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DEPLETION CALCULATIONS FOR THE MCCLELLAN NUCLEAR RADIATION CENTER

BY

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Depletion Calculations for the McClellan Nuclear Radiation Center

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Abstract

Depletion calculations have been performed for the McClellan reactor history from January 1990 through August 1996. A database has been generated for continuing use by operations personnel which contains the isotopic inventory for all fuel elements and fuel-followed control rods maintained at McClellan.

The calculations are based on the three-dimensional diffusion theory code REBUS-3 which is available through the Radiation Safety Information Computational Center (RSICC). Burnup-dependent cross-sections were developed at zero power temperatures and full power temperatures using the WIMS code (also available through RSICC). WIMS is based on discretized transport theory to calculate the neutron flux as a function of energy and position in a one-dimensional cell.

Based on the initial depletion calculations, a method was developed to allow operations personnel to perform depletion calculations and update the database with a minimal amount of effort. Depletion estimates and calculations can be performed by simply entering the core loading configuration, the position of the control rods at the start and end of cycle, the reactor power level, the duration of the reactor cycle, and the time since the last reactor cycle. The depletion and buildup of isotopes of interest (heavy metal isotopes, erbium isotopes, and fission product poisons) are calculated for all fuel elements and fuel-followed control rods in the MNRC inventory. The reactivity loss from burnup and buildup of fission product poisons and the peak xenon buildup after shutdown are also calculated. The reactivity loss from going from cold zero power to hot full power can also be calculated by using the temperature-dependent, burnup-dependent cross-sections. By calculating all of these reactivity effects, operations personnel are able to estimate the total excess reactivity necessary to run the reactor for the given cycle. This method has also been used to estimate the worth of individual control rods.

Using this approach, fuel management and core loading can be optimized such that each individual fuel element and fuel-followed control rod is used to its full potential before being replaced with fresh fuel. This fuel management strategy allows a significant cost saving to MNRC by reducing fuel replacement costs and maximizing the usefulness of each element in the inventory.

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Introduction

The McClellan Nuclear Radiation Center (MNRC) reactor is a TRIGA type reactor with four radiography beams. It was formerly known as the Stationary Neutron Radiography System (SNRS). The MNRC provides the capability to radiograph a wide variety of objects ranging in size from small parts up to aircraft wings. The facility includes four radiography bays with one reactor beam directed into each bay tangentially from the reactor core. Each bay has the necessary equipment to position parts for inspection.

To continue to achieve a high utilization factor for the facility, additional missions are being identified. These missions include, but are not necessarily limited to, the examination of advanced design turbine blades, boron neutron capture therapy research, and doping of silicon-based solid state materials by neutron irradiation to improve their properties. For these expanded missions, an increased neutron flux is required. To achieve this, the reactor has been upgraded from 1 MW to 2 MW.

Operation at 2 MW requires a much higher power density per fuel element which translates into significantly more burnup on average per reactor cycle. Correspondingly, the peak element power is much higher and this leads to a larger gradient in element burnup rates. Because of these much higher burnup rates and a heavier duty cycle than previous operations at 1 MW, it is necessary to track burnup of the fuel and its effects on reactivity. By accounting for the effects of burnup, fuel management can be performed. By varying core loading configurations, the lifetime of each fuel element can be extended by structuring the core such that fuel elements can be maximally depleted while still allowing ample excess reactivity to complete intended operating cycles.

A method has been developed to allow operations personnel to perform these calculations simply and easily. This method is based on the REBUS-3 code (ref. 1) and the WIMS code (ref. 2), which are both available through the Radiation Safety Information Computational Center (RSICC). The system requires a minimal amount of input to calculate the burnup by element and the effects on reactivity for a given core configuration. Armed with this depletion calculation system, operations personnel can tailor core loadings for specific purposes and be assured that there is enough excess reactivity to complete the reactor cycle.

This method was used to calculate the burnup for all elements in the MNRC inventory for reactor runs from January, 1990 through August, 1996. To do this, the reactor runs and core loadings were simplified into five different core configurations, as shown in Table 1. The core modelling was established as five different core loadings and reactor runs rather than modelling

Table 1 MNRC Operating History								
				Estimated Reactivity Loss ($)	Rod Positions (start-end)			
Core	Dates	Power Level	Burnup (MW-day)		Transient Rod	Shim 1	Shim 2	Reg Rod
1	1/90 - 7/90	250 kW	5.21	0.42	641-770	646-770	646-770	648-748
2	7/90 - 4/93	250 kW	36.11	0.43	555-720	555-720	555-720	564-720
3	5/93 - 10/94	250 kW	32.51	0.87	574-709	574-709	574-710	578-709
4	12/94 - 8/95	1 MW	69.66	1.50	627-800	999-999	627-800	626-796
5	8/95 - 8/96	1 MW	125.93	1.42	573-803	999-999	573-802	566-692

every single core loading and reactor run because it was not felt that the accuracy of the depletion estimates would suffer from these assumptions. Most of the reactor runs to date had very little burnup and were at lower power levels, so any error introduced from these assumptions would be slight or unnoticeable.

The REBUS and WIMS models used to develop this calculational system are discussed followed by the results for cores 1 through 5. The calculational method developed for operations personnel is then described followed by a summary of the operating experience with this system and the accuracy of the results. Conclusions based on the operating experience and applicability of this system to other facilities completes the paper.

Cross-Section Generation

Cross-sections were developed using the WIMS code (ref. 2). WIMS is a general lattice cell program which uses discretized transport theory to calculate the neutron flux as a function of energy and position in a one-dimensional cell. It is a widely accepted code that has been used extensively throughout the world for power and research reactor lattice physics analysis. Each fuel element and fuel-followed control rod type was modelled in WIMS. The model for a fuel element is shown in Figure 1. WIMS was then run to generate cross-sections for each isotope within the unit cell as a point-wise function of burnup over the range from fresh fuel to 75% depletion of the U-235. This range was developed by running numerous small incremental burn steps and generating the cross-sections for each step.

A generalized core geometry was then modelled using REBUS-3 (ref. 1). REBUS-3 is a system of programs designed for the analysis of fast reactor fuel cycles. The code solves two basic types of analysis problems: the infinite-time, or equilibrium, conditions of a reactor operating under a fixed fuel management scheme, and the explicit cycle-by-cycle, or non-equilibrium, operation of a reactor under a specified repetitive or non-repetitive fuel management problem. Admittedly, a TRIGA reactor is not a fast reactor, however, the code has been used

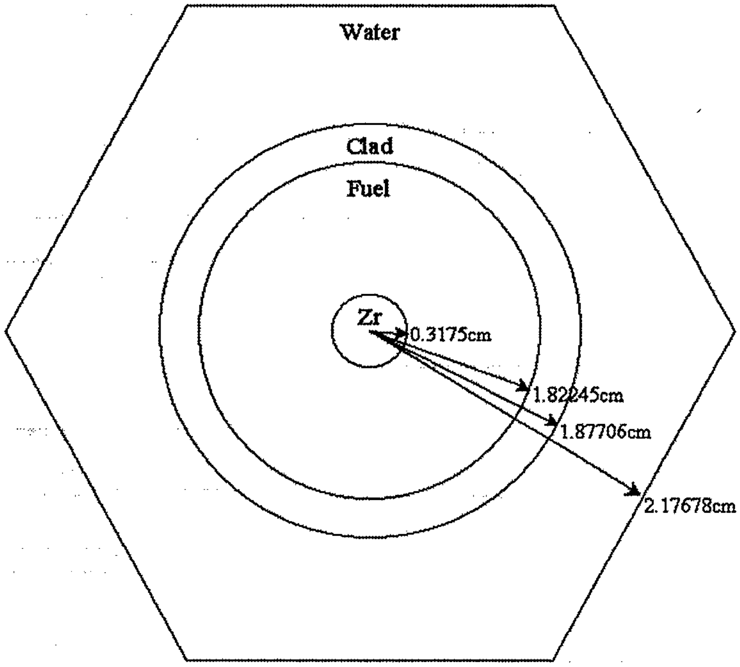

FIGURE 1 - TRIGA Fuel Cell

successfully to analyze thermal reactors. The code uses a three-dimensional nodal diffusion theory solver to calculate the flux within each region in the core. Many useful features of REBUS-3 are employed in the calculational method including burnup-dependent cross-sections, movable control rods, and variable cycle lengths and powers. These features are discussed in more detail in the section describing the calculational method.

To account for burnup-dependent cross-sections in REBUS-3, a polynomial function is generated for each cross-section of interest based on the values listed in the cross-section library. To obtain a cross-section at a given time, REBUS-3 tracks a specified isotope (U-235 in this case) and determines the cross-sections of interest at this U-235 concentration based on the polynomial fit to the point-wise cross-section values generated by WIMS.

Several cases were run to determine the effect that burnup-dependent cross-sections have on the reactivity calculations. To do this, 8.5/20 fuel (20 wt % uranium enriched to 8.5 wt % U-235) was loaded in an infinite array and burned for an extended period of time. Static cross-sections were used in one case and burnup-dependent cross-sections in the other. The results are shown in Figure 2. Burn step 10 corresponds to approximately 56 % depletion of the U-235. The reactivity loss is much faster when burnup-dependent cross-sections are used. This is due to the shift in spectra due to the burnout of U-235 and buildup of poisons and plutonium isotopes.

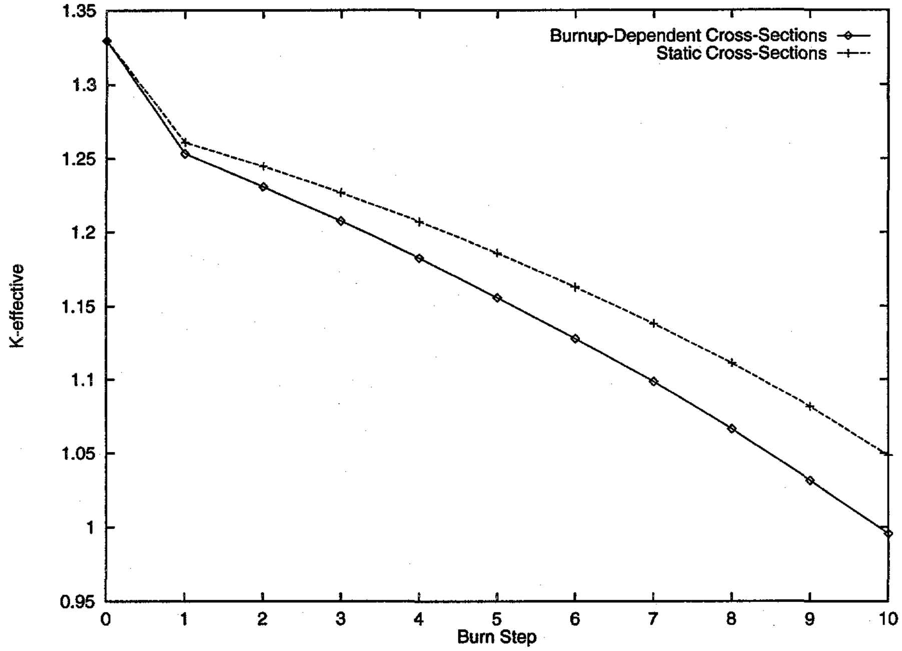

FIGURE 2 - Reactivity for 8.5/20 Fuel

A case was then run comparing the effects on reactivity for the 20/20 fuel element type. Once again, an infinite array of 20/20 elements was modelled. The result is shown in Figure 3. Burn step 18 correlates to approximately 40% depletion of the U-235. The effect is drastic for high burnup fuels. The case shows the nice effect of the burnable poison erbium. The erbium serves to counteract the effects of burnup and level out the reactivity. The erbium burns out faster than the U-235, and hence, an initial increase in reactivity is observed after the xenon has reached equilibrium (burn step 1). This is most notable for the static cross-sections. The same trend was also observed with the burnup-dependent cross-sections, however, to a much lesser degree. It is apparent that the erbium does not work as well or as long when the effects of burnup are considered. The erbium still serves to level out the reactivity. However, this reduced effectiveness is due to the shift in spectrum from the depletion of U-235, buildup of plutonium, and buildup of fission product poisons in the fuel. As the fuel is depleted the spectrum is shifted away from the lower erbium resonances which results in a loss in the effectiveness of the erbium poison and an increasingly negative effect on the reactivity. The reactivity drops off rapidly after burn step 8 (about 20% depletion of the U-235). At this point, the erbium is depleted and the reactivity effect is primarily the result of the depletion of the U-235.

Core Model

The initial calculations and the calculational method developed for operations personnel are based on a hexagonal array core model which matches the new MNRC grid spacing.

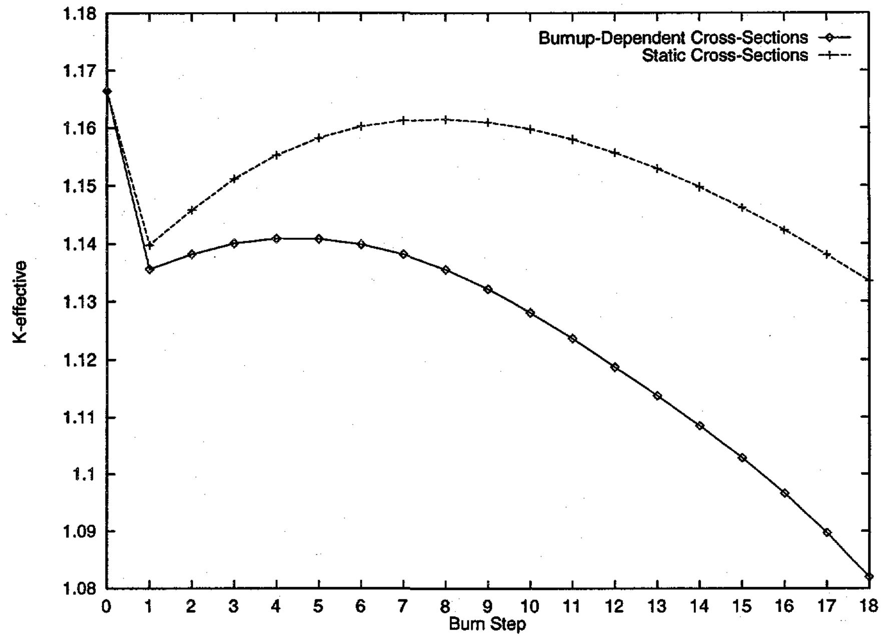

FIGURE 3 - Reactivity of 20/20 Fuel

Core loadings 1 through 5 were also modelled with this hexagonal geometry although the old grid plate used a circular array with non-uniform spacing throughout the core. Each fuel element and fuel-followed control rod was homogenized into ten different regions over its axial length. The fueled portion of each element (15 inches in length) was modelled as five different fuel regions. Accounting for the exact burnup profile for each element would have been ideal, however, it was not practical as computer memory and execution time were limiting factors. Five different fuel regions were chosen based on a scoping study involving several infinite array problems. For an infinite array of 8.5/20 fuel, the fuel portion of the elements were divided into 1, 3, 5, and 15 regions. The results are shown in Figure 4 for an extended run. Time step 12 corresponds to approximately 48% depletion of the U-235. From Figure 4, the results for 1 and 3 axial regions deviate from the result for 15 axial regions as the axial burnup profile becomes a factor. This is as expected. However, the results for 5 axial regions follow those for the 15 region case remarkably closely. Therefore, five axial regions were selected as a good trade-off between execution time and accuracy.

Each of the five fueled regions for each element is uniquely identified as a separate material type. These materials are then manipulated based on each reactor cycle. Isotopes of interest are built up (such as fission product poisons and plutonium) and others are depleted (such as U-235 and erbium isotopes). In addition, isotopes in the database are decayed from cycle to cycle. Figure 5 shows the core loading for core configuration 5. Every position in the core must be filled with a material for each reactor cycle. Therefore, besides the fuel elements and fuel-followed control rods, standard materials (boron carbide control rods, graphite dummy elements, irradiation positions, rabbit tubes, source locations, etc.) are included in the database as well.

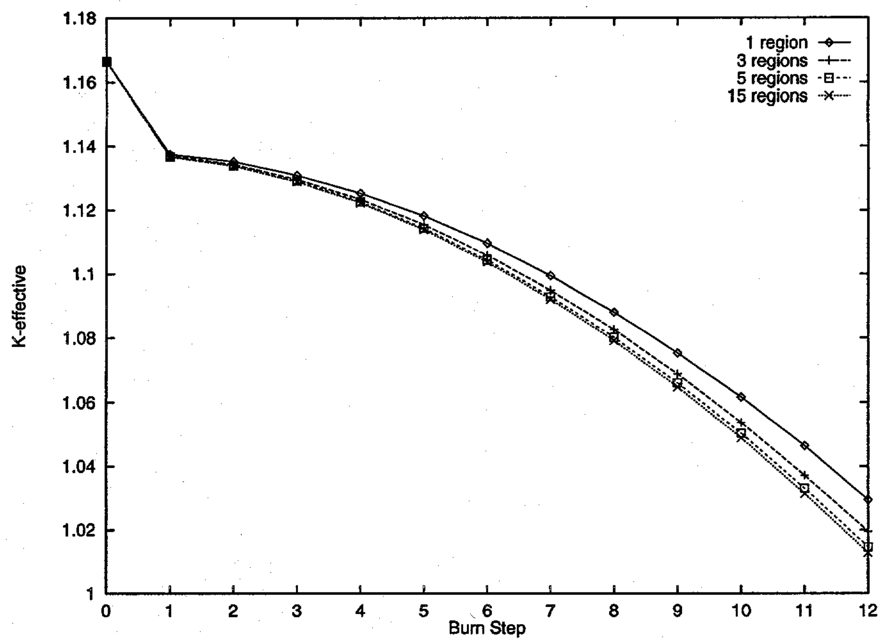

FIGURE 4 - Effect of Axial Regions on Reactivity

Depletion Results

As mentioned previously, the MNRC operating history was modelled as five different core configurations as shown in Table 1. The modelling of the day-to-day operation of the reactor was not attempted. Each core model was run for one continuous cycle at full power for a duration that would result in the equivalent burnup (in MW-days) to those listed in Table 1. In addition, a long shutdown time was chosen between the different cores to allow the xenon and samarium chains to decay.

The results for the five core configurations are shown in Table 2. It is immediately obvious from Table 2 that the reactivity loss estimates reported from the operations log books are not consistent with the burnup for the five core loadings. For example, the reactivity losses for cores 2 and 3 were noted as \$0.43 and \$0.87, respectively, although core 3 has a lower burnup in MW-days. These results are inconsistent as the cores are very similar. These inconsistencies are due to the nature of the operation. The reactor was not run for one continuous cycle, but instead for a few hours at a time each day. Therefore, when the rod positions were used to estimate the reactivity for the first and last reactor runs, fission product poisons were included to some extent. The extent is not known without reviewing the detailed operating history. In addition, the core configurations provided were only representative of the core configurations during the dates reported in Table 1 and do not necessarily reflect the actual core configurations at the start and end of each reactor cycle. Minor loading changes were made during the individual reactor runs resulting in small effects on reactivity when estimating the reactivity from the control rod positions.

Conf 5
Aug 95 - Aug 96
125.93 MWD
1MW Operation

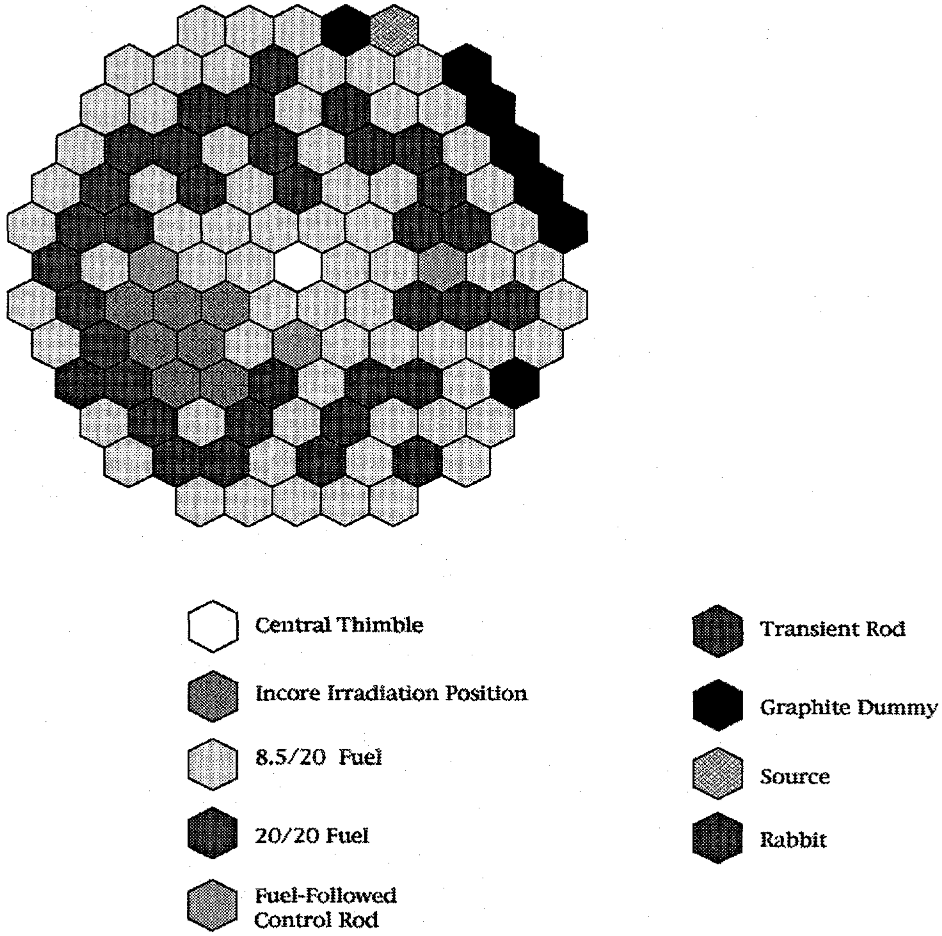

FIGURE 5 - Core Configuration 5

Table 2

Reactivity Losses from Cores 1 through 5

Core	Power Level	Burnup (MW/days)	Reactivity Loss from Table 1 ($)	Average Loss from Table 1 values (cents/MW-day)	Reactivity Loss from REBUS Calculations ($)	Average Loss from REBUS Calculations (cents/MW-day)
1	250 kW	5.21	0.42	8.06	0.132	2.53
2	250 kW	36.11	0.43	1.19	0.776	2.15
3	250 kW	32.51	0.87	2.68	0.669	2.06
4	1 MW	69.66	1.50	2.15	0.885	1.27
5	1 MW	125.93	1.42	1.13	1.517	1.20

The results reported from the REBUS calculations are much more consistent for the two different power levels. The average reactivity loss varies from 2.06 to 2.53 cents/MW-day for the 250 kW operations and varies from 1.20 to 1.27 cents/MW-day for the 1 MW operation. The results for the 1 MW operations are also in line with the GA rule-of-thumb of 1 cent/MW-day. The average reactivity loss for the 1 MW core loadings is significantly less than the average reactivity loss for the 250 kW loadings. The difference is not due to power level, but rather, it is due to fuel type. The 1 MW core loadings contain a mix of 20/20 and 8.5/20 fuel elements, whereas, the 250 kW core loadings only contain 8.5/20 fuel elements. The 20/20 fuel elements containing erbium as a burnable poison lose less reactivity as a function of burnup compared to the 8.5/20 fuel. This effect was demonstrated in Figures 2 and 3. It is expected that the average reactivity loss would be less than 1.20 cents/MW-day for a loading that contained only 20/20 fuel elements. Once the erbium is burned out, the reactivity loss as a function of burnup will be much greater for the 20/20 fuel. However, the fuel was not depleted to an extent where these effects were observable. A summary of the depletion of the fuel from the five core configurations is shown in Table 3. The 20/20 fuel was only depleted to a maximum of 4.42% (3.5% mean value) while the 8.5/20 fuel had a much higher depletion with a peak of 12.66% (6.57% mean value). These are small values for the depletion indicating that the fuel elements still have considerable life.

Calculational Method

A method has been developed so that operations personnel can perform depletion calculations simply and easily with minimal knowledge of the REBUS-3 code. This system has been developed by creating the cross-section libraries and interface files necessary to generate REBUS-3 input files, run the code, and perform post-processing functions. This allows the user to execute a problem and view a condensed version of the results, which includes the depletion by element for every element in the core, the power distribution profile by element, and the reactivity effect due to burnup, poison buildup, and peak xenon buildup after shutdown. The user can also determine the reactivity loss for going from cold start-up to hot-full power. In addition, the user can determine the reactivity worth of individual control rods and the reactivity effect from substitutions of material in the irradiation positions or other core loading changes. A method has also been included for adding new elements to the database and generating the element specific fuel regions necessary to perform depletion calculations.

The system operates on a UNIX platform and the operating file is written as a Bourne shell script. The script executes a series of FORTRAN programs to generate all of the necessary input and output files. The FORTRAN programs read the necessary information for the given reactor cycle from several data files that the user creates. These data files contain the necessary cycle specific information which includes: the core loading, the time since the last reactor cycle, the length of the reactor cycle, the power level of the reactor, and the position (in control rod units) of each control rod at the start and end of the reactor cycle. These are the only input parameters that are required to be entered by the user besides option choices for performing reactivity calculations and updating the database. Once these are entered, the script can be executed by the user by a single command line execution.

The script generates several output files of interest to MNRC personnel. The output files include a reactivity file, a burnup file, and a power profile file. The reactivity file includes the k-effective values for all cases run. Also included is the reactivity loss due to burnup effects,

Table 3 Depletion Results for Cores 1 through 5						
Core	Burnup (MW-days)	Fuel Type	Number of Elements	Cycle U-235 Burnup (%)		
				Maximum	Mean	Minimum
1	5.21	F8	75	0.333	0.211	0.138
		C8	3	0.290	0.219	0.217
2	36.11	F8	89	2.021	1.231	0.620
		C8	3	1.551	1.490	1.275
3	32.51	F8	95	1.728	1.065	0.588
		C8	3	1.317	1.229	1.173
4	69.66	F2	30	1.629	1.231	0.713
		F8	63	3.508	1.877	0.992
		C8	3	2.636	2.554	2.215
5	125.93	F2	30	2.789	2.042	1.259
		F8	70	5.906	3.198	1.709
		C8	3	4.329	4.228	3.669
Total	269.42	F2	30	4.418	3.512	1.972
		F8	95	12.657	6.572	1.679
		C8	3	9.941	9.632	8.821

Does not include fuel elements in inventory that have not been in the reactor.
Fuel Types are:
F2 - 20/20 fuel element
F8 - 8.5/20 fuel element
C8 - 8.5/20 fuel-followed control rod

xenon and samarium poisoning, and peak xenon buildup after shutdown. These values are computed from several REBUS-3 calculations within the script file.

The burnup file lists the burnup of each element in the core. The burnup is computed as a fraction of the initial heavy metal and as a fraction of the initial U-235 mass. In addition, the total burnup for each element and the incremental burnup for the reactor cycle are calculated. If the database is updated during the execution of the script, a burnup file is also created for the entire database.

The third output file is the power profile file. The power profile file lists the power (in kW) for every element in the core. In addition, the peak element, grid position, and power level are flagged. This allows the user to determine if a proposed core loading configuration is within the assumptions used in the Safety Analysis Report (SAR) regarding peak power per element. Typically, the safety analysis cases only include standard core configurations using cross-sections that do not depend on burnup. It has been found that power peaking can occur in cores containing mixed fuel types when fuel elements of different types are highly commingled with one another. So this feature is necessary to ensure that an intended core loading is within the bounds established by the SAR.

The method has been set up to allow the user to run test cases without updating the database. These test cases are typically excess reactivity calculations to determine if there is enough reactivity to operate for a given cycle. Once a cycle is completed, the user can run the same configuration and update the database to the current time. This allows the flexibility to perform both scoping calculations and inventory control calculations interchangeably while maintaining the integrity of the database and accuracy of the calculational results.

The programs contain a significant amount of error checking of the input values, run time checking for proper execution, and post-process error checking. If a fatal error is detected anywhere throughout the sequence, error messages are written and the database files will not be updated. If input errors (such as missing elements, multiple elements, invalid materials, or improper time periods) are detected, the script terminates and informs the user of the detected problem. To ensure recoverability from any point in the sequence, backup files are written throughout execution. These backup files include copies of the database files, REBUS-3 input files generated by the script, and files containing fuel region number densities at all stages in the process. In this way, unforeseen errors can be traced and corrected. In addition, these backup files allow the user to recover from unintended database manipulations.

Operations Experience

MNRC has had three different core configurations since the upgrade to 2 MW. The initial loading of a new core (Core 1) was operated primarily at 1 MW until all of the modifications necessary for 2 MW operation were in place. This core was then operated at 2 MW for a short period before shuffling the core. Core 1 was operated for a total of 11 MW-days.

Prior to reconfiguring the core, the excess was measured at \$7.77. After the core was reloaded (Core 2) the excess was measured at \$8.53 or a gain of \$0.76. Core loading cases using the database prior to Core 1 resulted in calculated k-effective values at hot full power for Core 1 and Core 2 of 1.03892 and 1.04482, respectively. The difference between the two cores was estimated to be an \$0.83 gain. The actual measurements were in good agreement with the predicted results. It should be noted that Cores 1 and 2 were mixed cores containing 8.5/20 and 20/20 fuel. The difference between the two cores was in the positioning of the 20/20 fuel (it was moved closer to the center of the core by one row) and shuffling some of the more reactive 8.5/20 fuel from the periphery of the core to the center.

After Core 2 was loaded the database was updated and the hot full power k-effective was calculated to be 1.04372. This represents a loss of 0.0011 (\$0.15) for this core loading when accounting for the effect of the Core 1 operation on the database. As mentioned, Core 1 operated for 11 MW-days, so a \$0.135 loss from burnup was expected based on previous calculations. Once again, the results were in good agreement.

Core 2 ran for 93.2 MW-days operating with a cycle of 5 days per week, 24 hours per day. It was originally intended to operate with this core loading for only 80 MW-days. Therefore, the estimates for Core 3 were based on a database that was only corrected for 80 MW-days of burnup. The calculation estimated a hot full power k-effective for Core 3 of 1.04195 or a 0.00177 (\$0.25) loss from the Core 2 estimate. The measured excess reactivity for Core 3 was

$8.30 which represents a loss of $0.23 from the Core 2 measured excess reactivity. Once again, the calculated values and estimated values were in good agreement.

The real strength of this method has been in the comparative estimates between core loading configurations. Two different core loadings can be directly compared to determine the change in reactivity from the reconfiguration. In addition, it provides an immediate check on the excess reactivity value to determine if there will be enough excess to run for a given cycle.

The worth of two of the control rods in the current configuration was also estimated from calculations. The transient rod was measured to be $2.37 and calculated to be $2.34. Shim rod 1 was measured at $2.67 and calculated to be $2.77. These two rods are in good agreement between the calculated and estimated values. However at this time, these are the only two rods that have been estimated.

Conclusions

This depletion calculation method has been shown to aid operations personnel in all aspects of fuel management operations. The system is easy to use and can be executed by operations personnel with little or no knowledge of the underlying codes REBUS-3 and WIMS. The method is intended for personnel with a nuclear engineering background who can read and interpret the input files and results from the calculations. Operations personnel can tailor core loading configurations to maximize the excess reactivity for a given reactor cycle or to maximize the lifetime of individual fuel elements. In addition, this method allows reactivity effects to be determined for core loading changes. It also allows the reactivity worth of individual control rods and removable experiments to be estimated.

This method has been employed successfully at the McClellan Nuclear Radiation Center for all of their 2 MW core loadings to date. It has been written specifically for MNRC use; however, it could easily be adapted for other TRIGA reactors. To modify this method for use at other TRIGA facilities would require changes to the coding to potentially allow a different grid plate configuration and grid position labels. In addition, cross-sections for different fuel types may have to be generated. A new database would have to be created and benchmark calculations would have to be performed.

The small capital cost of this depletion calculation system would easily be recouped in replacement fuel cost savings from maximizing the useful life of every fuel element in inventory. An added benefit is the ease of operation of the system, requiring fewer man-hours to assess core loading changes and make predictions prior to actual operation.

Acknowledgements

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