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Civilian Radioactive Waste Management System
Management & Operating Contractor

Summary Report of Commercial Reactor Critical Analyses Performed for the Disposal Criticality Analysis Methodology

Revision 01

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TABLE OF CONTENTS

LIST OF FIGURES

Page

LIST OF TABLES

Page

1.0 INTRODUCTION

The "Summary Report of Commercial Reactor Critical Analyses Performed for the Disposal Criticality Analysis Methodology" contains a summary of the commercial reactor critical (CRC) analyses used to support the validation of the criticality models for spent commercial light water reactor (LWR) fuel.

1.1 Background

The United States Department of Energy (DOE) Office of Civilian Radioactive Waste Management (OCRWM) is developing a methodology for criticality analysis to support disposal of commercial spent nuclear fuel in a geologic repository. A topical report on the disposal criticality analysis methodology is scheduled to be submitted to the United States Nuclear Regulatory Commission (NRC) for formal review in 1998. This technical report in one of a series of reports that provides a summary of the analyses results that support the development of the disposal criticality analysis methodology.

1.2 Objective

The objective of this report is to present a summary of the CRC analyses results. Results from the CRC evaluations will support the development and validation of the neutronics models used for criticality analyses involving commercial spent nuclear fuel (SNF). These models and their validation will be discussed in the Disposal Criticality Analysis Methodology Topical Report.

1.3 Scope

The scope of this Summary Report is the CRC analyses results for the following reactors:

Pressurized Water Reactor (PWR):	Crystal River Unit 3 (CR3), McGuire Unit 1 (MCG1), Three Mile Island Unit 1 (TMI1), Sequoyah Unit 2 (SEQ2)
Boiling Water Reactor (BWR):	Quad Cities Unit 1 (QC1), Quad Cities Unit 2 (QC2)

Future revisions of this report may include CRC analyses results for the following reactors:

PWRs:
Davis Besse Unit 1 (DB1)*,
Catawba Unit 1 (CAT1)*,
Catawba Unit 2 (CAT2)*, and
Calvert Cliffs Unit 1 (CC1)*.

BWRs:
LaSalle County Unit 1 (LSC1)*,
Dresden Unit 2 (D2)*,
Washington Nuclear Plant Unit 2 (WNP2)**, and
Cooper**.

* Data is available/acquired from these reactors, but have not been analyzed
** Reactors being considered as a source of CRC data, but no data have been acquired

1.4 Quality Assurance

The Civilian Radioactive Waste Management System (CRWMS) Management and Operating Contractor (M&O) Quality Assurance (QA) program applies to the development of this report. The results provided in this report will be used to develop the methodology for evaluating the Monitored Geologic Repository (MGR) waste package and engineered barrier segment. The waste package and engineered barrier segment have been identified as items important to safety, waste isolation, and physical protection of materials in the QAP-2-3 evaluation entitled "Classification of the Preliminary MGDS (Mined Geologic Disposal System) Repository Design" (Reference 1, TBV-228). The Waste Package Operations Department (WPOD) responsible manager has evaluated the technical document development activity in accordance with QAP-2-0, "Conduct of Activities." The QAP-2-0 activity evaluation, "Develop Technical Documents" (Reference 2), has determined that the preparation and review of this technical document is subject to "Quality Assurance Requirements and Description" (QARD, Reference 3) requirements. As specified in NLP-3-18, "Documentation of QA Controls on Drawings, Specifications, Design Analyses, and Technical Documents," this activity is subject to QA controls.

The results provided in this report are drawn from various analyses developed under the CRWMS M&O QA program. The data used in these analyses is drawn from various reports, calculations, and drawings developed under the NRC accepted quality assurance program of Framatome Cogema Fuels, and the data has supported prior licensing submittals. The results, therefore, will be considered acceptable for quality affecting activities and for use in analyses affecting procurement, construction, or fabrication.

No computer software subject to the requirements of the QARD was used in the development of this report. The results reported in this document were drawn from various analyses which did use software subject to the requirements of the QARD. The details of the computer software approved for quality affecting work used to generate the results contained herein are provided in the various analyses referenced by this report.

2.0 ANALYSIS MODEL

This section provides a description of the models used in generating the supporting analysis results reported in this document. There are three types of CRC analyses: 1) Beginning-of-Life (BOL) cases with all fresh fuel, 2) Beginning-of-Cycle (BOC) cases with part fresh fuel and part irradiated fuel, and 3) Middle-of-Cycle (MOC) or restart cases with all irradiated fuel. The cases with all fresh fuel only require a criticality model. The cases with irradiated fuel also require an isotopics model. The criticality and isotopics models are collectively referred to as the neutronics models.

2.1 Isotopics Model

The isotopics model was used to determine material isotopic concentrations of the irradiated commercial LWR fuel in the CRC analyses. The CRC analyses will be used in developing and validating the isotopics model for use in determining commercial SNF fissionable and absorbing isotope concentrations for disposal criticality evaluations.

The code system used in the isotopics model is the SAS2H sequence of the SCALE 4.3 computer code system (Reference 5). SAS2H is the control module for the analytical sequence. The functional modules (or codes) within the sequence are BONAMI, NITAWL-II, XSDRNPM, COUPLE, and ORIGEN-S. SAS2H converts user input data into the forms required by the functional modules. BONAMI and NITAWL-II perform problem-dependent resonance self-shielding corrections of neutron cross sections. XSDRNPM is a one-dimensional discrete-ordinates code that produces a weighted cross section library and spectra data. This data is used by COUPLE to update an ORIGEN-S data library. ORIGEN-S is a point-depletion/decay code that computes the time-dependent isotopic concentrations using the matrix exponential expansion technique. For short-lived nuclides a form of the Bateman equation is used to ensure better accuracy. ORIGEN-S computes the isotopic concentrations (actinides and fission products) for all required conditions. This includes both power operation and shutdown intervals while the fuel is in the reactor. ORIGEN-S is also used in calculating radioactive decay and daughter isotope buildup after the fuel is withdrawn from the reactor core. Since ORIGEN-S is a point model, spatial and spectral effects are not explicitly modeled. However, spatial and spectral effects are incorporated in the model through the one-dimensional spatial and the 44-energy group spectral weighting of cross section data by XSDRNPM.

A fuel assembly is modeled with SAS2H in one-dimensional (1-D) cylindrical geometry. This modeling is a two step process. First, the fuel is represented as an infinite lattice of fuel rods with XSDRNPM, where resonance self-shielding effects are treated by BONAMI and NITAWL-II. Second, cell-spectrum-weighted cross sections from XSDRNPM are then applied to the fuel zone in a larger cell model representing part or all of a fuel assembly within an infinite lattice. Material and volume ratios for the zones must be appropriate for the physical system being represented. Weighted cross section and spectra data from this model are used by COUPLE to update the ORIGEN-S data library. ORIGEN-S performs point depletion calculations to provide updated isotopic concentrations that are fed back to the one-dimensional model. The first step is then repeated and new weighted cross section and spectra data are determined for the next depletion calculation. Updating of the ORIGEN-S library for depletion time steps is performed

to appropriately represent changes (with depletion) in the neutron energy spectrum within the fuel assembly. Once all of the ORIGEN-S data libraries are developed, a final ORIGEN-S case is run using all of the developed data libraries to determine the final depleted isotopic concentrations. A more detailed discussion of the SAS2H modeling for PWR and BWR fuel is presented below.

2.1.1 SAS2H

The description of the SAS2H control module and the associated calculational modules presented in this report is a summary of the discussion presented in the SCALE 4.3 user's manual (Reference 6).

In the CRC analysis procedure, the SAS2H control module of the SCALE modular code system is used to perform a sequence of fuel depletion and decay calculations required to obtain spent fuel isotopic compositions. The objectives of SAS2H with respect to CRC analyses include the following:

1) predict spent fuel characteristics for spent fuel assemblies having a specified reactor history;
2) apply standard analytical models that represent the physics of the system being analyzed (within the 1-D transport limits of the problem);
3) automate the use of known methods of calculating some of the input parameters and the selection of appropriate control options for the various codes applied in the analysis.

SAS2H calculates the time-dependent fuel composition of a reactor fuel assembly by sequencing five independently tasked calculational modules of the SCALE code system. The sequence implemented by SAS2H results in an iterative burnup calculational procedure in which two separate lattice-cell calculations are performed for each iteration to determine the appropriate neutron spectrum and associated nuclide cross sections. At specified times during the burnup, the cross sections are updated using resonance processing codes and 1-D transport analyses. These updated cross sections are used in point depletion calculations that produce the time-dependent fuel compositions required for the next cross section update. This procedure creates a number of weighted working cross section libraries that are each applicable to a specific burnup duration over the irradiation history of the assembly. A final ORIGEN-S depletion and decay calculation is performed over the entire irradiation history of the assembly using the various weighted working cross section libraries as appropriate.

The 1-D transport analyses required to obtain the relevant neutron flux distribution necessary for cross section weighting is performed on two separate lattice-cells to approximate two-dimensional (2-D) effects that are introduced in fuel assemblies containing guide tubes, burnable poison rods (BPRs), Gadolinia bearing rods, control rods, control blades, and axial power shaping rods (APSRs). A general description of the two-part neutron flux distribution calculation is provided in Section 2.1.4. The axial variation in the neutron flux is not considered in the SAS2H sequence. A multi-dimensional representation of the fuel assembly must be incorporated to account for axial variations in the neutron flux. To account for axial variations in the neutron flux, the CRC analyses incorporate a multiple axial node model (e.g., 16 or 18 axial

nodes for PWRs and 10 or 24 axial nodes from BWRs) in which a separate SAS2H sequence calculation is performed for each axial node of each assembly in the reactor core, taking advantage of core symmetry where appropriate.

The SAS2H control module reads a set of well defined user input. The user input is converted by the SAS2H control module into the data required by each calculational module to perform the necessary calculations relevant to fuel depletion and decay. The basic user input required by the SAS2H control sequence includes the following:

1) the material zone dimensions of the fuel-pin cell and the larger unit cell representation of the fuel assembly used to incorporate 2-D effects;
2) the material densities of the fresh fuel assembly;
3) the material temperatures;
4) the specific power, exposure time and shutdown time of the fuel assembly in each appropriate cycle of the reactor history;
5) the various control parameters used to select libraries, the optional parameters preferred over the defaults for each calculational module, the level of output printout, and the modifications to the transport computations (e.g., the fineness of mesh intervals or the problem convergence criteria).

The SCALE system driver invokes the execution of the various calculational modules in the SAS2H control sequence and returns control to the SAS2H sequence. The pertinent results obtained from the calculational modules are processed by SAS2H and used to generate the input data for subsequent calculational modules. Passes through the calculational modules are repeated until the case in completed.

2.1.2 Overview of Calculational Modules

The SAS2H control sequence accesses five calculational modules of the SCALE code system for performing fuel depletion and decay calculations. The five calculational modules include BONAMI, NITAWL-II, XSDRNPM, COUPLE, and ORIGEN-S. Each of the calculational modules have a specific purpose in the sequence to perform the fuel depletion and decay calculations.

The BONAMI calculational module applies the Bondarenko method of resonance self-shielding to nuclides for which Bondarenko data is available.

The NITAWL-II calculational module performs Nordheim resonance self-shielding corrections for nuclides that have resonance parameter data available.

The XSDRNPM calculational module performs a 1-D neutron transport calculation on a specified geometry to facilitate production of cell-weighted cross sections for fuel depletion calculations.

The COUPLE calculational module updates all cross section constants included on an ORIGEN-S working nuclear data library with data from the cell-weighted cross section library obtained

from the XSDRNPM calculation. Additionally, the weighting spectrum produced by XSDRNPM is applied to update all nuclides in the ORIGEN-S working library which were not included in the XSDRNPM calculation.

The ORIGEN-S calculational module performs point depletion, buildup, and decay calculations for the specified assembly irradiation history.

2.1.3 General Description of Method

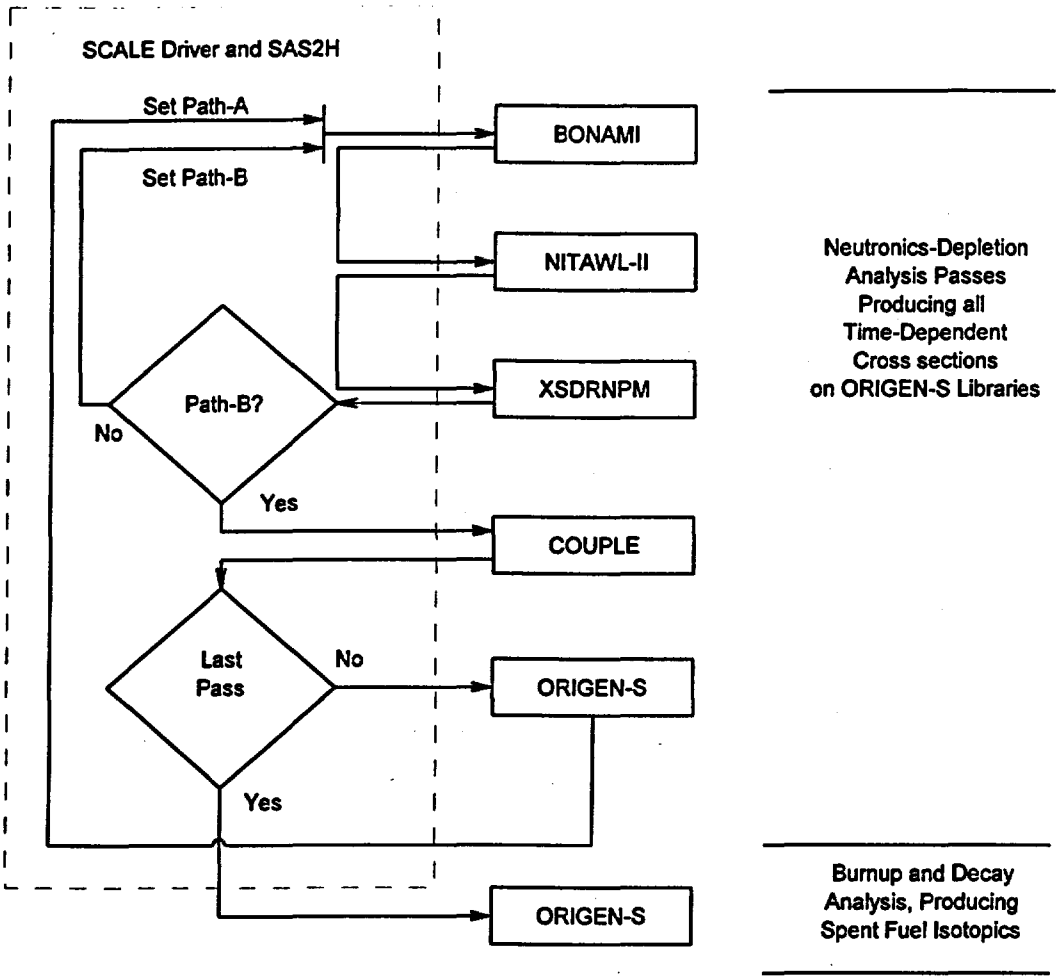
The method applied by SAS2H starts with the data describing a fuel assembly or node of a fuel assembly as it is initially loaded into a particular reactor. The composition, temperatures, geometry, and time-dependent specific power of the fuel assembly are required inputs to SAS2H. For each time-dependent fuel composition, the SAS2H sequence performs a 1-D neutron transport analysis using XSDRNPM and a two-part procedure incorporating two separate lattice-cell models. The first model is a unit fuel-pin cell from which cell-weighted cross sections are obtained. The second model represents a larger unit cell within an infinite lattice. The material zones of the larger unit cell can be structured for different types of light-water reactor fuel assemblies containing water holes, burnable poison rods, Gadolinia bearing fuel rods, etc. Problem-dependent resonance self-shielding of the cross sections is performed using the BONAMI and NITAWL-II calculational modules (Reference 6, p. S2.2.2).

The neutron flux spectrum obtained from the larger unit cell model is used to determine the appropriate nuclide cross sections for the specified burnup-dependent fuel composition. The cross sections derived from a transport analysis at each time step are used in an ORIGEN-S point-depletion calculation that produces the burnup-dependent fuel compositions to be used in the next spectrum calculation. This sequence is repeated over the prescribed operating history of the reactor. Ultimately, the nuclide inventory (actinides, fission products, and light elements) is computed at the burnup corresponding to the discharge of the assembly from the reactor. The buildup and decay of the nuclides in the fuel assembly are computed by ORIGEN-S for the down times relevant to reactor operation. For the CRC analyses, the ORIGEN-S buildup and decay calculations are performed for both the downtimes between cycles of insertion for an assembly and the downtime prior to the CRC statepoint (Reference 6, p. S2.2.2).

2.1.4 Preparation of Fuel Cross Sections

The preparation of fuel cross sections employs a two-part calculational procedure that utilizes two separate 1-D neutron transport calculations performed on different models to account for 2-D variations in fuel assembly design. The calculational flow path invoked by SAS2H is illustrated in Figure 2.1.4-1 (Reference 6, p. S2.2.4).

Figure 2.1.4-1 Calculational Flow Path Invoked by SAS2H



The upper part of Figure 2.1.4-1 shows the flow path for the neutronics-depletion analysis "passes" that create a cross section library at specified burnup intervals. First, in path A, BONAMI, NITAWL-II, and XSDRNPM are invoked to produce the cell-weighted cross sections of the fuel zone. The second return to the driver for path B of the reactor pass invokes all five calculational modules. The compositions for the first reactor pass are simply the nuclide mixtures of the new, or freshly loaded, fuel assembly. After completion of the path A and path B computations, execution continues with COUPLE updating an ORIGEN-S working library with data on the XSDRNPM weighted working library. The ORIGEN-S calculational module is invoked to compute the time-dependent densities of the nuclides in the fuel and burnable poison for the specified power and exposure times. SAS2H is then invoked for the next control function in the sequence.

The path A and path B models both utilize the same calculational modules (BONAMI, NITAWL-II, and XSDRNPM), but the models themselves are quite different. The path A model is simply a one-dimensional representation of an assembly unit cell containing a fuel rod. For BWRs, the explicit fuel in the path A model contains an average enrichment. The fuel rod is modeled explicitly with the square unit cell perimeter converted to a circular perimeter

maintaining the same total unit cell area. A white boundary condition is applied to the perimeter of the path A unit cell to simulate an infinite array of unit cells. The resonance self-shielding calculations and a 1-D neutron transport calculation are then performed on the path A model to obtain the neutron flux and cell-weighted cross sections for the fuel region. The cell-weighted cross sections from the path A model are then applied to the fuel region in the path B model.

The path B model is a larger unit cell representation of all or part of the fuel assembly in an infinite lattice. The path B model is used to calculate an "assembly-averaged" fuel region flux that considers the effects due to the path A model, the assembly lattice locations containing different types of rods or water holes, and the moderator present in assembly-to-assembly spacings. The use of cell-weighted cross sections from the path A model in the 1-D neutron transport calculation of the path B model is an approximate method for simulating 2-D effects present in fuel assemblies containing guide tubes and burnable poison rods. Examples of typical path B larger unit cell models for a PWR assembly containing guide tubes and a BWR assembly containing Gadolinia bearing fuel rods are shown in Figures 2.1.4-2 and 2.1.4-3, respectively.

Figure 2.1.4-2 Typical Path B Model for PWR Assembly Containing Guide Tubes

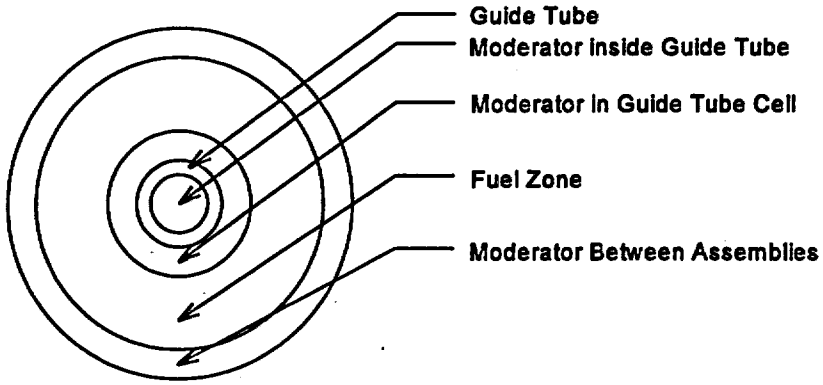
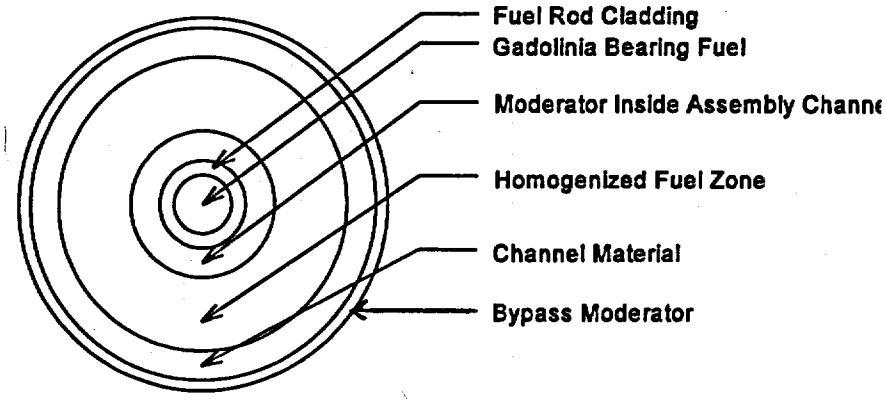


Figure 2.1.4-3 Typical Path B Model for BWR Assembly Containing Gadolinia Bearing Fuel Rods



The essential rule in deriving the zone radii is to maintain the mass ratios in the actual assembly.

The central region of the larger cell can be modeled as an assembly guide tube, a burnable poison rod containing no fuel, an orifice rod, an axial power shaping rod, a fuel rod containing a burnable poison, or almost any other pin-cell type rod. The moderator of the central region in the path B model is the moderator associated with the assembly lattice unit cell modeled in the central region. A homogenized region surrounds the central unit cell region in the path B model. This homogenized region represents the homogenization of a number of fuel pin lattice cells. The radius of the homogenized region is determined such that the mass ratios in the assembly are maintained in the path B model. In the case of the PWR assembly, the region surrounding the homogenized region in the path B model represents materials present between assemblies. In the case of the BWR assembly, the region surrounding the homogenized region in the path B model represents the assembly channel and bypass moderator. Typically, the assembly spacer grids and other hardware are either ignored or homogenized into the assembly moderator material due to their limited effect on the neutron flux and associated energy spectrum.

There are certain approximations made in developing the path B model. Two important approximations relevant to CRC analyses include:

1) the path B model assumes that the non-fuel pincell locations are equidistantly spaced throughout the assembly;
2) the path B model allows placement of an assembly-to-assembly moderator region around the homogenized region even though the path B model may only represent a portion of the entire fuel assembly.

The significance of these approximations appear to be less for the path B larger unit cell than if the additional zones representing the guide tubes or burnable poison rods were placed directly around the fuel rod unit cell (path A model).

The isotopic compositions for the various materials in the path A and path B models are prepared by the Material Information Processor of SCALE using both the user input and the SCALE Standard Composition Library. The user is required to input the compositions of the materials as defined in the fresh fuel assembly. Materials in the SAS2H input are defined by mixture numbers. The first three mixture numbers are reserved for the materials in the path A model. Mixture number one is reserved for the fresh fuel composition. Mixture number two is reserved for the cladding composition. Mixture number three is reserved for the moderator composition. Additional mixture numbers may then be defined to represent other materials (i.e., burnable absorber). The input mixtures define the "input mixing table" which is composed of the nuclide identification number, the mixture number, and the number density of the nuclide in the mixture. The input mixing table is combined with trace amounts (1E-20 atoms/b-cm) of nuclides present in the specified neutron cross section library used in the SAS2H sequence to produce a "master mixing table". Additional trace amounts of selected nuclides, presented in Table 2.1.4-1, are also included to ensure that appropriate resonance processing data is utilized for important nuclides that build up in the fuel during depletion. The user may also define additional trace nuclides in mixture one of the SAS2H input to ensure that appropriate resonance processing data is utilized.

Table 2.1.4-1 Selected Nuclides Automatically Added by SAS2H for Neutronics Processing

Xe-135	Cs-133	U-234	U-235
U-236	U-238	Np-237	Pu-238
Pu-239	Pu-240	Pu-241	Pu-242
Am-241	Am-242m	Am-243	Cm-242
Cm-243	Cm-244	1/v-absorbers	

After the neutronics code interfaces are completed, SAS2H generates interface files for codes that couple burnup-dependent densities into the model for producing time-dependent cross sections. First, an interface data set is produced for COUPLE, which updates cross section constants on libraries input to ORIGEN-S. Finally, an input data set for ORIGEN-S is developed to execute a depletion case in which computed densities of the fuel are saved in a data set at prescribed time intervals.

2.1.5 Time-Dependent Depletion and Decay Calculation

During the irradiation of fuel in a reactor, nuclide densities vary as a function of the neutron flux and its associated energy spectrum. SAS2H performs the neutronics depletion calculations previously described in Section 2.1.4 at a number of user-defined irradiation intervals to account for the variations in neutron flux and its associated energy spectrum that effect the fuel isotopic composition as a function of time. The irradiation intervals defined by the user are called "passes". The SAS2H sequence performs a number of passes prescribed by the user to adequately simulate the reactor history so that the resulting fuel isotopic composition for the assembly or node of the assembly may be determined. Referring to Figure 2.1.5-1, each pass through the procedure involves the following five steps:

1) preparation of new data interfaces by SAS2H;
2) return of control to the SCALE driver for execution of the three codes required in the path A model;
3) preparation of the data interfaces for the path B model by SAS2H;
4) return of control to the SCALE driver for execution of the five codes required in the path B model;
5) return to SAS2H.

The user inputs the number of irradiation steps requested, the number of cross section libraries to make per step, the specific power of the assembly or assembly node at each step, the total operation time of each step, and the downtime following each step. The irradiation steps are determined based on the reactor history and the detail required by the user. A single irradiation step or multiple irradiation steps may be specified to represent each reactor cycle. The irradiation-time associated with each cross section library is derived from the input.

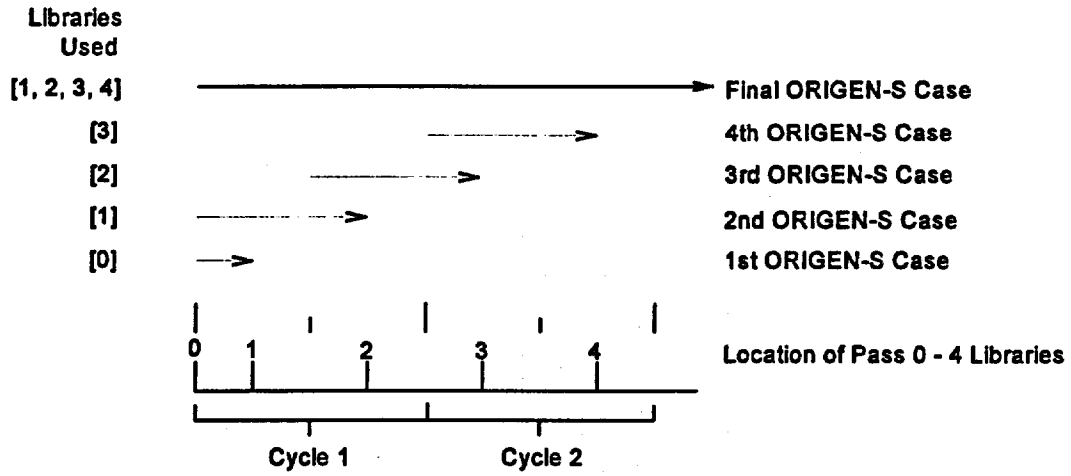
The moderator density does not change from the initial material specification (mixture 3) unless requested by the user. The CRC analyses change the moderator density (and fuel temperature)

for each axial node based on core-follow data. For BWR CRC analyses, the nodal moderator density is changed in a step-wise manner to track changes in the assembly void fraction profiles. In SAS2H, the fixed moderator density in mixture 3 is adjusted by user-defined fractions that follow the BWR void fraction profiles.

The PWR soluble boron fraction in the moderator varies with reactor operation time. To account for this the SAS2H sequence applies a default linear interpolation that specifies a boron concentration in the initial irradiation step which is 1.9 times the boron concentration specified in the mixture 3 material specification. The boron concentration in the final irradiation step is 0.1 times the boron concentration specified in the mixture 3 material specification. As an alternative to the default linear boron letdown scheme the user may specify a fraction of the mixture 3 boron concentration for each irradiation step in the depletion calculation with one cross section library update per irradiation step. If this is done, the boron concentration does not change as a function of time over each irradiation step. The specified boron concentration at each irradiation step should be the average boron concentration over the duration of the irradiation step. The CRC analyses employ the method of specifying boron fractions at each irradiation step so that the actual boron letdown curves for the reactor may be followed more accurately.

With the exception of the fresh fuel library, each cross section library is based on the number densities obtained for the midpoint of the irradiation-time interval. The midpoint number densities for an irradiation interval are computed from an ORIGEN-S case that uses the cross section library from the previous irradiation interval. This procedure is graphically illustrated in Figure 2.1.5-1 for a two-step case where two libraries are requested per step (Ref. 6, p. S2.2.8).

Figure 2.1.5-1 Schematic of Successive ORIGEN-S Cases Used to Produce the Burnup-Dependent Number Densities



In reference to Figure 2.1.5-1, the first step is to produce the "PASS 0" library prepared using the fresh-fuel isotopics. This initial library is used in the first ORIGEN-S case to generate number densities at the midpoint of the first irradiation interval. Next, the SAS2H module:

1) computes density-dependent parameters for the resonance calculations;
2) increments the required data set unit numbers;

3) adds "PASS 1" to the ORIGEN-S library title;
4) updates the ORIGEN-S input for the second case to save number densities for the starting point and the midpoint of the second irradiation interval;
5) rewrites all code interfaces using the new data.

Then the "PASS 1" library is produced by invoking execution of the five codes in both path A and path B a second time with the new input interfaces. Each additional pass applies the same procedure as used for "PASS 1". The midpoint densities are applied to the neutronics analysis to produce a new library. The depletion computation applies this library and the densities calculated for the start of the pass. A decay computation with zero power is applied for reactor downtime, if specified for the end of a cycle, before determining densities for the next pass. All ORIGEN-S libraries are saved, starting with the "PASS 1" library. The last pass is the only one in which there is a major difference in the procedure. After the completion of COUPLE in the final pass, the required libraries have been produced. The final ORIGEN-S case corresponding to the last pass uses all the libraries and runs through the entire assembly irradiation history input to the SAS2H module. All the cross section libraries produced during all previous passes are combined into the final multi-burnup-dependent ORIGEN-S working library made for the case.

The final ORIGEN-S case utilizes all previously determined cross section libraries for each step to deplete the fuel from its fresh fuel state to its discharge state. All ORIGEN-S cases include over 1600 nuclides in the ORIGEN-S library. Cross section constants are either updated directly from previous XSDRNPM output, or for those nuclides not included in the pin-cell analysis, from broad-group flux weighting factors.

In the example shown in Figure 2.1.5-1, the final ORIGEN-S case begins with the nuclide generation and depletion calculation performed using the cross section library, power, and time interval for "PASS 1". Four equal-size time steps are used during the irradiation time, followed by a single downtime interval. If no downtime was specified, a zero-time interval is applied. Next, a similar computation is performed using the compositions determined at the end of the "PASS 1" calculation and the cross section data on the "PASS 2" library. The analysis proceeds with each succeeding library and corresponding assembly power and time interval. Ultimately, the discharge composition of the fuel assembly is determined. Finally, a decay calculation using six equal-size time steps is performed for the final requested downtime.

2.1.6 SAS2H Results Relative to CRC Analyses

The number densities for the nuclides of the spent fuel are all computed by ORIGEN-S. In addition, ORIGEN-S calculates the depletion of most light elements, including the burnable poisons (e.g., boron). However, the densities of alloys or elements in the clad, moderator, or structural materials, and oxygen in the fuel remain constant (Reference 6, p. S2.2.9).

The fuel and burnable poison isotopic compositions are extracted from the final ORIGEN-S output for use in a subsequent MCNP criticality calculation. The fuel and burnable poison compositions are provided from ORIGEN-S in grams per assembly (or axial node of an assembly) and must be converted to material compositions in terms of isotopic weight percentages and overall material density such that the masses of material from ORIGEN-S are

preserved.

2.2 Criticality Model

The criticality model was used to calculate nuclear reactivity of fresh and irradiated commercial LWR fuel in the CRC analyses. The CRC analyses will be used in developing and validating the criticality model for use in determining reactivity for disposal criticality evaluations.

The code system used in the criticality model is the MCNP computer code system (Reference 7). MCNP is a general-purpose Monte Carlo N-Particle code that can be used for neutron, photon, electron, or coupled neutron/photon/electron transport calculations. MCNP also has the capability to calculate the effective neutron multiplication factor (k_{eff}) of systems containing fissionable material. The composition of the SNF materials, obtained from SAS2H (or ORIGEN-S), are used as input in the MCNP calculation. Nuclear interaction data is provided to MCNP in the form of cross section data tables for each isotope or element used in the calculation. The cross section data tables are presented in the ACE format after being processed from evaluations such as ENDF/B-V, ENDF/B-VI, LLNL, LANL:T-2, or LANL:XTM. The neutron cross section data tables are not collapsed into energy groups (as is done for the KENO code). Rather, the interaction data in the cross section tables is presented at a number of energy points on a main energy grid such that linear-linear interpolation within the main energy grid will reproduce the interaction data of the original evaluation to within approximately 1%. The energy range of the main energy grid in a cross section data table generally covers from very low neutron energies through 20 MeV.

2.2.1 MCNP General Description

A full description of the MCNP code is provided in the MCNP user manual (Reference 8). The following excerpts will provide a general description.

> "MCNP is a general-purpose Monte Carlo N-Particle code that can be used for neutron, photon, electron, or coupled neutron/photon/electron transport, including the capability to calculate eigenvalues for critical systems. The code treats arbitrary three-dimensional configurations of materials in geometric cells bounded by first-degree and second-degree surfaces and fourth-degree elliptical tori."
>
> "Pointwise cross-section data are used. For neutrons, all reactions given in a particular cross section evaluation (such as ENDF/B-V) are accounted for. Thermal neutrons are described by both the free gas and S(α,β) models."
>
> "Important standard features that make MCNP very versatile and easy to use include a powerful general source, criticality source, and surface source; both geometry and output tally plotters; a rich collection of variance reduction techniques; a flexible tally structure; and an extensive collection of cross section data" (Reference 8, p. ix).

2.2.2 Monte Carlo Method

The Monte Carlo method is a method of simulating and recording the behavior of individual particles within a system. The behavior of the simulated particles is extrapolated to describe the average behavior of all of the particles within the system. The Monte Carlo method as applied to neutrons in an MCNP criticality calculation is based upon following a number of individual neutrons through their various transport experiences such as scattering, fission, absorption, or leakage. The fission process is regarded as the birth event that separates generations of neutrons. A generation is the lifetime of a neutron from birth by fission to death by either escape, parasitic capture, or absorption leading to fission. The average behavior of the sample set of neutrons is used to describe the average behavior of the system with regard to the number of neutrons in successive generations (i.e., critical multiplication factor, k_{eff}).

2.2.3 MCNP's Critical Multiplication Factor (k_{eff}) Results

MCNP calculates three k_{eff} estimates for each cycle in a given problem:

1) the collision estimate;
2) the absorption estimate;
3) the track length estimate.

A detailed description of the three k_{eff} estimates may be found in Chapter 2, Section VIII, Part B, of the MCNP User Manual (Reference 8). The k_{eff} estimate used in the criticality analyses and in the bias value determination of this study is the statistical combination of all three k_{eff} estimates. According to statisticians at the Los Alamos National Laboratory, "the three-combined k_{eff} estimator is the best final estimate from an MCNP calculation" (Reference 8, p. 2-146). "The confidence interval based on the three statistically combined k_{eff} estimate is the recommended result to use for all final k_{eff} confidence interval quotations because all of the available information has been used in the final result" (Reference 8, p. 2-149).

2.2.4 Assessing the Validity of a Criticality Calculation

Before an MCNP k_{eff} result can be considered acceptable the validity of the calculation must be determined. Two minimum requirements for assessing the validity of an MCNP criticality calculation include:

1) all cells containing fissionable material should be adequately sampled, and
2) the fundamental spatial mode should be achieved before commencing the accumulation of data for calculation of the mean k_{eff}.

MCNP provides several features which help in assessing the validity of a k_{eff} calculation. To satisfy the first requirement, "MCNP verifies that at least one fission source point was generated in each cell containing fissionable material" (Reference 8, p. 2-150).

To satisfy the second requirement, MCNP provides several checks to determine if the fundamental spatial mode was achieved prior to the completion of the I_c cycles (I_c is the number

of source cycles that are skipped before k_{eff} data accumulation begins). One check is the comparison of the estimated three-combined k_{eff} and its standard deviation for the first and second half of the active k_{eff} cycles. If the difference between the average k_{eff} values for the two halves does not appear to be zero or if the ratio of the two standard deviations is larger than expected a "WARNING" message is provided in the output. MCNP determines the number of cycles which must be skipped to produce the minimum standard deviation for the three-combined k_{eff} estimate. If this result is larger than I_c it may be indicative that more cycles should be skipped before accumulating k_{eff} data. MCNP checks each k_{eff} estimate's cycle data to assure normality at the 95% and 99% confidence levels. If a k_{eff} estimate is not normally distributed with respect to the mean k_{eff} at the 99% confidence level a "WARNING" message is provided in the output. "Unless there is a high positive correlation among the three estimates, it is expected to be rare that all three k_{eff} estimates will not appear normally distributed at the 99% confidence level when the normal spatial mode has been achieved and maintained" (Reference 8, p. 2-150). Finally, MCNP tests for a monotonic trend of the three-combined k_{eff} estimate's results over the last ten active cycles. If the spatial mode is well converged and maintained, there should not be a monotonic trend within the last ten active cycles. A "WARNING" message is provided in the output if a monotonic trend is detected.

Compliance with the two minimum requirements addressed above should be verified for each criticality calculation using the checks provided by the MCNP code. If either of the two requirements appear to be violated, the k_{eff} results for the calculation should be evaluated further.

2.2.5 Cross Sections

Utilizing the appropriate material cross sections in an MCNP criticality calculation is essential to obtaining credible results. The cross sections for the various neutron interactions are used to determine the flow of the criticality calculation at each interaction site. The MCNP neutron interaction tables are processed from either the ENDF/B-V, ENDF/B-VI, LLNL, LANL:T-2, or LANL:XTM evaluations.

The following listing is a sample of the type of data provided on the MCNP cross section data tables:

1) all available cross section data;
2) angular distribution data for scattered neutrons;
3) energy distribution data for inelastically scattered neutrons;
4) data about secondary photon production;
5) Q-value data for each reaction;
6) the average number of neutrons per fission data for fissionable isotopes.

"For a particular table, the cross sections for each reaction are given on one energy grid that is sufficiently dense that linear-linear interpolation between points reproduces the evaluated cross sections within a specified tolerance that is generally within 1% or less" of the evaluated data (Reference 8, p. 2-18). A "thinned" neutron interaction table is available for some nuclides. The "thinned" tables have a significantly reduced size with a tolerance that is greater than 1%. The "thinned" tables are not recommended for use in calculations involving transport through the

resonance region (Reference 8, p. 2-18).

Neutron interaction table designations are included as part of the material composition input to MCNP. Each material composition is composed of one or more elements or isotopes designated by a ZAID identifier. The ZAID identifier takes the form "ZZZAAA.nnC" where "ZZZ" represents the atomic number of the element ("ZZZ" may be one or two digits), "AAA" represents the elemental isotope ("AAA" must be three digits incorporating leading zeros), and "nn" represents the neutron interaction table designation.

Calculations involving transport through the resonance region should use the most detailed neutron interaction tables available unless there is a valid reason to do otherwise, such as the availability of more appropriate temperature dependent cross sections. An MCNP k_{eff} calculation result is uniquely defined by the neutron interaction tables it utilizes. Table 2.2.5-1 contains a listing of elements and isotopes with their ZAIDs which have been selected for use by the Waste Package Design Department in criticality analyses. The cross section tables for the elements and isotopes of the CRC analyses summarized in this study are obtained from the set displayed in Table 2.2.5-1.

Table 2.2.5-1 Selected CRC Selected MCNP ZAIDs for Various Elements and Isotopes[1]

Element	Isotope	Selected Cross Section Library ZAID
Hydrogen	H-1	1001.50c
	H-2	1002.55c
	H-3	1003.50c
Helium	He-3	2003.50c
	He-4	2004.50c
Lithium	Li-6	3006.50c
	Li-7	3007.55c
Beryllium	Be-7	4007.35c
	Be-9	4009.50c
Boron	B-10	5010.50c
	B-10	5010.53c
	B-11	5011.56c
Carbon	C (natural)	6000.50c
	C-12	6012.50c
	C-13	6013.35c
Nitrogen	N-14	7014.50c
	N-15	7015.55c
Oxygen	O-16	8016.50c
	O-16	8016.53c
	O-16	8016.54c
	O-17	8017.60c
Fluorine	F-19	9019.50c
Sodium	Na-23	11023.50c
Magnesium	Mg (natural)	12000.50c
Aluminum	Al-27	13027.50c
Silicon	Si (natural)	14000.50c
Phosphorus	P-31	15031.50c
Sulfur	S (natural)	16000.60c
	S-32	16032.50c
Chlorine	Cl (natural)	17000.50c
Argon	Ar (natural)	18000.59c
Potassium	K (natural)	19000.50c
Calcium	Ca (natural)	20000.50c
	Ca-40	20040.21c
Scandium	Sc-45	21045.60c
Titanium	Ti (natural)	22000.50c
Vanadium	V (natural)	23000.50c
Chromium	Cr-50	24050.60c

Element	Isotope	Selected Cross Section Library ZAID
	Cr-52	24052.60c
	Cr-53	24053.60c
	Cr-54	24054.60c
Manganese	Mn-55	25055.50c
Iron	Fe-54	26054.60c
	Fe-56	26056.60c
	Fe-57	26057.60c
	Fe-58	26058.60c
Cobalt	Co-59	27059.50c
Nickel	Ni-58	28058.60c
	Ni-60	28060.60c
	Ni-61	28061.60c
	Ni-62	28062.60c
	Ni-64	28064.60c
Copper	Cu-63	29063.60c
	Cu-65	29065.60c
Gallium	Ga (natural)	31000.50c
Arsenic	As-74	33074.35c
	As-75	33075.35c
Bromine	Br-79	35079.55c
	Br-81	35081.55c
Krypton	Kr-78	36078.50c
	Kr-80	36080.50c
	Kr-82	36082.50c
	Kr-83	36083.50c
	Kr-84	36084.50c
	Kr-86	36086.50c
Rubidium	Rb-85	37085.55c
	Rb-87	37087.55c
Yttrium	Y-88	39088.35c
	Y-89	39089.50c
Zirconium	Zr (natural)	40000.60c
	Zr-93	40093.50c
Niobium	Nb-93	41093.50c
Molybdenum	Mo (natural)	42000.50c
	Mo-95	42095.50c
Technetium	Tc-99	43099.50c
Ruthenium	Ru-101	44101.50c
	Ru-103	44103.50c
Rhodium	Rh-103	45103.50c
	Rh-105	45105.50c
Palladium	Pd-105	46105.50c
	Pd-108	46108.50c
Silver	Ag-107	47107.60c
	Ag-109	47109.60c
Cadmium	Cd (natural)	48000.50c
Indium	In (natural)	49000.60c
Tin	Sn (natural)	50000.35c
Iodine	I-127	53127.60c
	I-129	53129.60c
	I-135	53135.50c
Xenon	Xe (natural)	54000.35c
	Xe-131	54131.50c
	Xe-134	54134.35c
	Xe-135	54135.50c
	Xe-135	54135.53c
	Xe-135	54135.54c
Cesium	Cs-133	55133.50c
	Cs-134	55134.60c
	Cs-135	55135.50c
	Cs-136	55136.60c
	Cs-137	55137.60c
Barium	Ba-138	56138.50c
Praseodymium	Pr-141	59141.50c
Neodymium	Nd-143	60143.50c

Element	Isotope	Selected Cross Section Library ZAID
	Nd-145	60145.50c
	Nd-147	60147.50c
	Nd-148	60148.50c
Promethium	Pm-147	61147.50c
	Pm-148	61148.50c
	Pm-149	61149.50c
Samarium	Sm-147	62147.50c
	Sm-149	62149.50c
	Sm-150	62150.50c
	Sm-151	62151.50c
	Sm-152	62152.50c
Europium	Eu-151	63151.55c
	Eu-152	63152.50c
	Eu-153	63153.55c
	Eu-154	63154.50c
	Eu-155	63155.50c
Gadolinium	Gd-152	64152.50c
	Gd-154	64154.50c
	Gd-155	64155.50c
	Gd-156	64156.50c
	Gd-157	64157.50c
	Gd-158	64158.50c
	Gd-160	64160.50c
Holmium	Ho-165	67165.55c
Thulium	Tm-169	69169.55c
Hafnium	Hf (natural)	72000.50c
Tantalum	Ta-181	73181.50c
	Ta-182	73182.60c
Tungsten	W (natural)	74000.55c
	W-182	74182.55c
	W-183	74183.55c
	W-184	74184.55c
	W-186	74186.55c
Rhenium	Re-185	75185.50c
	Re-187	75187.50c
Iridium	Ir (natural)	77000.55c
Platinum	Pt (natural)	78000.35c
Gold	Au-197	79197.50c
Lead	Pb (natural)	82000.50c
	Pb-206	82206.60c
	Pb-207	82207.60c
	Pb-208	82208.60c
Bismuth	Bi-209	83209.50c
Thorium	Th-230	90230.60c
	Th-231	90231.35c
	Th-232	90232.50c
	Th-233	90233.35c
Protactinium	Pa-231	91231.60c
	Pa-233	91233.50c
Uranium	U-232	92232.60c
	U-233	92233.50c
	U-234	92234.50c
	U-235	92235.50c
	U-235	92235.53c
	U-235	92235.54c
	U-236	92236.50c
	U-237	92237.50c
	U-238	92238.50c
	U-238	92238.53c
	U-238	92238.54c
	U-239	92239.35c
	U-240	92240.35c
Neptunium	Np-235	93235.35c
	Np-236	93236.35c
	Np-237	93237.50c

Element	Isotope	Selected Cross Section Library ZAID
	Np-238	93238.35c
	Np-239	93239.60c
Plutonium	Pu-236	94236.60c
	Pu-237	94237.35c
	Pu-238	94238.50c
	Pu-239	94239.55c
	Pu-240	94240.50c
	Pu-241	94241.50c
	Pu-242	94242.50c
	Pu-243	94243.60c
	Pu-244	94244.60c
Americium	Am-241	95241.50c
	Am-242m	95242.50c
	Am-243	95243.50c
Curium	Cm-241	96241.60c
	Cm-242	96242.50c
	Cm-243	96243.35c
	Cm-244	96244.50c
	Cm-245	96245.35c
	Cm-246	96246.35c
	Cm-247	96247.35c
	Cm-248	96248.60c
Berkelium	Bk-249	97249.60c
Californium	Cf-249	98249.60c
	Cf-250	98250.60c
	Cf-251	98251.60c
	Cf-252	98252.60c

[1] Detailed information about each of the selected cross section data tables is presented in Reference 18.

2.2.6 S(α,β) Thermal Treatment

The S(α,β) thermal treatment accounts for binding effects in molecules and crystalline solids. The S(α,β) thermal scattering treatment is a necessary requirement in a highly moderating medium where low-energy scattering may be dominant. S(α,β) thermal treatment tables are available for a limited number of materials. In the PWR CRC analyses, the thermal treatment is consistently applied to hydrogen in the borated moderator.

3.0 DESCRIPTION OF THE REACTOR SYSTEMS

This section provides a general description of the reactor systems analyzed in the supporting analyses.

3.1 PWRs

3.1.1 Crystal River Unit 3

Crystal River Unit 3 (CR3) operated by Florida Power Corporation is a 825 MWe Babcock and Wilcox (B&W) PWR with 177 fuel assemblies. CR3 is loaded with B&W 15 × 15 assemblies. A total of 33 CRC statepoints have been evaluated for CR3.

Table 3.1.1-1 provides some general information about the CR3 CRC statepoints. The

information includes the reactor fuel cycle the statepoint is in, the burnup of the core in effective full power days (EFPD), the initial weight percent (wt%) enrichments of fuel batches in the core during the statepoint (fresh fuel is identified by "()" around the enrichment values), and the downtime in days since the core was last at power before restarting.

The CRC statepoints for CR3 are generally modeled with 29 different assemblies taking advantage of eighth-core symmetry. The calculations for Cycles 1B and 9 required a full core model (all 177 assemblies). Calculations for Cycles 3 and 8 required a quarter core model (52 assemblies).

The CR3 reactor statepoints were modeled in great detail. Regions above, below, and around the active fuel were represented in the models. Figure 3.1.1-1 provides a radial view of the CR3 reactor internals. Figure 3.1.1-2 contains a radial view of a fuel assembly in the CR3 core. Figure 3.1.1-3 presents an axial dimension schematic of a fuel assembly in CR3. A more detailed description of CR3 including materials, geometry, and core follow data needed for the CRC evaluations is provided in Reference 11.

Table 3.1.1-1 General CR3 CRC Statepoint Information (pp. 30-40, 296, Ref. 11)

Case	Cycle	Cycle Length to Statepoint (EFPD)	Enrichments (wt% U-235 in U of UO_2)	Downtime (days)
1	1A	0.0	(1.93, 2.54, 2.83)	0.0
2	1B	268.8	1.93, 2.54, 2.83, 2.00	195.3
3	1B	411.0	1.93, 2.54, 2.83, 2.00	14.8
4	2	0.0	2.54, (2.64), 2.83	97.0
5	3	0.0	2.54, (2.62), 2.64, 2.83	164.0
6	3	168.5	2.54, 2.62, 2.64, 2.83	16.8
7	3	250.0	2.54, 2.62, 2.64, 2.83	12.3
8	4	0.0	2.62, (2.62), 2.64, (2.95)	73.0
9	4	228.1	2.62, 2.64, 2.95	15.2
10	4	253.0	2.62, 2.64, 2.95	24.0
11	5	0.0	2.62, 2.64, 2.95, (2.95, 3.29)	127.0
12	5	388.5	2.62, 2.64, 2.95, 3.29	5.0
13	6	0.0	2.62, 2.64, 2.95, 3.29, (3.49)	163.0
14	6	96.0	2.62, 2.64, 2.95, 3.29, 3.49	168.9
15	6	400.0	2.62, 2.64, 2.95, 3.29, 3.49	10.4
16	7	0.0	2.54, 2.62, 2.64, 3.29, 3.49, (3.84)	113.0
17	7	260.3	2.54, 2.62, 2.64, 3.29, 3.49, 3.84	18.9
18	7	291.0	2.54, 2.62, 2.64, 3.29, 3.49, 3.84	39.5
19	7	319.0	2.54, 2.62, 2.64, 3.29, 3.49, 3.84	109.5
20	7	462.3	2.54, 2.62, 2.64, 3.29, 3.49, 3.84	2.2
21	7	479.0	2.54, 2.62, 2.64, 3.29, 3.49, 3.84	7.2
22	8	0.0	1.93, 2.62, 3.29, 3.49, 3.84, (3.94)	99.0
23	8	97.6	1.93, 2.62, 3.29, 3.49, 3.84, 3.94	15.5
24	8	139.8	1.93, 2.62, 3.29, 3.49, 3.84, 3.94	6.2
25	8	404.0	1.93, 2.62, 3.29, 3.49, 3.84, 3.94	44.4
26	8	409.6	1.93, 2.62, 3.29, 3.49, 3.84, 3.94	4.9
27	8	515.5	1.93, 2.62, 3.29, 3.49, 3.84, 3.94	7.6
28	9	0.0	1.93, 3.84, (3.90), 3.94	75.0
29	9	158.8	1.93, 3.84, 3.90, 3.94	2.1
30	9	219.0	1.93, 3.84, 3.90, 3.94	53.1
31	9	363.1	1.93, 3.84, 3.90, 3.94	1.6
32	10	0.0	3.84, 3.90, 3.94, (4.167[1])	55.0
33	10	573.7	3.84, 3.90, 3.94, 4.167[1]	16.4

[1] Assemblies in this fuel batch contain 192 fuel rods at 4.19 wt% U-235 and 16 fuel rods at 3.89 wt% U-235, for an average assembly enrichment of 4.167 wt% U-235.

Figure 3.1.1-1 Radial View of the CR3 Reactor Internals

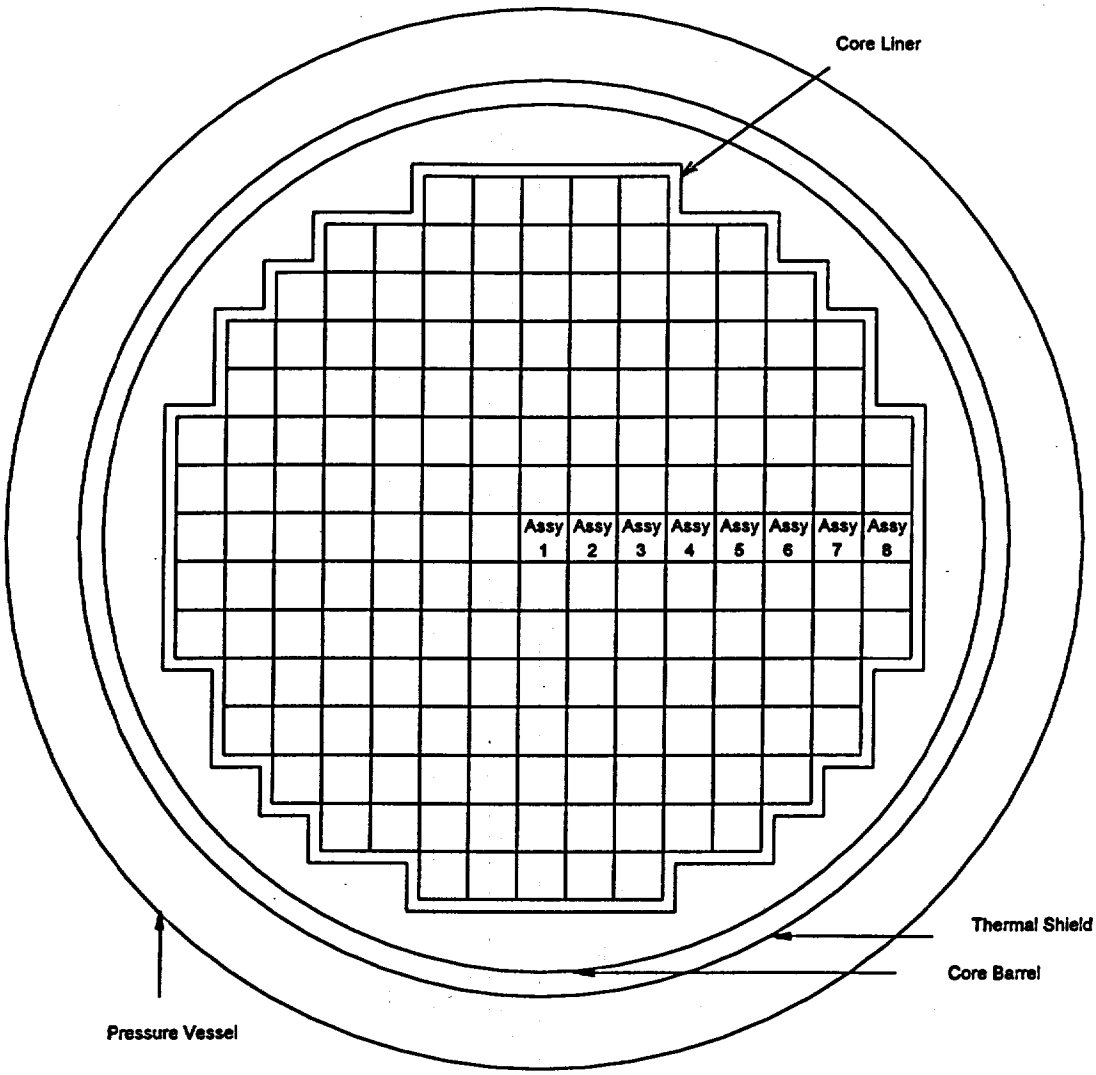


Figure 3.1.1-2 Radial View of a Single CR3 Fuel Assembly

Single Fuel Assembly Layout and Dimensions

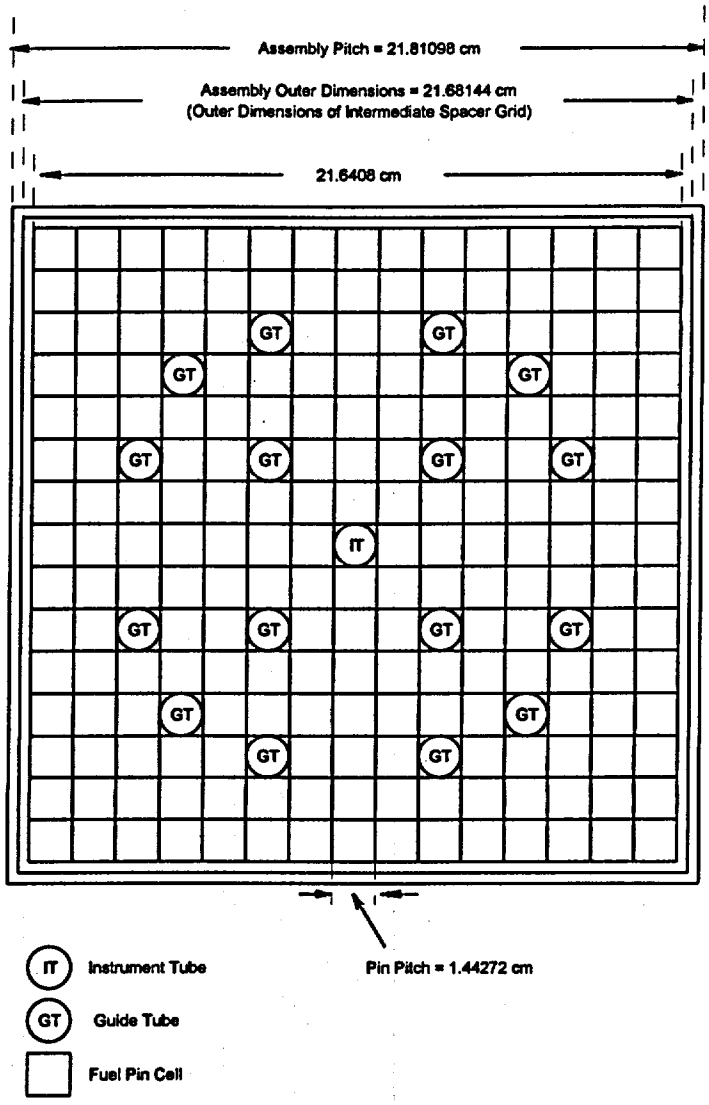
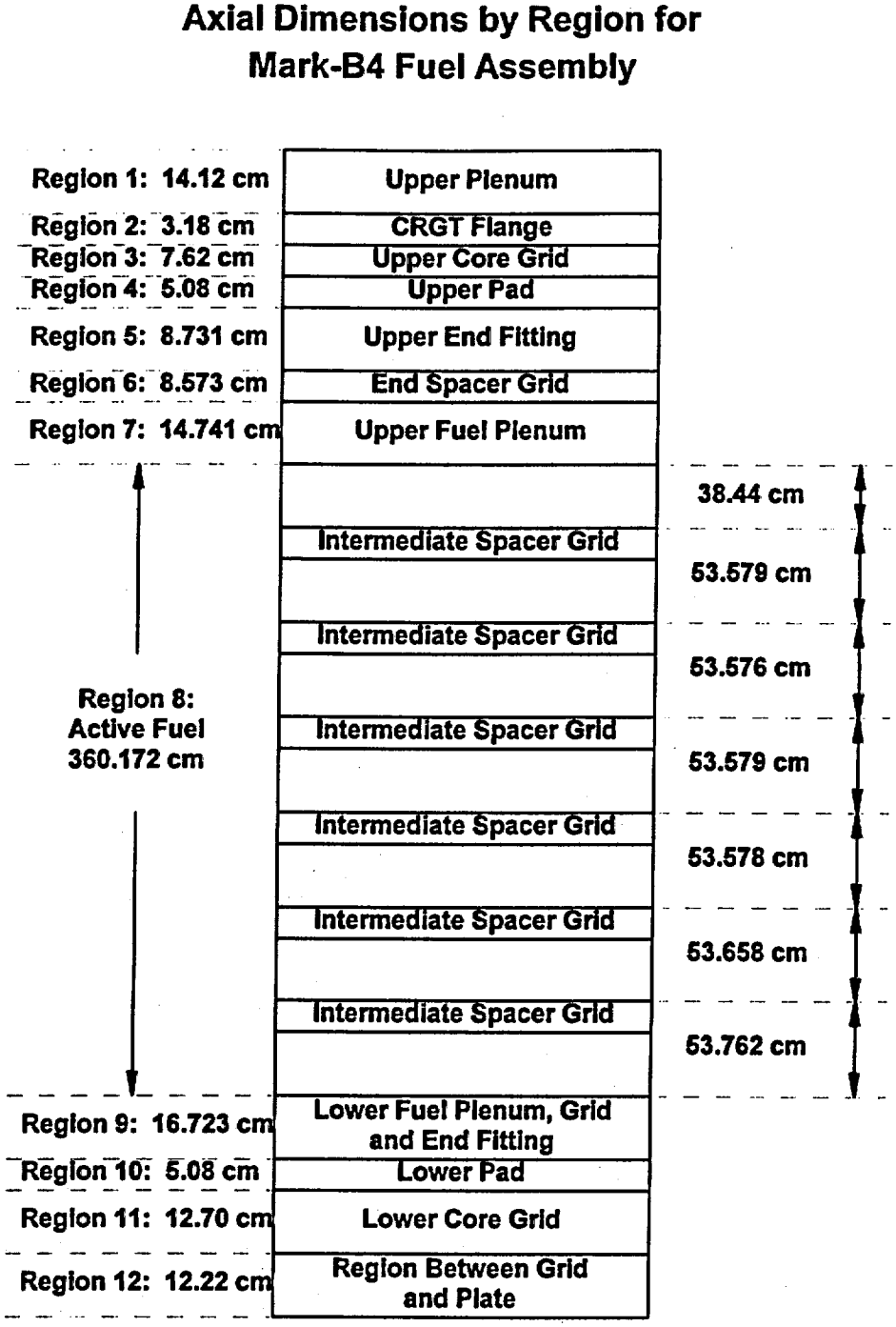


Figure 3.1.1-3 Axial Dimension Schematic for Fuel Assembly in CR3



3.1.2 McGuire Unit 1

McGuire Unit 1 (MCG1) operated by Duke Power Company is a 1129 MWe Westinghouse (W) PWR with 193 fuel assemblies. MCG1 is loaded with W 17 × 17 assemblies. A total of 6 CRC statepoints have been evaluated for MCG1.

Table 3.1.2-1 provides some general information about the available MCG1 CRC statepoints. The information includes the reactor fuel cycle the statepoint is in, the burnup of the core in effective full power days (EFPD), the initial weight percent (wt%) enrichments of fuel batches in the core during the statepoint (fresh fuel is identified by "()" around the enrichment values), and the downtime in days since the core was last at power before restarting.

The MCG1 CRC statepoints in Cycles 1 and 6 were modeled with 31 different assemblies taking advantage of eighth-core symmetry. The statepoints in Cycle 7 required a full-core model (all 193 assemblies).

The MCG1 reactor statepoints were modeled in great detail. Figure 3.1.2-1 presents a radial view of the MCG1 vessel internals including the stainless steel (SS304) core liner, core barrel, neutron pad, and vessel weld liner as well as the carbon steel (CS508) pressure vessel. Figure 3.1.2-2 presents a radial view of a W 17×17 fuel assembly used in MCG1. Figure 3.1.2-3 presents an axial dimension schematic of a fuel assembly in MCG1. A more detailed description of MCG1 including materials, geometry, and core loading data is provided in Reference 12.

Table 3.1.2-1 General MCG1 CRC Statepoint Information (pp. 40, 45, 46, 103, Ref. 12)

Case	Cycle	Cycle Length to Statepoint (EFPD)	Enrichments (wt% U-235 in U of UO_2)	Downtime (days)
1	1	0.0	(2.108, 2.601, 3.106)	0.0
2	6	0.0	2.92, 3.204, 3.40, (3.60)	78.0
3	6	62.4	2.92, 3.204, 3.40, 3.60	62.7
4	7	0.0	2.92, 3.204, 3.40, 3.60, (3.75)	130.0
5	7	129.0	2.92, 3.204, 3.40, 3.60, 3.75	29.6
6	7	282.3	2.92, 3.204, 3.40, 3.60, 3.75	18.8

Figure 3.1.2-1 Radial View of MCG1 Vessel Internals Along Core Midplane

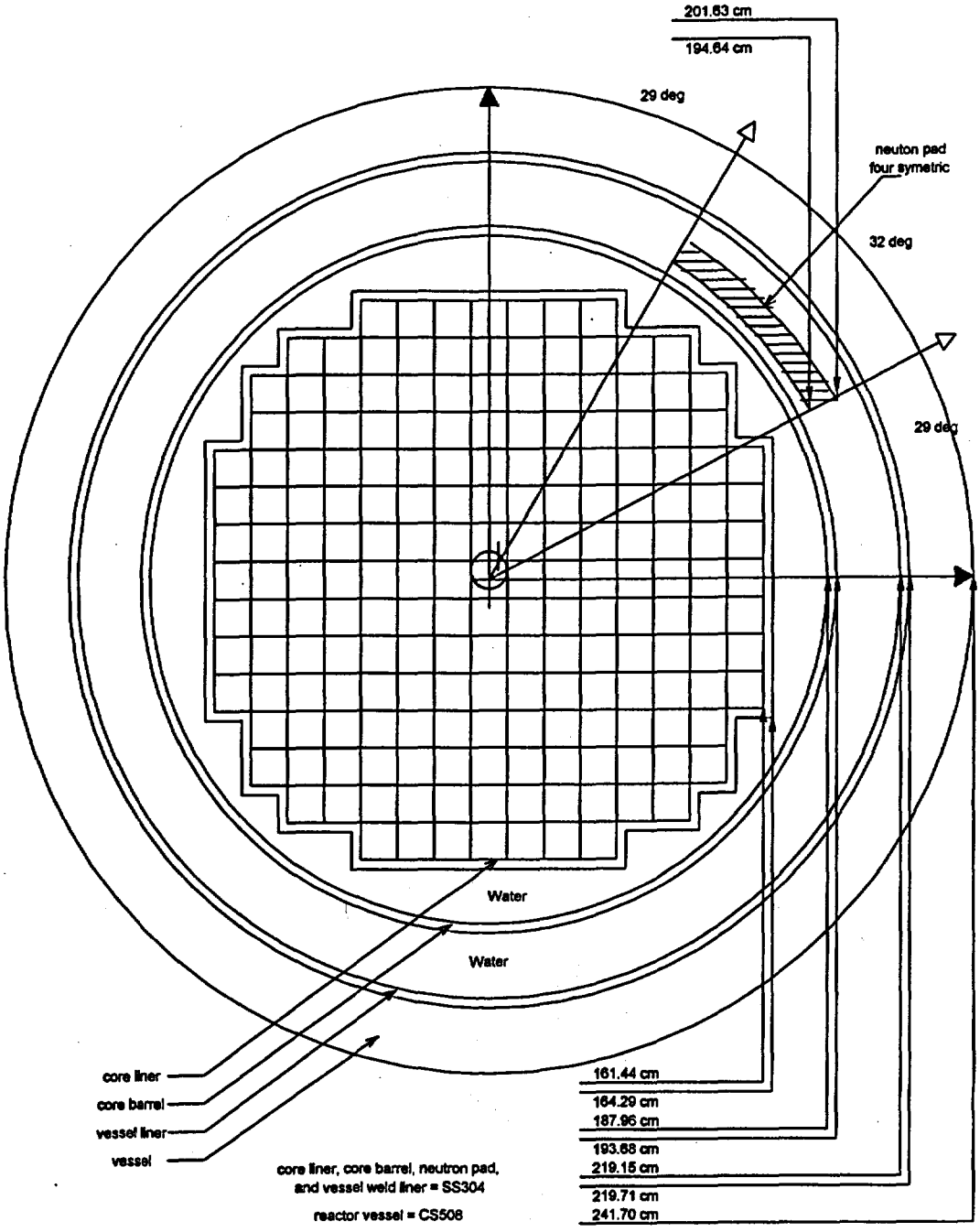


Figure 3.1.2-2 Radial View of a Single 17 x 17 Fuel Assembly for MCG1

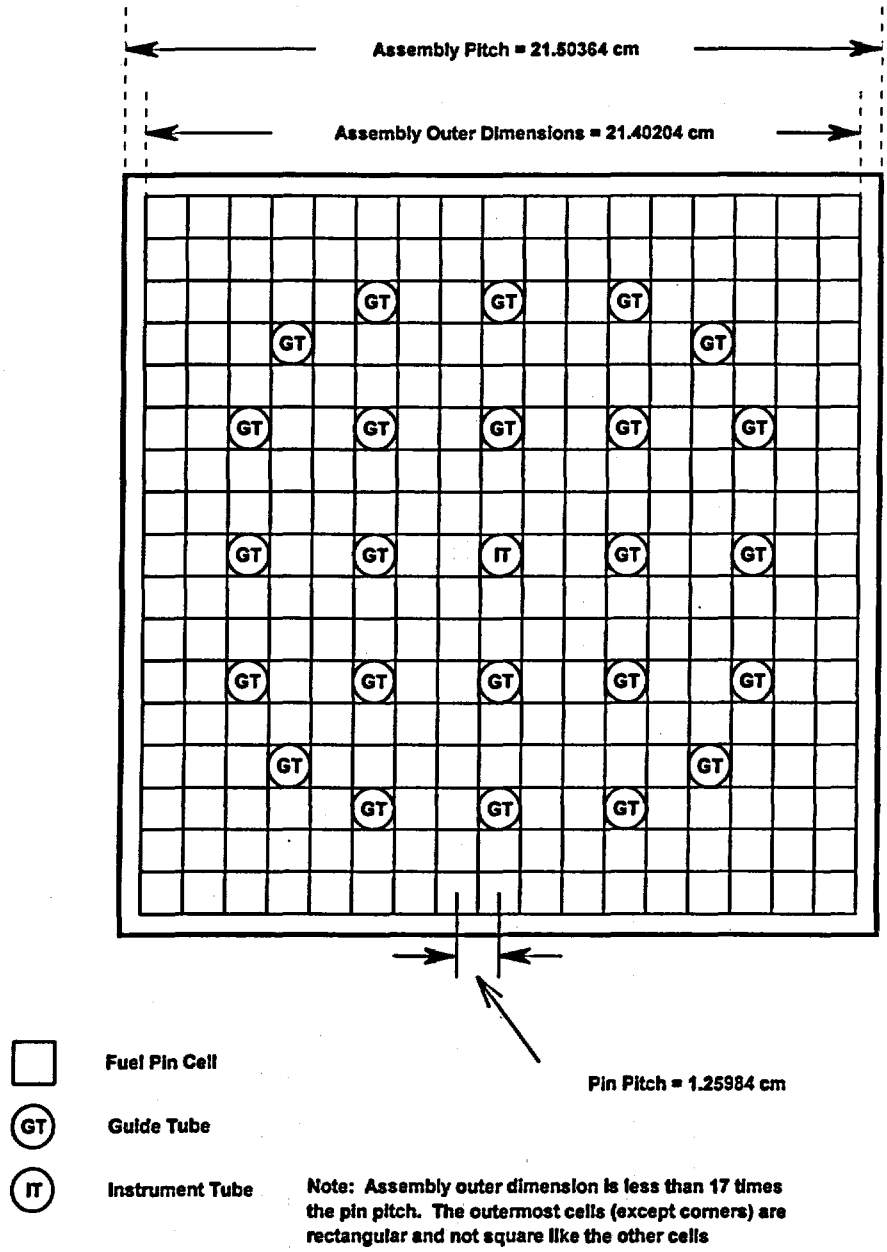
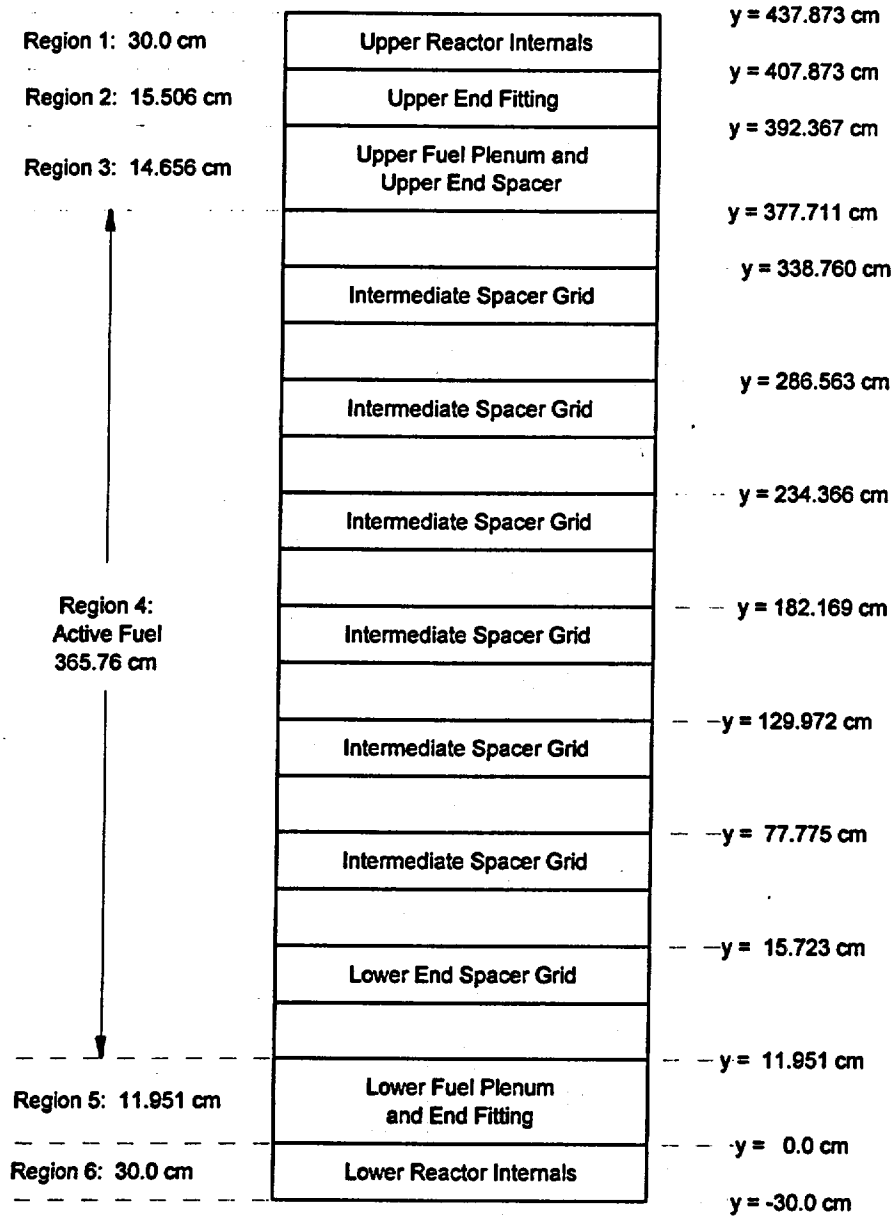


Figure 3.1.2-3 Axial Dimension Schematic for Fuel Assembly in MCG1



3.1.3 Sequoyah Unit 2

Sequoyah Unit 2 (SEQ2) operated by TVA Nuclear is a 1148 MWe W PWR with 193 fuel assemblies. SEQ2 is loaded with W 17 × 17 assemblies. A total of 3 statepoints have been evaluated for SEQ2.

Table 3.1.3-1 provides some general information about the SEQ2 CRC statepoints. The information includes the reactor fuel cycle the statepoint is in, the burnup of the core in effective full power days (EFPD), the initial weight percent (wt%) enrichments of fuel batches in the core during the statepoint (fresh fuel is identified by "()" around the enrichment values), and the downtime in days since the core was last at power before restarting.

The SEQ2 CRC statepoint in Cycle 1 was modeled with 31 different assemblies taking advantage of eighth-core symmetry. The statepoints in Cycle 3 required a quarter-core model containing 56 assemblies.

The SEQ2 reactor statepoints were modeled in great detail. Figure 3.1.3-1 presents a radial view of the SEQ2 vessel internals including the stainless steel (SS304) core liner, core barrel, neutron pad, and vessel weld liner as well as the carbon steel (CS508) pressure vessel. Figure 3.1.3-2 presents a radial view of a W 17×17 fuel assembly used in SEQ2. Figure 3.1.3-3 presents an axial dimension schematic of a fuel assembly in SEQ2. A more detailed description of SEQ2 including materials, geometry, and core loading data is provided in Reference 13.

Table 3.1.3-1 General SEQ2 CRC Statepoint Information (pp. 25, 27, 63, Ref. 13)

Case	Cycle	Cycle Length to Statepoint (EFPD)	Enrichments (wt% U-235 in U of UO_2)	Downtime (days)
1	1	0.0	(2.10, 2.60, 3.10)	0.0
2	3	0.0	2.60, 3.10,3.50, (3.60, 3.80)	81.0
3	3	210.9	2.60, 3.10,3.50, 3.60, 3.80	995.7

Figure 3.1.3-1 Radial View of SEQ2 Vessel Internals Along Core Midplane

Core Liner, Core Barrel, Thermal Shield, and
Vessel Liner = SS304. Reactor Vessel = CS508

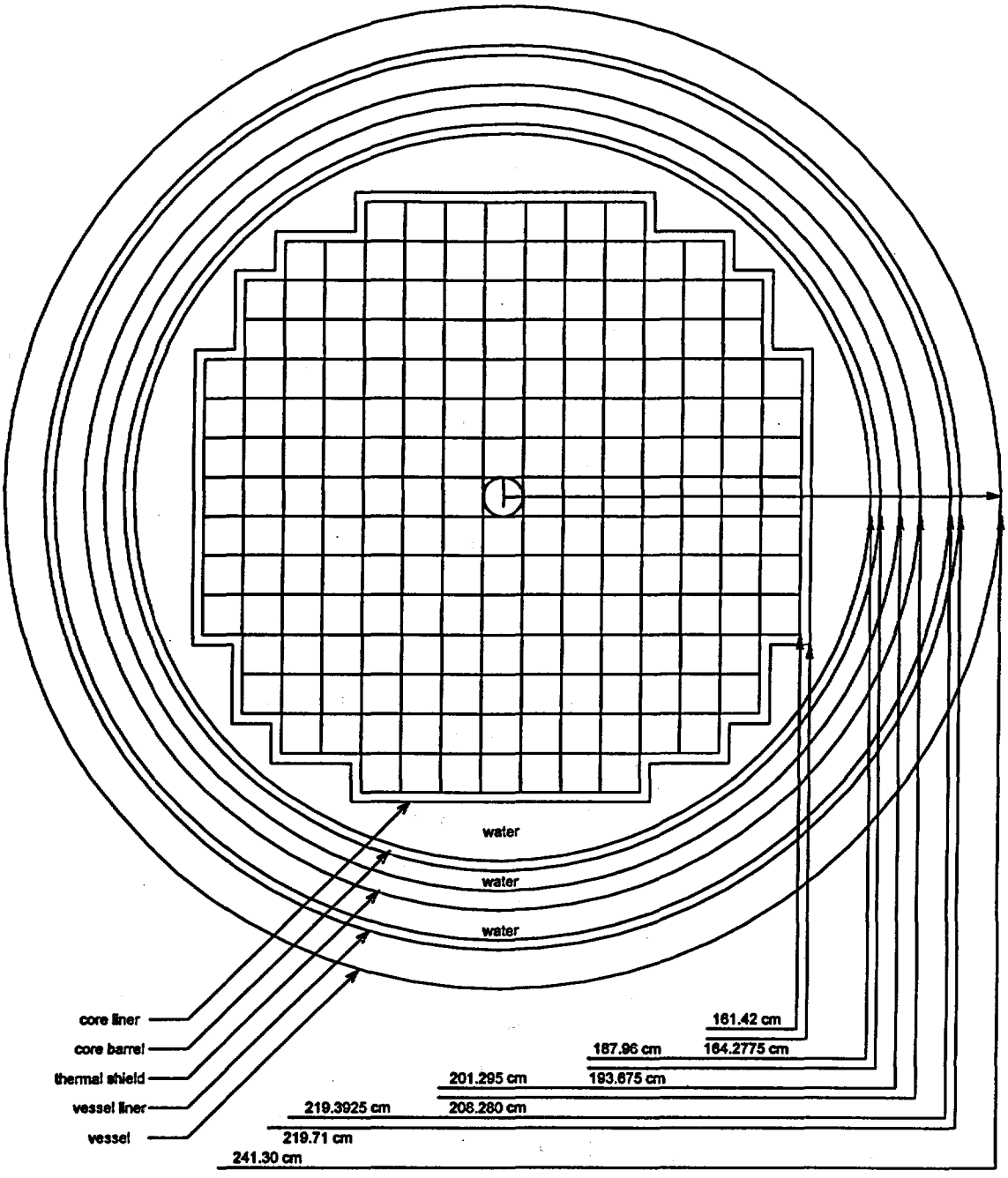


Figure 3.1.3-2 Radial View of a Single 17 x 17 Fuel Assembly for SEQ2

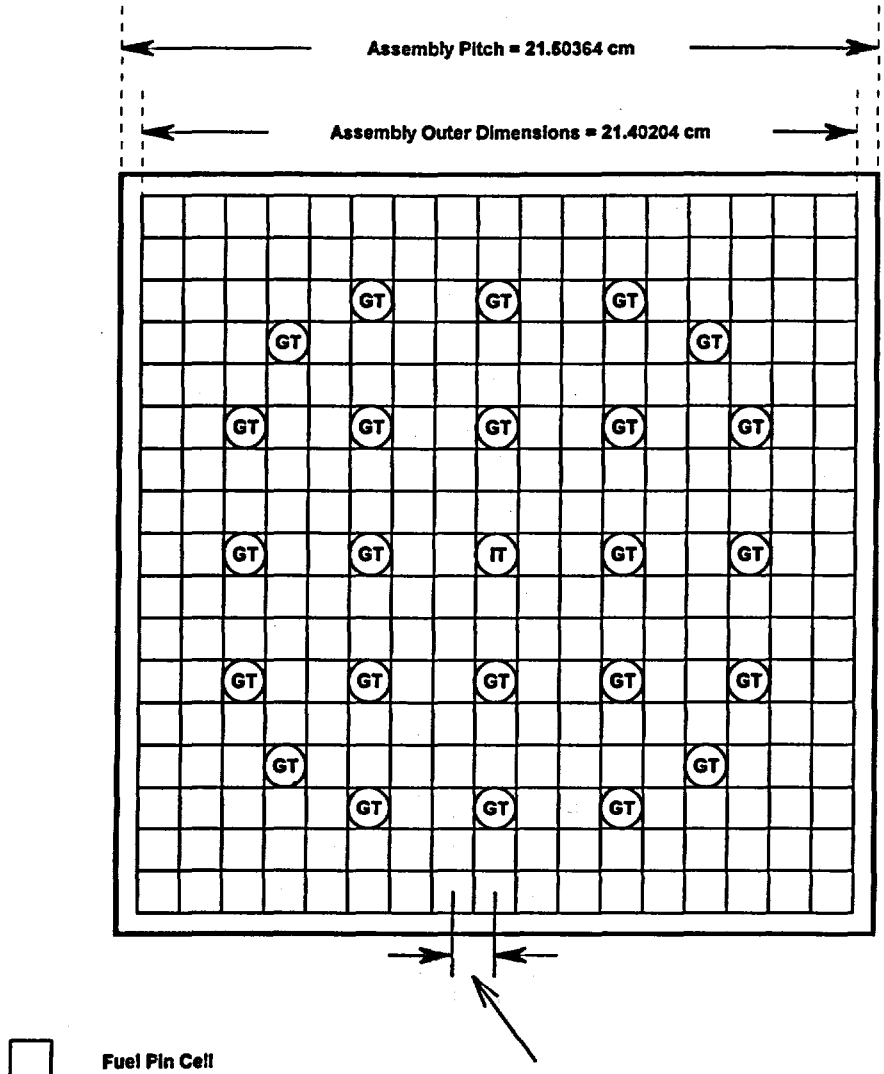
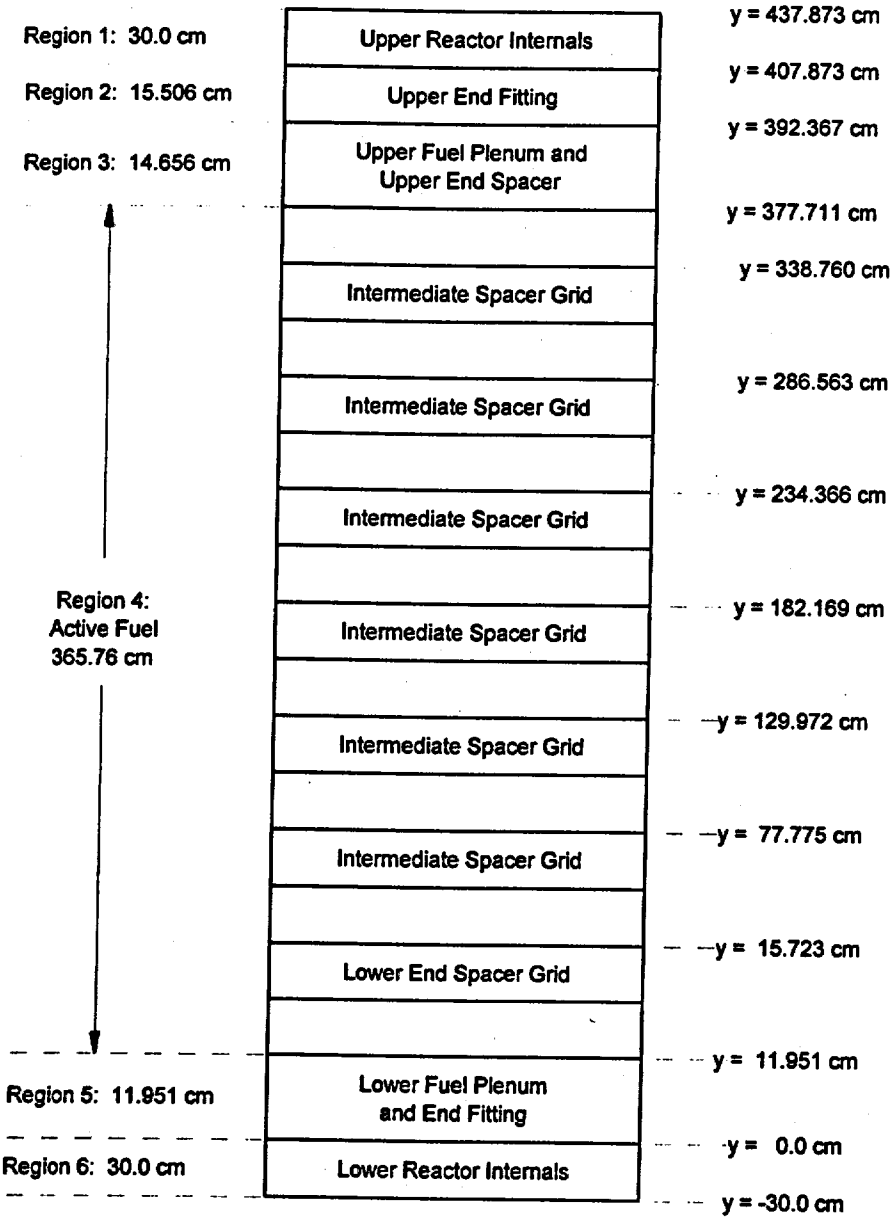


Figure 3.1.3-3 Axial Dimension Schematic for Fuel Assembly in SEQ2



3.1.4 Catawba Unit 1

Catawba Unit 1 (CAT1) operated by Duke Power Company is a 1129 MWe W PWR with 193 fuel assemblies. CAT1 is loaded with W 17 × 17 assemblies. No CRC statepoints have been evaluated at this time. CRC data from CAT1 is available for 3 statepoints.

3.1.5 Catawba Unit 2

Catawba Unit 2 (CAT2) operated by Duke Power Company is a 1129 MWe W PWR with 193 fuel assemblies. CAT2 is loaded with W 17 × 17 assemblies. No CRC statepoints have been evaluated at this time. CRC data from CAT2 is available for 4 statepoints.

3.1.6 Three Mile Island Unit 1

Three Mile Island Unit 1 (TMI1) operated by GPU Nuclear Corporation is a 786 MWe B&W PWR with 177 fuel assemblies. TMI1 is loaded with B&W 15 × 15 assemblies. A total of 3 statepoints have been evaluated for TMI1.

Table 3.1.6-1 provides some general information about the TMI1 CRC statepoints. The information includes the reactor fuel cycle the statepoint is in, the burnup of the core in EFPD, the initial wt% enrichments of fuel batches in the core during the statepoint (fresh fuel is identified by "()" around the enrichment values), and the downtime in days since the core was last at power before restarting.

The CRC statepoints for TMI1 were modeled with 29 assemblies taking advantage of eighth-core symmetry. The TMI1 reactor statepoints were modeled in great detail. Regions above, below, and around the active fuel were represented in the models. The radial view of the TMI1 reactor internals is the same as that shown for CR3 in Figure 3.1.1-1. The radial view of a fuel assembly in the TMI1 core is the same as that shown for CR3 in Figure 3.1.1-2. The axial dimension schematic of a fuel assembly in the TMI1 core is the same as that shown for CR3 in Figure 3.1.1-3. A more detailed description of TMI1 including materials, geometry, and core follow data needed for the CRC evaluations is provided in Reference 19.

Table 3.1.6-1 General TMI1 CRC Statepoint Information (pp. 25, 29, 66, Ref. 19)

Case	Cycle	Cycle Length to Statepoint (EFPD)	Enrichments (wt% U-235 in U of UO_2)	Downtime (days)
1	1	0.0	(2.06, 2.75, 3.05)	0.0
2	5	0.0	2.64, 2.85, (2.85)	2420.0
3	5	114.4	2.64, 2.85	32.2

3.1.7 Davis-Besse Unit 1

Davis-Besse Unit 1 (DB) operated by Toledo Edison Company is a 877 MWe B&W PWR with 177 fuel assemblies. DB is loaded with B&W 15 × 15 assemblies. No CRC statepoints have been evaluated at this time. CRC data from DB is available for 7 statepoints.

3.1.8 Calvert Cliffs Unit 1

Calvert Cliffs Unit 1 (CC1) operated by Baltimore Gas and Electric Company is a 825 MWe Combustion Engineering (CE) PWR. CC1 is loaded with CE 14 × 14 assemblies. CC1 is being considered as a source of CRC data. No CRC statepoints have been evaluated at this time. A partial set of CRC data from CC1 is available for 1 statepoint.

3.2 BWRs

3.2.1 Quad Cities Unit 1

Quad Cities Unit 1 (QC1) operated by Commonwealth Edison Company is a 789 MWe General Electric (GE) class 3 BWR. One CRC statepoint has been evaluated at this time. QC1 is being considered as a source of additional CRC data.

Table 3.2.1-1 provides some general information about the QC1 CRC statepoint. The information includes the reactor fuel cycle the statepoint is in, the burnup of the core in effective full power days (EFPD), the average bundle initial weight percent (wt%) enrichments of fuel batches in the core during the statepoint (fresh fuel is identified by "()" around the enrichment values), and the downtime in days since the core was last at power before restarting.

The QC1 core design consisting of 724 GE 7 × 7 fuel assemblies, 177 control blades, 41 in-core instrument assemblies, and 7 in-core neutron sources. The 1/4 core MCNP 4A core model consists of 181 assemblies, 15 in-core instrument tubes, 28 complete control blades, and 11 partial control blades. A core shroud surrounds the periphery fuel assemblies in the core. The periphery of the reactor consists of the core shroud, the pump region, and the pressure vessel. Each of these components are separated by a region of water. A radial view of the modeled reactor internals is shown in Figure 3.2.1-1. The core lattice model is filled with control cell quadrant regions that contain the correctly oriented assembly, channel, moderator, and control blade portions. The control cell view of four assemblies, a control blade, and an instrument tube is shown in Figure 3.2.1-2. A more detailed description of QC1 including materials, geometry, and core loading data is provided in Reference 17.

Table 3.2.1-1 General QC1 CRC Statepoint Information

Case	Cycle	Cycle Length to Statepoint (EFPD)	Average Bundle Enrichments (wt% U-235 in U of UO_2)	Downtime (days)
1	1	0.0	(2.12)	0.0

Figure 3.2.1-1 Radial View of QC1 - 1/4 Core

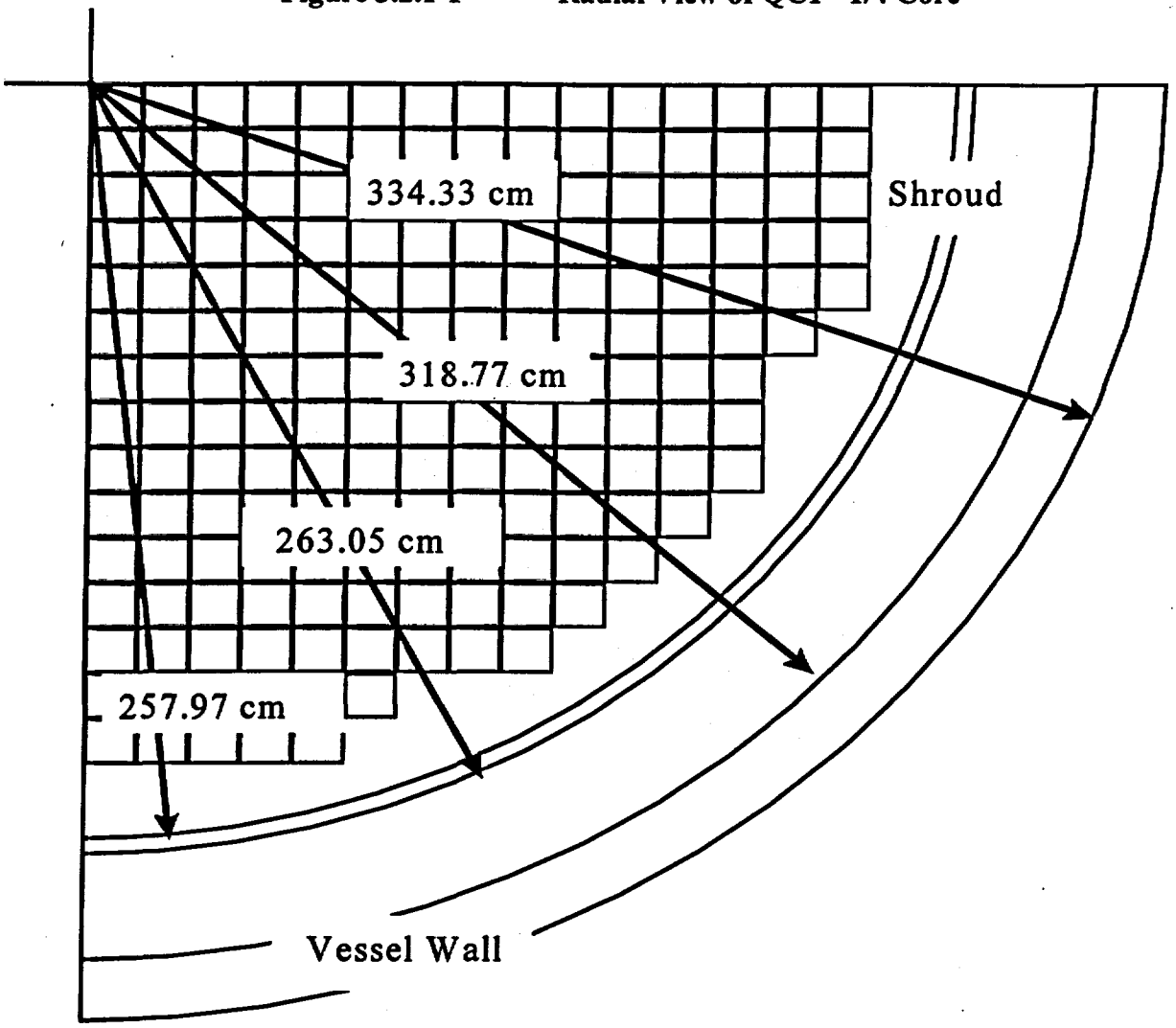
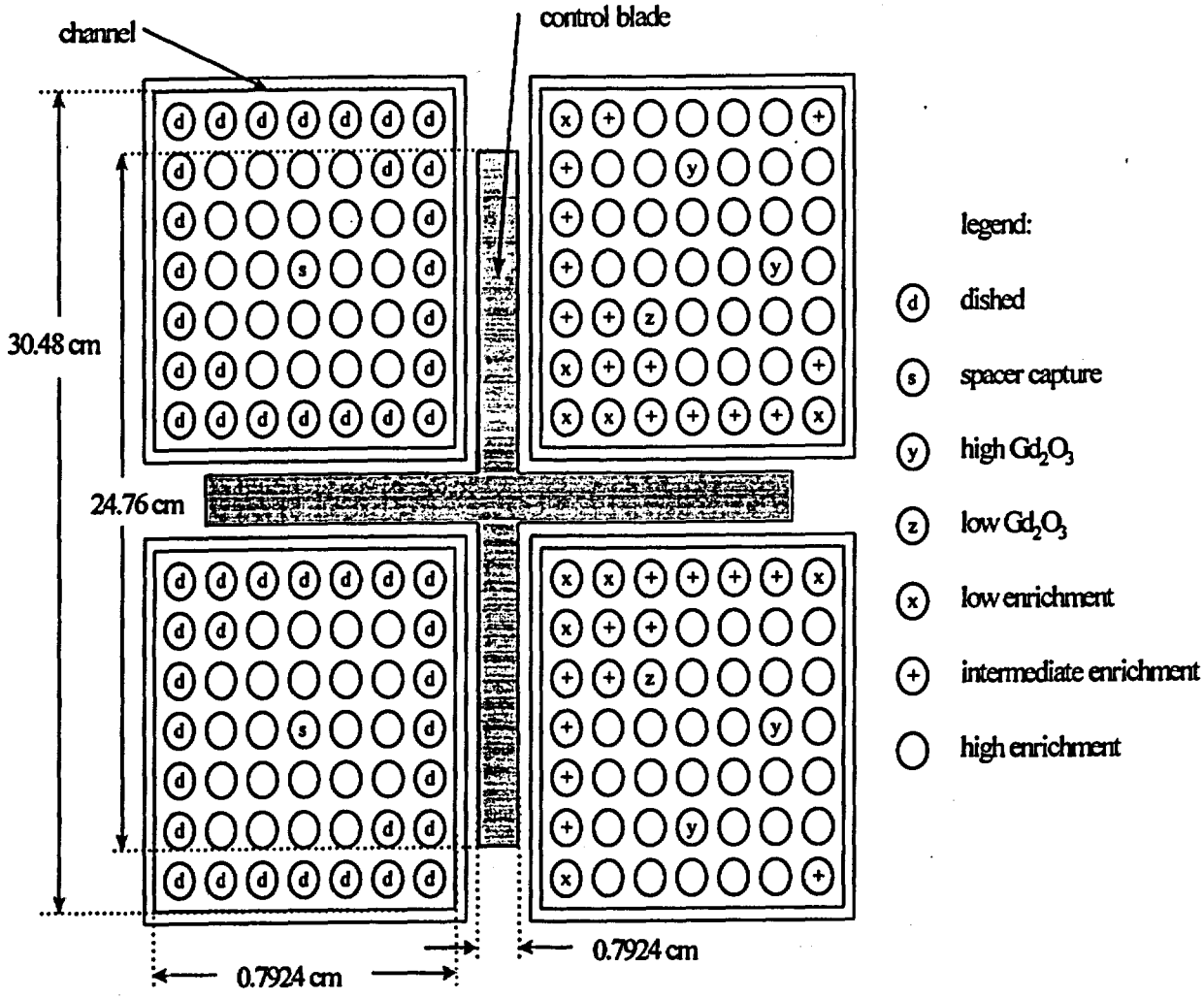


Figure 3.2.1-2 Control Cell Radial View



3.2.2 Quad Cities Unit 2

Quad Cities Unit 2 (QC2) is a sister plant to the Quad Cities Unit 1 so all dimensions and materials documented for the Quad Cities Unit 1 are equally applicable to the QC2 vessel and core internals. CRC analyses were performed for startup tests in cycles 13 and 14 of QC2 as shown in Table 3.2.2-1 and 3.2.2-2, respectively.

Table 3.2.2-1 Cycle Exposures for CRC Performed in Cycle 13

Cycle Exposure (MWd/MTU)	Identifier
0.0	QC2BOC13
201.61	QC2C13CP10
2257.20	QC2C13CP11
6489.46	QC2C13CP13

Table 3.2.2-2 Cycle Exposures for CRC Performed in Cycle 14

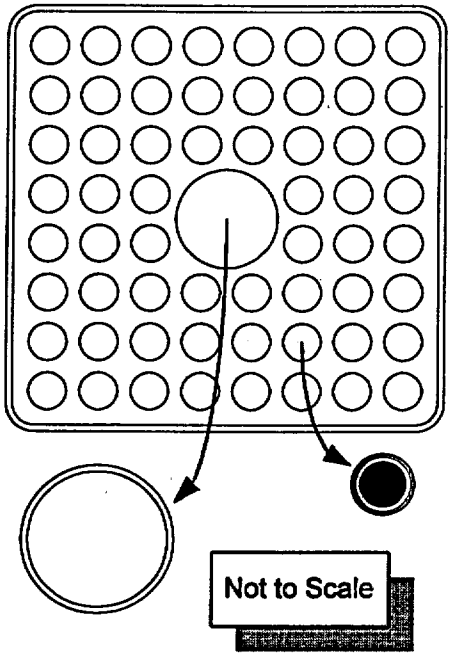
Cycle Exposure (MWd/MTU)	Identifier
0.0	QC2BOC14
4238.45	QC2C14CP16

Cycles 13 and 14 of QC2 were comprised entirely of 8 x 8 GE fuel with multiple fuel rod enrichments within planar slices through the assembly and axial zoning. Figure 3.2.2-1 provides an example of QC2 assembly axial zoning (Ref. 23). Integral burnable absorbers in the form of gadolinia were also distributed on a rod-by-rod basis and axial basis. The assemblies incorporated either two small water rods (i.e., with wall dimensions equal to empty cladding hulls) or a single large-central water rod (as illustrated in Figure 3.2.2-2). A detailed description of the fuel assemblies incorporated in the core during and before cycles 13 and 14, and their irradiation histories, as well as control blade locations are provided in Reference 21.

Figure 3.2.2-1 QC2 Fuel Assembly Axial Zoning

COLLAPSED NODES USED IN SAS2H	24 ACTUAL NODES	NODE DESCRIPTION	NODE HEIGHT
AXIAL NODE 10	24	URANIUM BLANKET	15.24CM
AXIAL NODE 9	23 22 21 20	3.37 WT% UO_2 / 7 - 4.0 WT% GD_2O_3	64.11CM
AXIAL NODE 8	19 18 17	3.37 WT% UO_2 / 7 - 40 WT% GD_2O_3	45.72CM
AXIAL NODE 7	16 15 14	3.37 WT% UO_2 / 7 - 4.0 WT% GD_2O_3	45.72CM
AXIAL NODE 6	13 12 11	3.37 WT% UO_2 / 7 - 4.0 WT% GD_2O_3	45.72CM
AXIAL NODE 5	10 9	3.37 WT% UO_2 / 7 - 4.0 WT% GD_2O_3	30.48CM
AXIAL NODE 4	8 7 6	3.37 WT% UO_2 / 7 - 4.0 WT% GD_2O_3	45.72CM
AXIAL NODE 3	5 4	3.37 WT% UO_2 / 7 - 4.0 WT% GD_2O_3	30.48CM
AXIAL NODE 2	3 2	3.37 WT% UO_2 / 7 - 4.0 WT% GD_2O_3	30.48CM
AXIAL NODE 1	1	URANIUM BLANKET	15.24CM

Figure 3.2.2-2 GE 8 x 8 Assembly with Large Central Water Rod



3.2.3 LaSalle County Unit 1

LaSalle County Unit 1 (LC1) operated by Commonwealth Edison Company is a 1078 MWe GE class 5 BWR with 764 fuel assemblies. LC1 is a source for GE 8 × 8 and 9 × 9 fuel assembly data. CRC data has been acquired for LC1 and is being considered as a source of CRC statepoints. No CRC statepoints have been evaluated at this time.

3.2.4 Dresden Unit 2

Dresden Unit 2 (D2) operated by Commonwealth Edison Company is a 794 MWe GE class 3 BWR with 724 fuel assemblies. D2 is a source for GE and Siemens 7 × 7 and 8 × 8 fuel assembly data. CRC data has been acquired for D2 and is being considered as a source of CRC statepoints. No CRC statepoints have been evaluated at this time.

3.2.5 Washington Nuclear Plant Unit 2

Washington Nuclear Plant Unit 2 (WNP2) operated by Washington Public Power Supply System is a 1112 MWe GE class 5 BWR with 764 fuel assemblies. WNP2 is a source for Siemens 8 × 8 and 9 × 9 fuel assembly data. WNP2 is being considered as a source of CRC data. No CRC statepoints have been evaluated at this time.

3.2.6 Cooper

Cooper operated by Nebraska Public Power District is a 764 MWe GE class 4 BWR with 548 fuel assemblies. Cooper is a source for GE 8 × 8 fuel assembly data. Cooper is being considered

as a source of CRC data. No CRC statepoints have been evaluated at this time.

4.0 CRC ANALYSES RESULTS

Each CRC reactivity calculation was performed four times using four different isotope sets to describe the spent fuel. The four different isotope sets are identified as best-estimate set, principal isotope set, principal actinide set, and actinide-only set. These isotope sets are shown in Tables 4.1.1-5 through 4.1.1-8, respectively. The best-estimate isotope set included all of the isotopes from the ORIGEN-S output for which MCNP continuous-energy cross section data tables were available. The k_{eff} results obtained from using the other three isotope sets serve to demonstrate the integral reactivity worth of the excluded isotopes with respect to the best-estimate k_{eff} result.

4.1 PWRs

4.1.1 Crystal River Unit 3, Three Mile Island Unit 1, McGuire Unit 1, and Sequoyah Unit 2

Table 4.1.1-1 contains the best-estimate CRC statepoint k_{eff} results for CR3, TMI1, MCG1, and SEQ2. The best-estimate calculations were performed with 85 fuel isotopes (as shown in Table 4.1.1-5), multiple axial zones in the fuel to account for axial profiles (18 zones for CR3 and TMI1, 16 zones for MCG1 and SEQ2), and reactor core-follow information (References 11, 12, 13, and 19) obtained with NRC accepted models for reload licensing. The full details of the analyses are documented in References 14, 15, 16, and 20.

Table 4.1.1-2 contains the principal isotope CRC statepoint k_{eff} results for CR3, TMI1, MCG1, and SEQ2. The principal isotope calculations were performed using only the 29 principal isotopes (as shown in Table 4.1.1-6) and oxygen to represent the depleted fuel. All other model characteristics were the same as in the best-estimate calculations.

Table 4.1.1-3 contains the principal actinide CRC statepoint k_{eff} results for CR3, TMI1, MCG1, and SEQ2. The principal actinide calculations were performed using only the 14 principal actinides (as shown in Table 4.1.1-7) and oxygen to represent the depleted fuel. All other model characteristics were the same as in the best-estimate calculations.

Table 4.1.1-4 contains the actinide-only CRC statepoint k_{eff} results for CR3, TMI1, MCG1, and SEQ2. The actinide-only calculations were performed using only 10 actinides (as shown in Table 4.1.1-8) and oxygen to represent the depleted fuel. All other model characteristics were the same as in the best-estimate calculations.

Table 4.1.1-1 CRC Best-Estimate k_{eff} Results for CR3, TMI1, MCG1, and SEQ2

Case	Cycle	Cycle Length to Statepoint (EFPD)	Enrichments (wt% U-235 in U of UO_2)	Downtime (days)	k_{eff} [σ]
Crystal River Unit 3					
1	1A	0.0	(1.93, 2.54, 2.83)	0.0	0.99601 [0.00043]
2	1B	268.8	1.93, 2.00, 2.54, 2.83	195.3	0.99285 [0.00040]

Case	Cycle	Cycle Length to Statepoint (EFPD)	Enrichments (wt% U-235 in U of UO_2)	Downtime (days)	k_{eff} [σ]
3	1B	411.0	1.93, 2.00, 2.54, 2.83	14.8	0.99502 [0.00046]
4	2	0.0	2.54, (2.64), 2.83	97.0	0.99282 [0.00044]
5	3	0.0	2.54, (2.62), 2.64, 2.83	164.0	0.99408 [0.00045]
6	3	168.5	2.54, 2.62, 2.64, 2.83	16.8	0.99304 [0.00045]
7	3	250.0	2.54, 2.62, 2.64, 2.83	12.3	0.99073 [0.00045]
8	4	0.0	2.62, (2.62), 2.64, (2.95)	73.0	0.99134 [0.00047]
9	4	228.1	2.62, 2.64, 2.95	15.2	0.99152 [0.00046]
10	4	253.0	2.62, 2.64, 2.95	24.0	0.99603 [0.00047]
11	5	0.0	2.62, 2.64, 2.95, (2.95, 3.29)	127.0	0.99479 [0.00047]
12	5	388.5	2.62, 2.64, 2.95, 3.29	5.0	0.99805 [0.00045]
13	6	0.0	2.62, 2.64, 2.95, 3.29, (3.49)	163.0	0.99561 [0.00043]
14	6	96.0	2.62, 2.64, 2.95, 3.29, 3.49	168.9	0.99579 [0.00047]
15	6	400.0	2.62, 2.64, 2.95, 3.29, 3.49	10.4	0.99273 [0.00044]
16	7	0.0	2.54, 2.62, 2.64, 3.29, 3.49, (3.84)	113.0	0.99324 [0.00052]
17	7	260.3	2.54, 2.62, 2.64, 3.29, 3.49, 3.84	18.9	0.99083 [0.00045]
18	7	291.0	2.54, 2.62, 2.64, 3.29, 3.49, 3.84	39.5	0.99222 [0.00049]
19	7	319.0	2.54, 2.62, 2.64, 3.29, 3.49, 3.84	109.5	0.98993 [0.00047]
20	7	462.3	2.54, 2.62, 2.64, 3.29, 3.49, 3.84	2.2	0.99321 [0.00042]
21	7	479.0	2.54, 2.62, 2.64, 3.29, 3.49, 3.84	7.2	0.99247 [0.00046]
22	8	0.0	1.93, 2.62, 3.29, 3.49, 3.84, (3.94)	99.0	0.99039 [0.00043]
23	8	97.6	1.93, 2.62, 3.29, 3.49, 3.84, 3.94	15.5	0.99021 [0.00046]
24	8	139.8	1.93, 2.62, 3.29, 3.49, 3.84, 3.94	6.2	0.99063 [0.00049]
25	8	404.0	1.93, 2.62, 3.29, 3.49, 3.84, 3.94	44.4	0.99054 [0.00042]
26	8	409.6	1.93, 2.62, 3.29, 3.49, 3.84, 3.94	4.9	0.99067 [0.00047]
27	8	515.5	1.93, 2.62, 3.29, 3.49, 3.84, 3.94	7.6	0.98772 [0.00044]
28	9	0.0	1.93, 3.84, (3.90), 3.94	75.0	0.99208 [0.00044]
29	9	158.8	1.93, 3.84, 3.90, 3.94	2.1	0.99311 [0.00050]
30	9	219.0	1.93, 3.84, 3.90, 3.94	53.1	0.99078 [0.00048]
31	9	363.1	1.93, 3.84, 3.90, 3.94	1.6	0.98837 [0.00048]
32	10	0.0	3.84, 3.90, 3.94, (4.167)	55.0	0.99164 [0.00052]
33	10	573.7	3.84, 3.90, 3.94, 4.167	16.4	0.98725 [0.00048]
Three Mile Island Unit 1					
1	1	0.0	(2.06, 2.75, 3.05)	0.0	1.00141 [0.00042]
2	5	0.0	2.64, 2.85, (2.85)	2420.0	0.99088 [0.00046]
3	5	114.4	2.64, 2.85	32.2	0.99162 [0.00048]
McGuire Unit 1					
1	1	0.0	(2.108, 2.601, 3.106)	0.0	0.99946 [0.00045]
2	6	0.0	2.92, 3.204,3.40, (3.60)	78.0	0.98541 [0.00050]
3	6	62.4	2.92, 3.204, 3.40, 3.60	62.7	0.98771 [0.00049]
4	7	0.0	2.92, 3.204, 3.40, 3.60, (3.75)	130.0	0.98954 [0.00047]
5	7	129.0	2.92, 3.204, 3.40, 3.60, 3.75	29.6	0.99175 [0.00046]
6	7	282.3	2.92, 3.204, 3.40, 3.60, 3.75	18.8	0.98723 [0.00049]
Sequoyah Unit 2					
1	1	0.0	(2.10, 2.60, 3.10)	0.0	0.99631 [0.00043]
2	3	0.0	2.60, 3.10,3.50, (3.60, 3.80)	81.0	0.99158 [0.00044]
3	3	210.9	2.60, 3.10,3.50, 3.60, 3.80	995.7	0.99180 [0.00050]

Table 4.1.1-2 CRC Principal Isotope k_{eff} Results for CR3, TMI1, MCG1, and SEQ2 [1]

Case Number	k_{eff} [σ]	Case Number	k_{eff} [σ]
Crystal River Unit 3			
2	1.00156 [0.00043]	18	1.00819 [0.00045]
3	1.00867 [0.00042]	19	1.00824 [0.00046]
4	1.00305 [0.00044]	20	1.01973 [0.00047]
5	1.00267 [0.00046]	21	1.01584 [0.00044]
6	1.00662 [0.00044]	22	0.99788 [0.00044]
7	1.00686 [0.00044]	23	1.00108 [0.00045]
8	0.99922 [0.00045]	24	1.00331 [0.00047]
9	1.00481 [0.00045]	25	1.01073 [0.00048]
10	1.01265 [0.00043]	26	1.01154 [0.00044]
11	1.00096 [0.00044]	27	1.01113 [0.00048]
12	1.01730 [0.00044]	28	1.00055 [0.00044]
13	1.00423 [0.00039]	29	1.01222 [0.00048]
14	1.00784 [0.00048]	30	1.00534 [0.00049]
15	1.01418 [0.00041]	31	1.01968 [0.00046]
16	1.00008 [0.00043]	32	1.00108 [0.00048]
17	1.00750 [0.00044]	33	1.01232 [0.00053]
Three Mile Island Unit 1			
2	1.00048 [0.00046]	3	1.00443 [0.00047]
McGuire Unit 1			
2	0.99428 [0.00043]	5	1.00565 [0.00043]
3	1.00013 [0.00045]	6	1.00786 [0.00047]
4	0.99755 [0.00049]		
Sequoyah Unit 2			
2	1.00109 [0.00047]	3	1.00679 [0.00046]

[1] The general characteristics of each case are shown in Table 4.1.1-1. Case number one always refers to the beginning-of-life of the reactor. Therefore, no depleted fuel is present in the CRC k_{eff} calculation.

Table 4.1.1-3 CRC Principal Actinide k_{eff} Results for CR3, TMI1, MCG1, and SEQ2 [1]

Case Number	k_{eff} [σ]	Case Number	k_{eff} [σ]
Crystal River Unit 3			
2	1.03798 [0.00045]	18	1.06102 [0.00046]
3	1.05566 [0.00043]	19	1.06279 [0.00047]
4	1.03870 [0.00046]	20	1.08088 [0.00045]
5	1.03623 [0.00041]	21	1.08063 [0.00047]
6	1.05296 [0.00048]	22	1.02540 [0.00046]
7	1.05823 [0.00045]	23	1.04021 [0.00046]
8	1.02417 [0.00045]	24	1.04377 [0.00047]
9	1.05470 [0.00044]	25	1.07206 [0.00046]
10	1.06342 [0.00044]	26	1.07188 [0.00049]
11	1.02244 [0.00046]	27	1.08122 [0.00044]
12	1.07724 [0.00047]	28	1.02577 [0.00049]
13	1.03381 [0.00046]	29	1.05266 [0.00056]
14	1.05189 [0.00048]	30	1.05564 [0.00047]
15	1.08174 [0.00044]	31	1.07823 [0.00047]
16	1.02339 [0.00047]	32	1.02682 [0.00049]
17	1.05652 [0.00050]	33	1.08862 [0.00047]

Three Mile Island Unit 1			
2	1.04072 [0.00045]	3	1.05157 [0.00048]
McGuire Unit 1			
2	1.02462 [0.00048]	5	1.04895 [0.00045]
3	1.04113 [0.00046]	6	1.06839 [0.00046]
4	1.02374 [0.00052]		
Sequoyah Unit 2			
2	1.03341 [0.00047]	3	1.06657 [0.00044]

[1] The general characteristics of each case are shown in Table 4.1.1-1. Case number one always refers to the beginning-of-life of the reactor. Therefore, no depleted fuel is present in the CRC k_{eff} calculation.

Table 4.1.1-4 CRC Actinide-Only k_{eff} Results for CR3, TMI1, MCG1, and SEQ2 [1]

Case Number	k_{eff} [σ]	Case Number	k_{eff} [σ]
Crystal River Unit 3			
2	1.03938 [0.00047]	18	1.06363 [0.00042]
3	1.05747 [0.00045]	19	1.06471 [0.00048]
4	1.04030 [0.00046]	20	1.08834 [0.00045]
5	1.03857 [0.00045]	21	1.08730 [0.00044]
6	1.05471 [0.00044]	22	1.02677 [0.00047]
7	1.06013 [0.00046]	23	1.04016 [0.00047]
8	1.02594 [0.00043]	24	1.04589 [0.00048]
9	1.05781 [0.00043]	25	1.07723 [0.00046]
10	1.06662 [0.00048]	26	1.07583 [0.00049]
11	1.02335 [0.00045]	27	1.08735 [0.00046]
12	1.08115 [0.00043]	28	1.02711 [0.00046]
13	1.03525 [0.00047]	29	1.05526 [0.00046]
14	1.05342 [0.00044]	30	1.05930 [0.00047]
15	1.08777 [0.00044]	31	1.08359 [0.00049]
16	1.02541 [0.00051]	32	1.02773 [0.00052]
17	1.05964 [0.00044]	33	1.09621 [0.00050]
Three Mile Island Unit 1			
2	1.04243 [0.00046]	3	1.05363 [0.00044]
McGuire Unit 1			
2	1.02664 [0.00047]	5	1.05303 [0.00047]
3	1.04387 [0.00042]	6	1.07259 [0.00043]
4	1.02556 [0.00050]		
Sequoyah Unit 2			
2	1.03468 [0.00047]	3	1.07018 [0.00042]

[1] The general characteristics of each case are shown in Table 4.1.1-1. Case number one always refers to the beginning-of-life of the reactor. Therefore, no depleted fuel is present in the CRC k_{eff} calculation.

Table 4.1.1-5 Isotope Set from which Best-Estimate MCNP Depleted Fuel Compositions are Developed

H-3	Ru-101	Nd-147	Gd-155	Np-238
He-4	Ru-103	Nd-148	Gd-156	Pu-237
Li-6	Rh-103	Pm-147	Gd-157	Pu-238
Li-7	Rh-105	Pm-148	Gd-158	Pu-239
Be-9	Pd-105	Pm-149	Gd-160	Pu-240
O-16	Pd-108	Sm-147	Ho-165	Pu-241
As-75	Ag-107	Sm-149	Th-232	Pu-242
Kr-80	Ag-109	Sm-150	Pa-233	Am-241
Kr-82	Xe-131	Sm-151	U-233	Am-242m
Kr-83	Xe-134	Sm-152	U-234	Am-243
Kr-84	Xe-135	Eu-151	U-235	Cm-242
Kr-86	Cs-133	Eu-152	U-236	Cm-243
Y-89	Cs-135	Eu-153	U-237	Cm-244
Zr-93	Ba-138	Eu-154	U-238	Cm-245
Nb-93	Pr-141	Eu-155	Np-235	Cm-246
Mo-95	Nd-143	Gd-152	Np-236	Cm-247
Tc-99	Nd-145	Gd-154	Np-237	Cm-248

Table 4.1.1-6 Isotope Set from which Principal Isotope MCNP Depleted Fuel Compositions are Developed

O-16	Nd-143	Sm-152	U-235	Pu-240
Mo-95	Nd-145	Eu-151	U-236	Pu-241
Tc-99	Sm-147	Eu-153	U-238	Pu-242
Ru-101	Sm-149	Gd-155	Np-237	Am-241
Rh-103	Sm-150	U-233	Pu-238	Am-242m
Ag-109	Sm-151	U-234	Pu-239	Am-243

Table 4.1.1-7 Isotope Set from which Principal Actinide MCNP Depleted Fuel Compositions are Developed

O-16	U-235	Np-237	Pu-240	Am-241
U-233	U-236	Pu-238	Pu-241	Am-242m
U-234	U-238	Pu-239	Pu-242	Am-243

Table 4.1.1-8 Isotope Set from which Actinide-Only MCNP Depleted Fuel Compositions are Developed

O-16	U-235	U-238	Pu-239	Pu-241	Am-241
U-234	U-236	Pu-238	Pu-240	Pu-242	

4.2 BWRs

4.2.1 Quad Cities Unit 1

Table 4.2.1-1 contains the CRC statepoint analyses results for QC1. The result reported is the best-estimate result using all the available data for the neutronics models. The full details of the analysis are documented in Reference 17.

Table 4.2.1-1 QC1 CRC Statepoint Result

Case	Cycle	Cycle Length to Statepoint (EFPD)	Average Bundle Enrichment (wt%)	Downtime (days)	Calculated k_{eff} (k_{eff} +/- σ)
1	1	0.0	(2.12)	0.0	1.00435 +/- 0.00040

4.2.2 Quad Cities Unit 2

Table 4.2.2-1 shows the CRC analysis results for the cycles 13 and 14 of the Quad Cities Unit 2 core. The details of the analysis are shown in Reference 22.

Table 4.2.2-1 CRC Results for Cycles 13 and 14 of QC2

Identifier	Cycle Length to Statepoint (EFPD)	Average Core Exposure (MWd/MTU)	k_{eff}	σ
Best Estimate				
QC2BOC13	0.00	0.00	1.02122	0.00027
QC2C13CP10	10.00	201.61	0.99525	0.00024
QC2C13CP11	123.00	2257.20	1.00254	0.00025
QC2C13CP13	325.00	6489.46	1.03404	0.00026
QC2BOC14	0.00	0.00	1.01929	0.00028
QC2C14CP16	211.00	4238.45	1.00176	0.00036
Principle Isotope				
QC2BOC13	0.00	0.00	1.03132	0.00027
QC2C13CP10	10.00	201.61	1.00927	0.00027
QC2C13CP11	123.00	2257.20	1.01553	0.00027
QC2C13CP13	325.00	6489.46	1.05023	0.00027
QC2BOC14	0.00	0.00	1.02855	0.00027
QC2C14CP16	211.00	4238.45	1.01582	0.00031
Principle Actinide				
QC2BOC13	0.00	0.00	1.07578	0.00027
QC2C13CP10	10.00	201.61	1.06023	0.00025
QC2C13CP11	123.00	2257.20	1.06666	0.00028
QC2BOC14	0.00	0.00	1.06990	0.00030
QC2C14CP16	211.00	4238.45	1.06440	0.00033
Actinide Only				
QC2BOC13	0.00	0.00	1.07690	0.00025
QC2C13CP10	10.00	201.61	1.06113	0.00026
QC2C13CP11	123.00	2257.20	1.06760	0.00027
QC2C13CP13	325.00	6489.46	1.10745	0.00024
QC2BOC14	0.00	0.00	1.07065	0.00027
QC2C14CP16	211.00	4238.45	1.06615	0.00032

5.0 CONCLUSIONS

The data reported herein are acceptable for quality affecting activities and for use in analyses affecting procurement, construction, or fabrication. The classification analysis for the repository (which includes the waste package) carries TBV-228 because of the preliminary status of the basis for the MGR design. This report conservatively assumes that the resolution of TBV-228 will find the waste package to be quality affecting; consequently, use of any of the data reported herein does not need to carry TBV-228.

6.0 REFERENCES

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