-1
Planar View of the Cooper Nuclear Station Reactor

3.2 Reactor System Material Compositions

Each region of the reactor is comprised of materials that include reactor fuel, steel, water, insulation, concrete, and air. Accurate material information is essential for the fluence evaluation as the material compositions determine the scattering and absorption of neutrons throughout the reactor system and, thus, affect the determination of neutron fluence in the reactor components.

Table 3-1 provides a summary of the material compositions in the various components and regions of the CNS. The attributes for the steel, insulation, concrete, and air compositions (i.e., material densities and isotopic concentrations) are assumed to remain constant for the operating life of the reactor. The attributes for the ex-core water compositions will vary with the operation of the reactor, but are generally represented at nominal hot operating conditions and are assumed to be constant throughout an operating cycle.

Table 3-1
Summary of Material Compositions by Region for Cooper Nuclear Station

Region	Material Composition
Reactor Core	^{235}U, ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, O_{fuel}, Zr, Water
Core Reflector	Water
Fuel Support Piece	Stainless Steel SS-304
Fuel Assembly Lower Tie Plate	Stainless Steel SS-304, ZR-2, Inconel-X
Fuel Assembly Upper Tie Plate	Stainless Steel SS-304, ZR-2, Inconel-X
Top Guide	Stainless Steel SS-304L
Shroud	Stainless Steel SS-304L
Jet Pump Riser and Mixer Flow Area	Water
Jet Pump Riser and Mixer Metal	Stainless Steel SS-304
Downcomer Region	Water
Surveillance Capsule	Carbon Steel
Reactor Pressure Vessel Clad	Stainless Steel SS-304
Reactor Pressure Vessel Wall	Carbon Steel SA-302B
Cavity Regions	Air (Oxygen)
Insulation	Stainless Steel SS-304
Biological Shield Clad	Carbon Steel
Biological Shield	Reinforced Concrete

The attributes of the fuel compositions in the reactor core region change continuously during an operating cycle due to changes in power level, fuel burnup, control rod movements, and changing moderator density levels (voids). Because of the dynamics of the fuel attributes with reactor operation, one to several data sets describing the operating state of the reactor core are used for each operating cycle. The number of data sets used in this analysis is presented in Section 3.3.2.

3.3 Reactor Operating Data Inputs

An accurate evaluation of fluence in the CNS requires an accurate accounting of the reactor operating history. The primary reactor operating parameters that affect neutron fluence evaluations for BWR's include the reactor power level, core power distribution, core void fraction distribution (or equivalently, water density distribution), and fuel material distribution. These items are described in the following subsections.

3.3.1 Power History Data

The reactor power history used in the CNS RPV fluence evaluation was obtained from daily power history edits provided by Nebraska Public Power District for operating cycles 15 through 21 [12] and estimated power levels for cycles 1 through 14 [13]. The daily power values represent step changes in power on a daily basis and are assumed to be representative of the power over the entire day. The RPV fluence evaluation for the CNS considers the complete daily operating history of the reactor from cycles 1 through 21. Also accounted for in the analysis are the shutdown periods. The shutdowns were primarily due to the refueling outages between cycles. Table 3-2 provides the effective full power years of power generation at the end of each cycle in this fluence evaluation.

3.3.2 Reactor State Point Data

CNS operating data for the RPV fluence evaluation was provided as state point data files by Nebraska Public Power District [12] and TransWare Enterprises Inc. [13]. The state point files provide a best-available representation of the operating conditions of the unit over the operating lifetime of the reactor. The data files include three-dimensional data arrays that describe the fuel materials, moderator densities, and relative power distribution in the core.

A total of 109 state point data files were used to represent the first twenty-one operating cycles of the CNS. Table 3-2 shows the number of state point data files for each cycle used in this fluence evaluation. A separate neutron transport calculation was performed for each state point. The calculated neutron flux for each state point was combined with the appropriate power history data described in Section 3.3.1 in order to predict the neutron fluence in the various reactor components.

For fluence predictions beyond the current plant life, the most recent completed operating cycle (i.e., cycle 21) is used. Due to changes in fuel designs throughout the operating life of the CNS, the most recent cycle is representative of a current "equilibrium" cycle. This cycle is, therefore, used as the basis for projecting plant operation from cycle 22 to the end of the plant design life of 32 EFPY. The rated thermal power output of the CNS for all operating cycles is specified as 2381 MWt as no power uprate is currently identified by Nebraska Public Power District for the reactor.

Table 3-2
Number of State-point Data Files for Each Cycle in Cooper Nuclear Station

Cycle Number	Number of State Point Data Files	Rated Thermal Power MWt	Effective Full Power Years (EFPY)
1	5	2381	1.4
2	2	2381	2.0
3	1	2381	2.4
4	1	2381	3.1
5	3	2381	3.8
6	2	2381	4.6
7	2	2381	5.3
8	2	2381	6.1
9	2	2381	6.8
10	2	2381	7.6
11	2	2381	8.5
12	2	2381	9.2
13	2	2381	9.9
14	2	2381	11.2
15	13	2381	12.3
16	11	2381	13.6
17	11	2381	14.8
18	12	2381	16.1
19	11	2381	17.2
20	11	2381	18.5
21	10	2381	19.6
22+	10	2381	32.0

3.3.3 Core Loading Pattern

It is common in BWRs that more than one fuel assembly design will be loaded in the reactor core in any given operating cycle. For fluence evaluations, it is important to account for the fuel assembly designs that are loaded in the core in order to accurately represent the neutron source distribution at the core boundaries (i.e., peripheral fuel locations, the top fuel nodes, and the bottom fuel nodes).

Four different fuel assembly designs are used in the CNS during cycles 1 through 21. Table 3-3 provides a summary of the fuel designs loaded in the reactor core for these operating cycles. The cycle core loading patterns provided by Nebraska Public Power District are used to identify the fuel assembly designs in each cycle and their location in the core loading pattern. For each cycle, appropriate fuel assembly models are used to build the reactor core region of the RAMA fluence model for the CNS.

Table 3-3
Summary of the Cooper Nuclear Station Core Loading Pattern

Cycle	General Electric (GE) 7x7 Fuel Assembly Designs	General Electric (GE) 8x8 Fuel Assembly Designs	General Electric (GE) 9x9 Fuel Assembly Designs	General Electric (GE) 10x10 Fuel Assembly Designs	Dominant Peripheral Fuel Design in the RAMA Model
1	X				GE 7x7
2	X	X			GE 7x7
3	X	X			GE 7x7
4	X	X			GE 7x7
5	X	X			GE 7x7
6	X	X			GE 7x7
7	X	X			GE 7x7
8		X			GE 8x8
9		X			GE 8x8
10		X			GE 8x8
11		X			GE 8x8
12		X			GE 8x8
13		X			GE 8x8
14		X	X		GE 8x8
15		X	X		GE 8x8
16		X	X		GE 8x8
17		X			GE 8x8
18		X			GE 8x8
19		X			GE 8x8
20		X		X	GE 8x8
21		X		X	GE 8x8
22+		X		X	GE 8x8

4
CALCULATION METHODOLOGY

The Cooper Nuclear Station fluence evaluations were performed using the RAMA Fluence Methodology software package. The Methodology and the application of the Methodology to the Cooper Nuclear Station reactor are described in this section.

4.1 Description of the RAMA Fluence Methodology

The RAMA Fluence Methodology software package is a system of codes that is used to perform fluence evaluations in light water reactor components. The significance of the Methodology is the integration of a three-dimensional arbitrary geometry modeling technique with a deterministic transport method to provide a flexible and accurate platform for determining neutron fluence in light water reactor systems. The Methodology is complemented with model building codes to prepare the three-dimensional models for the transport calculation and a post-processing code to calculate fluence from the neutron flux calculated by the transport code.

The primary inputs for the RAMA Fluence Methodology are mechanical design parameters and reactor operating history data. The mechanical design inputs are obtained from reactor design drawings (or vendor drawings) of the plant. The CNS operating history data is obtained from reactor core simulation calculations, system heat balance calculations, and daily operating logs that describe the operating conditions of the reactor.

The primary outputs from the RAMA Fluence Methodology calculations are neutron flux, neutron fluence, and uncertainty determinations. The RAMA transport code calculates the neutron flux distributions that are used in the determination of neutron fluence. Several transport calculations are typically performed over the operating life of the reactor in order to calculate neutron flux distributions that accurately characterize the operating history of the reactor. The post-processing code (RAFTER) is then used to calculate component fluence and nuclide activations using the neutron flux solutions from the transport calculations and daily operating history data for the plant. The fluence calculated by RAFTER may then be adjusted in accordance with the calculational bias to determine the best estimate fluence and uncertainty in accordance with the intent of U. S. Nuclear Regulatory Guide 1.190.

4.2 The RAMA Geometry Model for Cooper Nuclear Station

Figure 4-1 illustrates the planar configuration of the CNS model at an axial elevation near the core mid-plane of the reactor pressure vessel. In the radial dimension the model extends from the center of the RPV to the outside surface of the biological shield (399.4214 cm). Nine radial

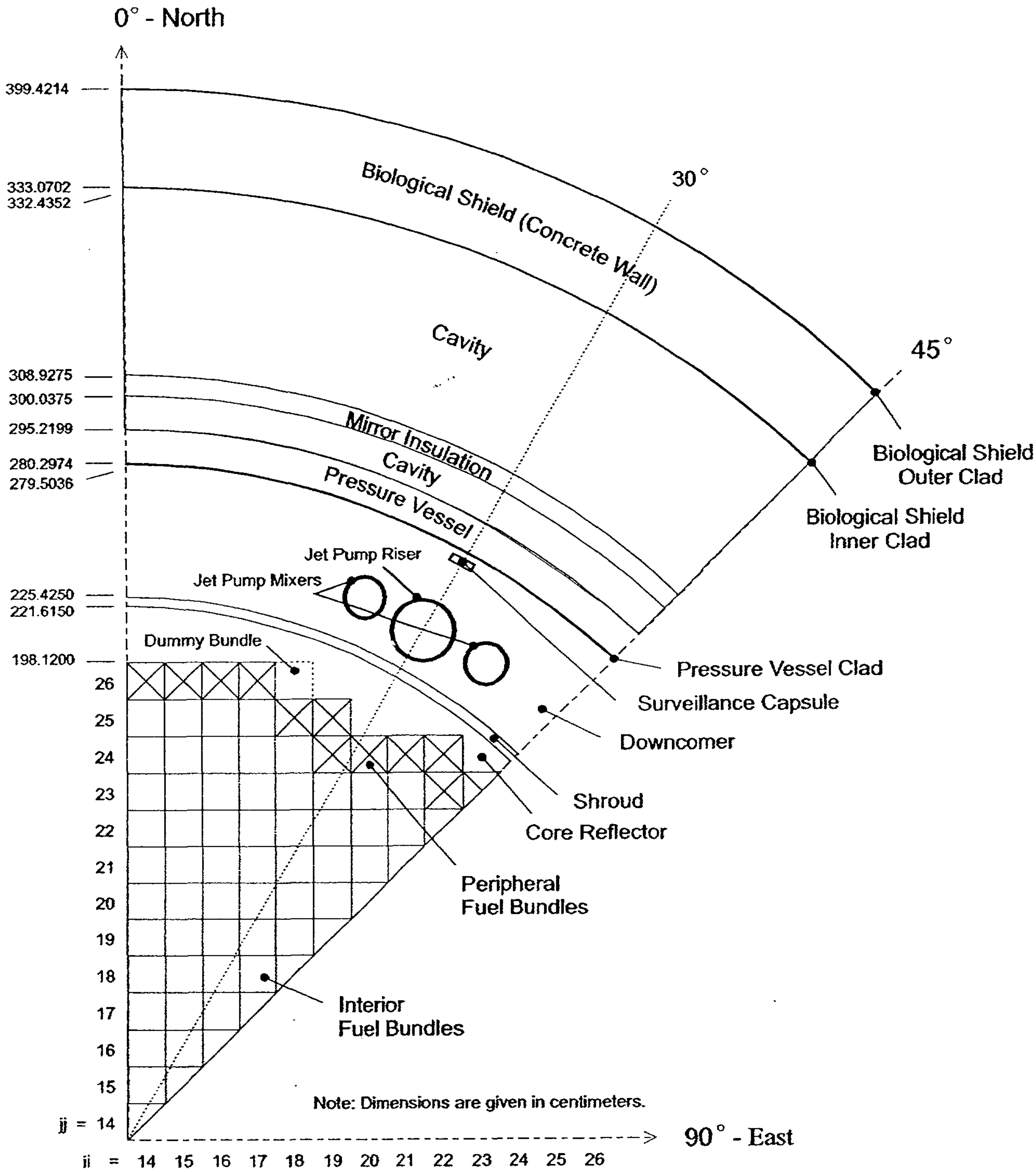

Figure 4-1
Planar View of the Cooper Nuclear Station RAMA Model

regions are defined in the CNS model: the core region (comprised of interior and peripheral fuel assemblies), core reflector, shroud, downcomer region with jet pumps, pressure vessel, mirror insulation, and biological shield, and inner and outer cavity regions. The pressure vessel has cladding on the wall inner surface. The biological shield has cladding on the inner and outer surfaces.

Figure 4-1 shows that the reactor core region is modeled with rectangular geometry to preserve the shape of the core region. The core region is characterized with two layers: the interior fuel assemblies and the peripheral fuel assemblies. The peripheral fuel assemblies are the primary contributors to the neutron source in the fluence calculation and are modeled to preserve the pin-wise source contribution at the core-core reflector interface.

Each of the components and regions that extend outward from the core region are modeled in their correct geometrical form. The core shroud, downcomer, RPV wall, mirror insulation, biological shield wall, and cavity regions are correctly modeled as cylindrical parts. The jet pump assembly design is properly modeled using cylindrical pipe elements for the jet pump riser and mixer pipes. The riser pipe is correctly situated between the mixer pipes. The surveillance capsule, which is rectangular in design, is modeled as an arc element in the geometry and is correctly positioned near the inner surface of the RPV wall. This model is an acceptable approximation since the capsule is a sufficient distance from the core center that the arc element closely approximates the shape of a rectangular element. Downcomer water surrounds the capsule on all sides.

The CNS has 10 jet pump assemblies that are positioned azimuthally at 30, 60, 90, 120, 150, 210, 240, 270, 300, and 330 degrees. One jet pump is modeled at the 30 degree azimuth which, when symmetry is applied, correctly represents the jet pumps at all positions except 90 and 270 degrees. There are no jet pumps present at the 0 and 180 degree azimuths. The surveillance capsules are shown as modeled at azimuth 30 degrees. When symmetry is applied to the model, this location represents each of the surveillance capsules loaded at 30, 210, and 300 degrees (see Figure 3-1).

As shown in Figure 3-1 of this document, the CNS geometry is quadrant symmetric, both in the core region and in the ex-core geometry. The quadrant core symmetry results from the presence of 12 dummy bundles, eight of which are located in octant symmetric locations, and four of which are located in quadrant symmetric locations. The ex-core symmetry results from the presence of ten jet pump assemblies that are located in quadrant-symmetric locations. For computational reasons, the RAMA model of the CNS core and ex-core geometry assumes octant symmetry. In the azimuthal dimension the model spans from 0 to 45 degrees where the 0 degree azimuth corresponds to the north compass direction that is specified in the reactor design drawings. The selection of this octant leads to a conservative estimate of the fluence in the various reactor components since the octant contains a single dummy assembly (versus two dummy assemblies in some of the octants) and the absence of a jet pump assembly and the corresponding shielding of the fluence at the 0 degree azimuth. Two-dimensional cases were run to determine the extent of conservatism introduced by the assumption of octant symmetry. It was determined that the assumption of octant symmetry resulted in less than 15% difference in fluence relative to a quadrant symmetric model, with the maximum difference confined to the local vicinity of the dummy bundle.

Figure 4-2 provides an illustration of the axial configuration of the CNS RAMA model for three significant components: a fuel column, the core shroud, and the reactor pressure vessel. Also shown in the figure are the relative axial positioning of the jet pumps, surveillance capsules, and core spray sparger pipes in the reactor model. The CNS fluence model spans axially from below the jet pump riser inlet to above the core shroud head flange for a total of 631.3488 cm in length.

The model consists of 70,903 to 71,022 mesh regions. Variations in the radial meshing are required to account for the removal and insertion of reactor dosimetry (i.e., surveillance capsules) in the various operating cycles. To the extent possible, the radial meshing is uniform between the planes. The axial planes are divided into several groups representing particular component regions of the model as follows: the core region, the top guide, the shroud head flange, the core spray spargers, the fuel support piece, core support plate, and core inlet region. Sub-planar meshing is used in the model, as needed, to properly represent the positioning of reactor components, such as the jet pump rams head and surveillance capsules.

As the primary interest in this fluence evaluation is the determination of the neutron fluence at specified RPV welds and shells, Figure 4-3 identifies these specific weld locations. These weld locations are referenced in the tables in Sections 2 and 7 of this report showing the RPV weld and shell fluence results by their identification numbers. Note that although circumferential welds VCB-BB-1 and VCB-BB-4 are shown in Figure 4-3 for completeness, these welds are not included in the RAMA model since their lifetime fluence is well below 1.00E+17 n/cm^2.

There are several key features of the RAMA code system that allow the CNS design to be accurately represented for component fluence evaluations. Following is a list of some of the key features of the model.

- Rectangular and cylindrical bodies are mixed in the model in order to provide an accurate geometrical representation of the components and regions in the reactor.
- The core geometry is modeled using rectangular bodies to represent the fuel assemblies in the reactor core region.
- Cylindrical bodies are used to represent the components and regions that extend outward from the core region.
- A combination of rectangular and cylindrical bodies is used to describe the transition parts that are required to interface the rectangular core region to the cylindrical outer core regions.
- The top guide is appropriately modeled by including a representation of the vertical fuel assembly parts and top guide plates. The upper fuel assembly parts that extend into the top guide region are modeled in three axial segments: the fuel rod plenum, fuel rod upper end plugs, and fuel assembly upper tie plate.

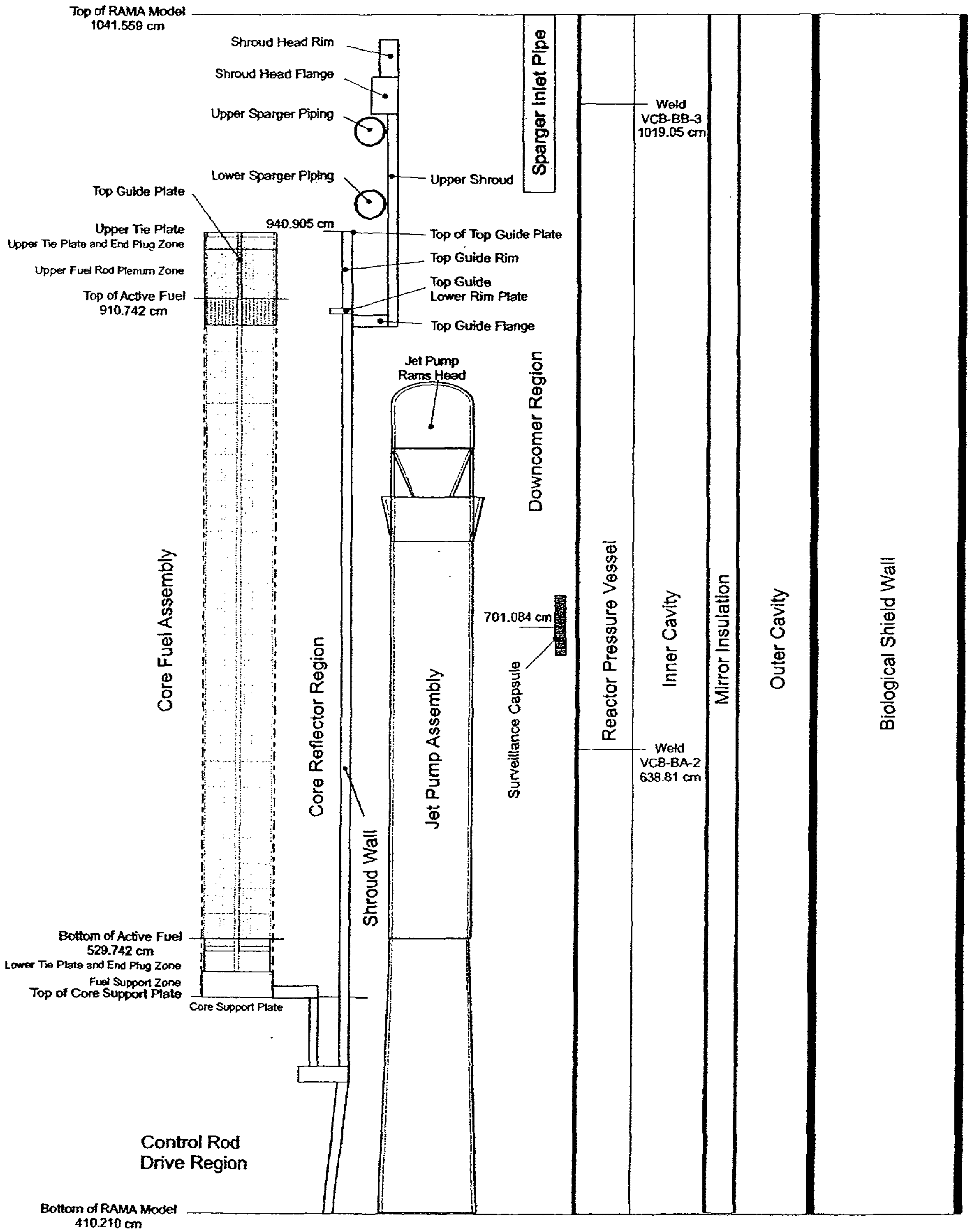

Figure 4-2
Axial View of the Cooper Nuclear Station RAMA Model

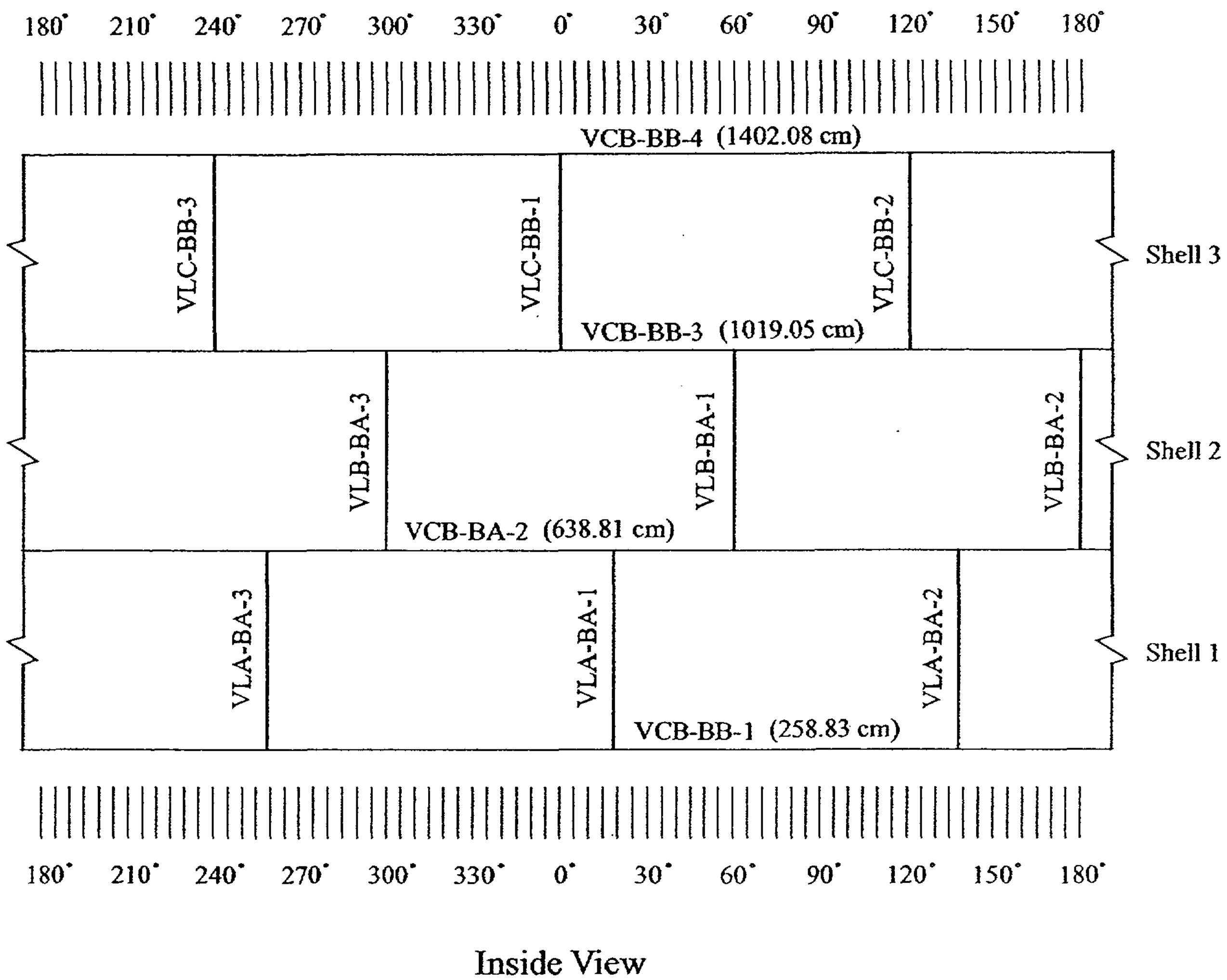

Figure 4-3
Cooper Nuclear Station RPV Weld and Shell Identifications

- The fuel support piece, core support plate, and core inlet regions appropriately include a representation of the cruciform control rod below the core region. The lower fuel assembly parts include representations for the fuel rod lower end plugs, lower tie plate, and nose piece.
- The surveillance capsules are represented in the downcomer region at the correct azimuth, at an axial elevation corresponding to the core mid-plane elevation, and radially near the inner surface of the pressure vessel wall.
- The core spray spargers are appropriately represented as toruses in the model. The sparger pipes reside inside the upper shroud wall above the top guide. The sparger model includes a representation of reactor coolant inside the pipes.

4.3 RAMA Calculation Parameters

The RAMA transport code uses a three-dimensional deterministic transport method to calculate neutron flux distributions in reactor problems. The transport method is based on a numerical integration technique that uses ray-tracing to form the integration paths through the problem geometry. The integration paths for the rays are determined using four parameters. The distance between parallel rays in the planar dimension is specified as 1.00 cm. The distance between parallel rays in the axial dimension is specified as 5.00 cm. The depth that a ray penetrates a reflective boundary is specified as 10 mean free paths. In accordance with the guidelines provided in [1], the angular quadrature for determining ray trajectories is specified as S8, which provides an acceptable compromise between computational accuracy and performance.

The RAMA transport calculation also uses information from the RAMA nuclear data library to determine the scope of the flux calculation. This information includes the Legendre order of expansion that is used in the treatment of anisotropy of the problem. By default, the RAMA transport calculation uses the maximum order of expansion that is available for each nuclide in the RAMA nuclear data library (i.e., through P_5 scattering for actinide and zirconium nuclides and through P_7 scattering for all other nuclides in the model).

The neutron flux is calculated using an iterative technique to obtain a converged solution for the problem. The convergence criterion used in the evaluation is 0.01 which provides an asymptotic solution.

The impact of these calculation parameter selections on the RAMA fluence evaluation for Cooper Nuclear Station is presented in Section 4.6.

4.4 RAMA Neutron Source Calculation

The neutron source for the RAMA transport calculation is calculated using the input relative power density factors for the different fuel regions and data from the RAMA nuclear data library.

The core neutron source was determined using the cycle-specific three-dimensional burnup distributions. The source distributions account for the radial power gradient in the fuel assemblies loaded near the core boundary by modeling the pin-wise source distributions in the outer three rows of fuel assemblies.

4.5 RAMA Fission Spectra

RAMA calculates a weighted fission spectrum, based on the relative contributions of the fuel isotopes, that is used in the transport calculation. The fission spectra for ^{235}U, ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu, and ^{242}Pu that are used in the RAMA transport calculations were taken directly from the latest release of the BUGLE-96 data library.

4.6 Parametric Sensitivity Analyses

Several sensitivity analyses were performed to evaluate the stability and accuracy of the RAMA transport calculation for the Cooper Nuclear Station model. Several parameters were evaluated including mesh size and the integration parameters discussed in Section 4.3. A summary of the analyses is presented in Table 4-1. As provided for in [1], two-dimensional models consisting of the detailed planar representation at the core mid-plane, typical of the model shown in Figure 4-1, are used to evaluate the sensitivities for those parameters that are insensitive to axial variations. Those parameters that are sensitive to axial variations are evaluated using detailed three-dimensional models typical of the model described in Section 4.2.

Table 4-1
Sensitivity Analyses

Case Description	Model Geometry	Varied Parameter	Maximum Absolute Deviation in the >1.0 MeV Capsule Flux Relative to the Production Model
Variation of the capsule planar mesh size	2-D	Mesh size is reduced to approximately one-fourth the production model mesh size	<2%
Variation of the distance between planar parallel rays	2-D	Distance between rays is reduced from 1.0 cm in the production model to 0.10 cm	<0.5%
Variation of the distance between axial parallel rays	3-D	Distance between rays is reduced from 5.0 cm in the production model to 3.0 cm	<0.02%
Variation of convergence criterion	2-D	Convergence criterion is reduced from 0.01 in the production model to 0.0001	≤0.02%
Variation of the angular quadrature set	2-D	Angular quadrature set is increased from S8 in the production model to S32	<7%
Variation of the maximum Legendre order of scattering	2-D	Scattering order is decreased from P_7 in the production model to P_3	<0.4%

5
SURVEILLANCE CAPSULE ACTIVATION RESULTS

This section contains the results from the CNS surveillance capsule activation analysis. The predicted activations (i.e., specific activities) generated by the RAMA Fluence Methodology were compared to the activation measurements for the capsule flux wires and are presented here. CNS surveillance Capsule 1, positioned in the reactor at azimuth 30°, was removed at the end of cycle 9 after being irradiated for a total of 6.8 effective full power years (EFPY). Capsule 2, positioned in the reactor at azimuth 300°, was removed at the end of cycle 14 after being irradiated for a total of 11.2 EFPY. Details of the dosimetry specimens and analysis are presented in the next subsection.

5.1 Comparison of Predicted Activation to Plant Specific Measurements

Copper flux wires, iron flux wires, and nickel flux wires were irradiated in both CNS surveillance capsules. Activation measurements were performed following irradiation for the following reactions [2,3]: $^{63}Cu(n,\alpha)^{60}Co$, $^{54}Fe(n,p)^{54}Mn$, and $^{58}Ni(n,p)^{58}Co$. The specific location of the individual wires within the capsules is not known so the RAMA calculation was performed using the average of the volume of the capsule for the flux wire location.

Table 5-1 provides a comparison of the RAMA calculated specific activities and the measured specific activities for the Capsule 1 flux wire specimens. Note that although there were nickel flux wires irradiated in Capsule 1, it is reported in [2] that too much time elapsed between the time the flux wires were removed from the CNS and the time the measurements were taken to produce any measurable activity for the nickel flux wires. Measurements are, therefore, only available for the copper and iron flux wires. The calculated-to-measured (C/M) results show good agreement between the RAMA calculated values and the measured values. The Capsule 1 total flux wire average C/M value is 0.96 with a standard deviation of ±0.09. The Capsule 1 average C/M value for the copper flux wire is 0.88 with a standard deviation of ±0.04. The Capsule 1 average C/M value for the iron flux wire is 1.03 with a standard deviation of ±0.01.

Table 5-1
Comparison of Specific Activities (in dps/g) for Cooper Nuclear Station Surveillance Capsule 1 Flux Wires (C/M)

Flux Wires (1)	Measured (dps/g)	Calculated (dps/g)	Calculated vs. Measured	Standard Deviation
Copper				
26146	1.05E+04	9.55E+03	0.91	
26194	1.15E+04	9.55E+03	0.83	
26195	1.06E+04	9.55E+03	0.90	
Average			**0.88**	**±0.04**
Iron				
26146	7.35E+04	7.48E+04	1.02	
26194	7.21E+04	7.48E+04	1.04	
26195	7.24E+04	7.48E+04	1.03	
Average			**1.03**	**±0.01**
Total Flux Wire Average	---	---	**0.96**	**±0.09**

1) Note that there were no measurements for the nickel flux wires so no comparisons could be made.

Table 5-2 provides a comparison of the RAMA calculated specific activities and the measured specific activities for the Capsule 2 flux wire specimens. The average calculated-to-measured (C/M) results for the Capsule 2 total flux wire is 1.14 with a standard deviation of ±0.15. The Capsule 2 average C/M value for the copper flux wire is 0.98 with a standard deviation of ±0.00. The Capsule 2 average C/M value for the iron flux wire is 1.13 with a standard deviation of ±0.03. The average C/M value for the nickel flux wire is 1.31 with a standard deviation of ±0.05.

Table 5-2
Comparison of Specific Activities (in dps/g) for Cooper Nuclear Station Surveillance Capsule 2 Flux Wires (C/M)

Flux Wires	Measured (dps/g)	Calculated (dps/g)	Calculated vs. Measured	Standard Deviation
Copper				
65310	9.93E+03	9.96E+03	1.00	
65311	1.04E+04	9.96E+03	0.96	
Average			**0.98**	**±0.00**
Iron				
65310	6.91E+04	7.79E+04	1.13	
65311	6.94E+04	7.79E+04	1.12	
Average			**1.13**	**±0.03**
Nickel				
65310	9.64E+05	1.23E+06	1.28	
65311	9.17E+05	1.23E+06	1.34	
Average			**1.31**	**±0.05**
Total Flux Wire Average	---	---	**1.14**	**±0.15**

Two additional surveillance capsule evaluations have been performed using the RAMA Fluence Methodology for BWR/4 reactors of similar design to the Cooper Nuclear Station. Summaries of these evaluations are presented below.

Three copper flux wires, three iron flux wires, and three nickel flux wires were irradiated in a BWR/4 surveillance capsule during the first five cycles of operation [8]. The calculated-to-measured (C/M) results show a very good agreement between the RAMA calculated values and the measured values. The total flux wire average C/M value is 0.98 with a standard deviation of ±0.09. The average C/M value for the copper flux wire is 0.88 with a standard deviation of ±0.04. The average C/M value for the iron flux wire is 1.01 with a standard deviation of ±0.05. The average C/M value for the nickel flux wire is 1.06 with a standard deviation of ±0.02.

Another BWR/4 evaluation consisted of comparing predicted activation to measurements for six flux wire specimens (three copper samples and three iron samples) that were retrieved from the surveillance capsule after the first cycle of reactor operation [10]. The total flux wire average

C/M value is 0.93 with a standard deviation of ±0.04. The average C/M value for the copper flux wire is 0.90 with a standard deviation of ±0.004. The average C/M value for the iron flux wire is 0.97 with a standard deviation of ±0.004.

Table 5-3 summarizes the results of the BWR/4 plant-specific surveillance capsule evaluations.

Table 5-3
Summary of Comparisons to Plant-Specific Surveillance Capsule Measurements

Benchmark	Number of Measurements	Calculated vs. Measured	Standard Deviation
Cooper Nuclear Station	12	1.05	±0.15
BWR/4 (specimens irradiated five cycles)	9	0.98	±0.09
BWR/4 (specimens irradiated one cycle)	6	0.93	±0.04
Total Plant-Specific Comparisons	**27**	**1.00**	**±0.12**

5.2 Comparison of Predicted Activation to Vessel Simulation Benchmark Measurements

In accordance with the guidelines provided in Regulatory Guide 1.190, and as specified in the RAMA Fluence Methodology theory and procedures manuals [5,6], it is appropriate to include comparisons of vessel simulation benchmark measurements in the overall fluence uncertainty evaluation whenever a statistically significant set of plant-specific comparison data is not available. The Pool Critical Assembly (PCA) and the VENUS-3 experimental benchmarks have been evaluated using the RAMA Fluence Methodology [7]. The PCA experimental benchmark includes 27 activation measurements at the mid-plane elevation in various simulated reactor components. The VENUS-3 experimental benchmark includes 385 activation measurements at a range of elevations in various simulated reactor components. Table 5-4 summarizes the reults obtained from the application of the RAMA Fluence Methodology to the vessel simulation benchmarks.

Table 5-4
Summary of Comparisons to Vessel Simulation Benchmark Measurements

Benchmark	Number of Measurements	Calculated vs. Measured	Standard Deviation
Pool Criticality Assembly	27	0.99	±0.05
VENUS-3	385	1.03	±0.05
Total Vessel Simulation Comparisons	**412**	**1.03**	**±0.05**

5.3 Comparison to other Fluence Evaluations

In addition to the vessel simulation benchmark evaluations, the RAMA Fluence Methodology has been used in the performance of three other fluence evaluations. These evaluations include: the BWR Pressure Vessel Numerical Benchmark (documented in [7]), the H. B. Robinson Unit 2 Pressure Vessel Benchmark (documented in [7]), and a BWR/2 reactor (documented in [9]). While the results of these other benchmarks do not contribute to the uncertainty evaluation, they do provide further confirmation of the adequacy of the RAMA Fluence Methodology to accurately predict surveillance capsule and vessel neutron flux distributions. A summary of the results of these other benchmarks is provided in the following paragraphs.

Comparison of the predicted reaction rates in the surveillance capsule region between the RAMA Fluence Methodology and the BWR Numerical Benchmark values showed an average deviation of approximately 1% with a standard deviation of 0.03. Comparison of the predicted neutron flux throughout the pressure vessel showed deviations ranging from a minimum of 0% to a maximum of 29%, with the majority of the deviations being on the order of 10% or less.

The H. B. Robinson Unit 2 evaluation was performed using two different data sets. One used the eight state-point operating data provided for cycle 9 while the other used the cycle 9 average operating data. For the eight state point data evaluation, the average C/M result for all the dosimeters was 0.95 with a comparison standard deviation of ±0.04. For the cycle 9 average data evaluation, the average C/M result for all the dosimeters was 0.98 with a comparison standard deviation of ±0.06. The measured values and RAMA values were corrected for photofission effects and ^{60}Co impurities.

The activation comparison for the BWR/2 reactor evaluation resulted in an average C/M for the total 309 capsule measurements of 0.98 with a comparison standard deviation of ± 0.07.

6
REACTOR PRESSURE VESSEL UNCERTAINTY ANALYSIS

The sources of the reactor pressure vessel uncertainty include analytic uncertainty and comparison uncertainty. These are combined to provide an estimate of the overall fluence bias and uncertainty (1σ). This subsection describes the parameters that were considered for the analytic uncertainty, the calculated comparison uncertainty, and the calculated combined uncertainty for the reactor pressure vessel fluence evaluation. The calculated combined uncertainty is used in Section 7 to calculate the reactor pressure vessel neutron fluence and corresponding standard deviation of the fluence.

6.1 Comparison Uncertainty

The reported measurement uncertainty (1σ) for the plant-specific activity measurements is ±2.5% [2,3]. Combining this measurement uncertainty with the standard deviation for the activity comparisons from Table 5-3 results in a plant-specific comparison uncertainty (1σ) of ±12% with no bias (i.e., a C/M of 1.00). The simulation benchmark comparison statistics are dominated by the large number of measurement samples from the VENUS-3 benchmark which have a reported measurement uncertainty (1σ) of ±5%. Combining this measurement uncertainty with the measurement statistics presented in Table 5-4 results in a benchmark comparison uncertainty (1σ) of ±7% with a bias of –3%.

6.2 Analytic Uncertainty

The analytic uncertainty is determined by estimating the uncertainty in calculated reactor pressure vessel >1.0 MeV neutron fluence resulting from uncertainties in the values of more than two dozen analytical parameters. The results of the analytic uncertainty evaluation are summarized in Table 6-1. The analytic parameters are grouped in the following categories: geometry, material composition, fission source, nuclear cross section data, and modeling inputs.

The uncertainty values for the geometry parameters are based upon the geometric (i.e., mechanical drawing) tolerances for the various reactor components, except for the RPV clad inner radius which is based upon as-built measurements. Geometrical tolerance ranges are assumed to represent $\pm 2\sigma$. The material composition uncertainty parameters are based upon typical steel composition. As with the geometric tolerance, the composition tolerance range is assumed to represent $\pm 2\sigma$. A nuclear cross section parameter uncertainty is approximated by varying the number density of the affected nuclide throughout the problem. This is equivalent to

Table 6-1
Reactor Pressure Vessel Analytic Uncertainty

Parameter	Parameter Uncertainty (1σ)	Energy >1.0 MeV Flux Uncertainty (% (1σ))
Geometry (Total)		**12.3**
Capsule Radial Distance to RPV	0.64 cm	4.0
Capsule Azimuthal Location	0.50 degrees	1.7
Flux Wire Radial Position	0.65 cm	8.0
Flux Wire Azimuthal Position	25.00%	6.5
Flux Wire Axial Position	7.62 cm	1.8
RPV Inner Radius	0.15 cm	1.5
RPV Clad Thickness	0.08 cm	0.7
Shroud Thickness	0.08 cm	0.7
Shroud Inner Radius	0.64 cm	4.3
Jet Pump Azimuthal Location	0.25 degrees	0.1
Jet Pump Riser/Mixer Spacing	0.32 cm	0.1
Jet Pump Mixer Thickness	2.50 cm	0.2
Material Composition (Total)		**2.5**
Core Void Fraction	2.50%	2.0
Reflector Water Density	0.16%	0.2
Downcomer Water Density	0.16%	0.8
Fuel Stack Density	0.50%	0.2
Stainless Steel Cr Composition	0.50%	0.1
Stainless Steel Fe Composition	2.26%	1.2
Stainless Steel Ni Composition	0.63%	0.0
Carbon Steel Fe Composition	0.50%	0.0
Fission Source (Total)		**3.8**
Core Exposure	500 MWd/T	0.2
Peripheral Bundle Power	5.00%	3.8
Nuclear Cross Sections (Total)		**3.4**
Hydrogen	1.00%	0.6
Oxygen (Water)	1.00%	1.6
Oxygen (Fuel)	1.00%	0.1
Fuel (^{238}U)	5.00%	1.5
Chromium	5.00%	0.7
Iron	5.00%	2.4
Nickel	5.00%	0.3
^{235}U Fission/Nufission	5.00%	0.2
^{239}Pu Fission/Nufission	5.00%	0.2
Modeling Input (Total)		**3.4**
Total Analytic Uncertainty		**14.0**

specifying an uncertainty in the macroscopic cross section. The range of uncertainty is assumed to be ±5% for nuclides with resonances and ±1% for nuclides without resonances (predominately hydrogen and oxygen). The nuclear cross section data uncertainties are determined for each nuclide individually, but are reported as a combined value in Table 6-1. The modeling input uncertainty parameters include meshing and integration parameters associated with the RAMA model. The sensitivity of these modeling input parameters is also discussed in Section 4.6 of this report, *Parametric Sensitivity Analyses*.

6.3 Combined Uncertainty

The combined reactor pressure vessel uncertainty is the weighted sum of the analytic, plant-specific computational, and benchmark computational uncertainties from Sections 6.1 and 6.2, as described in the RAMA Fluence Methodology Theory Manual [5] and the Response to NRC Request for Additional Information on RAMA Fluence Methodology [14]. Table 6-2 indicates that the combined uncertainty (1σ) in pressure vessel fluence is 9.2% for energy >1.0 MeV. The biases resulting from the plant-specific measurement comparisons and the benchmark measurement comparisons implicitly include any analytical bias contribution so no explicit analytical bias is included in the uncertainty evaluation. Since the computational bias terms from Section 6.1 are observed to be smaller than the overall combined uncertainty from Table 6-2, it is not necessary to adjust the predicted vessel fluence for bias effects. It should be noted that the combined uncertainty is within the limits prescribed in Regulatory Guide 1.190 for application of the methodology to the prediction of pressure vessel fluence.

Table 6-2
Combined Reactor Pressure Vessel Uncertainty

Energy Range	Analytic Weight Factor	Plant-Specific Comparison Weight Factor	Benchmark Comparison Weight Factor	Combined Uncertainty (% (1σ))
>1.0 MeV	0.16	0.21	0.63	9.2

7 CALCULATED NEUTRON FLUENCE FOR REACTOR PRESSURE VESSEL

The neutron fluence for the reactor pressure vessel at the inner vessel wall (0T), at 1/4T, and at 3/4T is determined by the RAMA Fluence Methodology for the end of cycle 21 and is projected to the end of CNS design life of 32 EFPY. The results of the fluence evaluation are presented in the tables that follow. Values are presented for energy >1.0 MeV. Tables 7-1 and 7-2 report the >1.0 MeV fluence at the end of operating cycle 21 in the RPV shell and weld locations that are included in the axial height of the RAMA fluence model. The location and identification of the RPV welds and shells are shown in Figure 4-3. Tables 7-3 and 7-4 report the >1.0 MeV fluence at 32 EFPY in the RPV shell and weld locations.

It is observed that the fluence is greater at the inner diameter at the point closest to the edge of the core (0T) for all RPV welds and shells. The maximum fluence for the RPV welds is at the inner diameter of circumferential weld VCB-BA-2 with a value of 1.22E+18 n/cm^2 at 32 EFPY. The maximum fluence for the RPV shells is at shell ring #2 with a value of 1.67E+18 n/cm^2 at 32 EFPY.

Note that fluence has exceeded 1.00E+17 n/cm^2 in all welds located in shell rings 1 and 2 as of the end of cycle 21. No welds in shell ring #3 are expected to exceed 1.00E+17 n/cm^2 during the reactor design life. The elevations at which the fluence value first exceeds 1.00E+17 n/cm^2 are 511.73 cm (201.47 inches) for EOC 21 and 502.85 cm (197.97 inches) for 32 EFPY.

Table 7-1
Peak >1.0 MeV Neutron Fluence in Reactor Pressure Vessel Shells at End of Cycle 21

Shell Location	Fluence (n/cm^2) 0T at weld	Fluence (n/cm^2) 1/4T at weld	Fluence (n/cm^2) 3/4T at weld
Shell Ring #1	8.11E+17	5.52E+17	2.16E+17
Shell Ring #2	1.13E+18	7.65E+17	2.96E+17
Shell Ring #3	7.38E+15	4.99E+15	2.06E+15

Table 7-2
Peak >1.0 MeV Neutron Fluence in Reactor Pressure Vessel Welds at End of Cycle 21

Shell Location	Weld	Fluence (n/cm^2) 0T at weld	Fluence (n/cm^2) 1/4T at weld	Fluence (n/cm^2) 3/4T at weld
Shell 1	VLA-BA-1	3.76E+17	2.56E+17	1.04E+17
	VLA-BA-2	7.63E+17	5.16E+17	2.01E+17
	VLA-BA-3	4.52E+17	3.10E+17	1.23E+17
Shell 2	VLB-BA-1	5.64E+17	3.87E+17	1.56E+17
	VLB-BA-2	5.64E+17	3.88E+17	1.56E+17
	VLB-BA-3	5.64E+17	3.87E+17	1.56E+17
Shell 3	VLC-BB-1	6.61E+15	4.53E+15	1.85E+15
	VLC-BB-2	6.93E+15	4.73E+15	1.93E+15
	VLC-BB-3	6.93E+15	4.73E+15	1.93E+15
Shell 1-2	VCB-BA-2	8.11E+17	5.52E+17	2.16E+17
Shell 2-3	VCB-BB-3	7.38E+15	4.99E+15	2.06E+15

Table 7-3
Peak >1.0 MeV Neutron Fluence in Reactor Pressure Vessel Shells at 32 EFPY

Shell Location	Fluence (n/cm^2) 0T at weld	Fluence (n/cm^2) 1/4T at weld	Fluence (n/cm^2) 3/4T at weld
Shell Ring #1	1.22E+18	8.29E+17	3.24E+17
Shell Ring #2	1.67E+18	1.14E+18	4.43E+17
Shell Ring #3	1.25E+16	8.45E+15	3.49E+15

Table 7-4
Peak >1.0 MeV Neutron Fluence in Reactor Pressure Vessel Welds at 32 EFPY

Shell Location	Weld	Fluence (n/cm^2) 0T at weld	Fluence (n/cm^2) 1/4T at weld	Fluence (n/cm^2) 3/4T at weld
Shell 1	VLA-BA-1	5.80E+17	3.96E+17	1.60E+17
	VLA-BA-2	1.15E+18	7.77E+17	3.03E+17
	VLA-BA-3	6.73E+17	4.61E+17	1.84E+17
Shell 2	VLB-BA-1	8.75E+17	6.00E+17	2.42E+17
	VLB-BA-2	8.13E+17	5.60E+17	2.25E+17
	VLB-BA-3	8.75E+17	6.00E+17	2.42E+17
Shell 3	VLC-BB-1	1.12E+16	7.68E+15	3.13E+15
	VLC-BB-2	1.18E+16	8.06E+15	3.29E+15
	VLC-BB-3	1.18E+16	8.06E+15	3.29E+15
Shell 1-2	VCB-BA-2	1.22E+18	8.29E+17	3.24E+17
Shell 2-3	VCB-BB-3	1.25E+16	8.45E+15	3.49E+15

8
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7. *BWRVIP-115: BWR Vessel and Internals Project, RAMA Fluence Methodology Benchmark Manual – Evaluation of Regulatory Guide 1.190 Benchmark Problems*, EPRI, Palo Alto, CA: 2003 1008063.
8. *BWRVIP-117: BWR Vessel and Internals Project, RAMA Fluence Methodology Plant Application – Susquehanna Unit 2 Surveillance Capsule Fluence Evaluation for Cycles 1-5*, EPRI, Palo Alto, CA: 2003. 1008065.
9. "Fluence Evaluation for Oyster Creek Reactor Pressure Vessel", TransWare Enterprises Inc., EXL-FLU-001-R-002, Revision 0, March 29, 2005.
10. "Hope Creek Flux Wire Dosimeter Activation Evaluation for Cycle 1", TransWare Enterprises Inc, TWE-PSE-001-R-001, Revision 0, September 24, 2003.
11. Nebraska Public Power District Calculation Number NEDC 04-018, "RPV Fluence Mechanical Design Inputs", April 30, 2004.
12. Nebraska Public Power District Calculation Number NEDC 04-019, "Cycle 1 to 21 Design Inputs for RAMA Fluence Analysis", July 23, 2004.

13. "Cooper Cycles 1-14 State Point Data", TransWare Enterprises Inc., EPR-VIP-003-A-001, Revision 0, 2004.

14. "Letter from William A. Eaton (BWRVIP) to Meena Khanna (NRC), "Project No. 704 – BWRVIP Response to NRC Request for Additional Information on BWRVIP-114, -115, -117, and -121," dated September 29, 2004.

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The following table identifies those actions committed to by Nebraska Public Power District (NPPD) in this document. Any other actions discussed in the submittal represent intended or planned actions by NPPD. They are described for information only and are not regulatory commitments. Please notify the Licensing Manager at Cooper Nuclear Station of any questions regarding this document or any associated regulatory commitments.

COMMITMENT	COMMITMENT NUMBER	COMMITTED DATE OR OUTAGE
None		