ATTACHMENT 8

LaSalle County Generating Station Unit 1 Fluence Methodology Report," LAS-FLU-001-R010, Rev. 0 (Non-Proprietary Version)

Topical Report

LASALLE COUNTY GENERATING STATION UNIT 1 FLUENCE METHODOLOGY REPORT

Document Number: LAS-FLU-001-R-010
Revision 0
September 2021

Prepared by: **TransWare Enterprises Inc.**

Prepared for: **Exelon Generation Company, LLC**
LaSalle County Generating Station
2601 N 21st Rd
Marseilles, IL 61341

Contract Number: 00808371

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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.

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1
INTRODUCTION

This report provides an overview of the LaSalle County Generating Station Unit 1 (LaSalle Unit 1) fast neutron fluence model and fluence methodology. LaSalle Unit 1 is a single reactor unit of the General Electric BWR/5 class design with a core loading of 764 fuel assemblies. The reactor began commercial operation in 1982 with a rated thermal power of 3323 MWth. Midway through Cycle 9 the rated power was increased to 3489 MWth. Another power uprate occurred midway through Cycle 14 to increase the rated power to 3546 MWth. The LaSalle Unit 1 reactor is owned and operated by Exelon Generation Company, LLC (Exelon).

Fluence evaluations performed for the LaSalle Unit 1 reactor are based upon the RAMA Fluence Methodology software [1], the RAMA Fluence Methodology Procedures Manual [2], and the RAMA Fluence Methodology Theory Manual [3]. The RAMA Fluence Methodology (hereinafter referred to as "RAMA") was developed by TransWare Enterprises Inc. under sponsorship of the Electric Power Research Institute, Inc. (EPRI) and the Boiling Water Reactor Vessel and Internals Project (BWRVIP).

All data used to construct the LaSalle Unit 1 fluence model, define the structural and fuel materials, and develop the lifetime operating history of the reactor was provided by Exelon.

1.1 Regulatory Requirements for Determining Fluence in Light Water Reactors

Part 50 of Title 10 of Code of Federal Regulations (10CFR50), which is issued by the federal agencies of the United States of America, provides requirements for establishing irradiated material monitoring programs that serve to ensure the integrity of the reactor coolant pressure boundary of light water nuclear power reactors. Two appendices to Part 50 present requirements that guide fluence determinations: Appendix G, "Fracture Toughness Requirements" [4], and Appendix H, "Reactor Vessel Material Surveillance Program Requirements" [5].

Appendix G specifies fracture toughness requirements for the carbon and low-alloy ferritic materials of the pressure-retaining components of the reactor coolant pressure boundary. These requirements are to ensure adequate margins of safety during any condition of normal operation including anticipated conditions for system hydrostatic testing, to which the pressure boundary may be subjected over its service lifetime. These requirements apply to base metal, welds, and weld heat-affected zones in the materials within the reactor pressure vessel (RPV) beltline region.

Appendix H specifies the requirements for a material surveillance program that serves to monitor changes in the fracture toughness properties of the ferritic materials in the reactor beltline region. The changes in fracture toughness properties of ferritic materials are attributed to the exposure of the material to neutron irradiation and the thermal environment. Section III of Appendix H

specifies that a material surveillance program is required for light water nuclear power reactors if the peak fast neutron fluence with energy greater than 1 MeV (E > 1 MeV) at the end of the design life of the vessel is expected to exceed 10^{17} n/cm^2.

In compliance with Appendix H requirements, fracture toughness test data are obtained from material specimens that are exposed to neutron irradiation in surveillance capsules installed at or near the inner surface of the reactor pressure vessel. These capsules are withdrawn periodically from the reactor for measurement and analysis. Fast neutron fluence is not a measurable quantity and must be determined using analytical methods. It must be demonstrated that the analytical method used to determine the fast neutron fluence provides a conservative prediction over the beltline region of the pressure boundary when compared to the measurement data with allowances for all uncertainties in the measurement. Section III of Appendix H also allows for an Integrated Surveillance Program (ISP) in which representative materials for the reactor are irradiated in one or more other reactors of sufficiently similar design and operating features to permit accurate comparisons of the predicted amount of radiation damage.

Implementing guidelines addressing the requirements of Appendices G and H are provided in U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials" [6], and Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" [7]. Regulatory Guide 1.99 addresses the requirements of Appendix G for determining the damage fluence that is used in the evaluation of fracture toughness in light water nuclear reactor pressure vessel ferritic materials. Regulatory Guide 1.190 addresses the requirements for determining the fast neutron fluence and uncertainty in the fluence predictions that are used in fracture toughness evaluations. RAMA is qualified against industry standard benchmarks for both boiling water reactor (BWR) and pressurized water reactor (PWR) designs. The RAMA methodology, as well as TransWare's application of the methodology, have been reviewed by the NRC and given generic approval for determining fast neutron fluence in both BWR and PWR pressure vessels [8] with no discernable bias in the computed results.

The RAMA methodology has also received conditional approval for determining fast neutron fluence in light water reactor vessel internals (RVI). The Safety Evaluation (SE) issued by the U.S. Nuclear Regulatory Commission for EPRI report BWRVIP-145 [9] concludes that "*for applications such as IASCC, crack propagation rates and weldability determinations, the RAMA methodology can be used in determining fast neutron fluence values in the core shroud and top guide…for licensing actions provided that the calculational results are supported by sufficient justification that the proposed values are conservative for the intended application*".

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The fast neutron fluence methods discussed in this report meet the requirements of 10CFR50 Appendices G and H and Regulatory Guides 1.190 and 1.99 Revision 2. The NRC has not issued a regulatory guide of similar scope to Regulatory Guide 1.190 for determining fluence in RVI components. In the absence of regulatory guidance, the intent of Regulatory Guide 1.190, particularly with regards to conservatism in constructing and evaluating reactor components, is used by TransWare in the determination of fast neutron fluence throughout a reactor pressure vessel.

1.2 Quality Assurance

The implementation and validation of the fluence methodology presented in this report complies with the quality assurance requirements of 10CFR50 Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants" [11], and to 10CFR21, "Reporting of Defects and Noncompliance" [12].

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2
DESCRIPTION OF THE REACTOR SYSTEM

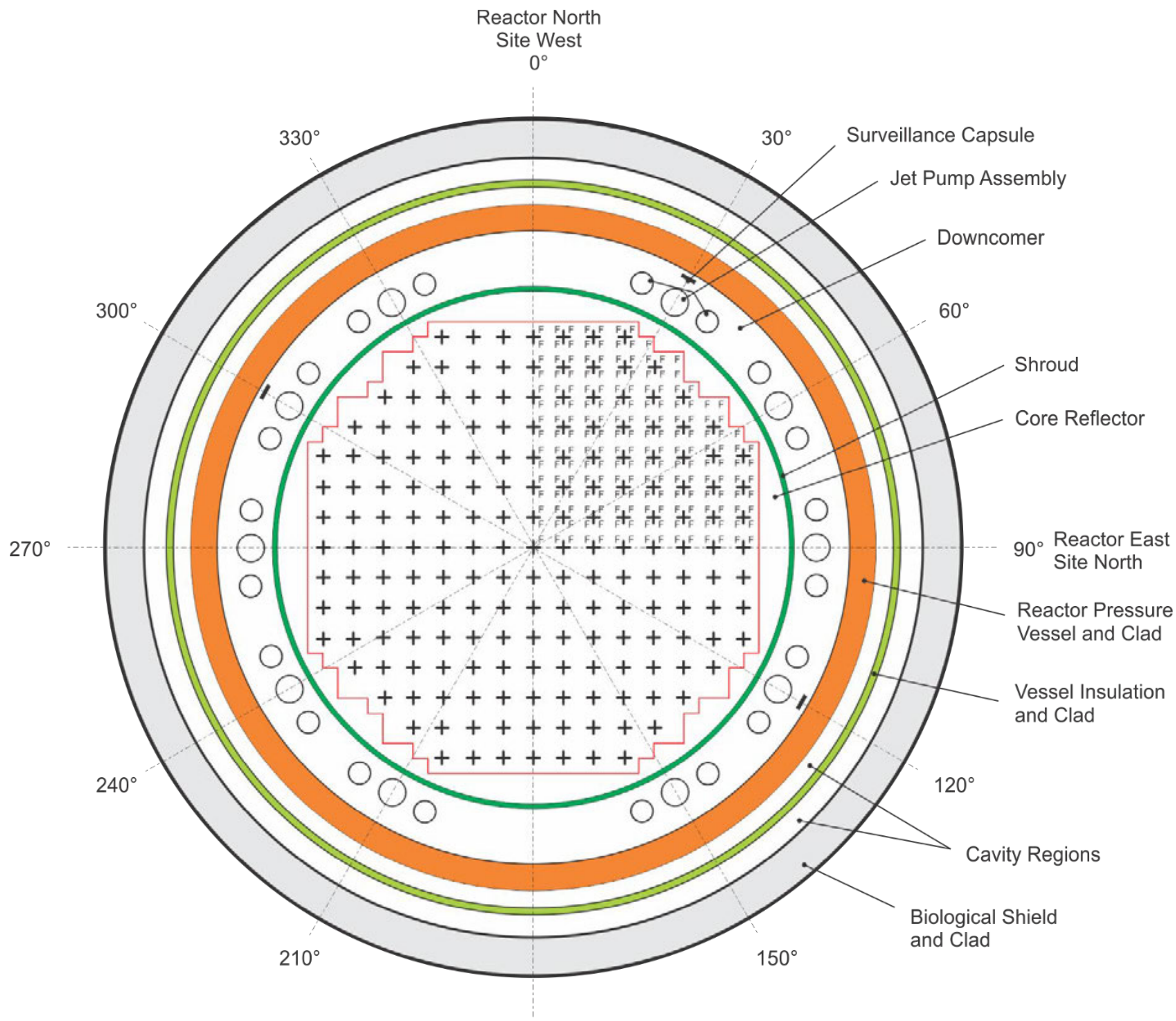
This section provides an overview of the reactor design and operating data inputs that were used to develop the computational fluence model for the LaSalle Unit 1 reactor. All reactor design and operating data inputs used to develop the model are plant-specific and were provided by Exelon. The inputs for the fluence geometry model were developed from nominal and as-built drawings for the reactor pressure vessel, vessel internals, fuel assemblies, and containment regions.

Several modifications were made to the LaSalle Unit 1 RAMA geometry model since the previous fluence evaluation performed by TransWare in 2014 [13]. [[

]]

2.1 Overview of the Reactor System Design

LaSalle Unit 1 is a General Electric BWR/5 class reactor with a core loading of 764 fuel assemblies. Figure 2-1 illustrates the basic planar configuration of the LaSalle Unit 1 reactor model at an axial elevation near the reactor core mid-plane. All of the radial regions of the reactor that are required for fluence evaluations are shown. Beginning at the center of the reactor and projecting outward, the regions include: the core region; core reflector region (bypass water); shroud wall; downcomer water region including the jet pumps; RPV wall; cavity region between the RPV wall and insulation; insulation; cavity region between the insulation and biological shield; and the biological shield wall. Cladding is included on the inner RPV surface as well as the inner and outer surfaces of the biological shield wall. Also represented in Figure 2-1 are notations indicating the control rod and fuel assembly locations within the core. Note that the fuel locations are shown only for the northeast quadrant of the core region.



Notes: This drawing is not to scale.
F = Fuel bundle locations.
(Locations shown only for the northeast quadrant.)
\+ = Control rod locations.

Figure 2-1
Planar View of the LaSalle Unit 1 Reactor at the Core Mid-Plane Elevation

2.2 Reactor System Mechanical Design Inputs

The mechanical design inputs used to construct the LaSalle Unit 1 fluence geometry model are based upon nominal design dimensional information. As-built data is always preferred when constructing plant-specific reactor fluence models; however, as-built data is not always available and nominal dimensions are used.

For the LaSalle Unit 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 inner radius

An important component of a computational reactor pressure vessel fluence model is the accurate description of the surveillance capsules installed in the pressure vessel. Figure 2-1 shows that the LaSalle Unit 1 reactor was initially equipped with three surveillance capsules. The capsules were installed at an elevation around the reactor core mid-plane. Each capsule was mounted radially near the inside surface (0T) of the RPV wall. The surveillance capsules were distributed around the pressure vessel at the 30°, 120°, and 300° azimuths relative to the reactor north 0° angular direction. 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. Activation comparisons will be listed for the 120° capsule container, the 300° capsule container, and the 30° capsule flux wire holder. These measurements are used to validate the fluence model. Presently, only the 30° capsule container remains inside the reactor system.

2.3 Reactor System Material Compositions

Each region of the reactor is comprised of materials that include reactor fuel, metal, 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 RPV, surveillance capsules, vessel internal components, and ex-vessel structures.

Table 2-1 provides a summary of the materials for the principal components and regions of the LaSalle Unit 1 reactor. The material attributes for the metal, 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 bulk water coolant properties throughout the reactor system, except for the core region, are determined assuming rated power and flow conditions. The coolant properties remain constant unless there is a reported change in system heat balance conditions that affect the water properties in the reactor. The nuclear fuel compositions and coolant properties in the reactor core region change continuously during reactor operation. The fuel and coolant properties in the core region are updated for each reactor statepoint condition based on the actual or predicted operating states of the reactor. Water properties immediately above and below the core region are updated on a cycle-by-cycle basis based on average cycle operating conditions.

Table 2-1
Summary of Material Compositions by Region for LaSalle Unit 1

Region	Material Composition
Biological Shield Clad	Low-Alloy Steel
Biological Shield Wall	Reinforced Concrete
Cavity Regions	Air
Control Rod Guide Tubes	Stainless Steel
Control Rods	Stainless Steel, B_4C, Water
Core Exit	Steam
Core Reflector	Water
Core Spray Sparger Nozzles	Steam, Stainless Steel
Core Spray Sparger Piping	Stainless Steel
Core Spray Sparger Flow Areas	Steam
Core Support Plate	Stainless Steel
Core Support Plate Rim	Stainless Steel
Core Support Plate Rim Bolts	Stainless Steel
Downcomer Region	Water
Fuel Support Pieces	Stainless Steel
Fuel Hardware Regions	Stainless Steel, Zircaloy, Inconel
Jet Pump Hold Down Beams	Inconel
Jet Pump Hold Down Brackets	Stainless Steel
Jet Pump Riser and Mixer Flow Areas	Water
Jet Pump Riser and Mixer Metal	Stainless Steel
Jet Pump Riser Brace	Stainless Steel, Water
Jet Pump Riser Brace Pad	Stainless Steel
Thermal Insulation	Air, Aluminum, Stainless Steel
Reactor Coolant / Moderator	Water
Reactor Core	^{235}U, ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, O_{fuel}, Zircaloy
Reactor Pressure Vessel Clad	Stainless Steel
Reactor Pressure Vessel Nozzle Forgings	Low-Alloy Steel
Reactor Pressure Vessel Wall	Low-Alloy Steel
Shroud	Stainless Steel
Sparger Inlet Piping	Stainless Steel
Steam Separator Standpipes	Stainless Steel
Surveillance Capsule Flux Wire Holder	Stainless Steel
Surveillance Capsule Specimen	Low-Alloy Steel
Top Guide	Stainless Steel

2.4 Reactor Operating Data Inputs

An accurate evaluation of reactor vessel and component fluence requires an accurate accounting of the reactor's operating history. The principal operating parameters that affect the determination of neutron fluence in light water reactors include the following: core configurations and fuel assembly designs, power history, exposure and isotopic distributions, and water density distributions. The following subsections provide additional information on the characterization of reactor operating data for fluence evaluations.

2.4.1 Core Configuration and Fuel Design

The reactor core configuration and the fuel assembly designs loaded in the reactor determine the neutron source and spatial source distribution contributing to the irradiation of the pressure vessel, vessel internals and ex-vessel supporting structures. The LaSalle Unit 1 core is comprised of 764 fuel assemblies in a fixed configuration. Several designs of fuel assemblies may be loaded in the reactor core in any given operating cycle. In order to determine accurate spatial fluence profiles throughout the reactor system, it is important to account for the different fuel designs loaded in the reactor over the operating lifetime of the reactor, especially those designs that reside in the peripheral locations of the core region.

Attachment 1, herein made a part of this report, provides a summary of the different fuel assembly designs that have been loaded in the LaSalle Unit 1 reactor core for each operating cycle.

2.4.2 Reactor Power History

Reactor power history is the measure of reactor power levels, reactor power spatial distributions, fuel exposure distributions, and fuel isotopic distributions that a reactor experiences over its operating life. The power history data used in the LaSalle Unit 1 fluence evaluation includes daily power levels for each cycle. The power history for LaSalle Unit 1 also accounts for periods of reactor shutdown due to refueling outages and other events that affect the activation and decay of dosimetry data. Power history data is also needed to predict the reactor fluence at the end of 60 years of operation.

Attachment 1 provides a summary of the operating history of the LaSalle Unit 1 reactor for each operating cycle. Attachment 1 also shows the EFPY accumulated at the end of each cycle. The accumulated EFPY is computed from the operating data provided by Exelon and is verified against power production and exposure data obtained separately for the plant.

2.4.3 Reactor Statepoint Data

Statepoints are snapshots in time that characterize the power-flow conditions of a reactor at a moment in time. Typically, several statepoints are used to represent the different operating conditions experienced by the reactor over the course of an operating cycle. The number of statepoints used to characterize a cycle of operation may vary between cycles for several reasons, including changes in core power, changes in core flow, and movement of control rods. In some cases, the number of statepoints for a cycle is also be affected by the availability of data to represent the operating conditions of the reactor for that cycle.

Core simulator data was provided by Exelon to characterize the historical operating conditions of the LaSalle Unit 1 reactor. 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. For the LaSalle Unit 1 reactor fluence model, core simulator data was provided by Exelon for all operating cycles beginning with the start of commercial operation; thereby providing the best representation of the reactors operating life and for the determination of neutron fluence.

Because core simulator codes are used for a variety of core analysis functions, 10's to 100's of core calculations may be performed to track and monitor the operation of a reactor over the course of an operating cycle. Not all core calculations are suitable for use in fluence evaluations. Therefore, each cycle of operating data is investigated to select the statepoints that are suitable for use in fluence evaluations. When all reactor conditions are considered, the number of core simulator statepoints selected for a fluence evaluation can vary from cycle to cycle.

A separate neutronics transport calculation is performed for each selected statepoint. The neutron fluxes calculated for each statepoint are then combined with the appropriate daily power history data described in Section 2.4.2 and Attachment 1 to provide an accurate accounting of the neutron fluence for the reactor pressure vessel, reactor vessel internals, and surveillance capsules. The periods of reactor shutdown are also accounted for in this process, particularly to allow for an accurate calculation of irradiated surveillance capsule activities.

2.4.4 Reactor Coolant Properties

The reactor coolant water densities used in the fluence model are determined using combinations of core simulator codes and reactor heat balance data.

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3
METHODOLOGY

This section provides an overview of the methodology and modeling approach used to determine fast neutron fluence for the LaSalle Unit 1 reactor pressure vessel (RPV), reactor vessel internals (RVI), and the fast neutron fluence and activations for the reactor surveillance capsules.

The fluence model for LaSalle Unit 1 is a plant-specific model that is constructed from the design inputs described in Section 2, *Description of the Reactor System*. The computational tools used in the fluence and activation analyses are based on the RAMA Fluence Methodology (RAMA) [3]. A general approach for using the toolset is presented in the RAMA Procedures Manual [2].

3.1 Computational Fluence Method

The RAMA Fluence Methodology is a system of computer codes, a data library, and an uncertainty methodology that determines best-estimate fluence and activations in light water reactor pressure vessels and vessel internal components. The primary software that comprises the methodology includes model builder codes, a particle transport code, and a fluence calculator code.

The primary inputs for the fluence methodology 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 multiple sources, such as core simulator software, system heat balance calculations, daily operating logs, and cycle summary reports. A variety of outputs are available from the fluence methodology that include neutron flux, fast neutron fluence, dosimetry activation, and an uncertainty analysis.

The model builder codes consist of geometry and material processor codes that generate input for the RAMA 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 and material property 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 RAMA transport code performs three-dimensional neutron flux calculations using a deterministic, multigroup, particle transport theory method with anisotropic scattering [[
]] The transport solver is coupled with a general geometry modeling capability based on combinatorial geometry techniques. The coupling of general (arbitrary) geometry with a deterministic transport solver provides a flexible, efficient, and stable method for calculating neutron flux in light water reactor pressure vessels, vessel components, and structures. The primary inputs 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 primary output from the transport code is the neutron flux in multigroup form for every material region mesh in the fluence model.

The fluence calculator code determines fluence and activation in the reactor pressure vessel, surveillance specimens, and vessel components over specified periods of reactor operation. The fluence calculator also includes treatments for isotopic production and decay that are required to calculate specific activities for irradiated materials, such as the dosimetry specimens in the surveillance capsules. 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, specification of which components to evaluate, and the energy ranges of interest for evaluating neutron fluence. The reactor operating history is generally represented with several reactor statepoints that represent the core power and core power distributions of the reactor over the operating life of the reactor. These statepoints are integrated with the daily variations in reactor power levels to predict the fluence and activations accumulated throughout the reactor system.

The RAMA nuclear data library contains atomic mass data, nuclear cross-section data, response functions, and other nuclear constants that are needed for each of the code tools. The structure and contents of the data contained within the nuclear data file are based on the BUGLE-96 nuclear data library [15], with extended data representations derived from the VITAMIN-B6 data library [16].

The uncertainty methodology provides an assessment of the overall accuracy of the fluence and activation calculations. Variations in the [[ ]] 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 [7].

3.2 Fluence Model

Section 2.2 describes the design inputs that were provided by Exelon for constructing the LaSalle Unit 1 reactor fluence model. These design inputs are used to develop a plant-specific, three-dimensional computational model of the LaSalle Unit 1 reactor for determining fast neutron fluence in the RPV and RVI components and for determining activation and fluence in reactor dosimetry for validating the RPV fluence predictions.

Figure 3-1 and Figure 3-2 provide general illustrations of the primary components, structures, and regions developed for the LaSalle Unit 1 fluence model. Figure 3-1 shows the planar configuration of the reactor model at an elevation corresponding to the reactor core mid-plane. Figure 3-2 shows an axial configuration of the reactor model. Note that the figures are not drawn to scale and do not include representations of the meshing developed for this evaluation. The figures 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). Additional detail is beyond the scope of this document.

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Figure 3-1
Planar View of the LaSalle Unit 1 Fluence Model at the Core Mid-Plane Elevation in Quadrant Symmetry

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Figure 3-2
Axial View of the LaSalle Unit 1 Fluence Model

3.2.1 Geometry Model

The LaSalle Unit 1 fluence model is constructed on a Cartesian coordinate system using a generalized three-dimensional geometry modeling technique based on combinatorial geometry. The 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. This allows any point in the reactor model to be referenced by specifying the (x,y,z) coordinates for that point.

[[

]] This modeling approach permits a model to be developed in any level of high-definition detail, such as is necessary for fluence and activation evaluations.

Figure 2-1 illustrates a planar cross-section view of the LaSalle Unit 1 reactor design at an axial elevation corresponding to the reactor core mid-plane. It is shown for this one elevation that the reactor design is a complex geometry [[
]] 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, surveillance capsules, vessel internals, and the supporting structures inside and outside of the reactor vessel.

Figure 3-1 and Figure 3-2 provide general illustrations of the planar and axial geometry complexities that are represented in the fluence model. For comparison purposes, the planar view illustrated in Figure 3-1 corresponds to the core elevation illustrated in Figure 2-1. [[

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As previously noted, Figure 3-1 and Figure 3-2 are not drawn precisely to scale and 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. The following subsections provide additional information on the constituent models developed for the individual components, structures, and regions of the fluence model.

3.2.2 Reactor Core and Core Reflector

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 coolant-containing region between the core and the core shroud is the core reflector. The reactor core geometry is rectangular in design [[

]]

3.2.3 Reactor Core Shroud

The core shroud is a canister-like structure that surrounds the reactor core. It 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 and is divided into three sections: lower, central, and upper. The lower shroud extends from the bottom of the model to the core support plate flange, the central shroud extends from the core support plate flange to the top guide flange, and the upper shroud extends from top guide flange to the top of the shroud head rim.

[[

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Above the shroud wall is the shroud head which is penetrated by numerous steam separator standpipes. [[

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3.2.4 Downcomer Region

The downcomer region lies between the core shroud and the reactor pressure vessel. The downcomer is effectively cylindrical in design, but with geometrical complexities created by the presence of jet pumps and surveillance capsules in the region. [[

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3.2.4.1 Jet Pumps

LaSalle Unit 1 has ten jet pump assemblies in the downcomer region, which provide the main recirculation flow for the core. [[

]]

3.2.4.2 Surveillance Capsules

The three (3) OEM surveillance capsules installed in the LaSalle Unit 1 reactor are positioned in close proximity to the RPV inner wall surface. The capsules are positioned at 30°, 120°, and 300°. [[

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3.2.5 Reactor Pressure Vessel

The reactor pressure vessel and vessel cladding lie outside the downcomer region, [[

]]

3.2.6 Thermal Insulation

The reactor vessel thermal insulation lies in the cavity region outside the pressure vessel wall. The insulation is composed of three layers: an inner cladding, a center insulating layer composed of an array of foils, and an outer cladding. [[

]]

3.2.7 Inner and Outer Cavity Regions

There are effectively two cavity regions represented in the model. The inner cavity region lies between the outer surface of the pressure vessel wall and the inner surface of the vessel insulation. The outer cavity region lies between the outer surface of the vessel insulation and inner surface of the biological shield wall cladding. [[

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3.2.8 Biological Shield Model

The biological shield (concrete) defines the outermost region of the fluence model. [[

]]

3.2.9 Above-Core Components

Figure 3-2 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, core spray spargers, upper core shroud wall, shroud head, and steam separator standpipes. The shroud regions and standpipes are mentioned in further detail in Section 3.2.3.

3.2.9.1 Top Guide

The top guide component lies above the core region and is appropriately modeled to include discrete representations of the top guide plates. The top guide model also accounts for the fuel assembly parts and coolant flow between the plates. [[

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3.2.9.2 Core Spray Spargers and Piping

The core spray spargers include upper and lower sparger annulus pipes and a vertical inlet pipe. [[

]]

3.2.10 Below-Core Components

Figure 3-2 includes illustrations of other components and regions that lie below the reactor core region. The predominant below core (i.e., below active fuel) components represented in the fluence model include the lower fuel assembly parts, fuel support pieces, core support plate, core support plate rim bolts, cruciform control rods, control rod guide tubes, and lower shroud wall. The lower shroud wall and fuel assembly components are described in previous sections, with the remaining components described in the following subsections.

3.2.10.1 Core Support Plate and Rim Bolts

The core support plate includes appropriate penetrations for the fuel support pieces, control rod guide tubes, cruciform control rods, and the core support plate rim bolts. Core support plate rim bolts protrude from the top of the core support plate and traverse through the plate, rim, and core shroud lower flange. [[

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3.2.10.2 Fuel Support Pieces

The nuclear fuel assemblies loaded in the reactor are seated on fuel support pieces, which then rest in the core support plate and control blade guide tubes. [[

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3.2.10.3 Control Blades and Guide Tubes

The fluence model allows for the representation of cruciform-shaped control blades and tubular control blade guide tubes in the below-core regions of the reactor. Coolant flow paths are included in the model [[

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3.2.11 Summary of the Geometry Modeling Approach

To summarize the reactor modeling process, there are several key features that allow the reactor design to be accurately represented for RPV and RVI fluence evaluations. [[

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3.3 Particle Transport Calculation Parameters

The accuracy of the transport method is based on a numerical integration technique [[

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3.4 Fission Spectrum and Neutron Source

Modern core simulator software is capable of providing three-dimensional core power distributions and fuel isotopics in high-definition detail, viz., on a pin-by-pin basis. This allows fluence models to be constructed with a high-level of modeling detail for representing unique fission spectrum and neutron source terms for the transport calculation. [[

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3.5 Parametric Sensitivity Analyses

Several plant-specific sensitivity analyses are performed to evaluate the accuracy and predictability of the neutral particle transport methodology for determining RPV and RVI component fluence. [[

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4

SURVEILLANCE CAPSULE EVALUATIONS AND COMBINED UNCERTAINTY ANALYSIS

U.S. NRC Regulatory Guide 1.190 [7] requires that fluence calculational methods be validated by comparison to operating reactor dosimetry measurements. It is preferred that measurement comparisons apply to the host reactor; however, provisions are made that allow comparison to reactor dosimetry from reactors of similar design. The acceptance criteria provided in Regulatory Guide 1.190 is that standard deviations determined from the calculated-to-measurement comparison ratios (C/M) fall within a computed standard deviation of ± 20%.

Attachment 1, herein made a part of this document, presents the computed activation and fluence results determined for the LaSalle Unit 1 reactor surveillance capsules and flux wires that were removed from the reactor at different exposures in the reactor's operating life. The activation results form the basis for the validation and qualification of the fluence methodology for the LaSalle Unit 1 reactor in accordance with requirements of Regulatory Guide 1.190. Attachment 1 also presents the results of the combined uncertainty analysis that was performed for the LaSalle Unit 1 reactor.

It is determined from the surveillance capsule flux wire evaluations and the combined uncertainty analysis that the LaSalle Unit 1 reactor fluence model meets the requirements of Regulatory Guide 1.190. Therefore, the computed fluence for the LaSalle Unit 1 reactor pressure vessel is the best-estimate fluence with no discernible bias in the results. For more information concerning the uncertainty evaluations, please see Attachment 1.

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5
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5.2 Glossary

AZIMUTHAL QUADRANT SYMMETRY – A type of core and pressure vessel azimuthal representation that represents a single quadrant of the reactor that can be rotated and mirrored to represent the entire 360-degree geometry. For example, 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.

BOC – An acronym for beginning-of-cycle.

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 produce the best-estimate neutron fluence.

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

DAMAGE FLUENCE – See Neutron Fluence.

DPA – An acronym for displacements per atom which is typically used to characterize material damage in ferritic steels due to neutron exposure.

EFFECTIVE FULL POWER YEARS (EFPY) – A unit of measurement representing one full year of operation at the reactor's rated power level. For example, if a reactor operates for 12 months at full rated power, this represents 1.0 EFPY. 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 also represents 1.0 EFPY.

EOC – An acronym for end-of-cycle.

EXTENDED BELTLINE REGION – See RPV beltline.

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

NEUTRON FLUENCE – Time-integrated neutron flux reported in units of n/cm^2. The term "best-estimate" fluence refers to the fast neutron fluence that is computed in accordance with the requirements of U.S. Nuclear Regulatory Commission Regulatory Guide 1.190. The term "damage fluence", which is required for material embrittlement evaluations, refers to an adjusted fast neutron fluence that is determined using damage functions specified in U.S. Nuclear Regulatory Commission Regulatory Guide 1.99.

OEM – An acronym for Original Equipment Manufacturer.

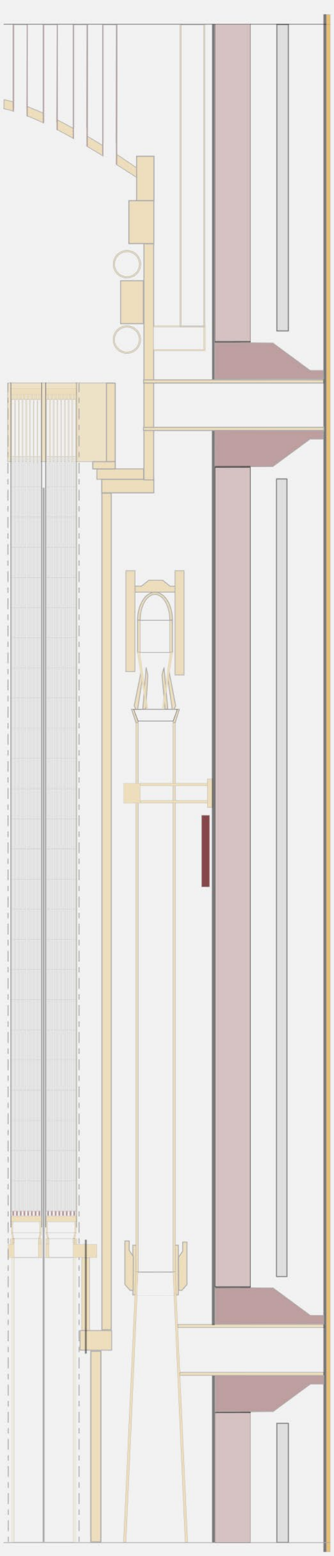
RPV – An acronym for reactor pressure vessel. Unless otherwise noted, the reactor pressure vessel refers to the base metal material of the RPV wall (i.e., excluding clad/liner).

RPV BELTLINE – The RPV beltline is defined as that portion of the RPV adjacent to the reactor core that attains sufficient neutron radiation damage that the integrity of the pressure vessel could be compromised. For purposes of this evaluation, the fast neutron fluence threshold used to

define the traditional RPV beltline is 1.0E+17 n/cm^2. The axial span of the RPV that can exceed this threshold includes the RPV shells, welds, and heat-affected zones. An "extended beltline" is also defined to include lower fluence regions of the pressure vessel but with higher stresses than the traditional beltline region, such as RPV nozzles. The combination of fluence and stress may result in a limiting location in the pressure vessel for determining pressure-temperature limits.

RPV ZERO ELEVATION – The RPV zero elevation is defined at the inside surface of the lowest point in the vessel bottom head, which is typically the bottom drain plug location. Axial elevations presented in this report are relative to RPV zero.

RVI – An acronym for reactor vessel internals.



Topical Report

LASALLE COUNTY GENERATING STATION UNIT 1 FLUENCE METHODOLOGY REPORT

Attachment 1

Qualification of the LaSalle Unit 1 Reactor Fluence Model – Cycles 1 to 18

Document Number: LAS-FLU-001-R-010
Attachment 1, Revision 0
September 2021

Prepared by: **TransWare Enterprises Inc.**

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Topical Report

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Prepared By: **TransWare Enterprises Inc.**

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DISCLAIMER OF WARRANTIES AND LIMITATION OF LIABILITIES

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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.

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1
INTRODUCTION

This attachment provides the reactor operating history, comparisons to activation measurements, and uncertainty analysis that are essential for validating the LaSalle County Generating Station Unit 1 (LaSalle Unit 1) fluence methodology. The methodology that is used for determining the neutron fluence in the LaSalle Unit 1 reactor is detailed in the LaSalle County Generating Station Unit 1 Fluence Methodology Report [1].

The power history data presented in this report covers the time period from start of commercial operation (circa 1982) to the end of operating cycle 18. All surveillance dosimetry removed from the reactor over that time period, and which is available in the form of activation measurements, is evaluated. A combined uncertainty factor for the fluence model based on the modeling approach and measurement comparisons is determined which demonstrates that the computational fluence method used by TransWare Enterprises Inc. is qualified for use in determining neutron fluence for the LaSalle Unit 1 reactor pressure vessel in accordance with U.S. Nuclear Regulatory Commission (U.S. NRC) Regulatory Guide 1.190 [2].

In compliance with Regulatory Guide 1.190, it is shown in this report that the calculated-to-measured (C/M) ratio and standard deviation is 0.99 ± 0.10 for all reactor dosimetry evaluated for the LaSalle Unit 1 reactor. The combined uncertainty for the LaSalle Unit 1 reactor is determined to be 9.15%. Based upon these results, there is no discernable bias in the computed reactor pressure vessel fluence for the period Cycle 1 through the end of Cycle 18 for the LaSalle Unit 1 reactor.

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2
REACTOR OPERATING HISTORY

Reactor operating history is the measure of daily reactor power levels that characterize the radiation exposure history of a reactor over its operating life. The daily power history data for the LaSalle Unit 1 reactor was provided by Exelon in discrete form for Cycles 1 through 18.

Another important element of the reactor operating history is the fuel designs that were loaded in the reactor for each operating cycle. Each fuel design has a different power signature in the core and, therefore, results in different spatial power, exposure, and fuel isotopic distributions throughout the core region.

Table 2-1 and Table 2-2 provide a summary of the fuel designs that were loaded in the LaSalle Unit 1 reactor for each operating cycle. Table 2-1 lists the fuel designs that were loaded in the reactor for Cycles 1 through 9. Table 2-2 lists the fuel designs that were loaded for Cycles 10 through 18. The dominant fuel design that was loaded in the core for each cycle is shown in **bold** font. The dominant fuel design that was loaded on the core periphery is identified in blue font.

Table 2-1
Summary of LaSalle Unit 1 Core Loading Inventory for Cycles 1 to 10

Cycle	Fuel Designs					
	8x8				9x9	10x10
	GE5	GE7B	GE8B	GE9	ATRIUM-9	ATRIUM-10
1	**764**					
2	**532**	232				
3	**308**	232	224			
4	137	**231**	224	172		
5		176	224	**364**		
6		44	156	**564**		
7				**764**		
8				**764**		
9A				**392**	372	
9B				**392**	372	
10A				46	**372**	346
10B				49	**369**	346

Table 2-2
Summary of LaSalle Unit 1 Core Loading Inventory for Cycle 11 to 18

Cycle	Fuel Designs				
	9x9	10x10			
	ATRIUM-9	ATRIUM-10	GE14	GNF2	GNF3
11	130	344	290		
12		474	290		
13		603	161		
14A		764			
14B		764			
15		468		296	
16		180		584	
17				764	
18				764	
19prj(1)				764	
20+(2)					764

1) Projection Cycle 19 consists partially of historical data and partially of projected data.
2) For purposes of fluence projections, operation beyond projection Cycle 19 uses an equilibrium projection cycle featuring a full core loading of GNF3 (Cycle 20+). This cycle was provided by Exelon to predict fluence at the end of the extended plant license period.

3
REACTOR STATEPOINT DATA

The reactor operating history is defined in discrete exposure steps referred to as statepoints. A statepoint is defined as a snapshot in time that characterizes an equilibrium power-flow condition of a reactor core at a moment in time. The importance of statepoints is the affect that changes in reactor core power and core flow have on the granular radial and axial power distributions in the reactor core region. In particular, the changes in the granular power distributions have a proportional effect on the neutron flux that is calculated in the structural components adjacent to the core region, including the reactor pressure vessel. As power and flow do not vary in a proportional manner, several statepoints are needed to accurately characterize the operating states of the reactor core and the integral power that affects a fluence calculation.

Several statepoints are generally used to represent the different operating states of a reactor core over the course of an operating cycle. The core power distribution for a statepoint is generally determined using core simulator software. Because core simulator codes are used for a variety of core analysis functions, 10's to 100's of core calculations may be performed to track and monitor the operation of a reactor. Not all core calculations are suitable for use in fluence evaluations. Therefore, each cycle of operating data is investigated to select the statepoints that are suitable for use in fluence calculations. When all reactor conditions are considered, the number of core simulator statepoints selected for a reactor fluence evaluation can vary from cycle to cycle.

Core simulator data was provided by Exelon to characterize the historical operating conditions for the LaSalle Unit 1 reactor for Cycles 1 through 18. Table 3-1 shows that a total of [[ ]] statepoints were used to represent the operating states of the LaSalle Unit 1 reactor for the first 18 cycles of operating history. It is also shown in Table 3-1 that the number of statepoints used per cycle in the fluence calculation varied. [[

]] Table 3-1 also shows the rated thermal power of the reactor for a cycle and the accumulated effective full power years (EFPY) of exposure accumulated for that cycle.

A separate neutronics transport calculation is performed for each statepoint listed in Table 3-1. The neutron fluxes calculated for each statepoint are then combined with daily thermal power information to provide an integral accounting of the neutron fluence for the reactor pressure vessel, reactor vessel internals, and surveillance capsules. The periods of reactor shutdown are also accounted for in this process, particularly to allow for an accurate calculation of irradiated surveillance capsule activities.

In addition to the 18 cycles of operating history data, two projection cycles, Cycle 19 and Cycle 20+, are also shown in Table 3-1. Cycle 19 was provided by Exelon as a transition cycle to Cycle 20+. Cycle 20+ was provided by Exelon and represents an equilibrium cycle that will be used to project fast neutron fluence to the end of 60 years of operation.

Table 3-1
Statepoint Data for LaSalle Unit 1 per Cycle Basis

Cycle Number	Number of Reactor Statepoints	Rated Thermal Power (MWt)	Accumulated EFPY
1	[[	3323	1.38
2		3323	2.24
3		3323	3.29
4		3323	4.32
5		3323	5.60
6		3323	6.47
7		3323	7.82
8		3323	9.21
9A		3323	9.69
9B(1)		3489	11.31
10A		3489	11.57
10B		3489	13.15
11		3489	15.16
12		3489	16.98
13		3489	18.87
14A		3489	19.42
14B(2)		3546	20.77
15		3546	22.59
16		3546	24.51
17		3546	26.40
18		3546	28.28
19(3)		3546	30.22
20+(4)	]]	3546	54.0

(1) A power uprate was implemented midway through cycle 9 from 3323 MWth to 3489 MWth.
(2) A power uprate was implemented midway through cycle 14 from 3489 MWth to 3546 MWth.
(3) Cycle 19 is comprised of partial historical and partial projection data. This cycle is used as a transition cycle to a GNF3 projection cycle.
(4) Cycle 20+ is a projection cycle using a full core of GNF3 fuel. This cycle will be used to project reactor fluence at the end of the extended license period.

4
SURVEILLANCE CAPSULE DOSIMETRY EVALUATION

This section presents the results of the activation analysis and the determination of fast neutron fluence for the LaSalle Unit 1 surveillance capsule dosimetry. Lead factors associating the peak RPV fluence with the capsule fluence are also reported. The results presented in this section form the basis for the validation and qualification of the fluence methodology as applied to the LaSalle Unit 1 reactor in accordance with the Regulatory Guide 1.190 [1].

Regulatory Guide 1.190 requires that fluence calculational methods to be validated by comparisons with activation measurements from operating reactor dosimetry. It is preferred that the activation data be taken from the reactor being evaluated. However, comparative data from plants of similar design may be used in cases where insufficient dosimetry measurements exist. In the case for the LaSalle Unit 1 reactor, there is sufficient plant-specific measurements available to qualify the calculational method without reference to other plants of similar design.

To report computed fluence as the best-estimate fluence, Regulatory Guide 1.190 requires that the standard deviation resulting from the comparison of calculated to measurement data should be ≤ 20%. It is determined that overall calculated-to-measured (C/M) comparison ratio and standard deviation for the LaSalle Unit 1 reactor is 0.99 ± 0.10. Therefore, the computational fluence model for the LaSalle Unit 1 reactor meets the Regulatory Guide 1.190 criteria and, as such, no bias adjustment is required to be applied to the computed RPV fluence.

4.1 Summary of the Flux Wire Activation Analysis

The LaSalle Unit 1 reactor hosted three (3) surveillance capsules that are mounted near the reactor core mid-plane elevation. The capsules are positioned near the inner surface of the reactor pressure vessel wall at the 30°, 120°, and 300° azimuths around the circumference of the reactor pressure vessel. Figure 4-1 illustrates the positioning of the surveillance capsules in the LaSalle Unit 1 reactor.

The 120° and 300° surveillance capsule were removed from the LaSalle Unit 1 reactor for testing of radiation effects on the Charpy specimens and activation analysis of the flux wires that were contained in the capsules. In addition, one (1) set of flux wires were extracted from the flux wire holder attached to the 30° surveillance capsule for activation analysis.

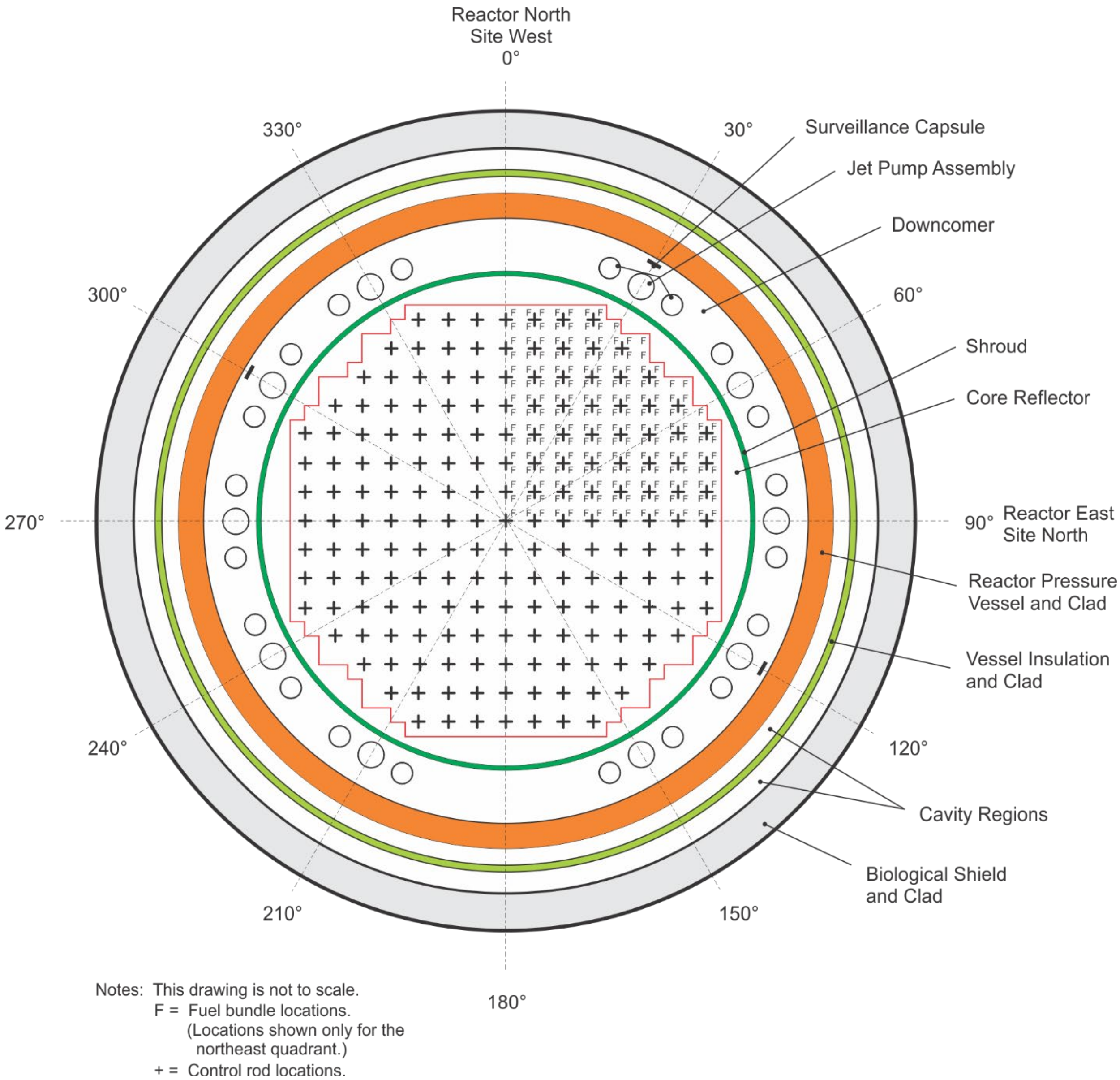

Figure 4-1
Positioning of the Surveillance Capsules Installed in the LaSalle Unit 1 Reactor

Table 4-1 provides a summary of the activation comparisons for each set of flux wire that were irradiated in LaSalle Unit 1 reactor. The table shows that the overall calculated-to-measured (C/M) ratio and associated standard deviation for 24 irradiated specimens was determined to be 0.99 ± 0.10.

Table 4-1 also provides the irradiation period in terms of cycle exposure, the accumulated exposure in terms of Effective Full Power Years (EFPY) of reactor operation, the average fast neutron fluence (E > 1.0 MeV) for each set of flux wires, and the number of specimens (viz., flux wires) evaluated for each surveillance capsule and set of flux wires removed from the LaSalle Unit 1 reactor.

Table 4-1
Summary of the Fluence and Activity Comparisons for the LaSalle Unit 1 Dosimetry

Dosimeter	Cycles of Exposure	Accumulated Exposure (EFPY)	Fast Neutron Fluence (>1 MeV, n/cm^2)	Number of Specimens	Calculated vs. Measured (C/M)	Standard Deviation (σ)
30° Flux Wire	1 - 1	1.38	2.34E+16	6	0.92	0.04
300° Capsule	1 - 6	6.47	1.05E+17	9	0.96	0.09
120° Capsule	1 - 13	18.87	3.79E+17	9	1.07	0.06
Overall Average C/M and Standard Deviation				**24**	**0.99**	**0.10**

Table 4-2 provides the overall calculated-to-measured ratios and standard deviations for the copper, iron, and nickel flux wires that were irradiated in the LaSalle Unit 1 reactor.

Table 4-2
Summary of the Activity Comparisons for the Iron and Copper Flux Wires Removed From the LaSalle Unit 1 Reactor

Flux Wire	Number of Specimens	C/M	σ
Copper	9	0.90	0.07
Iron	9	1.02	0.06
Nickel	6	1.08	0.06
Overall Average C/M and Standard Deviation	**24**	**0.99**	**0.10**

4.2 Comparison of Predicted Activation to Plant-specific Measurements

The comparison of calculated activations to measurements for the dosimetry are presented in this subsection. Fluence and lead factors for each capsule are reported in Subsection 4.3, *Reactor Pressure Vessel Lead Factors*.

4.2.1 Flux Wire Activation Analysis for the LaSalle Unit 1 30° Capsule

Table 4-3 shows the activation measurements, periods of irradiation, computed activations, and the computed-to-measured (C/M) ratios for the flux wires irradiated in the 30° flux wire holder. Activation calculations were performed for the following reactions: ^{54}Fe (n,p) ^{54}Mn and ^{63}Cu (n,α) ^{60}Co.

The average C/M ratio and associated standard deviation determined for all of the flux wires irradiated in the LaSalle Unit 1 30° flux wire holder is 0.92 ± 0.04.

Table 4-3
Comparison of the Calculated-to-Measured Activities for the Iron Flux Wires Removed From the LaSalle Unit 1 30° Flux Wire Holder

Flux Wire	Dosimeter	Measured (dps/g) [3]	Calculated (dps/g)	C/M	σ
Iron	Fe-1	3.17E+04	3.03E+04	0.96	-
	Fe-2	3.16E+04	3.03E+04	0.96	-
	Fe-3	3.19E+04	3.03E+04	0.95	-
Iron Average				**0.96**	**0.00**
Copper	Cu-1	1.81E+03	1.61E+03	0.89	-
	Cu-2	1.84E+03	1.61E+03	0.88	-
	Cu-3	1.84E+03	1.61E+03	0.88	-
Copper Average				**0.88**	**0.01**
Average C/M and Standard Deviation				**0.92**	**0.04**

4.2.2 Cycle 6 Surveillance Capsule Activation Analysis

Iron, nickel, and copper flux wires were irradiated in the LaSalle Unit 1 300° surveillance capsules during the first six (6) cycles of reactor operation. At the time of their removal, the flux wires had been irradiated for a total of 6.47 EFPY.

Activation calculations were performed for the following reactions: ^{54}Fe (n,p) ^{54}Mn, ^{58}Ni (n,p) ^{58}Co, and ^{63}Cu (n,α) ^{60}Co. The precise location of the individual wires within the capsule is not known; therefore, the activation calculations were performed at the center of the capsule container.

Table 4-4 provide comparisons of the calculated-to-measured specific activities for each iron, nickel, and copper flux wire removed from the LaSalle Unit 1 reactor at end of Cycle 6. The average of the calculated-to-measured ratio for each flux wire is reported in the table below. It is noted that only one measurement is provided for each flux wire. Therefore, a computed standard deviation for each flux wire is not determined.

Table 4-4
Comparison of the Calculated-to-Measured Activities for the Iron and Copper Flux Wires Removed From the LaSalle Unit 1 300° Surveillance Capsule at EOC 6

Flux Wire	Measured (dps/g) [4]	Calculated (dps/g)	C/M	σ
Iron	3.65E+04	3.75E+04	1.03	-
Nickel	5.17E+05	5.32E+05	1.03	-
Copper	5.38E+03	4.50E+03	0.84	-
Average C/M and Standard Deviation			**0.96**	**0.09**

4.2.3 Cycle 13 Surveillance Capsule Activation Analysis

Iron, nickel, and copper flux wires were irradiated in the LaSalle Unit 1 120° surveillance capsules during the first thirteen (13) cycles of reactor operation. At the time of their removal, the flux wires had been irradiated for a total of 18.87 EFPY.

Activation calculations were performed for the following reactions: ^{54}Fe (n,p) ^{54}Mn, ^{58}Ni (n,p) ^{58}Co, and ^{63}Cu (n,α) ^{60}Co. The precise location of the individual wires within the capsule is not known; therefore, the activation calculations were performed at the center of the capsule container.

Table 4-5 provide comparisons of the calculated-to-measured specific activities for each iron, nickel, and copper flux wire removed from the LaSalle Unit 1 reactor at end of Cycle 13. The average of the calculated-to-measured ratio for each flux wire is reported in the table below.

Table 4-5
Comparison of the Calculated-to-Measured Activities for the Iron and Copper Flux Wires Removed From the LaSalle Unit 1 120° Surveillance Capsule at EOC 13

Flux Wire	Dosimeter	Measured (dps/g) [5]	Calculated (dps/g)	C/M	σ
Iron	G016 Iron	6.34E+04	7.00E+04	1.10	-
	G017 Iron	6.52E+04	7.00E+04	1.07	
	G018 Iron	6.48E+04	7.00E+04	1.08	
Iron Average				1.09	0.02
Nickel	G016 Nickel	8.94E+05	1.01E+06	1.13	-
	G017 Nickel	8.94E+05	1.01E+06	1.13	
	G018 Nickel	8.91E+02	1.01E+06	1.14	
Nickel Average				1.13	0.00
Copper	G016 Copper	1.02E+04	1.03E+01	1.01	-
	G017 Copper	1.05E+04	1.03E+01	0.98	
	G018 Copper	1.05E+04	1.03E+01	0.98	
Copper Average				0.99	0.02
Average C/M and Standard Deviation				**1.07**	**0.06**

4.3 Reactor Pressure Vessel Lead Factors

Table 4-6 and Table 4-7 provide the best-estimate fast neutron fluence determined for the reactor pressure vessel and dosimeters and the associated lead factors for each evaluated time period for the LaSalle Unit 1 reactor.

Table 4-6
Best-Estimate Fluence Determined for the LaSalle Unit 1 Surveillance Capsules

Evaluated Time Period	Dosimeter Fluence (n/cm^2)			Peak RPV Fluence (n/cm^2)	
	30°	300°	120°	0T	1/4T
EOC 6	--	1.05E+17	--	1.16E+17	8.01E+16
EOC 13	--	--	3.79E+17	3.81E+17	2.63E+17
32 EFPY	6.07E+17	--	--	6.25E+17	4.30E+17
54 EFPY	9.37E+17	--	--	9.78E+18	6.74E+17

Table 4-7
Lead Factors Determined for the LaSalle Unit 1 Surveillance Capsules

Evaluated Time Period	Lead Factor					
	30°		300°		120°	
	0T	1/4T	0T	1/4T	0T	1/4T
EOC 6	--	--	0.90	1.31	--	--
EOC 13	--	--	--	--	0.99	1.44
32 EFPY	0.97	1.41	--	--	--	--
54 EFPY	0.96	1.39	--	--	--	--

1) The lead factor is defined as the ratio of the fast neutron fluence at the center of the surveillance capsule to the peak fast neutron fluence at the base metal inner surface (0T) of the RPV. A second lead factor is also provided assuming the peak damage fluence at the 1/4T depth of the RPV wall.

The calculated-to-measured activation comparisons for the surveillance capsules presented in the previous sections show no discernable bias in the computational fluence method. Therefore, the best-estimate fluence reported for each capsule in Table 4-6 is the fast neutron fluence computed by the fluence methodology.

5 REACTOR PRESSURE VESSEL FLUENCE UNCERTAINTY ANALYSIS

This section presents the combined uncertainty analysis and the determination of bias for the LaSalle Unit 1 reactor pressure vessel (RPV) fluence evaluation. The combined uncertainty is comprised of two components. One component is the uncertainty factors developed from plant-specific measurements and the other is an analytic uncertainty factor. When combined, these components provide a basis for determining the combined uncertainty (1σ) and bias in the computed RPV fluence.

The requirements for determining the combined uncertainty and bias for light water reactor pressure vessel fluence evaluations are provided in Regulatory Guide 1.190 [2]. The approach for determining combined uncertainty and bias for reactor pressure vessel fluence is demonstrated in Reference 6.

For pressure vessel fluence evaluations, two uncertainty factors are considered: comparison factors and uncertainty introduced by the measurement process. After analysis of these factors, it is determined that the combined uncertainty for LaSalle Unit 1 RPV fluence is 9.15%, and that no adjustment for bias is required for the RPV fast neutron fluence determined for the period Cycle 1 through the end of Cycle 18.

5.1 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: operating reactor comparison factors and benchmark comparison factors.

5.1.1 Operating Reactor Comparison Uncertainty

TransWare has evaluated activation measurements for several BWR plants ranging from BWR/2-class plants to BWR/6-class plants. Each class of BWRs can have one or several variations of reactor core configurations, each having different radial diameters for the core shroud, reactor pressure vessel, and biological shield components. In addition, each can have different placements of the jet pumps and surveillance capsules in the reactor vessels.

The LaSalle Unit 1 reactor is a BWR/5 class design. [[

]] The overall comparison

ratio for all BWR class plants evaluated as of the date of this report is 1.01 ± 0.10. [[

]]

5.1.2 Benchmark Comparison Uncertainty

The benchmark comparison uncertainty is based on a set of industry standard simulation benchmark comparisons. In accordance with the guidelines provided in Regulatory Guide 1.190, it is appropriate to include comparisons of vessel simulation benchmark measurements in the overall fluence uncertainty evaluation. Two vessel simulation benchmarks are evaluated: the Pool Critical Assembly (PCA) and VENUS-3 experimental benchmarks.

The PCA experimental benchmark includes [[ ]] activation measurements at the mid-plane elevation in various simulated reactor components. The VENUS-3 experimental benchmark includes [[ ]] activation measurements at a range of elevations in various simulated reactor components. Table 5-1 summarizes the calculated-to-measurement (C/M) results determined for these vessel simulation benchmarks.

Table 5-1
Summary of Comparisons to Vessel Simulation Benchmark Measurements

Benchmark	Number of Measurements	Average Calculated-to-Measured (C/M)	St. Dev. (1σ)
Pool Critical Assembly	[[		
VENUS-3			
Total Simulated Vessel Comparisons			]]

5.2 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.

[[

]]

5.3 Combined Uncertainty

The combined uncertainty for the reactor pressure vessel fluence evaluation is determined with a weighting function [[

]] Table 5-2 shows that the combined uncertainty (1σ) determined for the LaSalle Unit 1 reactor pressure vessel fluence is 9.15% for neutron energy exceeding 1.0 MeV.

It is shown in Table 5-2 that the combined uncertainty is well below the 20% uncertainty limit specified in Regulatory Guide 1.190. In accordance with Regulatory Guide 1.190, there is no discernable bias in the computed RPV fluence. Therefore, no adjustment to the RPV fast neutron fluence for the period corresponding to Cycle 1 through the end of Cycle 18 is required.

Table 5-2
LaSalle Unit 1 RPV Combined Uncertainty for Energy > 1.0 MeV

Uncertainty Term	Value
Combined Uncertainty (1σ)	9.15%
Bias	None(1)

1) The bias term is less than its constituent uncertainty values, concluding that no statistically significant bias exists.

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6
REFERENCES

6.1 References

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2. U.S. Nuclear Regulatory Commission. Office of Nuclear Regulatory Research. *Regulatory Guide 1.190: Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence*. Washington, D.C.: Office of Nuclear Regulatory Research: 2001
3. "Flux Wire Dosimetry Evaluation for LaSalle Nuclear Power Station, Unit 1," General Electric Company, MDE-89-0786, Rev. 0, July 1986.
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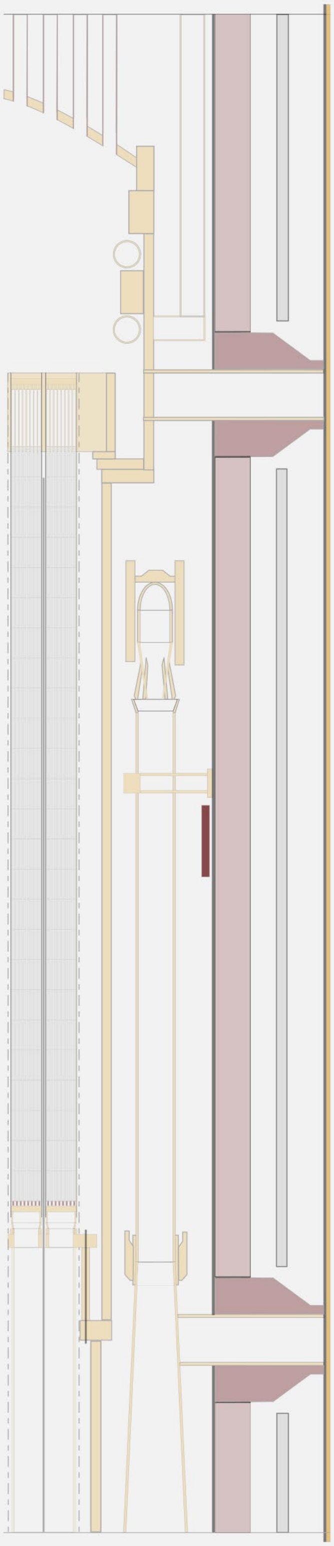
5. "Testing and Evaluation of the LaSalle 120 Degree Surveillance Capsule," MP Machinery & Testing, LLC, MPM-411997. April 2011.
6. BWRVIP-189: BWR Vessel and Internals Project, *Evaluation of RAMA Fluence Methodology Calculation Uncertainty*, EPRI, Palo Alto, CA: 2008. 1016938.



Topical Report

LASALLE COUNTY GENERATING STATION UNIT 1 REACTOR PRESSURE VESSEL FLUENCE EVALUATION - END OF CYCLE 18

Document Number: LAS-FLU-001-R-005
Revision 0
August 2021

Prepared by: **TransWare Enterprises Inc.**

Prepared for: **Exelon Generation Company, LLC**
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Contract Number: 00808371

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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.

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1
INTRODUCTION

This report presents the results of the reactor pressure vessel (RPV) fast neutron fluence evaluation that was performed for the LaSalle County Generating Station Unit 1 (LaSalle Unit 1) reactor. The LaSalle Unit 1 reactor is owned and operated by Exelon Generation Company, LLC (Exelon). The information in this report was generated with the methods described in "LaSalle County Generating Station Unit 1 Fluence Methodology Report" [1].

In compliance with Regulatory Guide 1.190 [2], TransWare Enterprises Inc. (TransWare) has benchmarked the RAMA Fluence Methodology against industry standard benchmarks and plant-specific dosimetry measurements for boiling water reactors and pressurized water reactors. The results of the benchmarking show that the fluence methodology implemented by TransWare predicts specimen activities with no discernable bias in the computed fluence. The combined uncertainty for the LaSalle Unit 1 reactor pressure vessel is determined to be 9.15%. Based upon these results, there is no discernable bias in the computed reactor pressure vessel fluence for the period of Cycle 1 through the end of Cycle 18 for the LaSalle Unit 1 reactor. The details of the uncertainty analysis can be found in Attachment 1 to the "LaSalle Unit 1 Power Plant Methodology Report" [3].

The fluence provided in this report for the LaSalle Unit 1 reactor pressure vessel and nozzles supersedes the fluence provided in the TransWare Report EXL-LSA-001-R-001 [4].

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2
SUMMARY OF RESULTS

This section provides a summary of the fast neutron fluence determined for the LaSalle Unit 1 RPV. Details of the RPV fluence evaluation are presented in Section 3, *Reactor Pressure Vessel Fast Neutron Fluence*.

LaSalle Unit 1 is a BWR/5 class reactor with a core loading of 764 fuel assemblies. The fluence evaluation for this reactor is based on historical and projected operating data through Cycle 18 (28.3 EFPY). Fluence evaluations are also performed at 32 EFPY and 54 EFPY.

Table 2-1 presents a summary of the maximum fast neutron fluence determined for the RPV shell plates, welds and nozzles at EOC 18 (28.3 EFPY), 32 EFPY, and 54 EFPY. The significant fluence for the evaluated areas of the RPV occur at the inside surface of the RPV base metal, which is denoted as the 0T depth in the pressure vessel wall. The N2 nozzles remain under the fluence threshold of 1.0E+17 n/cm^2. All other evaluated areas of the RPV have exceeded the fluence threshold of 1.0E+17 n/cm^2. It is shown in Table 2-1 that the maximum fluence is determined to occur at the 0T location of the Shell Ring 2 with a value of 9.78E+18 n/cm^2 at 54 EFPY. Note in Table 2-1 that all fluence that has exceeded the fluence threshold of 1.0E+17 n/cm^2 are shown in red font and that the maximum fluences in the RPV are additionally shown in **bold** font.

Table 2-2 shows the axial span of the RPV beltline region that was determined for LaSalle Unit 1 at EOC 18 (28.3 EFPY) , 32 EFPY, and 54 EFPY. The reactor beltline region is defined in Appendices G [5] and H [6] of 10CFR50 to include those regions that directly surround 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. This definition of the RPV beltline is considered to include all materials that exceed a fast neutron fluence of 1.0E+17 n/cm^2. At 54 EFPY the RPV beltline covers 416.86 cm, or approximately 13.7 ft of the reactor vessel. The scope of the fluence model was developed to provide an evaluation of the reactor pressure vessel over the full height of the RPV extended beltline region.

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

Component	Maximum Fast Neutron Fluence (n/cm^2)		
	EOC 18 (28.3 EFPY)	32 EFPY	54 EFPY
RPV Beltline Welds			
AB	2.05E+17	2.23E+17	3.44E+17
AC	4.60E+17	5.17E+17	8.42E+17
BA	1.72E+17	1.90E+17	3.05E+17
BB	1.72E+17	1.91E+17	3.08E+17
BC	1.57E+17	1.71E+17	2.59E+17
BD	4.17E+17	4.61E+17	7.19E+17
BE	3.28E+17	3.66E+17	5.87E+17
BF	**5.29E+17**	**5.87E+17**	**9.27E+17**
BG	4.46E+17	5.01E+17	8.20E+17
BH	3.48E+17	3.90E+17	6.36E+17
BJ	2.49E+17	2.82E+17	4.69E+17
Nozzle Forging-to-Base-Metal Welds			
Nozzle Weld N2	1.02E+16	1.12E+16	1.81E+16
Nozzle Weld N6	**2.17E+17**	**2.46E+17**	**4.17E+17**
Nozzle Weld N12	1.43E+17	1.64E+17	2.85E+17
Shell Plates			
Shell Ring 1	2.05E+17	2.23E+17	3.44E+17
Shell Ring 2	**5.64E+17**	**6.25E+17**	**9.78E+17**
Shell Ring 3	4.60E+17	5.17E+17	8.42E+17

Table 2-2
RPV Beltline Elevation Range for LaSalle Unit 1

Reactor Lifetime	Lower Elevation [in (cm)]	Upper Elevation [in (cm)]	Axial Span of the RPV Beltline [in (cm)]
EOC 18 (28.3 EFPY)	216.49 (549.87)	368.21 (935.26)	151.72 (385.39)
32 EFPY	215.42 (547.16)	369.48 (938.49)	154.06 (391.33)
54 EFPY	210.31 (534.19)	374.43 (951.05)	164.12 (416.86)

Section 3 provides detailed results for the RPV fast neutron fluence evaluation. RPV damage fluence is reported at the 0T, 1/4T, and 3/4T depths of the RPV wall for each horizontal (circumferential) weld, vertical (axial) weld, shell plate, and nozzle in the RPV beltline. Figure 3-1 illustrates the location of the welds, shell plates, and nozzles in the RPV. Fluence damage through the thickness of the RPV wall is determined using the displacements-per-atom (dpa) attenuation method prescribed in Regulatory Guide 1.99 [7].

Attachment 1 to the "LaSalle Unit 1 Power Plant Methodology Report" [3] presents the results for the calculated-to-measurement (C/M) activities determined for the surveillance capsule and flux wire dosimetry removed LaSalle Unit 1. The total average C/M for LaSalle Unit 1 is determined to be 0.99 with a standard deviation of ± 0.10.

3

REACTOR PRESSURE VESSEL FAST NEUTRON FLUENCE

This section presents the predicted best-estimate fast neutron fluence (energy > 1.0 MeV) for the LaSalle Unit 1 reactor pressure vessel (RPV) at EOC 18 (28.3 EFPY) , 32 EFPY, and 54 EFPY. It is reported in Reference 3 that the calculated fluence does not require a bias adjustment; therefore, the calculated fluence is the best-estimate fluence for LaSalle Unit 1.

The reactor pressure vessel fast neutron fluence is determined at the interface of the RPV base metal and cladding, which is denoted as the 0T location of the RPV wall. Damage fluence through the RPV wall is reported at the 1/4T and 3/4T depths in the wall. These values are determined based on the minimal RPV wall thicknesses of 18.0975 cm (7.125 in) and 15.5575 cm (6.125 in).

The fast neutron fluence that is used in material embrittlement evaluations should be determined using an appropriate damage function (such as displacements-per-atom of iron) rather than the computed fast neutron fluence obtained from transport calculations. Two acceptable methods for estimating the damage fluence are prescribed in Regulatory Guide 1.99 [7]. One method is based on a generic fluence attenuation formulation and the other is a plant-specific fluence attenuation.

The generic fluence attenuation formulation is used when computational fluence methods do not support plant-specific damage functions. The equation for generic attenuation is given in Regulatory Guide 1.99 as:

$$f_x = f_{surf} * e^{-0.24*x}$$

where x is the depth in the RPV wall, given in inches, and f_{surf} is the fast neutron fluence at the vessel wetted, 0T surface.

Plant-specific material damage assessments may be used to obtain a more accurate estimate of the damage fluence throughout the RPV wall in accordance with the following equation:

$$f_x = f_{surf} * \frac{dpa_x}{dpa_{surf}}$$

where dpa_x is the damage expressed as displacements-per-atom of iron (dpa) at depth x in the RPV wall, and dpa_{surf} is the damage at vessel 0T. In these evaluations, 0T represents the inner surface of the RPV base metal. It has been demonstrated that the generic attenuation approach can become increasingly non-conservative at increasing axial distances from the reactor core mid-plane [8] elevation; therefore, plant-specific damage assessments are recommended for use in RPV material embrittlement evaluations.

Plant-specific fast neutron damage fluence for LaSalle Unit 1 is presented in this report for the RPV horizontal (circumferential) welds, vertical welds, shell plates, and nozzles that reside in the

RPV beltline region. The location and naming convention of the shell plates, welds, and nozzles are shown in Figure 3-1. Also shown in Figure 3-1 is the calculated RPV beltline region for the LaSalle Unit 1 reactor at 54 EFPY.

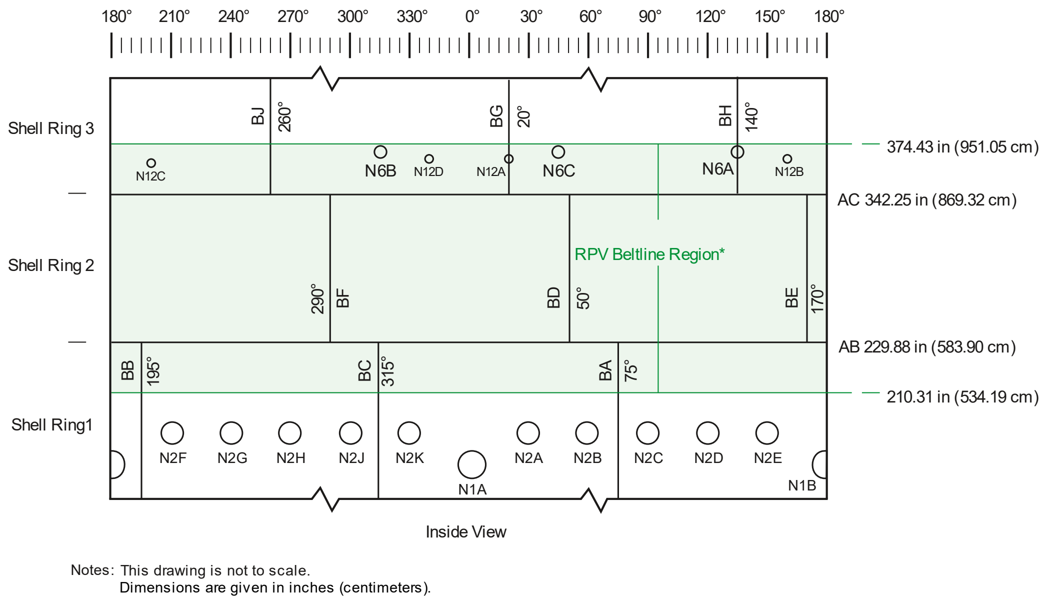

Figure 3-1
LaSalle Unit 1 RPV Beltline Region at 54 EFPY

Table 3-1 through Table 3-7 report the maximum fast neutron damage fluence that is determined for each of the RPV welds, shell plates, and nozzles residing in the RPV beltline. Damage fluence is reported at the 0T, 1/4T, and 3/4T depths in the RPV wall for the reporting periods of interest. In all tables, the damage fluence that exceeds the threshold fluence of 1.0E+17 n/cm^2 is shown in red font. The maximum damage fluence determined for the welds, shells and nozzles is also shown in **bold** font. It is observed in the tables that all evaluated components other than the N2 Nozzles will have exceeded the fluence threshold of 1.0E+17 n/cm^2 before 54 EFPY.

Table 3-1 through Table 3-3 report the maximum damage fluence that is determined for the RPV horizontal (circumferential) and vertical welds at EOC 18 (28.3 EFPY), 32 EFPY, and 54 EFPY, respectively. The maximum damage fluence for each weld is determined to occur at the 0T depth, with the maximum fluence occurring in horizontal weld BF with a value of 9.27E+17 n/cm^2 at 54 EFPY.

Table 3-4 reports the maximum damage fluence that is determined for each RPV shell plate at EOC 18 (28.3 EFPY), 32 EFPY, and 54 EFPY. The maximum damage fluence for each shell plate is determined to occur at the 0T depth, with the maximum fluence occurring in Shell Ring 2 with a value of 9.78E+17 n/cm^2 at 54 EFPY.

Table 3-5 through Table 3-7 report the maximum damage fluence that is determined for the RPV nozzles at EOC 18 (28.3 EFPY), 32 EFPY, and 54 EFPY, respectively. Damage fluence for the nozzles is presented along two paths: one along the nozzle forging-to-base-metal weld and the other along an extraction path that extends from the inside corner of the forging to the outside surface of the RPV wall. Both paths are illustrated for the N2 and N6 nozzles in Figure 3-2 and

for the N12 nozzles in Figure 3-3. It is noted that the extraction path is evaluated around the full circumference of the nozzle forging to determine the maximum fluence. The maximum damage fluence for the beltline nozzles is determined to occur at the 0T depth in the metal weld of the N6 nozzle with a value of 4.17E+17 n/cm^2 at 54 EFPY.

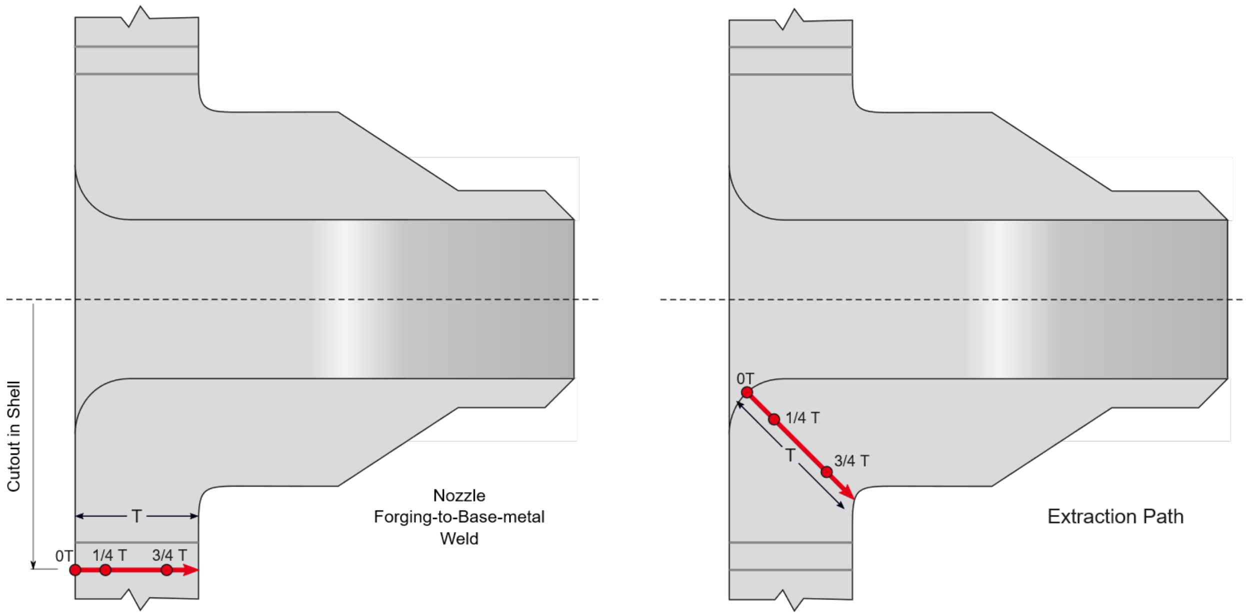

Figure 3-2
Nozzle Fluence Edit Locations for N2 and N6 Nozzles

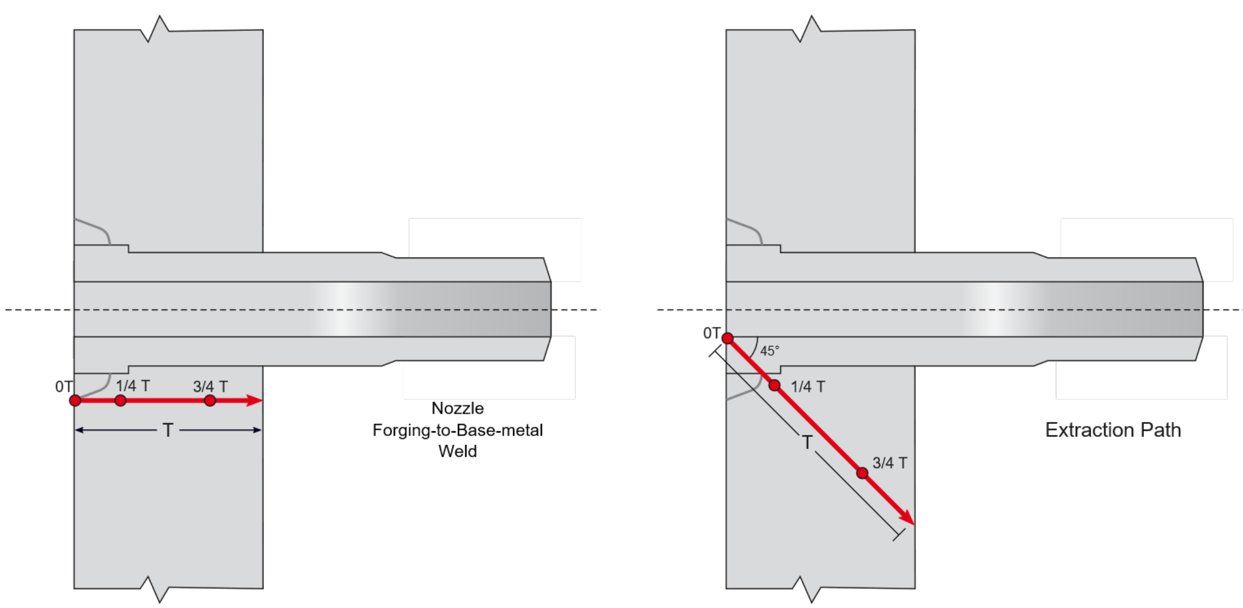

Figure 3-3
Nozzle Fluence Edit Locations for N12 Nozzles

Table 3-8 reports the elevations that define the RPV beltline at EOC 18 (28.3 EFPY), 32 EFPY, and 54 EFPY. It is shown that the RPV beltline at 54 EFPY covers 416.86 cm, or approximately 13.8 ft of the reactor vessel.

Table 3-1
Maximum Fast Neutron Damage Fluence for LaSalle Unit 1 RPV Beltline Welds at EOC 18 (28.3 EFPY)

Weld	Azimuth(1)	Elevation [in (cm)](1)	Fast Neutron Fluence (n/cm^2)		
			0T	1/4T	3/4T
Horizontal Welds					
AB	62°	229.9 (583.9)	2.05E+17	1.42E+17	6.21E+16
AC	68°	342.2 (869.3)	**4.60E+17**	3.17E+17	1.32E+17
Shell Ring 1 Vertical Welds					
BA	75.0°	229.9 (583.9)	**1.72E+17**	1.20E+17	5.23E+16
BB	195.0°	229.9 (583.9)	1.72E+17	1.20E+17	5.29E+16
BC	315.0°	229.9 (583.9)	1.57E+17	1.10E+17	4.90E+16
Shell Ring 2 Vertical Welds					
BD	50°	317.4 (806.2)	4.17E+17	2.91E+17	1.26E+17
BE	170°	311.4 (791.0)	3.28E+17	2.30E+17	1.01E+17
BF	290°	317.4 (806.2)	**5.29E+17**	3.67E+17	1.55E+17
Shell Ring 3 Vertical Welds					
BG	20°	342.3 (869.3)	**4.46E+17**	3.07E+17	1.28E+17
BH	140°	342.3 (869.3)	3.48E+17	2.41E+17	1.03E+17
BJ	260°	342.3 (869.3)	2.49E+17	1.74E+17	7.59E+16

(1) Azimuth and elevation values are listed for the 0T location only.

Table 3-2
Maximum Fast Neutron Damage Fluence for LaSalle Unit 1 RPV Beltline Welds at 32 EFPY

Weld	Azimuth(1)	Elevation [in (cm)](1)	Fast Neutron Fluence (n/cm^2)		
			0T	1/4T	3/4T
Horizontal Welds					
AB	62°	229.9 (583.9)	2.23E+17	1.55E+17	6.78E+16
AC	68°	342.2 (869.3)	**5.17E+17**	3.55E+17	1.48E+17
Shell Ring 1 Vertical Welds					
BA	75.0°	229.9 (583.9)	1.90E+17	1.32E+17	5.78E+16
BB	195.0°	229.9 (583.9)	**1.91E+17**	1.33E+17	5.85E+16
BC	315.0°	229.9 (583.9)	1.71E+17	1.19E+17	5.34E+16
Shell Ring 2 Vertical Welds					
BD	50°	317.4 (806.2)	4.61E+17	3.22E+17	1.39E+17
BE	170°	311.4 (791.0)	3.66E+17	2.57E+17	1.13E+17
BF	290°	317.4 (806.2)	**5.87E+17**	4.07E+17	1.72E+17
Shell Ring 3 Vertical Welds					
BG	20°	342.3 (869.3)	**5.01E+17**	3.45E+17	1.43E+17
BH	140°	342.3 (869.3)	3.90E+17	2.70E+17	1.16E+17
BJ	260°	342.3 (869.3)	2.82E+17	1.97E+17	8.57E+16

(1) Azimuth and elevation values are listed for the 0T location only.

Table 3-3
Maximum Fast Neutron Fluence for LaSalle Unit 1 RPV Beltline Welds at 54 EFPY

Weld	Azimuth(1)	Elevation [in (cm)](1)	Fast Neutron Fluence (n/cm^2)		
			0T	1/4T	3/4T
Horizontal Welds					
AB	62°	229.9 (583.9)	3.44E+17	2.39E+17	1.05E+17
AC	22°	342.2 (869.3)	**8.42E+17**	5.79E+17	2.41E+17
Shell Ring 1 Vertical Welds					
BA	75.0°	229.9 (583.9)	3.05E+17	2.12E+17	9.26E+16
BB	195.0°	229.9 (583.9)	**3.08E+17**	2.14E+17	9.44E+16
BC	315.0°	229.9 (583.9)	2.59E+17	1.81E+17	8.11E+16
Shell Ring 2 Vertical Welds					
BD	50°	317.4 (806.2)	7.19E+17	5.02E+17	2.18E+17
BE	170°	311.4 (791.0)	5.87E+17	4.12E+17	1.81E+17
BF	290°	317.4 (806.2)	**9.27E+17**	6.43E+17	2.71E+17
Shell Ring 3 Vertical Welds					
BG	20°	342.3 (869.3)	**8.20E+17**	5.64E+17	2.35E+17
BH	140°	342.3 (869.3)	6.36E+17	4.41E+17	1.88E+17
BJ	260°	342.3 (869.3)	4.69E+17	3.27E+17	1.42E+17

(1) Azimuth and elevation values are listed for the 0T location only.

Table 3-4
Maximum Fast Neutron Damage Fluence for LaSalle Unit 1 RPV Beltline Shell Plates

Shell Plate	Azimuth(1)	Elevation [in (cm)](1)	Fast Neutron Fluence Damage (n/cm^2)		
			0T	1/4T	3/4T
EOC 18 (28.3 EFPY)					
Shell Ring 1	62°	229.9 (583.9)	2.05E+17	1.33E+17	4.89E+16
Shell Ring 2	65°	317.4 (806.2)	**5.64E+17**	3.89E+17	1.62E+17
Shell Ring 3	68°	342.2 (869.3)	4.60E+17	3.17E+17	1.32E+17
32 EFPY					
Shell Ring 1	62°	229.9 (583.9)	2.23E+17	1.46E+17	5.35E+16
Shell Ring 2	65°	317.4 (806.2)	**6.25E+17**	4.30E+17	1.80E+17
Shell Ring 3	68°	342.2 (869.3)	5.17E+17	3.55E+17	1.48E+17
54 EFPY					
Shell Ring 1	62°	229.9 (583.9)	3.44E+17	2.24E+17	8.27E+16
Shell Ring 2	65°	317.4 (806.2)	**9.78E+17**	6.74E+17	2.82E+17
Shell Ring 3	22°	342.2 (869.3)	8.42E+17	5.79E+17	2.41E+17

(1) Azimuth and elevation values are listed for the 0T location only.

Table 3-5
Maximum Fast Neutron Damage Fluence for LaSalle Unit 1 RPV Beltline Nozzles at EOC 18 (28.3 EFPY)

Location	Azimuth	Elevation [in (cm)][(1)]	Fast Neutron Fluence at (n/cm^2)		
			0T	1/4T	3/4T
N2 Nozzles					
Weld	30°	181.0 (459.7)	1.01E+16	8.47E+15	5.83E+15
Extraction Path			1.73E+15	1.75E+15	2.93E+15
Weld	60°		**1.02E+16**	8.55E+15	5.85E+15
Extraction Path			1.74E+15	1.76E+15	2.93E+15
Weld	90°		5.30E+15	4.66E+15	3.74E+15
Extraction Path			9.78E+15	1.08E+15	2.07E+15
N6 Nozzle					
Weld	45°	372.5 (946.2)	**2.17E+17**	1.58E+17	7.20E+16
Extraction Path			7.01E+16	6.15E+16	4.71E+16
N12 Nozzles					
Weld	20°	366.0 (929.6)	**1.43E+17**	1.03E+17	4.72E+16
Extraction Path			1.32E+17	1.02E+17	6.35E+16

(1) Elevation values correspond to each nozzle centerline elevation.

Table 3-6
Maximum Fast Neutron Damage Fluence for LaSalle Unit 1 RPV Beltline Nozzles at 32 EFPY

Location	Azimuth	Elevation [in (cm)][(1)]	Fast Neutron Fluence at (n/cm^2)		
			0T	1/4T	3/4T
N2 Nozzles					
Weld	30°	181.0 (459.7)	1.11E+16	9.33E+15	6.42E+15
Extraction Path			1.90E+15	1.93E+15	3.23E+15
Weld	60°		**1.12E+16**	9.40E+15	6.43E+15
Extraction Path			1.92E+15	1.94E+15	3.22E+15
Weld	90°		5.98E+15	5.23E+15	4.17E+15
Extraction Path			1.10E+15	1.21E+15	2.31E+15
N6 Nozzle					
Weld	45°	372.5 (946.2)	**2.46E+17**	1.79E+17	8.15E+16
Extraction Path			8.05E+16	7.05E+16	5.37E+16
N12 Nozzles					
Weld	20°	366.0 (929.6)	**1.64E+17**	1.18E+17	5.38E+16
Extraction Path			1.52E+17	1.16E+17	7.23E+16

(1) Elevation values correspond to each nozzle centerline elevation.

Table 3-7
Maximum Fast Neutron Damage Fluence for LaSalle Unit 1 RPV Beltline Nozzles at 54 EFPY

Location	Azimuth	Elevation [in (cm)][(1)]	Fast Neutron Fluence at (n/cm^2)		
			0T	1/4T	3/4T
N2 Nozzles					
Weld	30°	181.0 (459.7)	**1.81E+16**	1.51E+16	1.03E+16
Extraction Path			3.11E+15	3.14E+15	5.18E+15
Weld	60°		1.81E+16	1.51E+16	1.02E+16
Extraction Path			3.10E+15	3.12E+15	5.12E+15
Weld	90°		1.03E+16	8.91E+16	6.90E+15
Extraction Path			1.90E+15	2.05E+15	3.81E+15
N6 Nozzle					
Weld	45°	372.5 (946.2)	**4.17E+17**	3.01E+17	1.36E+17
Extraction Path			1.40E+17	1.23E+17	9.17E+16
N12 Nozzles					
Weld	20°	366.0 (929.6)	**2.85E+17**	2.05E+17	9.25E+16
Extraction Path			2.64E+17	2.01E+17	1.23E+17

(1) Elevation values correspond to each nozzle centerline elevation.

Table 3-8
Reactor Beltline Elevation Range for LaSalle Unit 1

Reactor Lifetime	Lower Elevation [in (cm)]	Upper Elevation [in (cm)]	Axial Span of the RPV Beltline [in (cm)]
EOC 18 (28.3 EFPY)	216.49 (549.87)	368.21 (935.26)	151.72 (385.39)
32 EFPY	215.44 (547.22)	369.49 (938.51)	154.05 (391.29)
54 EFPY	210.31 (534.19)	374.43 (951.05)	164.12 (416.86)

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REFERENCES

4.1 References

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4.2 Glossary

AZIMUTHAL QUADRANT SYMMETRY – A type of core and pressure vessel azimuthal representation that represents a single quadrant of the reactor that can be rotated and mirrored to represent the entire 360-degree geometry. For example, 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.

BOC – An acronym for beginning-of-cycle.

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 produce the best-estimate neutron fluence.

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

DAMAGE FLUENCE – See Neutron Fluence.

DPA – An acronym for displacements per atom which is typically used to characterize material damage in ferritic steels due to neutron exposure.

EFFECTIVE FULL POWER YEARS (EFPY) – A unit of measurement representing one full year of operation at the reactor's rated power level. For example, if a reactor operates for 12 months at full rated power, this represents 1.0 EFPY. 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 also represents 1.0 EFPY.

EOC – An acronym for end-of-cycle.

EXTENDED BELTLINE REGION – See RPV beltline.

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

NEUTRON FLUENCE – Time-integrated neutron flux reported in units of n/cm^2. The term "best-estimate" fluence refers to the fast neutron fluence that is computed in accordance with the requirements of U. S. Nuclear Regulatory Commission Regulatory Guide 1.190. The term "damage fluence", which is required for material embrittlement evaluations, refers to an adjusted fast neutron fluence that is determined using damage functions specified in U. S. Nuclear Regulatory Commission Regulatory Guide 1.99.

NOZZLE EXTRACTION PATH – The path, or trajectory through the nozzle blend radius along which fluence is determined for the nozzle.

NOZZLE FORGING-TO-BASE-METAL WELD – The weld between the nozzle forging and the RPV base metal materials. This is sometimes referred to as the Nozzle-to-Shell Weld

OEM – An acronym for Original Equipment Manufacturer.

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

RPV BELTLINE – The RPV beltline is defined as that portion of the RPV adjacent to the reactor core that attains sufficient neutron radiation damage that the integrity of the pressure vessel could be compromised. For purposes of this evaluation, the fast neutron fluence threshold used to define the traditional RPV beltline is 1.0E+17 n/cm^2. The axial span of the RPV that can exceed this threshold includes the RPV shells, welds, and heat-affected zones. An "extended beltline" is also defined to include lower fluence regions of the pressure vessel but with higher stresses than the traditional beltline region, such as RPV nozzles. The combination of fluence and stress may result in a limiting location in the pressure vessel for determining pressure-temperature limits.

RPV ZERO ELEVATION – The RPV zero elevation is defined at the inside surface of the lowest point in the vessel bottom head, which is typically the bottom drain plug location. Axial elevations presented in this report are relative to RPV zero.

RVI – An acronym for reactor vessel internals.