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RS-14-054 10 CFR 50.90
February 26, 2014

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

LaSalle County Station, Unit 1
Facility Operating License No. NPF-11
NRC Docket No. 50-373

Subject: Supplemental Information Supporting License Amendment Request to Revise Reactor Coolant System (RCS) Pressure and Temperature (P/T) Curves for LaSalle County Station, Unit 1

References:
1) Letter from D. M. Gullott (Exelon Generation Company, LLC) to U. S. Nuclear Regulatory Commission, "License Amendment Request to Revise Reactor Coolant System (RCS) Pressure and Temperature (P/T) Curves for LaSalle County Station, Unit 1," dated December 20, 2013

2) Letter from B. Purnell (U. S. Nuclear Regulatory Commission) to M. J. Pacilio (Exelon Generation Company, LLC), "LaSalle County Station, Unit 1 – Supplemental Information Needed for Acceptance of License Amendment Request to Revise Pressure and Temperature Limit Curves (TAC Nos. MF3270 and MF3271)," dated February 18, 2014

In Reference 1, Exelon Generation Company, LLC (EGC) requested an amendment to Facility Operating License No. NPF-11 for LaSalle County Station (LSCS), Unit 1. Specifically, the proposed change would revise Technical Specifications (TS) 3.4.11, "RCS Pressure and Temperature (P/T) Limits," Figures 3.4.11-1 through 3.4.11-3. The changes to TS 3.4.11 are necessary to address the discovery of a non-conservative TS.

In Reference 2, the NRC provided the results of the acceptance review of the license amendment request. The NRC has determined that additional information is needed before the staff can begin its detailed review. The NRC requested that EGC supplement the application to address the requested information by February 27, 2014. In response to this request, EGC is providing the attached information.

EGC has reviewed the information supporting a finding of no significant hazards consideration, and the environmental consideration, that were previously provided to the NRC in Attachment 1 of Reference 1. The additional information provided in this submittal does not affect the bases for concluding that the proposed license amendment does not involve a significant hazards consideration. In addition, the additional information provided in this submittal does not affect the bases for concluding that neither an environmental impact statement nor an environmental assessment needs to be prepared in connection with the proposed amendment.

There are no regulatory commitments contained within this letter. Should you have any questions concerning this letter, please contact Ms. Lisa A. Simpson at (630) 657-2815.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 26th day of February 2014.

Respectfully,

David M. Gullott
Manager – Licensing
Exelon Generation Company, LLC

Attachments:

1) Supplemental Information Supporting License Amendment Request to Revise RCS Pressure and Temperature (P/T) Curves for LaSalle County Station, Unit 1
2) TransWare Reactor Pressure Vessel Fluence Evaluation Performed for LaSalle County Station, Unit 1

cc: NRC Regional Administrator, Region III
NRC Senior Resident Inspector, LaSalle County Station
Illinois Emergency Management Agency – Division of Nuclear Safety

By letter dated December 20, 2013 (Reference 1), Exelon Generation Company, LLC (EGC) submitted a license amendment request (LAR) to revise Technical Specifications (TS) 3.4.11, "RCS Pressure and Temperature (P/T) Limits," Figures 3.4.11-1 through 3.4.11-3, for LaSalle County Station (LSCS), Unit 1. The changes to TS 3.4.11 are necessary to address the discovery of a non-conservative TS.

By letter dated February 18, 2014 (Reference 2), the NRC provided the results of the acceptance review of the license amendment request. The NRC has determined that additional information is needed before the staff can begin its detailed review. EGC is providing the requested information in Attachments 1 and 2.

NRC Request:

Background

The LAR relies on methods described in General Electric-Hitachi (GEH) licensing topical report (TR) NEDC-33178P-A, "GE Hitachi Nuclear Energy Methodology for Development of Reactor Pressure Vessel Pressure-Temperature Curves" (ADAMS Accession No. ML092370487), to determine the revised P/T limits. As a condition of TR NEDC-33178P-A, the licensee needs to determine the fluence values using an NRC-approved methodology.

The LAR states that two different methods were combined to determine the fluence values. The LAR states, "EGC has determined the dual methodology approach utilized to support this LAR results in more conservative fluence input to the P/T curves." However, the LAR states this determination was made using draft calculations.

The NRC staff has not approved the combination of two methodologies for determining fluence values. Therefore, the staff considers this a deviation from TR NEDC-33178P-A, as EGC was previously informed in a letter dated December 5, 2013 (ADAMS Accession No. ML13269A323). The LAR does not provide adequate justification for combining two methods to determine the fluence, since the staff cannot rely on draft information in its safety reviews.

Request

Provide adequate justification for the combination of two methods for determining fluence. Also, provide calculations and other supporting documents used for this justification.

EGC Response:

EGC is providing in Attachment 2 the TransWare Enterprises Inc. (TransWare) calculation EXL-LSA-001-R-001, Revision 0, "LaSalle County Generating Station Unit 1 Reactor Pressure Vessel Fluence Evaluation at End of Cycle 15 with Projections to 32 and 54 EFPY," dated January 20, 2014. This calculation, developed for a future LSCS License Renewal licensing activity, utilized the NRC approved RAMA methodology alone to calculate fluence. As discussed during the NRC and EGC teleconference on February 7, 2014, the proposed LSCS Unit 1 P/T curves provided in Reference 1 do not rely upon the attached TransWare calculation.

The comparison of the fluence values for 32 EFPY between the dual calculation approach (RAMA followed by GEH) used to calculated the revised P/T curves in the LAR and the calculation of RAMA alone performed by TransWare (Attachment 2) indicates that the dual calculation bounds the single calculation (i.e., the peak fluence results for the dual methodology approach utilized to support Reference 1 are greater than the results for RAMA alone). Therefore, EGC has determined that the dual methodology approach utilized to support Reference 1 results in more conservative fluence input to the P/T curves.

<u>EGC Position Regarding Use of Dual Fluence Methodologies</u>

In Reference 3, the NRC provided a closeout letter regarding the LSCS surveillance capsule report and non-conservative Unit 1 P/T limit curves. As stated in Reference 3:

> The licensee used methods described in General Electric-Hitachi (GEH) licensing report TR NEDO-33178-A, "GE Hitachi Nuclear Energy Methodology for Development of Reactor Pressure Vessel Pressure-Temperature Curves" (ADAMS Accession No. ML092370487), to assess the operability of LaSalle, Unit 1, P-T limits. A limitation and condition of the GEH P-T method requires that fluence must be calculated with a single, NRC-approved, fluence method.

EGC's position is that TR NEDO-33178-A does not provide a limitation requiring fluence to be calculated with a single, NRC-approved, fluence method. The Safety Evaluation (SE) for TR NEDO-33178-A (Reference 4) states the following:

> Neutron fluence calculations are acceptable if they are performed with approved methodologies or with methods which are shown to conform to the guidance of RG 1.190.

Attachment 1 of Reference 4 provides an example of a P/T curve report template. Also provided in the SE for TR NEDO-33178-A (Reference 4) is the following statement:

> Section 4.2.1.2 of Attachment 1 [to Reference 4] requires the licensee to identify the report used to calculate the neutron fluence and to document that the plant-specific neutron fluence calculation will be performed using an approved neutron fluence calculation methodology.

The use of singular nouns in the SE does not impose a condition limiting the fluence input to that from a single fluence methodology.

The EGC position is further supported by the following excerpt from the NRC Final SE of BWRVIP-86, Revision 1, the Updated BWR Integrated Surveillance Program Implementation Plan (Reference 5):

> The staff finds that the [Technical Report] fluences (as addressed in the attached SE) are acceptable for use in the TR provided that all licensees continue to use one or more compatible neutron fluence methodologies acceptable to the NRC staff, i.e., which comply with the guidance in Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," to determine surveillance capsule and RPV neutron fluences. Compatible in this case may be understood to mean neutron fluence methodologies which provide results that are within acceptable levels of uncertainty for each calculation.

The above EGC position is also supported by the following excerpt from NRC Regulatory Issue Summary (RIS) 2002-05, "NRC Approval of Boiling Water Reactor Pressure Vessel Integrated Surveillance Program" (Reference 6):

> The NRC staff reviewed the BWRVIP-78 report, the BWRVIP-86 report, and the associated RAI responses. The staff concluded that the final proposed BWRVIP ISP was acceptable for BWR licensee implementation provided that all participating licensees use one or more compatible neutron fluence methodologies acceptable to the NRC staff for determining surveillance capsule and RPV neutron fluences. "Compatible," in this case, means neutron fluence methodologies that provide results that are within acceptable levels of uncertainty for each calculation. This condition of ISP implementation ensures that data from surveillance capsules included in the ISP may be appropriately shared between BWR facilities and that the basis for the neutron fluence determined for a specific capsule is comparable to the RPV which it is intended to represent. This issue is related to the requirements for an ISP found in items a, b, and c of 10 CFR Part 50, Appendix H, paragraph III.C.1.

In conclusion, EGC's position regarding the use of dual fluence methodologies is that TR NEDO-33178-A does not provide a limitation requiring fluence to be calculated with a single, NRC-approved, fluence method, and the use of singular nouns in the SE (Reference 4) does not impose a condition limiting the fluence input to a single fluence methodology.

References

1) Letter from D. M. Gullott (Exelon Generation Company, LLC) to U. S. Nuclear Regulatory Commission, "License Amendment Request to Revise Reactor Coolant System (RCS) Pressure and Temperature (P/T) Curves for LaSalle County Station, Unit 1," dated December 20, 2013

2) Letter from B. Purnell (U. S. Nuclear Regulatory Commission) to M. J. Pacilio (Exelon Generation Company, LLC), "LaSalle County Station, Unit 1 – Supplemental Information Needed for Acceptance of License Amendment Request to Revise Pressure and Temperature Limit Curves (TAC Nos. MF3270 and MF3271)," dated February 18, 2014

3) Letter from N. J. DiFrancesco (U. S. Nuclear Regulatory Commission) to M. J. Pacilio (Exelon Generation Company, LLC), "LaSalle County Station, Unit 1 and 2 – Review Closeout of Surveillance Capsule Report and Non-Conservative Pressure - Temperature Limit Curve Technical Specification (TAC No. MF0500)," dated December 5, 2013

4) Letter from Thomas B. Blount (U. S. Nuclear Regulatory Commission) to Mr. Doug Coleman, (Boiling Water Reactor Owners' Group), "Final Safety Evaluation for Boiling Water Reactors Owners' Group Licensing Topical Report NEDC-33178P, General Electric Methodology for Development of Reactor Pressure Vessel Pressure-Temperature Curves (TAC No. MD2693)," dated April 27, 2009

5) Letter from R. A. Nelson (U. S. Nuclear Regulatory Commission) to D. Czufin, "Final Safety Evaluation for Electric Power Research Institute Boiling Water Reactor Vessel and Internals Project Technical Report 1016575, 'BWRVIP-86, Revision 1: BWR Vessel and Internals Project, Updated BWR Integrated Surveillance Program (ISP) Implementation Plan' (TAC No. ME2190)," dated October 20, 2011

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

ATTACHMENT 2

TransWare Reactor Pressure Vessel Fluence Evaluation

Performed for LaSalle County Station, Unit 1

EXL-LSA-001-R-001
Revision 0
January 2014

62 pages follow

LASALLE COUNTY GENERATING STATION UNIT 1 REACTOR PRESSURE VESSEL FLUENCE EVALUATION AT END OF CYCLE 15 WITH PROJECTIONS TO 32 AND 54 EFPY

Document Number: EXL-LSA-001-R-001
Revision 0

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DISCLAIMER OF WARRANTIES AND LIMITATION OF LIABILITIES

QUALITY REQUIREMENTS

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

ACKNOWLEDGMENTS

The undersigned wish to acknowledge Christine Kinkead, Mike Guthrie, and JoAnn Shields for their management support at Exelon. In addition, I wish to acknowledge Dale Bradish, Letty Guitierrez, and Pete Weggeman for their support and assistance in providing the mechanical design and operating data for this reactor pressure vessel fluence evaluation.

Eric N. Jones, TransWare Enterprises Inc.

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GLOSSARY OF TERMS

AZIMUTHAL QUADRANT SYMMETRY – A type of core and pressure vessel configuration that can be represented by a single quadrant that can be rotated and mirrored to represent the entire 360-degree geometry. The northeast quadrant can be mirrored to represent the northwest and southeast quadrants and can be rotated to represent the southwest quadrant.

BEST-ESTIMATE NEUTRON FLUENCE – See Neutron Fluence.

CALCULATED NEUTRON FLUENCE – See Neutron Fluence.

CALCULATIONAL BIAS – A calculational adjustment based on comparisons of calculations to measurements. If a bias is determined to exist, it may be applied as a multiplicative correction to the calculated fluence to determine the best-estimate neutron fluence.

CORE BELTLINE – The axial elevations corresponding to the active fuel region of the core.

EFFECTIVE FULL POWER YEARS (EFPY) – A unit of measurement representing one full year at the reactor's rated power level for that time period. If the reactor operates for 10 months at full rated power, then goes into a power uprate and continues operating for another 2 months at the new full rated power, this represents 1 EFPY.

FAST NEUTRON FLUENCE – Fluence accumulated by neutrons with energy greater than 1.0 MeV (>1.0 MeV).

NEUTRON FLUENCE – Time-integrated neutron flux reported in units of n/cm^2. The term "best-estimate" fluence refers to the values that are computed in accordance with the requirements of U. S. Nuclear Regulatory Commission Regulatory Guide 1.190 for use in material embrittlement evaluations. The term "calculated" fluence refers to the values that are predicted by the RAMA Fluence Methodology. In this report the best-estimate fluence is synonymous with the calculated fluence since the RAMA Fluence Methodology requires no bias correction to the calculated fluence.

RPV – An abbreviation for reactor pressure vessel. Unless otherwise noted, the reactor pressure vessel refers to the base metal material (i.e., excluding clad/liner).

RPV BELTLINE – The region of the reactor vessel (shell material, including welds, heat-affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage. For purposes of this evaluation, the adjacent regions predicted to experience sufficient neutron radiation damage are considered to be those within the axial elevations corresponding to the regions where the RPV exceeds a fast neutron fluence of 1.0E17 n/cm^2.

RPV ZERO ELEVATION – Axial elevations cited in this report assume that RPV zero is at the inside surface of the pressure vessel at the bottom head nozzle.

1
INTRODUCTION

This report presents the results of the reactor pressure vessel fluence evaluation performed for the LaSalle County Generating Station Unit 1 reactor (LaSalle 1). In this analysis, maximum fast neutron (energy > 1.0 MeV) fluence is reported for the reactor pressure vessel (RPV) plates, welds, and nozzles throughout the RPV beltline region at the interface of the RPV base metal and cladding, hereafter denoted as the 0T location of the RPV wall. Fluence attenuations are performed through the RPV wall to the 1/4 T and 3/4 T locations using the displacements per atom (DPA) attenuation method prescribed in U. S. Nuclear Regulatory Commission (NRC) Regulatory Guide 1.99, "*Radiation Embrittlement of Reactor Vessel Materials*" [1]. The fluence is determined at the end of cycle (EOC) 15 and projected to 32 effective full power years of operation (EFPY) and 54 EFPY, which is assumed to represent the end of the reactor's extended operating license.

The fluence evaluation was performed in accordance with guidelines presented in U. S. NRC Regulatory Guide 1.190, "*Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence*" [2]. In compliance with these guidelines, comparisons to surveillance capsule flux wire and dosimeter measurements were performed to determine the accuracy of the RPV fluence model. An uncertainty analysis was also performed to determine if a statistical bias exists in the model. It was determined that the LaSalle 1 fluence model does not have a statistical bias and that the best-estimate fluence presented in this report is suitable for use in evaluating the effects of embrittlement on RPV material as specified in U. S. Code of Federal Regulations (CFR) 10CFR50 Appendix G, "*Fracture Toughness Requirements*" [3] and U. S. NRC Regulatory Guide 1.99.

The fluence values presented in this report were calculated using the RAMA Fluence Methodology [4]. Under funding from Electric Power Research Institute, Inc. (EPRI) and the Boiling Water Reactor Vessel and Internals Project (BWRVIP), the RAMA Fluence Methodology was developed by TransWare Enterprises Inc. for the purpose of calculating fast neutron fluence in nuclear reactor pressure vessels and vessel internal components. The RAMA Fluence Methodology (hereinafter referred to as RAMA) has been reviewed by the U. S. NRC on two separate occasions, resulting in the issuance of Safety Evaluation Reports (SERs) providing conditional approval of the RAMA methodology for determining fast neutron fluence in BWR pressure vessels [5] and vessel internal components that include the core shroud and top guide [6]. As part of the license renewal application for the Seabrook Station Nuclear Power Plant [7], the U.S. NRC performed a subsequent review of the methodology, resulting in an extended approval of the RAMA methodology, specifically noting that "the RAMA methodology is generically approved by the NRC staff for [determining pressure vessel neutron fluence in] BWRs and PWRs." The conditional approval of the RAMA methodology for determining neutron fluence in reactor internals is specifically cited in [6] stating that the staff will "accept RAMA calculated fluence values for BWR core shrouds and top guide for licensing actions provided that the calculation results are supported by sufficient justification that the proposed values are conservative for the intended application."

As prescribed in Regulatory Guide 1.190, TransWare Enterprises Inc. has benchmarked the RAMA Fluence Methodology against industry standard benchmarks for both boiling water reactor (BWR) and pressurized water reactor (PWR) designs, and compared with plant-specific dosimetry measurements and reported fluence from numerous commercial operating reactors. The results of the benchmarks and comparisons to measurements that TransWare has performed show that the RAMA methodology accurately predicts specimen activities in all light water reactor types. The overall comparison results determined by TransWare for over 1,150 measurements is calculated to be 1.02 ± 0.09.

The information and associated evaluations provided in this report have been performed in accordance with the requirements of 10CFR50 Appendix B [8].

2
SUMMARY OF RESULTS

This section provides a summary of the fast neutron fluence evaluation performed for the LaSalle 1 pressure vessel based on operating data through cycle 14 and projected data for cycles 15 through 18. Fluence was calculated at EOC 15 (22.7 EFPY) and projected through the end of the reactor's design life of 32 EFPY and extended design life of 54 EFPY. The focus of the RPV evaluation was to determine the maximum fast neutron fluence accumulated in plates, welds, and nozzles of the RPV beltline region at the 0T, 1/4 T, and 3/4 T locations. Table 2-1 summarizes the maximum fast neutron fluence (0T only) for the RPV welds, plates, and nozzles included in this evaluation at the three reporting periods. Fluences that exceed the threshold fluence of 1.0E+17 n/cm^2, used to define the RPV beltline, prior to the end of the respective time period are shown in red. As is evident from the table, the threshold fluence value is exceeded in all welds, plates, and nozzles prior to the EOC 15, with the exception of the N2 and N12 nozzle locations. Note that the maximum fluence value is 1.06E+18 n/cm^2 at 54 EFPY and is highlighted in **bold** type in Table 2-1. It occurs at the RPV inner diameter in the lower intermediate shell plates G-5604-1, G-5604-2, and G-5604-3.

Figure 7-1 in Section 7 illustrates the location of the welds, plates, and nozzles in the RPV. Fluence attenuations through the RPV were performed using the DPA attenuation method specified in U. S. Regulatory Guide 1.99 [1].

Table 2-1
Maximum Fast Neutron Fluence for LaSalle 1 RPV Beltline Welds, Nozzles, and Plate Locations

Component	Heat No.	Fast Neutron Fluence at 0T (n/cm^2)		
		EOC 15 (22.7 EFPY)	32 EFPY	54 EFPY
RPV Beltline Welds				
Middle Shell Axial Weld 3-308A	305424 1P3571	3.57E+17	5.09E+17	8.66E+17
Middle Shell Axial Weld 3-308B		2.79E+17	4.03E+17	6.95E+17
Middle Shell Axial Weld 3-308C		2.02E+17	2.88E+17	4.92E+17
Lower Int. Shell Axial Weld 4-308A	12008 305414	3.44E+17	4.81E+17	8.00E+17
Lower Int. Shell Axial Weld 4-308B		2.71E+17	3.73E+17	6.17E+17
Lower Int. Shell Axial Weld 4-308C		4.38E+17	6.06E+17	1.00E+18
Lower Shell Axial Weld 2-307A	21935 12008	1.46E+17	1.97E+17	3.18E+17
Lower Shell Axial Weld 2-307B		1.45E+17	1.98E+17	3.27E+17
Lower Shell Axial Weld 2-307C		1.31E+17	1.79E+17	2.92E+17
Middle/Lower Int. Shell Girth Weld 6-308	6329637	3.70E+17	5.29E+17	9.01E+17
Lower Int./Lower Shell Girth Weld 1-313	4P6519	1.74E+17	2.34E+17	3.75E+17
Nozzle Forging-to-Base-Metal Welds				
Nozzle Weld N2	Q2Q22W	8.67E+15	1.19E+16	1.93E+16
Nozzle Weld N6		1.74E+17	2.60E+17	4.63E+17
Nozzle Weld N12		7.56E+16	1.14E+17	2.06E+17
Shell Plates				
Middle Shell Plate G-5605-1	A5333-1	3.70E+17	5.29E+17	9.01E+17
Middle Shell Plate G-5605-2	B0078-1	3.70E+17	5.29E+17	9.01E+17
Middle Shell Plate G-5605-3	C6123-2	3.70E+17	5.29E+17	9.01E+17
Lower Int. Shell Plate G-5604-1	**C6345-1**	**4.67E+17**	**6.45E+17**	**1.06E+18**
Lower Int. Shell Plate G-5604-2	**C6318-1**	**4.67E+17**	**6.45E+17**	**1.06E+18**
Lower Int. Shell Plate G-5604-3	**C6345-2**	**4.67E+17**	**6.45E+17**	**1.06E+18**
Lower Shell Plate G-5603-1	C5978-1	1.74E+17	2.34E+17	3.75E+17
Lower Shell Plate G-5603-2	C5978-2	1.74E+17	2.34E+17	3.75E+17
Lower Shell Plate G-5603-3	C5979-1	1.74E+17	2.34E+17	3.75E+17

The reactor beltline region, as defined in Appendices G [3] and H [9] of 10CFR50, includes the region that directly surrounds the effective height of the reactor core as well as those adjacent areas of the RPV that are predicted to experience sufficient neutron irradiation damage. As identified for the definition of the RPV beltline, this is considered to include all materials that exceed a fast neutron fluence of 1.0E+17 n/cm^2. The elevation ranges at which the RPV fluence exceeds 1.0E+17 n/cm^2 for the three reporting periods are given in Table 2-2. These elevations define the RPV beltline region for that time period.

Table 2-2
RPV Beltline Elevation Range for LaSalle 1

Reactor Lifetime	Lower Elevation [cm (in)]	Upper Elevation [cm (in)]
EOC 15 (22.7 EFPY)	556.50 (219.09)	929.95 (366.12)
32 EFPY	546.19 (215.03)	940.07 (370.11)
54 EFPY	533.08 (209.88)	953.82 (375.52)

Section 7 contains tables of results listing the fast neutron fluence for all of the RPV circumferential (girth) welds, vertical (axial) welds, nozzles, and plates included in this evaluation at 0T, 1/4 T and 3/4 T.

Comparisons were made between the calculated and measured specific activities for three sets of surveillance capsule dosimetry specimens removed from LaSalle 1. The total average calculated-to-measured (C/M) result for all LaSalle 1 dosimetry specimens is 1.02 with a standard deviation of ±8%. A more detailed assessment of the surveillance capsule activation analyses is presented in Section 5 of this report.

Another result from this evaluation is the calculated RPV fluence combined uncertainty value. By combining the measurement uncertainty and analytic uncertainty, the combined RPV fast neutron fluence uncertainty is determined to be 9.2% with no bias correction to the fluence.
In conclusion, it is determined that RAMA produces results that meet the requirements of U. S. NRC Regulatory Guide 1.190.

3
DESCRIPTION OF THE REACTOR SYSTEM

This section provides an overview of the reactor design and operating data inputs that were used to develop the LaSalle 1 reactor fluence model. All reactor design and operating data inputs used to develop the model were plant-specific and were provided by Exelon Generation Company, LLC [10, 11]. The inputs for the fluence geometry model were developed from design and as-built drawings for the reactor pressure vessel, vessel internals, fuel assemblies, and containment regions. Several modifications were made to the LaSalle 1 RAMA geometry model since the previous fluence evaluation performed by TransWare in 2011 [12]. These modifications include:

- Adding RPV nozzle penetrations including N2, N6 and N12 nozzles
- Adding jet pump repair assemblies, hold down beams and hold down brackets.
- Adding core support plate rim bolts

The reactor operating data inputs were developed from core simulator data that provided a historical accounting of how the reactor operated for cycles 4 through 14. At the time of this fluence analysis, a portion of cycle 15 had been completed. Data through February 2013 was based on actual operation with the remainder of cycle 15 being based on projected operation. Core simulator data was not available for cycles 1 through 3. Data for these cycles was approximated using information from cycle summary reports and nodal software combined with detailed operating data from cycles of comparable core design and energy production. Exelon provided projection cycle data for cycles 16 through 18 [10]. The operating data provided for projection cycle 18 was used to project fluence to the end of the reactor's operating life and extended operating life.

3.1 Overview of the Reactor System Design

LaSalle 1 is a General Electric BWR/5 class reactor with a core loading of 764 fuel assemblies. LaSalle 1 began commercial operation in 1982 with a design rated power of 3323 MWt. Midway through cycle 9 the rated power was increased to 3489 MWt. Another power uprate was implemented midway through cycle 14 to increase the rated power to 3546 MWt. At the time of this fluence analysis, LaSalle 1 had completed 14 cycles of operation and a portion of cycle 15.

Figure 3-1 illustrates the basic planar configuration of the LaSalle 1 reactor at an axial elevation near the reactor core mid-plane. All of the radial regions of the reactor that are required for fluence projections are shown. Beginning at the center of the reactor and projecting outward, the regions include: the core region, including control rod locations and fuel assembly locations (fuel locations are shown only for the 0-to-90-degree quadrant); core reflector region (bypass water); central shroud wall; downcomer water region including the jet pumps; reactor pressure vessel (RPV) wall; cavity region between the RPV wall and insulation; insulation; biological shield (concrete wall); and cavity regions between the RPV and biological shield.

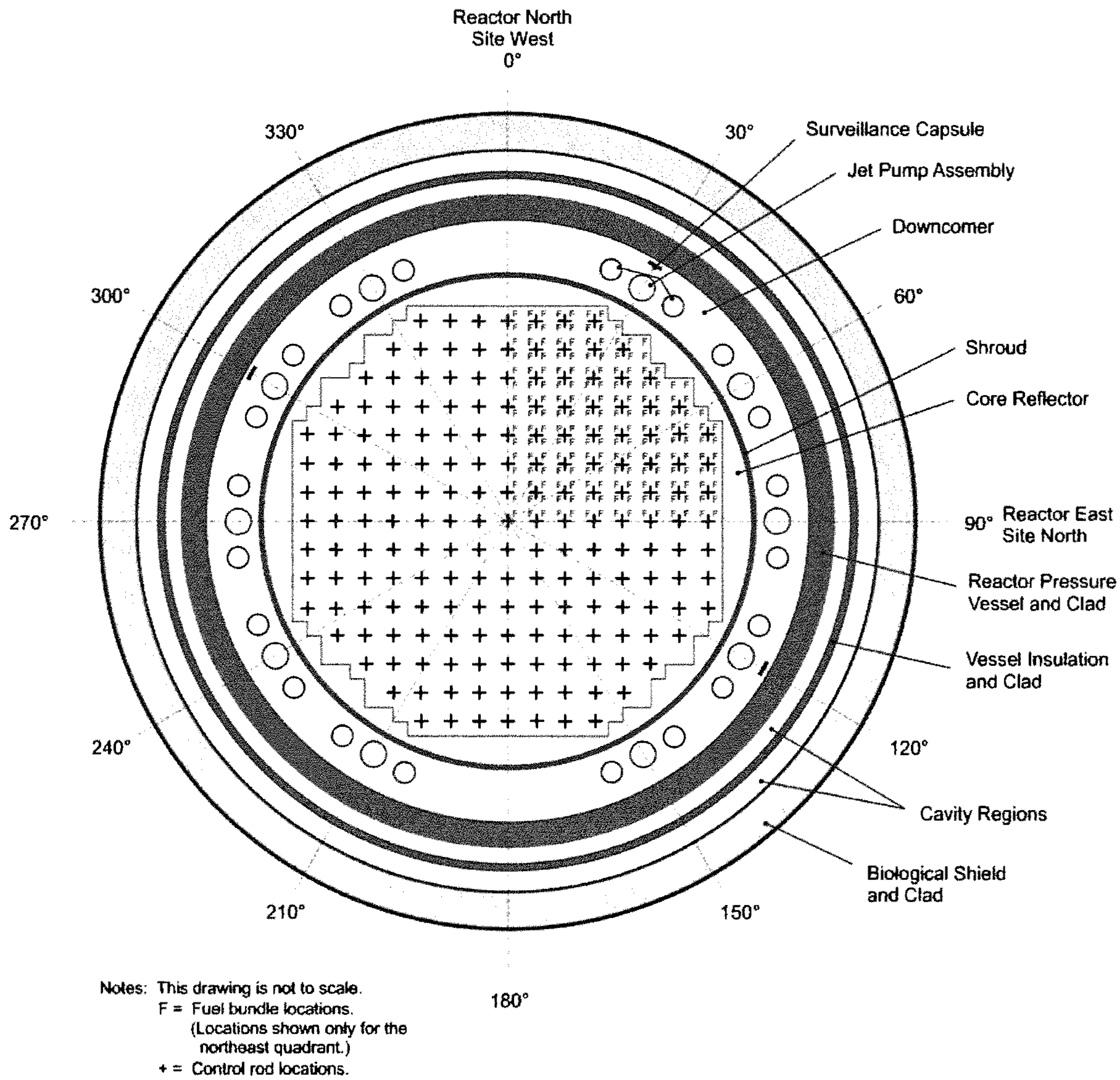

Figure 3-1
Planar View of LaSalle 1 at the Core Mid-Plane Elevation

3.2 Reactor System Mechanical Design Inputs

The mechanical design inputs that were used to construct the LaSalle 1 fluence geometry model included as-built and nominal design dimensional data. As-built data for the reactor components and regions of the reactor system is always preferred when constructing plant-specific models; however, as-built data is not always available. In these situations, nominal design information is used.

For the LaSalle 1 fluence model, the predominant dimensional information used to construct the fluence model was nominal design data. As-built data was used for the following dimensions:

- Core support plate outer radius
- Core support plate height
- Core support plate inner radius
- Core support height rim + plate

Another important component of the fluence analysis is the accurate description of the surveillance capsules in the reactor. It is shown in Figure 3-1 that three surveillance capsules were initially installed in the LaSalle 1 reactor. The capsules were attached radially to the inside surface of the RPV (looking outward from the core region) at the 30-, 120-, and 300-degree azimuths. Surveillance capsules are used to monitor the radiation accumulated in the reactor over a period of time. The importance of surveillance capsules in fluence analyses is that they contain flux wires that are irradiated during reactor operation. When a capsule is removed from the reactor, the irradiated flux wires are evaluated to obtain activity measurements. These measurements are used to validate the fluence model. Three sets of flux wires have been removed from the LaSalle 1 reactor and analyzed. (See Section 5, which presents a comparison of the calculated-to-measured capsule results.)

3.3 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 materials for the various components and regions of the LaSalle 1 reactor. The material 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 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, several state-point data sets are used to describe the operating states of the reactor for each operating cycle. The number of data sets used in this analysis is presented in Section 3.4.3.1.

3.4 Reactor Operating Data Inputs

An accurate evaluation of reactor vessel and component fluence requires an accurate accounting of the reactor's operating history. The primary reactor operating parameters that affect the determination of fast neutron fluence in light water reactors include reactor power levels, core power distributions, coolant water density distributions, and fuel material (isotopic) distributions.

Table 3-1
Summary of Material Compositions by Region for LaSalle 1

Region	Material Composition
Control Rods and Guide Tubes	Stainless Steel, B_4C
Core Support Plate	Stainless Steel
Core Support Plate Rim Bolts	Stainless Steel
Fuel Support Piece	Stainless Steel
Fuel Assembly Lower Tie Plate	Stainless Steel, Zircaloy, Inconel
Reactor Core	^{235}U, ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, O_{fuel}, Zircaloy
Reactor Coolant / Moderator	Water
Core Reflector	Water
Fuel Assembly Upper Tie Plate	Stainless Steel, Zircaloy, Inconel
Top Guide	Stainless Steel
Core Spray Sparger Pipes	Stainless Steel
Core Spray Sparger Flow Areas	Water
Shroud	Stainless Steel
Downcomer Region	Water
Jet Pump Riser and Mixer Flow Areas	Water
Jet Pump Riser and Mixer Metal	Stainless Steel
Jet Pump Riser Brace and Pads	Stainless Steel
Jet Pump Hold Down Beams	Inconel
Jet Pump Hold Down Brackets	Stainless Steel
Surveillance Capsule Specimens	Carbon Steel
Reactor Pressure Vessel Clad	Stainless Steel
Reactor Pressure Vessel Wall	Carbon Steel
RPV Nozzle Forgings	Carbon Steel
RPV Nozzle Forging Interior	Water
Cavity Regions	Air (Nitrogen)
Insulation Clad	Stainless Steel
Insulation	Aluminum Foil
Biological Shield Clad	Carbon Steel
Biological Shield Wall	Reinforced Concrete

3.4.1 Core Loading

It is common in BWRs that more than one fuel assembly design may 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 and the top and bottom fuel elevations).

Three different fuel assembly designs were loaded in the LaSalle 1 reactor during the period included in this evaluation. Table 3-2 provides a summary of the fuel designs loaded in the reactor core for these operating cycles. The cycle core loading provided by Exelon Generation Company, LLC was used to identify the fuel assembly designs in each cycle and their location in the core loading inventory. For each cycle, appropriate fuel assembly models were used to build the reactor core region of the LaSalle 1 RAMA fluence model.

3.4.2 Power History Data

Reactor power history is the measure of reactor power levels and core exposure on a continual or periodic basis. For this fluence evaluation, the power history for the LaSalle 1 reactor was developed from power history inputs provided by Exelon Generation Company, LLC [10, 11]. The power history data showed that LaSalle 1 started commercial operation with a design rated thermal power of 3323 MWt for cycles 1 through 8. Midway through cycle 9, a power uprate occurred bringing the thermal power to 3489 MWt. Another power uprate was implemented midway through cycle 14 raising the thermal power to 3546 MWt. It was assumed in this analysis that all future cycles would operate at the 3546 MWt power level.

The power history data for LaSalle 1 included daily power levels for most cycles. When daily power histories weren't available, average power levels were constructed based on exposure accumulation using the core simulator codes. This data was used to calculate the vessel fluence. Periods of reactor shutdown due to refueling outages and other events were also accounted for in the model. The power history data was verified by comparing the calculated energy production in effective full power years with power production records provided by Exelon Generation Company, LLC. Table 3-3 lists the accumulated EFPY at the end of each cycle for this fluence evaluation.

3.4.3 Reactor State-Point Data

Cycle 1 of LaSalle 1 was derived from a Cycle Summary Report (CSR), which contained bundle average exposures. Cycles 2 and 3 were generated from early PANACEA core simulator data that provided bundle average exposure data. Data from cycle 4 was used to provide axial power shapes, pin-by-pin data, fuel assembly orientation and control rod patterns throughout the cycle for cycles 1 through 3. This represented the best available information for these cycles.

Table 3-2
Summary of LaSalle 1 Core Loading Inventory

Cycle	8x8 Designs				9x9 Designs	10x10 Designs			Dominant Peripheral Design
	GE5	GE7B	GE8B	GE9	ATRIUM-9B	ATRIUM-10	GE14	GNF2	
1	764								GE5
2	532	232							GE5
3	308	232	224						GE5
4	137	231	224	172					GE5
5		176	224	364					GE7
6		44	156	564					GE8
7				764					GE9
8				764					GE9
9A/9B				392	372				GE9
10A				46	372	346			GE9(1)
10B				49	369	346			GE9
11					130	344	290		ATRIUM-9B
12						474	290		ATRIUM-10
13						603	161		GE14
14						764			ATRIUM-10
15						468		296	ATRIUM-10
16						184		580	ATRIUM-10
17								764	GNF2
18								764	GNF2
19+(2)								764	GNF2

1) In cycle 10A, the periphery is composed of an even number of GE9 and ATRIUM-9B bundles. GE9 is chosen as the dominant fuel type because a few ATRIUM-9B leakers were found partway through the cycle and were swapped out for GE9 bundles.
2) Cycles 19 and beyond use cycle 18 projection data provided by Exelon for projecting fluence to the end of the extended plant license period.

Table 3-3
State-point Data for Each Cycle of LaSalle 1

Cycle Number	Number of State Point Data Files	Rated Thermal Power[1] (MWt)	Accumulated Effective Full Power Years (EFPY)
1	3	3323	1.4
2	3	3323	2.2
3	3	3323	3.3
4	12	3323	4.3
5	16	3323	5.6
6	8	3323	6.5
7	19	3323	7.8
8	17	3323	9.2
9A	4	3323	9.7
9B	19	3489	11.3
10A	3	3489	11.6
10B	20	3489	13.2
11	19	3489	15.2
12	17	3489	17.0
13	21	3489	18.9
14[2]	25	3489/3546	20.8
15	20	3546	22.7
16	18	3546	24.6
17	18	3546	26.5
18	17	3546	28.4
19+[3]	17	3546	54.0

1) The rated thermal power is listed for each cycle. The actual power levels were used for the individual state-point calculations for cycles 1-14. Projected power levels provided by Exelon Generation Company, LLC were used for cycles 15 through the end of the extended plant license period.
2) Power uprate to 3546 MWt was achieved on 9/26/2010 during cycle 14's operation.
3) Cycles 16-18 projected operating data was provided by Exelon Generation Company, LLC using PANAC11A simulator data. Cycles 19 and beyond use projected cycle 18 data for projecting fluence to the end of the extended plant license period.

Core simulator data was provided by Exelon Generation Company, LLC to characterize the historical operating conditions of LaSalle 1 for cycles 4 through 14 and the first portion of cycle 15. Additional core simulator data was provided to characterize the projected operation of the remainder of cycle 15 and cycles 16-18. The data calculated with core simulator codes represents the best-available information about the reactor core's operating history over the reactor's operating life. In this analysis, the core simulator data provided by Exelon was processed by TransWare to generate state-point data files for input to the RAMA fluence model [10, 11]. The state-point files included three-dimensional data arrays that described core power distributions, fuel exposure distributions, fuel materials (isotopics), and coolant water densities.

A separate neutron transport calculation was performed for each of the state points tallied in Table 3-3. The calculated neutron flux for each state point was combined with the appropriate power history data described in Section 3.4.2 in order to provide an accurate accounting of the fast neutron fluence for the reactor pressure vessel. Fluence projections to the end of the reactor's design life and extended design life were performed using projected equilibrium cycles. Equilibrium cycles are discussed in Section 3.4.3.2.

3.4.3.1 Beginning of Operation through Cycle 18 State Points

A total of 282 state points were used to represent the operating history for the first 14 completed operating cycles, partially completed cycle 15, and cycle projections through the end of cycle 18 of LaSalle 1. These state points were selected from hundreds of exposure points that were calculated with the core simulator code. The hundreds of exposure points were evaluated and grouped into a fewer number of exposure ranges in order to reduce the number of transport calculations required to perform the fluence evaluation. Several criteria were used in the determination of the exposure ranges, including evaluations of core thermal powers, core flows, core power profiles, and control rod patterns. In determining exposure ranges, it is assumed that there will be at least one exposure step in that range that would accurately represent the average operating conditions of the reactor over that range. This single exposure step is then referred to as the "state point". Table 3-3 shows the number of state points used for each cycle in this fluence evaluation.

3.4.3.2 Projected Reactor Operation

Projections of plant operations beyond February 2013 are represented by cycle projections through cycle 18 and an "equilibrium" cycle that incorporates the best-available information on expected cycle length, fuel bundle loading, and operating strategies for future cycles beyond cycle 18. Cycle 18, with a projected thermal power level of 3546 MWt, is used as the equilibrium cycle for this analysis to project fluence to the end of the extended plant design life of 54 EFPY. Note that at the time of this evaluation, cycle 15 had been partially completed. Actual operating history through February 2013 was incorporated for cycle 15 with the remainder of the cycle being based on projected operation.

3.4.3.3 Limitation of Fluence Projections

The fluence values presented in this report are based on projections of LaSalle 1's operations beyond the current operating cycle. Projections are performed using an assumed equilibrium cycle. The significance of the equilibrium cycle is that it defines the flux profiles that are used to project fluence into the future. Providing that the design basis for the equilibrium cycle does not change appreciably, projections based on the equilibrium cycle should remain bounding through 54 EFPY to support licensing, in-service inspection, and flaw evaluation activities.

If the design basis for the equilibrium cycle changes at any point in time, thereby producing a significant change to the flux profiles for the equilibrium cycle, then a new evaluation is needed. Operating conditions, if changed, that could impact the validity of the equilibrium cycle include power uprates, introduction of new fuel designs, changes in projected cycle lengths, changes in core loading strategies, changes in reactor flow, or other changes that could alter the flux profiles used in the fluence projections.

4
CALCULATION METHODOLOGY

This section provides an overview of the LaSalle 1 fluence model that was developed with the RAMA Fluence Methodology software [4]. The RAMA fluence model for LaSalle 1 is a plant-specific model that was constructed from the design inputs described in Section 3, *Description of the Reactor System.*

4.1 Description of the RAMA Fluence Methodology

The RAMA Fluence Methodology (RAMA) is a system of computer codes, a data library, and an uncertainty methodology that determines best-estimate fluence in light water reactor pressure vessels and vessel components. The primary codes that comprise the RAMA methodology include model builder codes, a particle transport code, and a fluence calculator code. The data library contains nuclear cross sections and response functions that are needed for each of the codes. The uncertainty methodology is used to determine the uncertainty and bias in the best-estimate fluence calculated by the software.

The primary inputs for RAMA are mechanical design parameters and reactor operating history data. The mechanical design inputs are obtained from plant-specific design drawings, which include as-built measurements when available. The reactor operating history data is obtained from reactor core simulator codes, system heat balance calculations, daily operating logs, and cycle summary reports that describe the operating conditions of the reactor over its operating lifetime. The primary outputs from RAMA calculations are neutron flux, neutron fluence, dosimetry activation, and an uncertainty evaluation.

The model builder codes consist of geometry and material processor codes that generate input for the particle transport code. The geometry model builder code uses mechanical design inputs and meshing specifications to generate three-dimensional geometry models of the reactor. The material processor code uses reactor operating data inputs to process fuel materials, structural materials, and water densities that are consistent with the geometry meshing generated by the geometry model builder code.

The particle transport code performs three-dimensional neutron flux calculations using a deterministic, multigroup, particle transport theory method with anisotropic scattering. The primary inputs prepared by the user for the transport code include the geometry and material data generated by the model builder codes and numerical integration and convergence parameters for the iterative transport calculation. The transport solver is coupled with a general geometry modeling capability based on combinatorial geometry techniques. The coupling of general geometry with a deterministic transport solver provides a flexible, accurate, and efficient tool for calculating neutron flux in light water reactor pressure vessels and vessel components. The primary output from the transport code is the neutron flux in multigroup form.

The fluence calculator code determines fluence and activation in the reactor pressure vessel and vessel components over specified periods of reactor operation. The primary inputs to the fluence calculator include the multigroup neutron flux from the transport code, response functions for the various materials in the reactor, reactor power levels for the operating periods of interest, the specification of which components to evaluate, and the energy ranges of interest for evaluating neutron fluence. The fluence calculator includes treatments for isotopic production and decay that are required to calculate specific activities for irradiated materials. The reactor operating history is generally represented with several reactor state points that represent the various power levels and core power shapes generated by the reactor over the life of the plant. These detailed state points are combined with the daily reactor power levels to produce accurate estimates of the fluence and activations accumulated in the plant.

The uncertainty methodology provides an assessment of the overall accuracy of the RAMA Fluence Methodology. Variances in the dimensional data, reactor operating data, dosimetry measurement data, and nuclear data are evaluated to determine if there is a statistically significant bias in the calculated results that might affect the determination of the best-estimate fluence for the reactor. The plant-specific results are also weighted with comparative results from experimental benchmarks and other plant analyses and analytical uncertainties pertaining to the methodology to determine if the plant-specific model under evaluation is statistically acceptable as defined in Regulatory Guide 1.190.

The RAMA nuclear data library contains atomic mass data, nuclear cross-section data, and response functions that are needed in the material processing, transport, fluence, and reaction rate calculations. The cross-section data and response functions are based on the BUGLE-96 nuclear data library [13] and the VITAMIN-B6 data library [14].

4.2 The RAMA Geometry Model for the LaSalle 1 Reactor

Section 3, *Description of the Reactor System*, describes the design inputs that were provided by Exelon for the LaSalle 1 reactor fluence evaluation. These design inputs were used to develop a plant-specific, three-dimensional computer model of the LaSalle 1 reactor with the RAMA Fluence Methodology.

Figures 4-1 and 4-2 provide general illustrations of the primary components, structures and regions developed for the LaSalle 1 fluence model. Figure 4-1 shows the planar configuration of the reactor model at an elevation corresponding to the reactor core mid-plane elevation. Figure 4-2 shows an axial configuration of the reactor model. Note that the figures are not drawn to scale. They are intended only to provide a perspective for the layout of the model, and specifically how the various components, structures, and regions lie relative to the reactor core region (i.e., the neutron source).

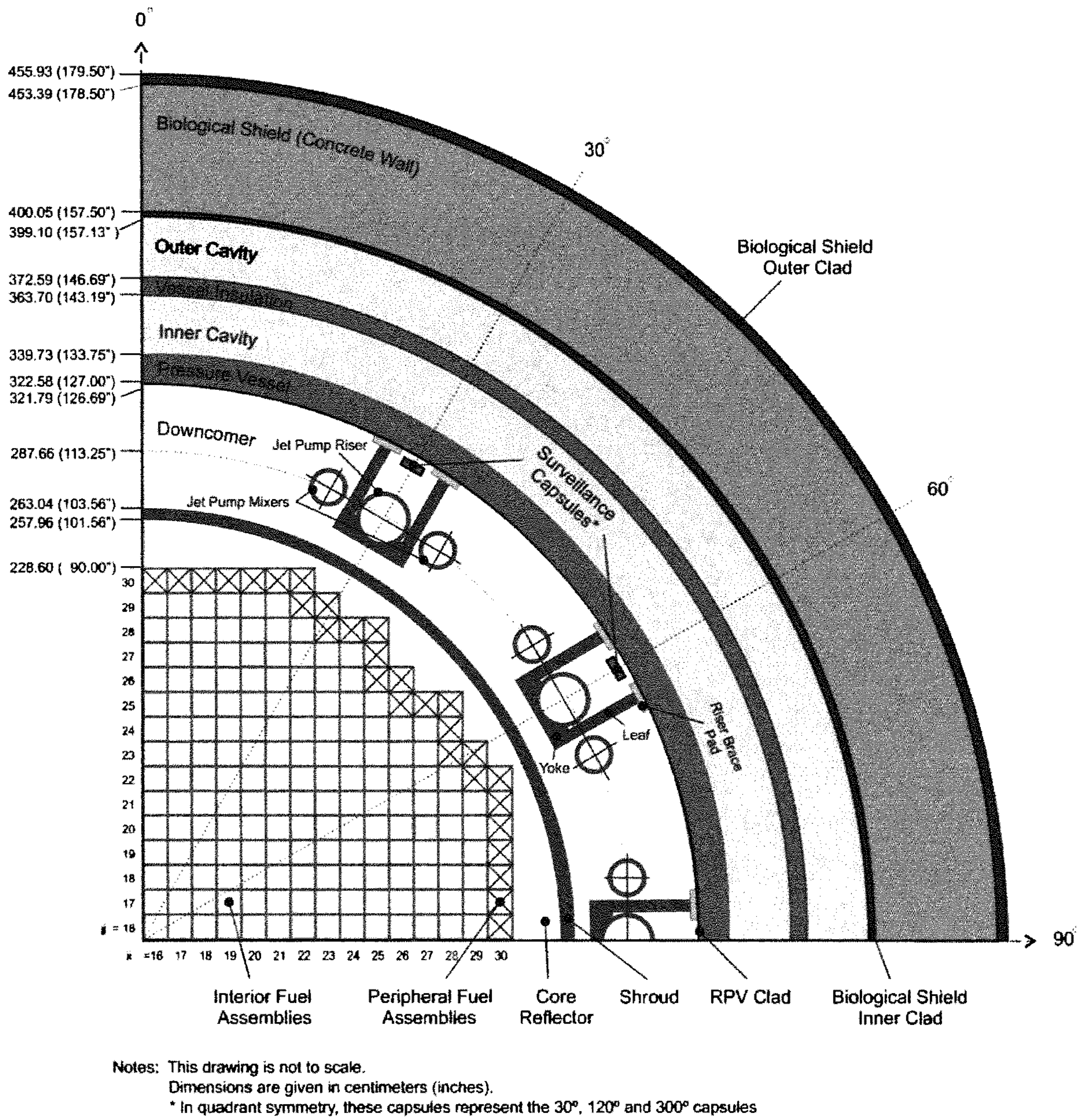

Figure 4-1
Planar View of the LaSalle 1 RAMA Quadrant Model at the Core Mid-Plane Elevation

Because the figures are intended only to provide a general overview of the model, they do not include illustrations of the geometry meshing developed for the model. To provide such detail is beyond the scope of this document.

The following subsections provide an overview of the computer models that were developed for the various components, structures, and coolant flow regions of the LaSalle 1 reactor.

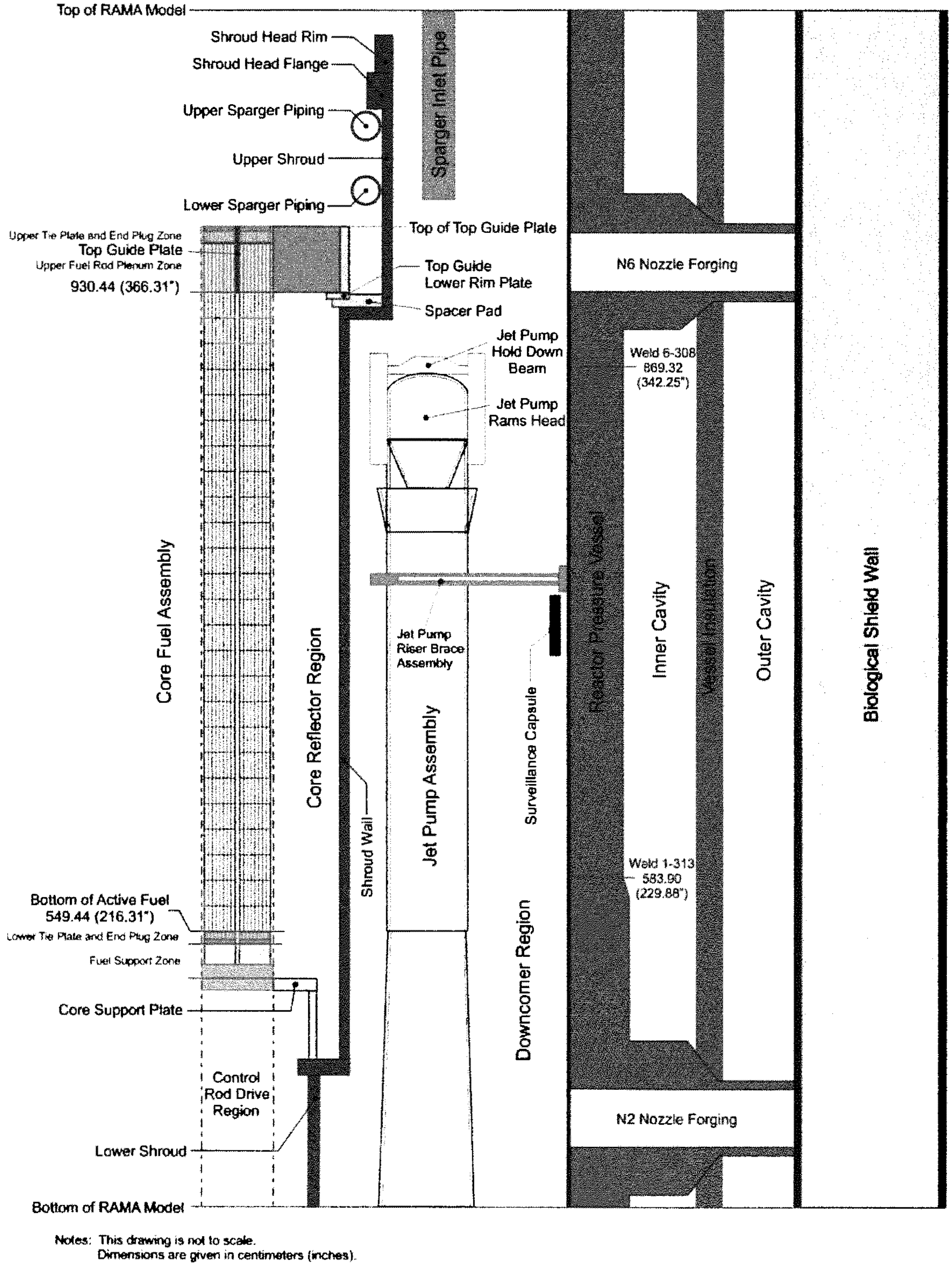

Figure 4-2
Axial View of the LaSalle 1 RAMA Model

4.2.1 The Geometry Model

RAMA uses a generalized three-dimensional geometry modeling system that is based on a combinatorial geometry technique, which is mapped to a Cartesian coordinate system. In this analysis, an axial plane of the reactor model is defined by the (x,y) coordinates of the modeling system and the axial elevation at which a plane exists is defined along a perpendicular z-axis of the modeling system. Thus, any point in the reactor model can be addressed by specifying the (x,y,z) coordinates for that point.

Figure 3-1 of Section 3 illustrates a planar cross-section view of the LaSalle 1 reactor design at an axial elevation corresponding to the reactor core mid-plane elevation. It is shown for this one elevation that the reactor design is a complex geometry composed of various combinations of rectangular, cylindrical, and wedge-shaped bodies. When the reactor is viewed in three dimensions, the varying heights of the different components, structures, and regions create additional geometry modeling complexities. An accurate representation of these geometrical complexities in a predictive computer model is essential for calculating accurate, best-estimate fluence in the reactor pressure vessel, the vessel internals, and the surrounding structures.

Figures 4-1 and 4-2 provide general illustrations of the planar and axial geometry complexities that are represented in the LaSalle 1 fluence model. For comparison purposes, the planar view illustrated in Figure 4-1 corresponds to the same core mid-plane elevation illustrated in Figure 3-1. The computer model for LaSalle 1 assumes azimuthal quadrant symmetry in the planar dimension.

Figure 4-1 illustrates the quadrant geometry that was modeled in this analysis. In terms of the modeling coordinate system, the "northeast" quadrant of the geometry is represented in the model. The 0-degree azimuth, which has a "north" designation, corresponds to the 0-degree azimuth referenced in the plan drawings for the reactor pressure vessel. Degrees are incremented clockwise. Thus, the 90-degree azimuth is designated as the "east" direction. All other components, structures, and regions have been appropriately mirror reflected or rotated to this quadrant based upon their relationship to the pressure vessel orientation to ensure that the fluence is appropriately calculated relative to the neutron source (i.e., the core region). Although symmetry is a modeling consideration, the results presented in this report for the different components and structures are given at their correct azimuths in the plant.

Figure 4-2 illustrates the axial configuration of the primary components, structures, and regions in the fluence model. For discussion purposes, the same components, structures, and regions shown in the planar view of Figure 4-1 are also illustrated in Figure 4-2. Figure 4-2 shows that the axial height of the fluence model spans from a lower elevation just below the jet pump riser inlet to above the core shroud head flange. This axial height covers all areas of the reactor pressure vessel that are expected to exceed a fluence threshold of 1.0E+17 n/cm^2 at 54 EFPY.

As previously noted, Figures 4-1 and 4-2 are not drawn precisely to scale. They are intended only to provide a perspective of how the various components, structures, and regions of the reactor are positioned relative to the reactor core region (i.e., the neutron source) and each other. The following subsections provide details on the modeling of individual components, structures, and regions. Please refer to the figures for visual orientation of the components and regions described in the following subsections.

4.2.2 The Reactor Core and Core Reflector Models

The reactor core contains the nuclear fuel that is the source of the neutrons that irradiate all components and structures of the reactor. The core is surrounded by a shroud structure that serves to channel the reactor coolant through the core region during reactor operation. The region between the core and the core shroud is the core reflector, and it contains coolant. The reactor core geometry is rectangular in design and is modeled with rectangular elements to preserve its shape in the analysis. The core reflector region interfaces with the rectangular shape of the core region and the curved shape of the core shroud. It is, therefore, modeled using a combination of rectangular and cylindrical elements.

The core region is centered in the reactor pressure vessel and is characterized in the analysis with two fundamental fuel zones: interior fuel assemblies and peripheral fuel assemblies. The peripheral fuel assemblies are the primary contributors to the neutron source in the fluence calculation. Because these assemblies are loaded at the core edge where neutron leakage from the core is greatest, there is a sharp power gradient across these assemblies that requires consideration. To account for the power gradient, the peripheral fuel assemblies are sub-meshed with additional rectangular elements that preserve the pin-wise details of the fuel assembly geometry and power distribution. The interior fuel assemblies make a lesser contribution to the reactor fluence and are, therefore, modeled in various homogenized forms in accordance with their contributions to the reactor fluence. For computational efficiency, homogenization treatments are used in the interior core region primarily to reduce the number of mesh regions that must be solved in the transport calculation. The meshing configuration for each fuel assembly location in the core region is determined by parametric studies to ensure an accurate estimate of fluence throughout all regions of the reactor system.

Each fuel assembly design, whether loaded in the interior or peripheral locations in the core, is represented with four axial material zones: the lower tie plate/end plug zone, the fuel zone, the fuel upper plenum zone, and the upper tie plate/end plug zone. The structural materials in the top and bottom nozzles for each unique assembly design are represented in the model to address the shielding effects that these materials have on the components above and below the core region. The fuel zone contains the nuclear fuel and structural materials for the fuel assemblies. The materials for each fuel assembly are unique during reactor operation and are incorporated into the model using reactor operating data from core simulator codes. The upper plenum region captures fission gases during reactor operation.

The LaSalle 1 reactor core region has a nominal elevation for the bottom of active fuel at 549.435 cm (216.31") and an active fuel height of 381.00 cm (150"). LaSalle 1 loaded fuel designs with active fuel heights ranging from 149" to 150". The core simulator codes used by LaSalle 1 modeled the core as 150" in all situations, so this value was also used in the RAMA model. Since the predominant peripheral fuel designs throughout LaSalle 1's history were 150" in height, the effect of this approximation on the RPV fluence is negligible.

From an isotopic standpoint, the core is modeled using quadrant symmetry. For the 30- and 300-degree capsule evaluations, as well as the peak RPV fluence calculations, the NE fuel quadrant was used. Due to suppressed power levels for the last several completed operating cycles prior to removal, the SE fuel quadrant was used for the 120-degree capsule evaluation.

4.2.3 The Core Shroud Model

The core shroud is a canister-like structure that contains the reactor core and channels the reactor coolant and steam produced by the core into the steam separators. Axially the shroud extends almost the entire height of the model, from the lower shroud wall to the top of the shroud head rim. There are several circumferential and vertical welds on the shroud. The core shroud is cylindrical in design and is modeled with pipe elements.

4.2.4 The Downcomer Region Model

The downcomer region lies between the core shroud and the reactor pressure vessel. It is basically cylindrical in design, but with some geometrical complexities created by the presence of jet pumps and surveillance capsules in the region. The majority of the downcomer region is modeled with pipe segments. The areas of the downcomer containing the jet pumps and specimen capsules are modeled with the appropriate geometry elements to represent their design features and to preserve their radial, azimuthal, and axial placement in the downcomer region. These structures are described further in the following subsections.

4.2.5 The Jet Pump Model

There are ten jet pump assemblies in the downcomer region of LaSalle 1, which provide the main recirculation flow for the core. The jet pumps are modeled at azimuths 30, 60, and 90 degrees in the downcomer region. When symmetry is applied to the model, the 30-degree location represents the jet pump assemblies that are positioned azimuthally at 30, 150, 210, and 330 degrees; the 60-degree location represents those at 60, 120, 240, and 300 degrees; and the 90-degree location represents the jet pump assemblies at 90 and 270 degrees. Note that there are no jet pumps present at the 0- and 180-degree azimuths of the reactor.

The jet pump model includes representations for the riser, mixer, and diffuser pipes; nozzles; rams head; hold-down beams and brackets; riser inlet pipe; and riser brace yoke, leafs, and pads. The jet pump assembly design is modeled using cylindrical pipe elements for the jet pump riser and mixer pipes. The riser pipe is correctly situated between the centers of the mixer pipes. Cylinders intersected with wedges are used to represent the rams head. The riser brace assembly model includes two leaf structures that attach to the yoke and pad elements. The jet pump assembly includes hold down beams and brackets, built with rectangular boxes, which are attached to the rams head. The jet pump repair hardware, including the auxiliary wedges, riser brace clamps, and slip joint clamps, are not explicitly included in the model as they have a negligible effect on the vessel fluence.

4.2.6 The Surveillance Capsule Model

Section 3 describes the three surveillance capsules installed in the LaSalle 1 reactor. The surveillance capsules are installed near the inner surface of the pressure vessel wall. The surveillance capsules are rectangular in design. Because of this shape, the capsules are not easily implemented in the otherwise cylindrical elements of the downcomer region model. With reference to Figure 3-1, it is observed that the capsules are of small dimensions in the planar geometry and they reside a long distance (view factor) from the core region. Based on these factors, the otherwise rectangular shape of the surveillance capsules can be reasonably approximated in the model with arc elements. The surveillance capsule model also includes a representation for the downcomer water that surrounds the capsule on all sides.

The surveillance capsules are correctly modeled behind the jet pump riser pipes at the 30- and 60-degree azimuths. When symmetry is applied to the model, the 30-degree location represents the capsule installed at 30 degrees, while the 60-degree location represents the capsules at 120 and 300 degrees.

The surveillance capsules are modeled at their correct axial position and height relative to the core region. The surveillance capsules cover about nine percent of the total core height.

4.2.7 The Reactor Pressure Vessel Model

The reactor pressure vessel and vessel cladding lie outside the downcomer region and each is cylindrical in design. Both are modeled with pipe elements. The cladding-pressure vessel interface is a key location for RPV fluence calculations and is preserved in the model. This interface defines the inside surface (0T) for the pressure vessel base metal where the RPV fluence is calculated. LaSalle 1 has cladding only on the inside surface of the pressure vessel wall. Representations of the forgings for the recirculation inlet (N2), RHR (N6) and instrumentation (N12) nozzles are also included in the model out to the biological shield radius. The nozzle representations are modeled in their true conical shape.

4.2.8 The Vessel Insulation Model

The vessel insulation lies in the cavity region outside the pressure vessel wall. The insulation is cylindrical in design and follows the contour of the pressure vessel wall. It is modeled with pipe elements.

4.2.9 The Inner and Outer Cavity Models

The cavity region lies between the pressure vessel and biological structures. As previously described, the vessel insulation lies in the cavity region; thus creating two cavity regions. The inner cavity region lies between the vessel and the insulation. The outer cavity region lies between the vessel insulation and biological shield cladding. The boundaries of the cavity regions follow the contours of the pressure vessel, vessel insulation, and biological shield. The cavity regions are essentially cylindrical in design and are modeled with pipe segments.

4.2.10 The Biological Shield Model

The biological shield (concrete) defines the outer most region of the fluence model. The biological shield is basically cylindrical in design and is modeled with pipe segments. There is cladding on the inside and outside surface of the biological shield.

4.2.11 The Above-Core Component Models

Figure 4-2 includes illustrations of other components and regions that lie above the reactor core region. The predominant above-core components represented in the model include the top guide and core spray spargers.

4.2.11.1 The Top Guide Model

The top guide component lies above the core region. The top guide is appropriately modeled by including representations for 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. The fuel assembly parts and top guide plates are modeled with rectangular elements.

4.2.11.2 The Core Spray Sparger Model

The core spray spargers include upper and lower sparger pipes and a vertical inlet pipe. The core spray spargers are appropriately represented as torus structures in the model. The sparger pipes reside inside the upper shroud wall above the top guide. The spargers are modeled as pipe-like structures and include a representation of reactor coolant inside the pipes.

4.2.12 The Below-Core Component Models

Figure 4-2 includes illustrations of other components and regions that lie below the reactor core region. 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 below-core components are modeled with rectangular elements with the exception of the core support plate. The core support plate is modeled using both rectangular and cylindrical elements to provide an appropriate representation of that component. Core support plate bolts are included.

4.2.13 Summary of the Geometry Modeling Approach

To summarize the reactor modeling process, there are several key features of the RAMA code system that allow the reactor design to be accurately represented for RPV fluence evaluations. Following is a summary of some of the key features of the model.

- Rectangular, cylindrical, and wedge bodies are mixed in the model in order to provide an accurate geometrical representation of the components and regions in the reactor.
- The reactor core geometry is modeled with rectangular bodies to represent its actual shape in the reactor. The fuel assemblies in the core region are also sub-meshed with additional rectangular bodies to represent the pin cell regions in the assemblies.
- A combination of rectangular and cylindrical bodies is used to describe the transition parts between the rectangular core region and the cylindrical outer core regions.

- Cylindrical and wedge bodies are used to model the components and regions that extend outward from the core region (core shroud, downcomer, RPV, etc.).
- The surveillance capsules are modeled at their correct radial, azimuthal, and elevational positions behind the jet pumps in the downcomer region.
- The above-core region includes accurate representations of the top guide and core spray spargers.
- The below-core region includes appropriate representations for the fuel support piece, core support plate and rim bolts, core inlet regions, cruciform control rods, and control rod drives.
- The biological shield is appropriately represented as a cylindrical body.

4.3 RAMA Calculation Parameters

The RAMA transport code uses a three-dimensional deterministic transport method to calculate the neutron flux. The accuracy of the transport method is based on a numerical integration technique that uses ray-tracing to characterize the geometry, anisotropy treatments to determine the directional flow of particles, and convergence parameters to determine the overall accuracy of the flux solution between iterates. The code allows the user to specify values for each of these parameters.

The primary input parameters that control the ray-tracing calculation are the distance between parallel rays in the planar and axial dimensions, the depth that a particle is tracked when a reflective boundary is encountered, and the number of equally spaced angles in polar coordinates for tracking the particles. Plant-specific values are determined for each of the parameters. The RAMA transport calculation employs a treatment for anisotropy that is based on a Legendre expansion of the scattering cross sections. By default, the RAMA transport calculation uses the maximum order of expansion that is available for each nuclide in the RAMA nuclear data library. For the actinide and zirconium nuclides, a P_5 expansion of the scattering cross sections is used. For all other nuclides, a P_7 expansion of the scattering cross sections is used.

The overall accuracy of the neutron flux calculation is determined using an iterative technique to converge the flux iterations. The convergence criterion used in the evaluation was determined by parametric study to provide an asymptotic solution for this model.

4.4 RAMA Neutron Source Calculation

RAMA calculates a unique neutron source distribution for each transport calculation using the input relative power density factors for the fuel region and data from the RAMA nuclear data library. The source distribution changes with fuel burnup; thus, the source is determined using core-specific three-dimensional burnup distributions at frequent intervals throughout a cycle. For the fluence model, the peripheral fuel assemblies are modeled to preserve the power gradient at the core edge that is formed from the pin-wise source distributions in these fuel assemblies.

4.5 RAMA Fission Spectra

RAMA calculates a weighted fission spectrum for each transport calculation that is based on the relative contributions of ^{235}U, ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu, and ^{242}Pu isotopes. The fission spectra for these isotopes are derived from the BUGLE-96 nuclear data library.

5
SURVEILLANCE CAPSULE ACTIVATION AND FLUENCE RESULTS

This section documents the activation results for the LaSalle 1 reactor. The activation results form the basis for the validation and qualification of the application of the RAMA Fluence Methodology to the LaSalle 1 reactor in accordance with the requirements of U. S. NRC Regulatory Guide 1.190 (Reg. Guide 1.190). Reg. Guide 1.190 requires fluence calculational methods to be validated by comparison with measurements from operating reactor dosimetry for the specific plant being analyzed or for reactors of similar design. The acceptance criteria provided in Reg. Guide 1.190 is that the comparison to measurement ratios (C/M) and standard deviation values must be $\leq 20\%$. All of the LaSalle 1 reactor capsule measurement comparisons to the RAMA predicted values meet the Reg. Guide 1.190 limits. The accuracy of the comparisons is additional confirmation that the RAMA Fluence Methodology provides unbiased fluence estimates for the LaSalle 1 reactor dosimetry. The activation comparisons presented in this section were generated as part of a previous evaluation [12] and are reproduced here for reference.

Three flux wire activation analyses were performed for the LaSalle 1 reactor. Flux wires were removed from the 30-degree capsule flux wire holder and analyzed at the end of cycle 1 (irradiated for 1.4 EFPY); surveillance capsule flux wires were removed at the end of cycle 6 from the 300-degree capsule (irradiated for 6.5 EFPY); and surveillance capsule flux wires were removed at the end of cycle 13 from the 120-degree capsule (irradiated for 18.9 EFPY). Details of the dosimetry specimens and analysis are presented in Section 5.1.

5.1 Comparison of Predicted Activation to Plant-Specific Measurements

The comparison of predicted activation for the LaSalle 1 cycles 1, 6, and 13 flux wires to measurements is presented in this subsection. Note that the precise location of the individual wires within the surveillance capsule flux wire holder and surveillance capsule holders is not known, therefore, the activation calculations were performed at the center of the respective holders.

5.1.1 Cycle 1 30-Degree Flux Wire Holder Activation Analysis

Copper and iron flux wires were irradiated in the LaSalle 1 surveillance capsule flux wire holder at the 30-degree azimuth during the first cycle of operation. The wires were removed after being irradiated for a total of 1.4 EFPY. Activation measurements were performed following irradiation for the following reactions [15]: $^{63}Cu\ (n,\alpha)\ ^{60}Co$ and $^{54}Fe\ (n,p)\ ^{54}Mn$.

Table 5-1 provides a comparison of the RAMA calculated specific activities and the measured specific activities for the flux wire specimens. The cycle 1 total flux wire average calculated-to-measured (C/M) value is 0.98 with a standard deviation of ± 0.03.

Table 5-1
Comparison of Specific Activities for LaSalle 1 Cycle 1 30-Degree Flux Wire Holder Wires (C/M)

Flux Wires	Measured (dps/g)	Calculated (dps/g)	Calculated vs. Measured	Standard Deviation
Iron				
A	3.17E+04	3.19E+04	1.01	---
B	3.16E+04	3.19E+04	1.01	---
C	3.19E+04	3.19E+04	1.00	---
Average	**3.17E+04**	**3.19E+04**	**1.01**	**0.00**
Copper				
A	1.81E+03	1.75E+03	0.96	---
B	1.84E+03	1.75E+03	0.95	---
C	1.84E+03	1.75E+03	0.95	---
Average	**1.83E+03**	**1.75E+03**	**0.95**	**0.01**
Total Flux Wire Average	**---**	**---**	**0.98**	**0.03**

5.1.2 Cycle 6 300-Degree Flux Wire Holder Activation Analysis

Copper, iron, and nickel flux wires were irradiated in the LaSalle 1 surveillance capsule at the 300-degree azimuth during the first 6 cycles of operation. The wires were removed after being irradiated for a total of 6.5 EFPY. Activation measurements were performed following irradiation for the following reactions [16]: ^{63}Cu (n,α) ^{60}Co, ^{54}Fe (n,p) ^{54}Mn, and ^{58}Ni (n,p) ^{58}Co.

Table 5-2 provides a comparison of the RAMA calculated specific activities and the measured specific activities for the surveillance capsule flux wire specimens. The cycle 6 capsule total flux wire average C/M value is 1.06 with a standard deviation of ±0.10.

Table 5-2
Comparison of Specific Activities for LaSalle 1 Cycle 6 300-Degree Surveillance Capsule Flux Wires (C/M)

Flux Wires	Measured (dps/g)	Calculated (dps/g)	Calculated vs. Measured	Standard Deviation
Iron				
Average(1)	3.653E+04	4.123E+03	1.13	---
Copper				
Average(1)	5.380E+03	4.964E+03	0.92	---
Nickel				
Average(1)	5.174E+05	5.840E+05	1.13	---
Total Flux Wire Average	---	---	**1.06**	**0.10**

1) The source document for the flux wire measurements only provided an average activity that represents the average of 3 wires for each wire type.

5.1.3 Cycle 13 120-Degree Flux Wire Holder Activation Analysis

Copper, iron, and nickel flux wires were irradiated in the LaSalle 1 surveillance capsule at the 120-degree azimuth during the first 13 cycles of operation. The wires were removed after being irradiated for a total of 18.9 EFPY. Activation measurements were performed following irradiation for the following reactions [17]: ^{63}Cu (n,α) ^{60}Co, ^{54}Fe (n,p) ^{54}Mn, and ^{58}Ni (n,p) ^{58}Co.

Table 5-3 provides a comparison of the RAMA calculated specific activities and the measured specific activities for the surveillance capsule flux wire specimens. The cycle 13 capsule total flux wire average C/M value is 1.02 with a standard deviation of ±0.06.

Table 5-3
Comparison of Specific Activities for LaSalle 1 Cycle 13 120-Degree Surveillance Capsule Flux Wires (C/M)

Flux Wires	Measured (dps/mg)	Calculated (dps/mg)	Calculated vs. Measured	Standard Deviation
Iron				
G016	63.37	66.26	1.05	---
G017	65.17	66.26	1.02	---
G018	64.76	66.26	1.02	---
Average	**64.43**	**66.26**	**1.03**	**±0.02**
Copper				
G016	10.19	9.84	0.97	---
G017	10.46	9.84	0.94	---
G018	10.46	9.84	0.94	---
Average	**10.37**	**9.84**	**0.95**	**±0.01**
Nickel				
G016	894.38	926.15	1.08	---
G017	893.78	962.15	1.08	---
G018	891.30	962.15	1.08	---
Average	**893.15**	**962.15**	**1.08**	**±0.01**
Total Flux Wire Average	**---**	**---**	**1.02**	**0.06**

5.1.4 Surveillance Capsule Activation Analysis Summary

Table 5-4 presents a summary of the total average calculated-to-measured result of specific activities for all LaSalle 1 flux wires. Combining all flux wires (copper, iron, and nickel), the total average C/M is 1.02 with a standard deviation of ±0.08.

Table 5-4
Comparison of Activities for LaSalle 1 Flux Wires

Flux Wire	Number of Measurements	Calculated vs. Measured	Standard Deviation
30-Degree Cycle 1 – Average	6	0.98	0.03
300-Degree Cycle 6 – Average	9	1.06	0.10
120-Degree Cycle 13 – Average	9	1.02	0.06
Total Flux Wire Average	**24**	**1.02**	**0.08**

5.2 Comparison of Predicted Activation to All BWR Plant Measurements

Several BWR reactor units have had surveillance capsule activation analyses performed with the RAMA Fluence Methodology software. The results of several of these analyses are documented in [18]. Some additional analyses have been performed since the publication of [18]. The summary of all BWR analyses is presented in Table 5-5. The table contains the number of surveillance capsule measurements taken from all BWR plants. The comparison of those measurements to the calculated specific activities generated by RAMA (C/M) values is shown in the last column of Table 5-5. The total C/M for all capsules evaluated with RAMA is 0.99 with a standard deviation of ±9%.

Table 5-5
Summary of BWR Operating Reactor Surveillance Capsule Measurement Comparisons

Reactor Class	Fuel Assembly Configuration	No. of Samples	Calculated vs. Measured
BWR/2	560	317	0.98 ±7%
BWR/3	484	18	1.03 ±10%
BWR/3	400	6	1.09 ±5%
BWR/4	368	27	1.03 ±8%
BWR/4	548	18	0.93 ±12%
BWR/4	560	39	0.98 ±12%
BWR/4 & BWR/5	764	117	1.02 ±11%
BWR/6	624	3	0.95 ±1%
Total	**----**	**545**	**0.99 ±9%**

6 REACTOR PRESSURE VESSEL FLUENCE UNCERTAINTY ANALYSIS

This section presents the combined uncertainty analysis and bias determination for the LaSalle 1 reactor pressure vessel fluence evaluation. The combined uncertainty is comprised of the comparison uncertainty factors developed in Section 5 and an analytic uncertainty factor developed in this section. When combined, these components provide a basis for determining the overall uncertainty (1σ) and bias in the RPV fluence for this analysis. The uncertainty values for LaSalle 1 were previously computed as part of a previous fluence evaluation [12]. The results of that evaluation are provided here for reference and are applicable to the current fluence evaluation.

The requirements for determining the combined uncertainty and bias for light water reactor fluence evaluations are provided in Regulatory Guide 1.190. The method implemented for determining the combined uncertainty and bias for reactor component fluence is described in the RAMA Theory Manual [4]. Regarding the determination of a bias in the pressure vessel fluence, Regulatory Guide 1.190 provides that an adjustment to the calculated vessel fluence for bias effects is needed if a statistically significant bias exists in the fluence computation.

The results presented in this section show that the combined uncertainty for the LaSalle 1 RPV fluence evaluation is 9.2% and that no adjustment for bias effects is required to the calculated RPV fluence reported in Section 7 of this report.

The following subsections describe the comparison uncertainties determined in Section 5, the determination of the analytic uncertainty, and the determination of the overall combined uncertainty and bias for the LaSalle 1 RPV fluence evaluation.

6.1 Comparison Uncertainty

Comparison uncertainty factors are determined by comparing calculated activities with activity measurements. For pressure vessel fluence evaluations, two comparison uncertainty factors are considered: an operating reactor comparison uncertainty factor and a benchmark comparison uncertainty factor. The determination of a comparison uncertainty factor based on measurements involves the combination of two measurement components. One component is the variation in the comparison of the calculated-to-measured (C/M) activity ratio and the other accounts for the uncertainty introduced by the measurement process.

6.1.1 Operating Reactor Comparison Uncertainty

The operating reactor, or plant-specific, comparison uncertainty for the LaSalle 1 reactor is determined by combining the standard deviation for the activity comparisons with the measurement uncertainty for the plant-specific activity measurements.

6.1.2 Benchmark Comparison Uncertainty

The benchmark comparison uncertainty used in the LaSalle 1 uncertainty analysis is based on a set of industry standard simulation benchmark comparisons.

6.2 Analytic Uncertainty

The calculational models used for fluence analyses are comprised of numerous analytical parameters that have associated uncertainties in their values. The uncertainty in these parameters needs to be tested for its contribution to the overall fluence uncertainty.

The uncertainty values for the geometry parameters are based upon uncertainties in the dimensional data used to construct the plant geometry model. The uncertainty values for the material parameters are based upon uncertainties in the material densities for the water and nuclear fuel materials and the compositional makeup of typical steel materials.

The uncertainty values for the fission source parameters are based upon uncertainties in the fuel exposure and power factors for the fuel assemblies loaded on the core periphery. The transport method used in the fluence analysis employs a fission source calculation that accounts for the relative contributions of the uranium and plutonium fissile isotopes in the fuel and the relative power density of the fuel in the reactor. Both fission source parameters are derived directly from information calculated by three-dimensional core simulator codes. The uncertainty values for the nuclear cross-section parameters are based upon uncertainties in the number densities for the predominant nuclides that make up the reactor materials.

The uncertainty parameters for the fluence model inputs are based upon geometry meshing and numerical integration parameters used in the neutron flux transport calculation. The process for determining the geometry meshing and numerical integration parameters involves an exhaustive sensitivity study that is described in the RAMA Procedures Manual [19].

6.3 Combined Uncertainty

The combined uncertainty for the reactor pressure vessel fluence evaluation is determined with a weighting function that combines the analytic, plant-specific comparison, and benchmark comparison uncertainty factors developed in Sections 6.1 and 6.2, above. Table 6-1 lists that the combined uncertainty (1σ) determined for the LaSalle 1 reactor pressure vessel fluence is 9.2% with energy >1.0 MeV.

Table 6-1 also shows that, in accordance with Regulatory Guide 1.190, no bias term exists and it is not necessary to adjust the RAMA predicted RPV fluence in this analysis for bias effects. It is also demonstrated in Table 6-1 that the combined uncertainty is within the limits prescribed in U. S. NRC Regulatory Guide 1.190 (i.e. $\leq$ 20%).

Table 6-1
LaSalle 1 Combined RPV Uncertainty for Energy >1.0 MeV

Uncertainty Term	Value
Combined Uncertainty (1σ)	9.2%
Bias	None(1)

1) The bias terms are less than their constituent uncertainty values, concluding that no statistically significant bias exists.

7
CALCULATED REACTOR PRESSURE VESSEL FAST NEUTRON FLUENCE

This section presents the predicted best-estimate fast neutron fluence (>1.0 MeV) for the LaSalle 1 reactor pressure vessel (RPV) at EOC 15 (22.7 EFPY), 32 EFPY, and 54 EFPY. It is reported in Section 6 that the RAMA-calculated pressure vessel fluence for the LaSalle 1 reactor requires no bias adjustment; therefore, the best-estimate fluence is the calculated fluence that was predicted with the RAMA Fluence Methodology.

The reactor pressure vessel fluence reported in this section was determined at the interface of the RPV base metal and cladding, denoted as the 0T location of the RPV wall as well as at the 1/4 T and 3/4 T locations. Fluence attenuations through the RPV wall are performed using the plant-specific DPA attenuation method specified by Reg. Guide 1.99 [1]. Fluence is presented for the RPV circumferential welds, vertical welds, plates, and nozzles residing in the RPV beltline region. The location and identification of the RPV plates, welds, and nozzles are shown in Figure 7-1. Also illustrated in Figure 7-1 is the calculated RPV beltline region for the LaSalle 1 reactor at 54 EFPY.

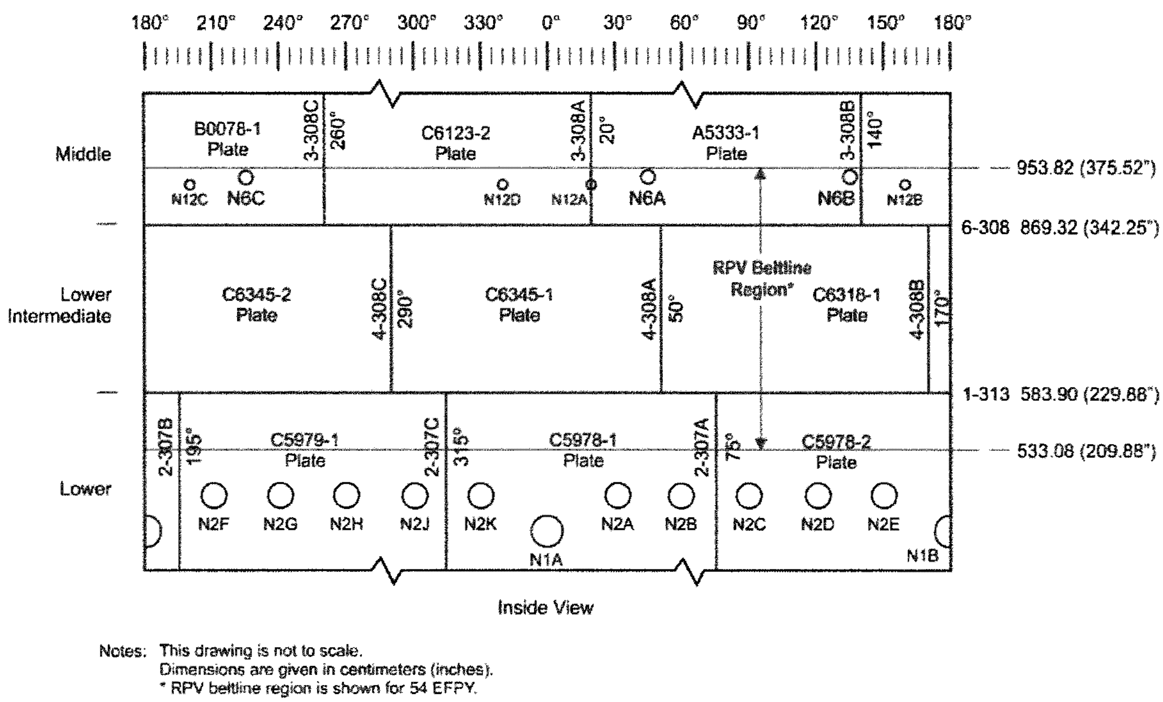

Figure 7-1
LaSalle 1 RPV Shell Plate, Weld, and Nozzle Identifiers

Fluence in the RPV welds is presented at the center of the weld. Fluence for the nozzles is presented along two paths, one representing the nozzle forging-to-base-metal weld in the RPV shell course, the other representing the fluence along a 45-degree extraction path from the inside corner of the forging. Both paths are illustrated in Figure 7-2 and were edited from the peak angle around the forging's circumference. The 0T and fractional depths are calculated based on the overall length of each respective path. Note that due to the shallow weld used for the N12 nozzle, only 0T fluence is provided.

Table 7-1 reports the maximum >1.0 MeV neutron fluence at 0T, 1/4 T, and 3/4 T for the RPV circumferential and vertical welds in the RPV beltline region at EOC 15. Table 7-2 reports the maximum fast neutron fluence for the RPV plates residing in the RPV beltline region at EOC 15. Table 7-3 reports the maximum fast neutron fluence for the RPV nozzles residing in the RPV beltline region at EOC 15. Tables 7-4 through 7-6 report the same fluence values as Tables 7-1, 7-2, and 7-3 for 32 EFPY and Tables 7-7 through 7-9 report fluence at 54 EFPY. Table 7-10 reports the elevations that define the RPV beltline at EOC 15, 32 EFPY, and 54 EFPY.

In all tables, fluences that exceed the threshold fluence of 1.0E+17 n/cm^2 are shown in red. The weld with the highest fluence value in each table is shown in **bold** type. Note that fluence at the inner surface of all welds, plates, and nozzles exceeds the threshold value of 1.0E+17 n/cm^2 prior to EOC 15, with the exception of the N2 and N12 nozzles. The maximum fluence value for all welds, plates, and nozzles is 1.06E+18 n/cm^2 at 54 EFPY at the RPV inner diameter in the lower intermediate shell plates G-5604-1, G-5604-2, and G-5604-3.

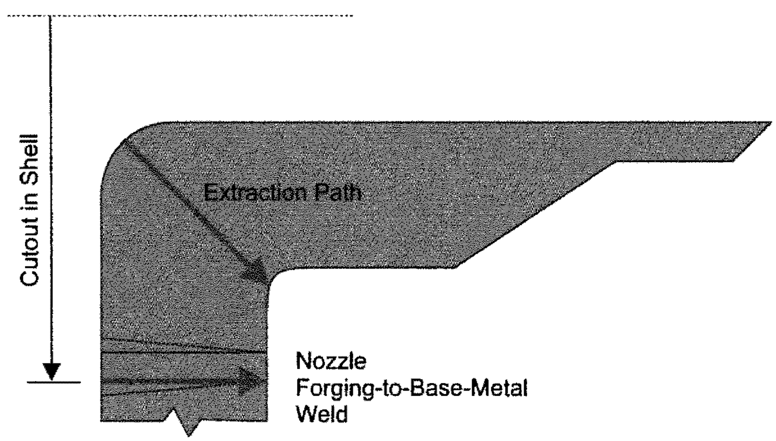

Figure 7-2 Nozzle Edit Locations for Sample Nozzle

Table 7-1
Maximum Fast Neutron Fluence for LaSalle 1 RPV Beltline Weld Locations at EOC 15 (22.7 EFPY)

Weld	Heat No.	Fast Fluence with Plant-Specific Attenuation (n/cm^2)		
		0T	1/4 T	3/4 T
Middle Shell Axial				
3-308A	305424 1P3571	3.57E+17	2.39E+17	9.31E+16
3-308B		2.79E+17	1.88E+17	7.52E+16
3-308C		2.02E+17	1.37E+17	5.56E+16
Lower-Intermediate Shell Axial				
4-308A	**12008 305414**	3.44E+17	2.33E+17	9.46E+16
4-308B		2.71E+17	1.84E+17	7.55E+16
4-308C		**4.38E+17**	**2.96E+17**	**1.17E+17**
Lower Shell Axial				
2-307A	21935 12008	1.46E+17	9.82E+16	3.98E+16
2-307B		1.45E+17	9.79E+16	4.00E+16
2-307C		1.31E+17	8.86E+16	3.71E+16
Lower-Intermediate and Lower Shell Girth				
6-308	6329637	3.70E+17	2.48E+17	9.63E+16
1-313	4P6519	1.74E+17	1.17E+17	4.75E+16

Table 7-2
Maximum Fast Neutron Fluence for LaSalle 1 RPV Beltline Shell Plate Locations at EOC 15 (22.7 EFPY)

Shell	Heat No.	Fast Fluence with Plant-Specific Attenuation (n/cm^2)		
		0T	1/4 T	3/4 T
Middle Shell				
G-5605-1	A5333-1	3.70E+17	2.48E+17	9.63E+16
G-5605-2	B0078-1	3.70E+17	2.48E+17	9.63E+16
G-5605-3	C6123-2	3.70E+17	2.48E+17	9.63E+16
Lower-Intermediate Shell				
G-5604-1	**C6345-1**	**4.67E+17**	**3.12E+17**	**1.22E+17**
G-5604-2	**C6318-1**	4.67E+17	**3.12E+17**	**1.22E+17**
G-5604-3	**C6345-2**	**4.67E+17**	**3.12E+17**	**1.22E+17**
Lower Shell				
G-5603-1	C5978-1	1.74E+17	1.17E+17	4.75E+16
G-5603-2	C5978-2	1.74E+17	1.17E+17	4.75E+16
G-5603-3	C5979-1	1.74E+17	1.17E+17	4.75E+16

Table 7-3
Maximum Fast Neutron Fluence for LaSalle 1 RPV Beltline Nozzle Locations at EOC 15 (22.7 EFPY)

Nozzle Location	Heat No.	Fast Fluence with Plant-Specific Attenuation (n/cm^2)		
		0T	1/4 T	3/4 T
N2 Forging Weld		8.67E+15	6.21E+15	3.94E+15
N2 Extraction Path		1.50E+15	1.69E+15	2.52E+15
N6 Forging Weld	**Q2Q22W**	1.74E+17	**1.16E+17**	**4.81E+16**
N6 Extraction Path		5.50E+16	4.88E+16	3.40E+16
N12 Forging Weld		7.56E+16	N/A	N/A
N12 Extraction Path		N/A	N/A	N/A

Table 7-4
Maximum Fast Neutron Fluence for LaSalle 1 RPV Beltline Weld Locations at 32 EFPY

Weld	Heat No.	Fast Fluence with Plant-Specific Attenuation (n/cm^2)		
		0T	1/4 T	3/4 T
Middle Shell Axial				
3-308A	305424 1P3571	5.09E+17	3.40E+17	1.33E+17
3-308B		4.03E+17	2.71E+17	1.08E+17
3-308C		2.88E+17	1.95E+17	7.94E+16
Lower-Intermediate Shell Axial				
4-308A	**12008 305414**	4.81E+17	3.26E+17	1.32E+17
4-308B		3.73E+17	2.54E+17	1.04E+17
4-308C		**6.06E+17**	**4.09E+17**	**1.62E+17**
Lower Shell Axial				
2-307A	21935 12008	1.97E+17	1.33E+17	5.40E+16
2-307B		1.98E+17	1.34E+17	5.49E+16
2-307C		1.79E+17	1.22E+17	5.09E+16
Lower-Intermediate and Lower Shell Girth				
6-308	6329637	5.29E+17	3.54E+17	1.37E+17
1-313	4P6519	2.34E+17	1.58E+17	6.43E+16

Table 7-5
Maximum Fast Neutron Fluence for LaSalle 1 RPV Beltline Shell Plate Locations at 32 EFPY

Shell	Heat No.	Fast Fluence with Plant-Specific Attenuation (n/cm^2)		
		0T	1/4 T	3/4 T
Middle Shell				
G-5605-1	A5333-1	5.29E+17	3.54E+17	1.37E+17
G-5605-2	B0078-1	5.29E+17	3.54E+17	1.37E+17
G-5605-3	C6123-2	5.29E+17	3.54E+17	1.37E+17
Lower-Intermediate Shell				
G-5604-1	**C6345-1**	**6.45E+17**	**4.31E+17**	**1.69E+17**
G-5604-2	**C6318-1**	**6.45E+17**	**4.31E+17**	**1.69E+17**
G-5604-3	**C6345-2**	**6.45E+17**	**4.31E+17**	**1.69E+17**
Lower Shell				
G-5603-1	C5978-1	2.34E+17	1.58E+17	6.43E+16
G-5603-2	C5978-2	2.34E+17	1.58E+17	6.43E+16
G-5603-3	C5979-1	2.34E+17	1.58E+17	6.43E+16

Table 7-6
Maximum Fast Neutron Fluence for LaSalle 1 RPV Beltline Nozzle Locations at 32 EFPY

Nozzle Location	Heat No.	Fast Fluence with Plant-Specific Attenuation (n/cm^2)		
		0T	1/4 T	3/4 T
N2 Forging Weld		1.19E+16	8.53E+15	5.41E+15
N2 Extraction Path		2.06E+15	2.33E+15	3.48E+15
N6 Forging Weld	**Q2Q22W**	**2.60E+17**	**1.73E+17**	**7.12E+16**
N6 Extraction Path		8.42E+16	7.46E+16	5.12E+16
N12 Forging Weld		1.14E+17	N/A	N/A
N12 Extraction Path		N/A	N/A	N/A

Table 7-7
Maximum Fast Neutron Fluence for LaSalle 1 RPV Beltline Weld Locations at 54 EFPY

Weld	Heat No.	Fast Fluence with Plant-Specific Attenuation (n/cm^2)		
		0T	1/4 T	3/4 T
Middle Shell Axial				
3-308A	305424 1P3571	8.66E+17	5.81E+17	2.26E+17
3-308B		6.95E+17	4.68E+17	1.87E+17
3-308C		4.92E+17	3.33E+17	1.35E+17
Lower-Intermediate Shell Axial				
4-308A	**12008** **305414**	8.00E+17	5.43E+17	2.20E+17
4-308B		6.17E+17	4.21E+17	1.73E+17
4-308C		**1.00E+18**	**6.77E+17**	**2.67E+17**
Lower Shell Axial				
2-307A	21935 12008	3.18E+17	2.15E+17	8.74E+16
2-307B		3.27E+17	2.21E+17	9.05E+16
2-307C		2.92E+17	1.98E+17	8.31E+16
Lower-Intermediate and Lower Shell Girth				
6-308	6329637	9.01E+17	6.03E+17	2.34E+17
1-313	4P6519	3.75E+17	2.54E+17	1.03E+17

Table 7-8
Maximum Fast Neutron Fluence for LaSalle 1 RPV Beltline Shell Plate Locations at 54 EFPY

Shell	Heat No.	Fast Fluence with Plant-Specific Attenuation (n/cm^2)		
		0T	1/4 T	3/4 T
Middle Shell				
G-5605-1	A5333-1	9.01E+17	6.03E+17	2.34E+17
G-5605-2	B0078-1	9.01E+17	6.03E+17	2.34E+17
G-5605-3	C6123-2	9.01E+17	6.03E+17	2.34E+17
Lower-Intermediate Shell				
G-5604-1	**C6345-1**	**1.06E+18**	**7.12E+17**	**2.79E+17**
G-5604-2	**C6318-1**	**1.06E+18**	**7.12E+17**	**2.79E+17**
G-5604-3	**C6345-2**	**1.06E+18**	**7.12E+17**	**2.79E+17**
Lower Shell				
G-5603-1	C5978-1	3.75E+17	2.54E+17	1.03E+17
G-5603-2	C5978-2	3.75E+17	2.54E+17	1.03E+17
G-5603-3	C5979-1	3.75E+17	2.54E+17	1.03E+17

Table 7-9
Maximum Fast Neutron Fluence for LaSalle 1 RPV Beltline Nozzle Locations at 54 EFPY

Nozzle Location	Heat No.	Fast Fluence with Plant-Specific Attenuation (n/cm^2)		
		0T	1/4 T	3/4 T
N2 Forging Weld		1.93E+16	1.39E+16	8.88E+15
N2 Extraction Path		3.36E+15	3.83E+15	5.74E+15
N6 Forging Weld	**Q2Q22W**	**4.63E+17**	**3.08E+17**	**1.26E+17**
N6 Extraction Path		1.55E+17	1.37E+17	9.24E+16
N12 Forging Weld		2.06E+17	N/A	N/A
N12 Extraction Path		N/A	N/A	N/A

Table 7-10
Reactor Beltline Elevation Range for LaSalle 1

Reactor Lifetime	Lower Elevation [cm (in)]	Upper Elevation [cm (in)]
EOC 15 (22.7 EFPY)	556.50 (219.09)	929.95 (366.12)
32 EFPY	546.19 (215.03)	940.07 (370.11)
54 EFPY	533.08 (209.88)	953.82 (375.52)

8
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