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THREE DIMENSIONAL DEPLETION ANALYSIS FOR THE "AS BUILT" FSV INITIAL CORE

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ABSTRACT

During fabrication of the initial FSV core fuel elements, some difficulties with assayed quantities of uranium and thorium were encountered. This problem became apparent when about 40% of the total fuel rod requirement was in some state of production and about 10% of the fuel elements had been assembled. As a result, some of the specified fuel blends were nearing completion and the uranium and/or thorium was 3 to 4% too high or low, and some of the already assembled fuel elements were outside the loading tolerances given in the fuel specification.

Resolution of this problem with the minimum perturbation to the fuel production and no loss of integrity on the fuel performance was achieved in the following manner:

1. Some already assembled fuel elements had to be unloaded and reassembled to meet the fuel loading requirements.
2. Some fuel elements were accepted for use after the in-core service limit was downgraded to a lower value (requiring a special in-core location).
3. A large number of the fuel elements yet to be assembled were identified for selective in-core locations.
4. The remainder of the fuel rod fabrication was completed with carefully controlled and in some cases specially specified fuel rod uranium and thorium loading requirements. In this manner, the total fuel blend loadings were brought within the required

loading tolerances. These specially specified fuel rods were then either homogenized in fuel elements to achieve correct fuel element loadings or were located in fuel elements identified for specific in-core locations.

A summary of the "as built" initial core fuel element loadings is as follows:

1. The total core uranium loading is about 0.1% high.
2. The total core thorium loading is about 0.4% low.
3. All but two of the required 13 fuel blends are within 1.0% of the specified metal loadings and the maximum blend deviation is 1.4% low.
4. The calculated power distribution for the as-built core using conventional design methods is within the limits given for the reference design.
5. The calculated core excess reactivity is slightly higher and the shutdown margin slightly lower than the reference design. One additional burnable poison rod has been selectively added to 144 fuel elements to make the expected performance consistent with the reference design.

Altogether, of the 1482 fuel elements required for the initial core assembly, 736 were identified for specially selected in-core locations. Since the analyses used to select the in-core locations and the specially specified fuel rod loadings consisted mostly of one dimensional perturbation calculations, it was considered desirable that the acceptability of the initial core performance be confirmed with a multidimensional depletion analysis. A detailed description of that analysis and the calculated

results is the subject of this report. It was considered important that each of the fuel elements in the core assembly was represented explicitly.

In order to adequately describe each of the fuel elements explicitly in the core assembly, that analysis was made using a three-dimensional diffusion theory code. A special data handling package that would initially load the fuel elements in the core and then perform a fuel depletion calculation was used. Fluxes as calculated from the three-dimensional analysis were input to the data handling package. Explicit material compositions using the as-built fuel and material loadings available from fuel element and burnable poison rod production were also used.

Various simulated fuel element loading patterns were investigated and in all of them the final loading for both the six axial core layers and the 37 refueling regions was generally within 1% of the loading specified in the reference design.

The results of the three-dimensional depletion analysis confirm the acceptability of the initial core fuel elements and ensure that the core performance for the as-built core assembly is consistent with that of the reference design. In addition, this analysis showed good correlation with results using less explicit design methods and increased the confidence in the use of those methods in determining the design performance.

Some significant conclusions from that analysis are:

1. The overall power distribution and core reactivity were within the design limits.
2. The predicted critical control rod configuration using previously calculated rod worths agreed closely with those obtained in the analysis, including partially inserted control rod groups.

3. It became evident that the reference control rod withdrawal sequence had to be modified to ensure that the calculated power peaking factors were within the limits given in the Technical Specifications for all possible operating control rod configurations. A new sequence was investigated and found to be acceptable.
4. The excess reactivity at the beginning of the cycle was about $0.005 \Delta k$ higher than in the design analysis. In addition, the maximum worth control rod in the cold, all-rods-in configuration was about $0.007 \Delta k$ higher, so the shutdown margin was reduced by a little more than $0.01 \Delta k$. Since there was still sufficient margin, an evaluation was made to determine whether the condition should be accepted or some burnable poison rods added to some fuel elements. By adding burnable poison rods to the buffered fuel elements in the regions of high worth, both the excess reactivity and the calculated maximum worth control rod were reduced so that the shutdown margin was the same as for the design and as stated in the FSAR. If this reactivity shim is added prior to core loading, the margin for both the excess reactivity and the core shutdown is nearly equal and the probability of having to add shim after core loading is reduced. Consequently, it was decided to add one burnable poison rod to each of 144 fuel elements that are in high worth fuel regions and which are located at the side reflector interface.
5. The axial power distribution throughout the first year of operation remained stable with very little difference between the beginning and end of cycle.

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1. INTRODUCTION AND SUMMARY

1.1. PURPOSE OF ANALYSIS

During fabrication of the initial core fuel elements, some difficulties with assayed quantities of uranium and thorium were encountered. These difficulties resulted from changes in the fuel rod standards used in assay equipment calibration and in the techniques employed for assaying fuel particles and fuel rods. As a result the apparent values for the average metal loadings for some fuel rods and fuel elements changed.

This problem first became apparent when about 40 percent of the total fuel rod requirement was in some state of production. Fuel blends 4, 6 and 8 were most directly affected because fuel rod production in these blends was further along and because they experienced greater changes in measured metal loadings. Blend 4 was 95 percent complete with an average uranium loading 3-4 percent high. Blends 6 and 8 were about 60 percent complete with average uranium loadings about 3 percent high. For these blends some already-assembled fuel elements were outside the tolerances given in the fuel specification.

A study was undertaken at this time to make recommendations on the disposition of already-fabricated fuel and adjustments for the remainder of fuel fabrication. This study, which comprises a large part of Reference 1, focused on utilizing, as much as possible, already-fabricated fuel without compromising the performance of the core. A variety of adjustments were considered and used in some instances, but the most successful method was the imposition of a layer location on fuel elements. By use of this additional core assembly requirement, a distribution of high metal and low metal load fuel elements was obtained that assured that power perturbations would be small and localized and that temperature perturbations

would be small and confined to axial layers where temperatures are nominally low.

Altogether, of the 1482 fuel elements required for the initial core assembly, 736 were identified for pre-selected core locations. Since the study mentioned above consisted mostly of one dimensional perturbation analysis, it was considered desirable that a multidimensional depletion analysis be made to confirm the acceptability of the initial core performance. It was considered important that each of the fuel elements in the core assembly could be represented explicitly. The results of that three dimensional analysis is the subject of this report.

1.2. GENERAL OUTLINE FOR ANALYSIS

The following is a general outline of the 3-D analysis performed:

1. Obtain mass data for each of the fuel elements from the RODS program.⁽²⁾ This data included the element identification, the loadings for carbon, silicon, uranium and thorium, and the number of fuel rods.
2. Write a data handling program to:
  - a. Load each fuel element into the core assembly in a specified array.
  - b. Compute the appropriate nuclear parameters for each element, such as atom densities, C/U ratios, thorium density in the fuel rod, burnable poison density, etc.
  - c. Determine the effective microscopic cross section for each nuclide in each fuel element with INTERP.⁽³⁾

- d. Combine cross section and atom density data to obtain macroscopic cross sections for each diffusion region which includes burnable poison rods and control rods.
  - e. Generate axial averaged data for each fuel column for use with either GAUGE⁽⁴⁾ or BUGTRI⁽⁵⁾ burnup codes.
3. Modify the 3-D GATT⁽⁶⁾ program so that the convergence for a specific problem is accelerated, appropriate power edits are made available on a power tape, more than 1100 regions are acceptable, data necessary for a burnup analysis is written on tape, and converged fluxes can be saved on tape for use as input to a later problem.
  4. Develop a burnup package to take input from a 3-D GATT calculation combined with additional microscopic data for the core regions, perform a depletion analysis and generate macroscopic data for use in a subsequent GATT analysis.
  5. Obtain "As Built" GAUGE data and do sufficient analysis to show that results are consistent with the design analysis. This included a comparison of the initial excess reactivity, radial power distribution, and some depletion results. Run a degenerate GATT problem with GAUGE library and compare to GAUGE results to debug the modified GATT package.
  6. Perform a three dimensional depletion analysis for the first year of operation. Use best estimated control rod patterns based on previously calculated rod worth and excess reactivity from GAUGE depletion studies. Make adjustments to the control rod pattern as necessary.

7. Perform BUGTRI-RZ analysis for the as built core cold fully rodded, cold unrodded, and hot partially rodded - essentially the same as the beginning of cycle GATT calculation. Determine control rod worth and compare BOC power distribution with that from 3-D GATT calculation.
8. Perform GAUGE analysis for the as built core to compare directly with that from the 3-D GATT calculation. Mockup partially inserted control rods with reduced control rod atom densities. Compare excess reactivity and end of cycle atom densities. Compare radial power distributions between GATT and GAUGE.

1.3. SUMMARY OF RESULTS

Using the as built fuel and material loadings available from fuel element and burnable poison rod production, a comprehensive analysis to determine the as built initial core performance was made. This included a detailed three dimensional depletion and power distribution calculation for the first year of operation. The multidimensional depletion analysis confirmed the acceptability of the initial core fuel elements from physics considerations. In addition, the 3-D analysis showed good correlation with results using simpler methods.

A brief summary of significant conclusions from this study follows:

- A. Overall power distribution and core reactivity were within design limits.
- B. The predicted critical control rod configuration using previously calculated rod worths agreed quite closely with that obtained in the analysis. This included partially inserted control rod pairs.

- C. The expected excess reactivity at the BOC is about $.005 \Delta k$ larger than the design value and at EOC about $.002 \Delta k$ smaller. The main contribution to the increase at BOC is the reduced poison in the "as built" burnable poison rods. The decrease at EOC is apparently due to axial depletion effects, i.e., burnup in preferred high worth zones.
- D. The axial distribution throughout depletion remains extremely stable with very little difference between BOC and EOC.
- E. For partially rodded patches, the use of the average axial patch power distribution in conjunction with the calculated maximum tilt will lead to an over-estimate of the maximum fuel temperature for that region.
- F. The radial region tilt factors as calculated with GAUGE agree within a few percent of those from the 3-D GATT calculation.

As a result of this analysis, it became evident that the reference control rod withdrawal sequence required a slight modification so that the power peaking factors are within the limits of the Reactor Technical Specifications for all possible operating control rod configurations. A new sequence was investigated and found to be acceptable.

As noted above, the excess reactivity at the beginning of the cycle was about $0.005 \Delta k$ higher than in the design analysis. In addition, the maximum worth control rod in the cold, all rods in configuration, was about $0.007 \Delta k$ higher, so that the shutdown margin was reduced by a little more than $0.01 \Delta k$. Since there was still sufficient margin, an evaluation was made to determine whether the condition should be accepted or some burnable poison rods added to some fuel elements. By adding burnable poison rods to the buffered fuel elements in the regions of high worth, both the excess reactivity and the calculated maximum worth control rod were reduced so that

the shutdown margin was the same as for the design and as stated in the FSAR. If this reactivity shim is added prior to core loading, the margin for both the excess reactivity and the core shutdown is nearly equal and the probability of having to add shim after core loading is reduced. Consequently, it was decided to add one burnable poison rod to each of 144 fuel elements that are in high worth fuel regions and which are located at the side reflector interface.

A summary of some of the important calculated data is given in Tables 1 and 2. In Table 1, the agreement between the predicted critical control rod configurations and those obtained from the three dimensional analysis is shown. In Table 2, some important nuclear parameters as calculated for the design and the "as built" core are given. The effect of adding shim poison rods is also shown.

Reactor core impurities - A final estimate of the impurities present in the initial core, both hydrogen and parasitic poisons, has shown that:

Total hydrogen in fuel rods (residual hydrogen)	worth + .0024 Δk
Total hydrogen as water in fuel blocks	worth + .0022 Δk
Residual chlorine in fuel rods	worth - .0011 Δk
Parasitic boron equivalents	
Fuel rods (1.3 ppm-as assumed in design)	worth - .0016 Δk
Fuel blocks (.9 ppm-1.0 ppm assumed in design)	worth - .0032 Δk
Total, net	- .0013 Δk

Since the boron equivalent is essentially that assumed in the nuclear design and the water in the fuel blocks will be removed before power operation, the net effect is to add only about .001 Δk to the calculated excess reactivity.

TABLE 1
PREDICTED CRITICAL CONTROL ROD CONFIGURATIONS DURING FIRST
YEAR AND CALCULATED k_{eff} FROM GATT ANALYSIS

Effective Full Power Days Operation	Predicted Critical Control Rod Configuration ^(a)			Calculated k_{eff} From GATT Analysis
	Rod Groups Full In	Rod Groups Part In	Total No. Rod Pairs	
0 ^(b)	4C, 3D, 4A, 4F	1, 3B	16	1.003
2	4C, 3D	1, 4A	10	1.004
5	4C, 3D	1, 4A	10	1.000
10	4C	1, 3D	7	1.003
50	4C	1, 3D	7	1.000
100	4C	1, 3D	7	0.997
150	4C	1, 3D	7	0.996
200 ^(c)	4C	1, 3D	7	1.000
255 ^(c)		1, 4C	4	1.000
310 ^(c)		1, 4C	4	0.999

(a) The predicted critical configurations were based on group rod worths as given in the Quarterly Progress Report (Gulf-GA-A12200) and a rod withdrawal curve (Z) as calculated with FEVER for startup tests A6 and B9.

(b) Clean core, no Xe or Sm.

(c) The preselected configuration was modified to account for three-dimensional depletion results by withdrawing more control rods than initially predicted.

TABLE 2
IMPORTANT NUCLEAR PARAMETERS (INITIAL CORE)

	Reference Design (FSAR)	As-Built Loadings	As-Built With Poison Rods Added
k_{ex} (BOC, room temperature)	0.133	0.139 (no Sm)	0.136 (no Sm)
k_{ex} (hot operating, 150 days)	0.020	0.020	0.019
Control rod bank worth (Δk), room temperature	0.223	0.221	0.221
Maximum worth rod (Δk), room temperature	0.045	0.051	0.044
Maximum worth second rod (Δk)	0.043	0.051	0.043
Value used in accident analysis	0.047		
Maximum worth of two rods (Δk), refueling temperature	0.088	0.098	0.087
Minimum shutdown margin at 150 days			
Cold, one rod out	0.037	0.030	0.039
Refueling temperature, two rods out, 1-week decay	0.033	0.021	0.032
Reactivity worth of burnable poison (Δk)	0.060	0.056	0.059

2. DESCRIPTION OF INPUT

2.1. "AS BUILT" DATA FROM FUEL ELEMENTS

The expected material composition for the initial core assembly is given in Table 3. A summary of the total measured material loading for carbon, silicon, uranium and thorium for the as fabricated fuel elements is shown along with a comparison for the anticipated loadings that were used in the design analyses. In addition, some information is given on the known measured impurities and the effect on the expected beginning of cycle reactivity.

Using statistical sampling techniques, the average material composition for the fuel rods was determined and this in turn was used to determine the metal loading, uranium and thorium, for each of the fuel elements. Each element was weighed before and after the addition of the fuel rods so that the total mass of the fuel block was also known. The average silicon and carbon contents of the fuel rods were also measured, and except for the silicon content, it can be seen that the measured loadings are very close to those expected. Since the nominal thickness of the silicon carbide layer within the fissile and fertile particles is only 25 microns, it does not require much of an increase in thickness to yield the additional silicon measured in the as built core.

The net effect on the expected initial core excess reactivity due to these loading differences is also shown in Table 3. Because the differences are small, this net effect is also small, only $+0.0025 \Delta k/k$ for the total loadings. This reactivity change was estimated from sensitivity studies in which small changes in the material composition were introduced homogeneously in the reactor core.

TABLE 3
MATERIAL COMPOSITION OF AS-BUILT INITIAL CORE ASSEMBLY

Material	As-Built Loading (kg)	Nominal Load for Design Analysis (kg)	As-Built Difference (%)	Change in Core Reactivity ($\Delta k/k$)
Uranium	774.1	773.2	+0.1	+0.0005
Thorium	15,905	15,971	-0.4	+0.0015
Silicon	6,099	5,458	+10.5	-0.0010
Carbon (fuel rods)	24,048	28,600	+1.5	--
Carbon (fuel block)	132,539	131,890	+0.5	+0.0010
Residual hydrogen (ppm in fuel rods)	110	0	--	+0.002
Residual chlorine (ppm in fuel rods)	60	0	--	-0.001
Other impurities (ppm boron equivalent in fuel rods)	1<X<2	1.3	--	-0.0005
			Total	+0.0025

The core is a patch-loaded design with about one-sixth of the total fuel elements replaced at each refueling interval. By using 13 fuel blends, each having different uranium and thorium loadings, in the initial core fuel elements, a partial mockup of the equilibrium core was obtained. Those fuel elements to be removed at the first refueling interval contained a light uranium loading and a heavy thorium loading so that they mocked up depleted fuel with large concentrations of fission products, while those to be removed at the later reloadings had a higher than average uranium loading. To ensure that the power distribution in the initial core would be about the same as that calculated in the design analyses, a tolerance of $\pm 1\%$ of the total uranium and thorium loading for each fuel blend was required during the fuel element fabrication.⁽⁷⁾ The intent of this requirement was to assure that the fuel loading in each of the 37 refueling regions would generally be within 1% of that specified. A summary of the fuel loading for each of the 13 fuel blends is shown in Table 4, and it can be seen that this requirement was generally satisfied. The percent deviation for the total uranium and thorium loading from that specified for each fuel blend is shown in parentheses; for all blends except 4 and 6 this deviation was less than 1%, and for those blends it was just barely over 1%.

TABLE 4
FUEL LOADING SUMMARY FOR ALL INITIAL CORE FUEL ELEMENTS

Fuel Blend	Number of Fuel Blocks	U Weight ^(a) (kg)	Th Weight ^(a) (kg)
1	255	106.0 (+0.4%)	2882.6 (-0.8%)
2	255	80.2 (-0.3%)	2595.6 (0.0%)
3	63	39.5 (+0.8%)	631.8 (-0.7%)
4	63	28.5 (-1.4%)	541.0 (-0.6%)
5	126	89.2 (+0.4%)	1325.2 (+0.1%)
6	126	65.1 (-1.2%)	1159.0 (+0.1%)
7	162	112.1 (+0.5%)	1442.6 (-0.3%)
8	162	84.9 (+0.5%)	1277.9 (-0.7%)
9	63	36.3 (+0.4%)	717.9 (-0.3%)
10	63	25.8 (+0.1%)	594.3 (-0.8%)
11	54	32.2 (+0.3%)	550.0 (+0.2%)
12	54	23.7 (-0.2%)	471.6 (-0.5%)
13	36	50.8 (+0.4%)	1724.4 (-0.9%)

(a) Percent deviation in blend loading from specification is shown in parantheses.

2.2. AS BUILT BURNABLE POISON RODS

Based on the results of sample analysis documented in Reference 8, the geometrical description and boron density for the "as built" burnable poison rods for use with the initial FSV core was obtained and is shown in Table 5.

The Type I poison rods were selected as the reference to achieve the reactivity control desired during the initial cycle. (2 rods/fuel block in 24 refueling regions). Types II and III were requested primarily as a shim provision, in which about .01 Δk additional reactivity control can be achieved by exchanging all of the type I rods with type II rods or by adding some type II or III rods to regions not containing burnable poison.

2.3. DATA HANDLING CODES

As mentioned in Section 1.2 of this report, a significant data handling package had to be developed to prepare as built material loading data (fuel elements and poison rods) for use in a three dimensional GATT calculation. In addition, since GATT is not a depletion code, a burnup package had to be developed utilizing calculated data from the time independant GATT calculation and the initial core loading data. A flow chart for the data handling and fuel depletion programs used with the 3-D GATT depletion analysis is given in Fig. 1. A detailed description of all the data handling programs is given in Reference 9, and for the GATT flux and power edits and burnup program in Reference 10.

A brief description of these programs, and GATT program modifications, is as follows:

Data Handling⁽⁹⁾

1. RODS program - lists the material composition for each of the as built fuel elements. Each fuel element is identified by fuel element type, fuel service limit, and axial

TABLE 5
GEOMETRICAL DESCRIPTION OF AS BUILT BURNABLE POISON RODS

Type and Designation	Average Boron Density (gm BN/cc)		Average Length (in.)		Average OD (in.)	
	As Built	Requested	As Built	Requested	As Built	Requested
I (Low Boron)	.02194	.025	28.40	28.50	.413	.405
II (Medium Boron)	.05458	.059	28.41	28.50	.300	.300
III (High Boron)	.1117	.114	28.42	28.50	.300	.300

Fig. 1. Flow chart for data handling programs used with three dimensional (3-D) GATT analysis

layer designator. The number of fuel rods for each element is given.

2. SCATER program - randomly loads all of the as built fuel elements into specific core locations. Those fuel elements which have to be used in a pre-selected location are utilized first and then the remaining core locations filled with the appropriate fuel element. A number of internal tests are performed to insure that all of the fuel elements are used and used only once and no element is misplaced in the core assembly. Total loadings for each refueling region, each fuel column and each axial fuel layer are calculated and compared with those in the reference design.
3. WONDER program - uses designated core assembly loading from SCATER and mass loadings from RODS to compute the following data:
  - a. Axial smeared atom densities for use with GAUGE or BUGTRI.
  - b. Atom densities for each GATT region.
  - c. Carbon/uranium ratio and thorium lump densities for use with INTERP to obtain effective microscopic cross sections.

3-D GATT Modifications⁽¹⁰⁾

1. The edit of GATT was modified to write the following data on a results tape:
  - a. Average core axial power distribution.

- b. Refueling region radial peaking factors (37) and the average axial power distribution in each refueling region.
- c. Radial column peaking factors (7 per region) and the average axial power distribution for each column (7 per region).
- d. Local radial peaking factors for each radial meshpoint and the axial power distribution for each radial meshpoint. (7 per hex column).

This data was arranged on results tapes by patch, associated columns and associated radial meshpoints.

2. Additional requirements for GATT were:

- a. Macroscopic cross section input - read from tape.
- b. Average region fluxes written on tape.
- c. Region volumes - written on tape.
- d. GATT should allow for ≥ 1100 regions.
- e. Converged fluxes - saved on tape.

3. Data transfer between GATT and the burnup package was by tape. Data for the various time steps was stacked on the tapes as follows:

Tape 1	Power distribution results (written by GATT)
Tape 2	Flux distribution results (written by GATT)
Tape 3	Average region fluxes, volumes (written by GATT)

Tape 4   Macroscopic x-sections (written by burnup package)
Tape 5   Atom densities (written by burnup package)
Tape 6   Microscopic cross sections

Burnup Package - (BUGATT)⁽¹⁰⁾

A burnup package, compatible with the GATT tapes, was developed with input to this package as follows:

- a. Region fluxes from GATT.
- b. Microscopic cross sections.
- c. Region volumes from GATT.
- d. Region atom densities from WONDER.
- e. Usual burnup input - same format as in GAUGE.

The code computed new atom densities and macroscopic cross sections for all GATT regions.

2.4. FUEL LOADING FOR AS BUILT CORE ASSEMBLY

As mentioned in Section 2.1, a tolerance of $\pm 1\%$ was required for the total fuel loading of each of the 13 different fuel blends to ensure that the uranium and thorium loading in each of the 37 refueling regions would be essentially the same as in the design analyses. This was satisfied as shown in Table 4.

However, during the production of the initial core fuel elements, it was necessary to assign an axial layer designation to about half of the 1482 required fuel elements. This was necessary because a large number of the fuel elements had been loaded with a uranium loading as much as 5% greater than nominal. By specifically locating these fuel elements in layers where the fuel temperature was the lowest, and by assembling some fuel elements with lower fuel loadings to be located in layers where the fuel temperature was the highest, the design axial fuel zoning and resulting axial power distribution were essentially maintained.

Using the as-built mass loading data for each fuel element and several random core loading patterns, in which the designated core layer elements were selected first, a computer study was made using the SCATER program to determine the anticipated initial core assembly. The deviations for the uranium and thorium from the specified loading used in the design analysis were calculated for each of the 37 refueling regions and for each of the six core layers for the various random patterns. The results of this analysis are shown in Table 6. It can be seen that little change occurred between the various random patterns. This might well be expected since nearly half of the fuel elements have been designated for use in specific core layers and in some cases, because of their blend and fuel service limit, in specific refueling regions. The uranium loading exceeds the design value by 1% in only 6 of the 37 refueling regions; in no region is the design value exceeded by 2%. The same is true for the thorium loading. The maximum deviation occurs in refueling region 1 and will have a minimal effect on the radial power distribution.

The loading for the six axial core layers can be seen to have a maximum deviation in core layer 9, where the fuel temperature is the highest and the average uranium loading is almost 4% lower than nominal. The top three layers constitute one axial zone and contain the highest uranium loading while the bottom three layers are another zone with a lower uranium loading. It can be seen that the deviations for uranium loading complement the axial zoning and that the thorium loading deviations are nearly uniform. The results of this study show the degree of success obtained in making fuel rod loading adjustments and selectively locating the fuel elements in the reactor.

2.5. EXPECTED IMPURITY LEVELS

Many tests were made on the impurity levels in the initial core fuel elements. A summary of these measurements is discussed in Reference 11 and the results shown in Table 3 of Section 2.1.

TABLE 6
PROJECTED URANIUM AND THORIUM LOADING DEVIATIONS
(PERCENT FROM NOMINAL) FOR REFUELING REGIONS AND
AXIAL CORE LAYERS IN INITIAL CORE ASSEMBLY

	Random Loading 1		Random Loading 2		Random Loading 3	
	Uranium	Thorium	Uranium	Thorium	Uranium	Thorium
Axial layer						
4 (a)	0.40	0.19	0.37	0.26	0.22	0.20
5	1.22	-0.49	1.48	-0.65	1.11	-0.71
6	-0.16	-0.95	-0.40	-0.33	0.14	-0.73
7	2.13	-0.35	2.29	-0.56	2.13	-0.35
8	0.32	-0.50	0.43	-0.77	0.47	-0.59
9 (b)	-3.53	-0.38	-3.81	0.11	-3.68	-0.29
Refueling region						
1	1.96	-1.39	1.96	-1.39	1.96	-1.39
2	-1.55	-0.21	-0.46	-0.81	-0.46	-0.61
3	-0.26	-0.56	0.41	-0.79	-0.22	-0.56
4	1.08	-2.83	0.91	-1.32	1.72	-1.43
5	-0.33	-0.40	-0.96	0.35	-0.30	-0.53
6	0.33	-1.03	0.22	-0.85	0.58	-0.74
7	0.93	-0.63	-0.06	-0.41	-0.41	-0.52
8	1.40	-1.34	1.04	-0.61	1.05	-0.87
9	0.64	-0.38	-0.68	0.58	0.36	-0.29
10	0.27	-0.29	-0.11	-0.20	-0.05	0.16
11	0.02	-0.02	0.47	-0.47	0.02	-0.02
12	-0.85	0.30	-0.23	0.45	-0.79	0.28
13	0.45	-0.32	0.37	-0.26	0.45	-0.32
14	-0.20	0.42	-0.43	0.22	0.07	0.34
15	0.81	0.65	0.67	-0.76	0.41	0.29
16	-0.17	-0.45	0.84	-0.29	-0.17	-0.45
17	-0.37	-0.70	-0.22	-0.87	-0.27	-0.70
18	0.59	-0.70	0.00	-0.53	0.55	-0.70
19	-1.10	0.61	-0.63	0.00	-1.14	0.69
20	0.36	-1.38	0.44	-1.10	0.39	-1.20
21	-0.29	-0.04	-0.18	-0.06	-0.39	-0.14
22	0.34	-0.63	-0.81	-0.21	0.34	-0.63
23	0.80	-1.06	0.60	-0.73	0.81	-0.86
24	0.15	-0.87	0.71	-1.17	0.13	-1.86
25	-0.18	0.00	-0.20	-0.13	-0.38	-0.25
26	-0.02	-0.55	0.27	-0.05	-0.02	-0.55
27	1.02	-0.57	1.07	-0.58	0.92	-0.49
28	-0.84	0.16	-0.34	0.23	-0.73	0.04
29	0.66	-0.44	0.66	-0.44	0.66	-0.44
30	0.08	-0.64	-0.06	-0.50	0.08	-0.64

TABLE 6 (continued)

	Random Loading 1		Random Loading 2		Random Loading 3	
	Uranium	Thorium	Uranium	Thorium	Uranium	Thorium
31	0.78	-0.86	0.45	-0.95	0.77	-0.94
32	-0.24	0.18	-0.06	-0.12	-0.27	0.11
33	-0.66	-0.28	0.29	-0.70	-0.66	-0.28
34	0.78	-0.28	0.53	-0.48	0.88	-0.37
35	-0.60	0.71	-0.24	0.54	-0.45	0.68
36	-0.86	0.18	-0.31	-0.08	-0.46	-0.27
37	-0.01	-0.49	0.00	-0.91	-0.01	-0.49

(a) Top.

(b) Bottom.

The RODS program maintained a running inventory of the metallic impurities for each batch of fuel rods that were made and computed a core average for each impurity. A boron equivalent of all the metallic impurities was computed as a function of the total fuel rods made. Although the program rounds off to the nearest one ppm, the total boron equivalent is estimated to be less than 1.6 ppm since it was 1 ppm with 90% of the rods accepted. This is consistent with the value assumed during the design analysis. (See Table 3.) Consequently the analysis for the as built core used the same parasitic impurity level as was used in the original design.

The H_2 content in the graphite blocks is derived totally from the H_2O that has been adsorbed into the blocks from the surrounding atmosphere. Many tests have been made to determine the water sorption and hydrogen content of the graphite blocks and completed fuel elements. Results of these tests are in reasonably good agreement and show a maximum H_2 of 5.1 kg (45 kg H_2O) in the graphite blocks. A good portion of this moisture will be driven off during the loading of the fuel elements where the temperature of the elements will be greater than 100°F and the relative humidity will be less than 70%. Since this analysis was made to determine the operating performance at power, and all moisture will be removed by then, it was not included in the analysis.

During fuel rod fabrication a hot hydrochloric acid environment was used to leach out surface contamination of thorium. This resulted in some deposit of chlorine in the fuel rods. In addition, the use of a hydrocarbon pitch in the fuel rod matrix resulted in the deposition of some residual hydrogen in the rods. Both of these may be driven off during power operation; however, as seen in Table 3, the reactivity worth of these impurities is insignificant, and opposite in value. Consequently, the presence of these was ignored in the as built analysis.

The other impurities in the graphite blocks were determined using Deltainhour (DIH) measurements which are a reactivity worth comparisons to a known standard. The three graphite samples that were used in these measurements give an impurity level equivalent to 0.9 ppm (by weight) of

natural boron. A boron equivalent of 1 ppm for the impurities in the fuel element blocks was used in all of the design calculations, as well as the as built analysis.

3. ANALYTICAL MODEL (GATT-2X)

3.1. GENERAL GEOMETRIC DESCRIPTION

As discussed in Reference 1, almost half of the as built initial core fuel elements were identified for specific in core locations. This was due to the fact that the majority of the fuel elements had fuel loadings that deviated from the nominal specified loading by more than ± 1 percent. A distribution of the fabricated fuel elements and their deviation from the nominal loading for both uranium and thorium is shown in Fig. 2. As seen there, only about 20 percent of the fuel elements had a uranium loading within 1 percent of that specified and 12 percent had loading deviations from -3 to -7 percent.

Consequently it was considered important to do a multidimensional analysis in which each fuel element could be represented explicitly. There are 1482 fuel elements in the FSV core assembly, 6 layers of 247 elements per layer. In the GAUGE nuclear model, by combining fuel columns of similar loading and radial position, it was possible to represent the core assembly with 169 burnup regions. By assuming the same GAUGE representation for each of the 6 axial core layers, it was possible to geometrically describe the core with 1014 burnup regions. A plan view of the lowest axial fuel layer, core layer 9, using this burnup region description is given in Fig. 3. The next fuel layer up, core layer 8, would have the same burnup region layout except the central column would be burnup region 170 and the highest region number would be 338 (i.e., all numbers are increased by 169). This numbering sequence was repeated for each of the axial fuel layers. As seen in Fig. 3, most of the fuel elements are explicit although in some of the outer regions a burnup region will consist of as many as 3 fuel elements. This radial description was used for each of the top and bottom reflector columns as well.

Fig. 2. Percent deviation from nominal loading

Fig. 3. Burnup region description (GATT-3-D) for axial layer 9

The GATT model is limited to 8 slices axially, although each slice can be subdivided into two subslices having different macroscopic properties. Consequently, each fuel layer was represented by one slice, and the top and bottom reflector by a separate and distinct slice. This allowed for explicit representation in the axial direction. A cross section of the core assembly showing the top to bottom axial description as well as the side reflector is given in Fig. 4. The axial dimensions and mesh point description are also shown.

In each refueling region, the central control rod fuel column is offset axially from the surrounding fuel columns to prevent cross flow of the coolant and prevent undue block motion during earthquakes. By having two subslices in each axial fuel layer, it was possible to describe this offset explicitly, and include both the short control rod fuel element and control rod fuel elements separately. This can be seen in core layers 8 and 9 of Fig. 4. It was also possible to represent the significantly different flux spectrum at the core reflector interface by having two regions in the top and bottom reflector. The region within 10 cm of the core was carried separately.

As seen by the dimensions in Fig. 4, the top reflector is 100 cm thick while the bottom reflector is only 60 cm thick. The bottom reflector actually consists of three layers of reflector elements and is about 120 cm thick, but in the lower half of the second layer some borated rods are loaded to reduce the thermal flux at the core support floor. The amount of boron in that region is sufficient to reduce the incoming thermal current to zero. This can be achieved analytically by forcing the flux to be zero at that boundary. Consequently, both the top and bottom outer boundaries in the GATT analysis were located at the edge of the borated region.

A complete summary of all the core and reflector burnup regions required for this depletion calculation is given in Table 7. The depletion analysis was carried out for the core regions only.

Fig. 4. Geometrical description for core assembly GATT model 3-D burnup analysis

TABLE 7
BURNUP REGION DESCRIPTION FOR GATT-3-D ANALYSIS

Burnup Region Number	Description
1-169	Fuel elements for axial layer 9
170-338	Fuel elements for axial layer 8
339-507	Fuel elements for axial layer 7
508-676	Fuel elements for axial layer 6
677-845	Fuel elements for axial layer 5
846-1014	Fuel elements for axial layer 4
1015-1016	Bottom reflector (within 10 cm of core and region down to borated reflector)
1017-1018	Top reflector (within 10 cm of core and region up to borated reflector)
1019	Side reflector region
1020	Side reflector with borated rods
1021-1022	Not assigned
1023	Top reflector with 30 w/o control rods homog patch
1024	Top reflector with 40 w/o control rods homog patch
1025	Control rods homogenized in central hex reflector above control rod column
1026	Side reflector with light carbon and heavy boron to provide boundary condition as in GAUGE

3.2. IN CORE CROSS SECTIONS

There were a total of 1014 burnup regions in the core assembly for the 3-D GATT analysis as described earlier and summarized in Table 7. Effective microscopic cross sections for each of these regions was obtained in the same way that they were for all of the design GAUGE analyses.

1. For each of the regions, the carbon/uranium (C/U) ratio and thorium density in the fuel rod (N_{th}) were obtained with the WONDER data handling program.
2. Using the fuel and moderator temperature data (T_{fuel} and T_{mod}) as given in Reference 12, the average fuel and moderator temperatures were determined for each of the axial fuel layers. These values are given for the 6 core layers in Table 8.
3. The Interp Program was used to obtain effective microscopic cross sections for each region selecting the appropriate C/U ratio, N_{th} , T_{fuel} and T_{mod} . The effective thermal cross section is dependent upon the C/U ratio and T_{mod} and the fast and resonance cross section upon the N_{th} and T_{fuel} .⁽¹³⁾
4. These effective microscopic cross sections were utilized in the burnup program BUGATT for the depletion analysis and combined with the core calculated atom densities to obtain macroscopic data for GATT initially.

NOTE: Particle self shielding factors for the uranium in the fuel rods were not included in this "as built" core analysis. The reactivity worth of ignoring this effect is discussed later in this report and the basis is summarized in Reference 14.

TABLE 8
AVERAGE FUEL AND MODERATOR TEMPERATURES FOR EACH OF THE AXIAL CORE LAYERS

Axial Core Layer Number	Average Fuel Temperature (°K)	Average Moderator Temperature (°K)
4 (top)	835	791
5	964	901
6	1088	1014
7	1133	1072
8	1177	1125
9 (bottom)	1184	1148

3.3. REFLECTOR CROSS SECTIONS

The following is a summary of the procedures and approximations that were used in obtaining effective reflector cross section for use with the GATT analysis.

Side Reflector

The GAUGE model macroscopic cross sections for the three side reflector regions were used. This utilizes CREF 1⁽¹⁵⁾ for the unborated region adjacent to the core which consists of the removable reflector elements and the unborated permanent side reflector. This was used instead of previous GATT side reflector cross sections. They had been

obtained by collapsing and flux weighting CREF 1 and CREF 2 from an RZ calculation into one region. It is thought to be more important to have the cross sections immediately adjacent to the core be correct. The borated side reflector combines a heavy boron loading with CREF 3 and the region outside the core assembly consists of a dilute carbon loaded region.

CREF 1, CREF 2, CREF 3, CBOT 1, CBOT 2, and CTOP 3 refer to cross section sets which take into account the thermalization (Reflector Spectrum Softening) as the distance from the core reflector interface increases. They were obtained using a one-dimensional multigroup calculation and then collapsing the multigroup cross sections to effective few group sets.⁽¹⁵⁾

Bottom Reflector

As mentioned in Section 3.1, the bottom reflector borated region was not included but the outer boundary of the model was located at the inner edge of the borated region. Consequently the bottom reflector consisted of two sets of macroscopic data: CBOT 1⁽¹⁵⁾ for the 10 cm adjacent to the core and CBOT 2⁽¹⁵⁾ for the region extending to the edge of the borated region. All of these were for a moderator temperature of 900°K.

Top Reflector

The top reflector was treated similarly to the bottom reflector in that the outer boundary of the problem was located at the inner edge of the top borated reflector region. In addition the 10 cm region adjacent to the core consisted of the CTOP 3 cross section set.⁽¹⁵⁾ However the region immediately above that was obtained by flux weighting and collapsing the macroscopic data from a previous explicit RZ calculation.

3.4. CONTROL ROD CROSS SECTIONS

In all regions where control rods were located the effective macroscopic cross sections for that region were obtained by homogenizing control rod cross sections as given in Reference 16 with those previously obtained for the region without control rods.

When control rods were inserted in the core assembly, the control rods were homogenized within the central fuel element column of the refueling region. When they were partially withdrawn into the top reflector, they were homogenized within the central fuel element column for the first 10 cm above the core and in that reflector region which extends into the core assembly, because of the axial offset for the control rod column. For the upper part of the top reflector, they were homogenized within all seven fuel element columns of that refueling region.

When the control rods were fully inserted, they were smeared over all six layers of the core assembly and were not included in the upper reflector region. When they were fully withdrawn, they were smeared over the upper part of the top reflector region only; that is, that region that extends within 10 cm of the top interface of the core. Whenever they were partially inserted, they were smeared into both regions of the top reflector and the appropriate regions of the core assembly. Since there were six axial core layers, they could be partially inserted in single layer increments only. That is, they had to be one-sixth, one-third or one-half withdrawn depending on the desired axial position. They could not be partially inserted within an axial core layer. A summary of the burnup regions which could contain control rods in the core assembly is given in Table 9.

3.5. DEPLETION MODEL (BUGATT)

The same depletion scheme as used in GAUGE was incorporated into BUGATT. Consequently the same fission yields and fission products and nuclide scheme as for GAUGE was used. This made comparisons with GAUGE analysis simple and direct.

TABLE 9
BURNUP REGIONS IN GATT WITH CONTROL RODS

	Axial Layer 9	Axial Layer 8	Axial Layer 7	Axial Layer 6	Axial Layer 5	Axial Layer 4
30 w/o control rods	1	170	339	508	677	846
	2	171	340	9	8	7
	3	172	341	10	9	8
	4	173	342	11	680	9
	5	174	343	12	1	850
	6	175	344	13	2	1
	7	176	345	14	3	2
	8	177	346	15	4	3
	9	178	347	16	5	4
	10	179	348	17	6	5
	11	180	349	18	7	6
	12	181	350	19	8	7
	13	182	1	20	9	8
	14	183	2	21	690	9
	15	184	3	22	1	860
	16	185	4	23	2	1
	17	186	5	24	3	2
	18	187	6	25	4	3
	19	188	7	26	5	4
40 w/o control rods	20	189	8	7	6	5
	21	190	9	8	7	6
	22	191	360	9	8	7
	23	192	1	530	9	8
	24	193	2	1	700	9
	25	194	3	2	1	870
	26	195	4	3	2	1
	27	196	5	4	3	2
	28	197	6	5	4	3
	29	198	7	6	5	4
	30	199	8	7	6	5
	31	200	9	8	7	6
	32	201	370	9	8	7
	33	202	1	540	9	8
	34	203	2	1	710	9
	35	204	3	2	1	880
	36	205	4	3	2	1
	37	206	375	4	3	2

4. PRELIMINARY AND INTERNAL CALCULATIONAL CHECKS

4.1. FUEL ELEMENT MASS DATA

It was shown in Section 2.1, Table 1 of this report that the total core loading for the "as built" fuel elements is essentially the same as that used in the design analyses. This was true not only for the uranium, thorium and carbon loading but for the core impurities also.

It was also shown in Section 2.4, Table 6 of this report that, since almost half of the as built fuel elements were required to be placed in pre-selected core locations, the actual loading pattern used made little difference in the loading variations around the core. For this analysis, the first random loading pattern was selected for use.

Some internal tests were made to insure that the projected as built core loading for the analysis was representative of what would be expected. These were as follows:

1. Total core loadings for carbon, uranium and thorium in the as built assembly were computed from the analytical atom densities and volumes and compared against those obtained from the RODS program.
2. Each location in the core assembly requires a specific type of fuel element.⁽¹⁷⁾ Since there are 1482 locations and only 301 types of fuel elements, including service limit and layer designator specification, each type can be loaded into many different locations. Listings were made and tested to see that each location contained an acceptable fuel element.

3. These same listings were tested to insure that each fuel element was used only once.
4. Total burnable poison loading was calculated internally and checked against that expected.
5. The fuel element and blend loading adjustments discussed in Reference 1 were acceptable because some fuel elements were being used in preferred in-core locations, where the temperature is high or low depending on the loading. Knowing these adjustments, the loadings in specific core regions and core layers in the as built assembly were inspected to see that they were consistent with the expected results. This is discussed and shown in Table 6 of Section 2.4.

4.2. DEGENERATE GATT ANALYSIS

Before the full core 3-D GATT depletion analysis was performed, a degenerate GATT calculation was run for one axial layer of a single refueling region and compared with the results from a GAUGE calculation. This calculation included a one month depletion and was made with core average densities. In this manner, the GATT modifications and the burnup calculation loop using BUGATT, and the appropriate flux and power tapes, were tested. The results showed complete agreement between the two calculations.

4.3. GAUGE COMPARISON WITH REFERENCE DESIGN

Besides the degenerate GATT analysis mentioned above, a full core GAUGE depletion calculation was run with the as built loading and compared with a similar reference design case. The atom densities for the as built GAUGE burnup regions were computed in the same manner as for the GATT analysis; namely, RODS input for fuel element masses, SCATER for fuel element loading in the core, and WONDER for the average axial atom

densities. WONDER also computed the C/U atom ratio and the thorium lump density for each fuel type and cross sections were computed from INTERP for the appropriate nuclear data and fuel and moderator temperature. Values for the top and bottom core halves and the full core are given in Tables 10, 11 and 12. For all of these cases, the fuel loading for each fuel column was axially averaged.

The reference design model for the full core GAUGE is shown in Table 13. In this model, the buffer fuel element is described using one cross section from the thorium lump in the buffer region which is then smeared over the outer fuel element, and another cross section from the thorium lump not in the buffer region, by fuel blend, which is also smeared over the outer fuel element.

In the as-built model, an average thorium lump was computed for the buffer fuel elements, by fuel element type, which was used to compute an average thorium cross section for the buffer element. C/U ratios and thorium lump densities for the various fuel compositions in the reference design and the as built core are shown in Tables 10 thru 13.

A sequence of calculations was performed to check the as built input, both cross sections and core atom densities. The results of these calculations are summarized in Table 14. They were all performed for the beginning of cycle, hot operating, clean condition. In essence they showed that the as built data and whatever assumptions associated with the as built model were consistent with the reference design.

The summary given in Table 14 is pretty much self explanatory, but a few comments should be added. The difference between cases 1 and 2 is described in great detail in Reference 18 and the only reason it is shown here is because all of the as built cross sections were determined in the same way as used in case 2, namely using C/U rather than C/U-235. The adequacy of the as built cross section model, in particular for the treatment of the two thorium densities in the outer buffer fuel elements, is

TABLE 10
MODEL FOR TOP HALF GAUGE (AS BUILT)

X-Section Set	Description (Composition)	Refueling Regions	C/U _{tot}	N _{Th} x 10 ⁻³
1	1,2 (inner)	1,4,5,8,10,15,17	4772	1.697
2	1,2 (outer non-buffer)	21,25,28,32,35,36	4917	1.714
3	3,4	2,6,7	3407	1.418
4	5,6	9,11,12,13,16,19	3019	1.487
5	9,10	3,13,18	3702	1.611
6	7,8 (non-buffer)	20,23,24,26,27,30,31,34,37	2677	1.468
7	7,8 (buffer)	Same	3244	1.660
8	11,12 (non-buffer)	22,29,33	3129	1.685
9	11,12 (buffer)	22,29,33	3537	1.780
10	1,2 (buffer)	21,25,28,32,35,36	4901	1.816
11	Average core	All	3577	1.612

Temperature = 965-900

TABLE 11
MODEL FOR BOTTOM HALF GAUGE (AS BUILT)

X-Section Set	Description (Composition)	Refueling Regions	C/U _{tot}	N _{Th} x 10 ⁻³
1	1,2 (inner)	1,4,5,8,10,15,17	6238	1.541
2	1,2 (outer non-buffer)	21,25,28,32,35,36	6437	1.540
3	3,4	2,6,7	4670	1.224
4	5,6	9,11,12,14,16,19	4095	1.311
5	9,10	3,13,18	5200	1.344
6	7,8 (non-buffer)	20,23,24,26,27,30,31,34,37	3526	1.317
7	7,8 (buffer)	Same	3940	1.544
8	11,12 (non-buffer)	22,29,33	4184	1.462
9	11,12 (buffer)	22,29,33	4507	1.631
10	1,2 (buffer)	21,25,28,32,35,36	5870	1.719
11	Average core	All	4697	1.445

Temperature = 1155-1105

TABLE 12
MODEL FOR FULL-CORE GAUGE (AS BUILT)

X-Section Set	Description (Composition)	Refueling Regions	C/U _{tot}	N _{Th} x 10 ⁻³
1	1,2 (inner)	1,4,5,8,10,15,17	5405	1.619
2	1,2 (outer non-buffer)	21,25,28,32,35,36	5572	1.628
3	3,4	2,6,7	3936	1.321
4	5,6	9,11,12,14,16,19	3474	1.399
5	9,10	3,13,18	4325	1.478
6	7,8 (non-buffer)	20,23,24,26,27,30,31,34,37	3041	1.393
7	7,8 (buffer)	Same	3553	1.602
8	11,12 (non-buffer)	22,29,33	3579	1.574
9	11,12 (buffer)	22,29,33	3964	1.706
10	1,2 (buffer)	21,25,28,32,35,36	5340	1.768
11	Average core	All	4059	1.529

Temperature = 1060-1003

TABLE 13
MODEL FOR FULL CORE GAUGE (REFERENCE DESIGN)

X-Section Set	Composition (Fuel Type)	Refueling Regions	C/U _{tot}	N _{Th} x 10 ⁻³
1	1,2	1,4,5,8,10,15,17,21,25,28,32,35,36	5430	1.633
2	3,4	2,6,7	3937	1.330
3	5,6	9,11,12,14,16,19	3459	1.399
4	7,8	20,23,24,26,27,30,31,34,37	2951	1.402
5	9,10	3,13,18	4318	1,487
6	11,12	22,29,33	3452	1,575
7	13	Buffer	5725	1.960
8	Average	All	4063	1.565

Temperature = 1060-1003

Note: The buffer composition was ignored for determining the C/U in all the outer regions.

TABLE 14
CALCULATED GAUGE RESULTS FOR AS BUILT AND REFERENCE DESIGN

	Begin of Cycle, Hot, Unrodded, K_{eff}
1. Reference Design (Table 13)	1.0728
2. Reference Design with consistent x-sections (use C/U_{tot} rather than C/U_{235})	1.078
3. Reference Design densities - as-built x-sections	1.0785
4. Same as <u>3</u> except buffer Th is explicit	1.0784
Models 3 and 4 indicate that the smeared thorium in the buffer is a good model.	
5. As-built loadings and x-sections	1.0801
The k_{eff} in model 5 is $\sim .002 \Delta k$ higher than the reference design which was expected from the higher as-built loadings.	
6. Same as <u>5</u> except choose separate C/U and N_{th} for inner and outer hex blocks in outer ring. (Types 1&2, 7&8, 11&12) also choose separate C/U and N_{th} for Types 1&2 in central 19 regions.	1.0807
7. Adjust transverse leakage to be consistent with SCANAL results and make small correction in computing C/U and N_{th} . See Table 10.	1.0731

substantiated by the results of cases 2, 3 and 4. As noted in Table 3 of the summary, because of the loading differences for carbon, silicon, uranium and thorium, it would be expected that the excess reactivity for the as built loadings would be about $0.002 \Delta k$ larger. This agreed almost exactly with the difference between cases 4 and 5 and verified the adequacy of the as built input data.

The calculated radial power distribution for these different cases is shown in Fig. 5. The primary difference occurs in the outer refueling regions and the largest variation is found in those regions containing fuel compositions 7 and 8. As seen by the slight variation between cases 5, 6 and 7 of Fig. 5, this calculated difference appears to be due to the difference in atom densities for those regions and not the cross sections. This is what might be expected if the as built loadings are inspected (see Table 6 of Section 2.4).

The control rod bank worth for the reference sequence to be used with the GATT analysis for the as built core was also calculated with the GAUGE full core model. The results are summarized in Table 15 and Fig. 6. As seen by the results in Table 15, the calculated worth for each of the groups is essentially the same as that obtained for the reference design. This is important, because the anticipated critical control rod patterns for GATT were based on reference design analysis. In addition, the effect of the equilibrium xenon on the worth for 16 control rods was calculated and is given in Table 15.

All of the as built GAUGE calculations were done without self shielding factors (SSF) for U-235; however, the unrodded-hot case was run with SSF to determine the effect on the BOC core. With SSF, the k_{eff} was reduced by $0.0028 \Delta k$. See Table 15.

A summary of the calculated radial power distribution for the as built core with various control rod configurations is given in Fig. 6. It can be seen that as long as there are less than 16 control rods inserted, the

Fig. 5. Calculated radial power distribution for as built and reference design core (GAUGE-BOC)

TABLE 15
CONTROL ROD GROUP WORTH FOR AS BUILT GAUGE MODEL
(OPERATING TEMPERATURE)

Rod Configuration	# Rods In	As-Built		Reference Design	
		k_{eff}	Δk	Δk	k_{eff}
1 All rods out	0	1.0731			1.0691
2 All rods in	37	.8444	.2287	.2307	.8384
3 3B,4F,4A,3D,4C,1	16	.992			.9877
4 4F,4A,3D,4C,1	13	1.0147	.0227	.0234	1.0111
5 4A,3D,4C,1	10	1.0253	.0106	.0100	1.0211
6 3D,4C,1	7	1.0364	.0111	.0102	1.0313
7 4C,1	4	1.0586	.0222	.0232	1.0545
8 1	1	1.0674	.0088	.0086	1.0631
9 -	0	1.0731	.0057	.0060	1.0691
			.0811	.0814	
10 16 rods in @ 2 days	16	.9619			
11 All rods out - 2 days	0	1.0406		.0787	} Equilibrium Xenon
12 All rods out with SSF for U-235	0	1.0703		.0028	

Fig. 6. Calculated radial power distribution with as built core for reference control rod sequence (Sheet 1 of 2)

Fig. 6. Calculated radial power distribution with as built core for reference control rod sequence (Sheet 2 of 2)

radial peaking factors are within the limits of the Technical Specifications (i.e., less than 1.83).

More details on the GAUGE comparison for the as built and reference design core are given in Reference 19.

4.4. U-233, U-235 AND BURNABLE POISON DATA DURING DEPLETION

To insure that no errors occur to the sequenced GATT analysis during burnup, several checks were performed while the analysis was in progress. These checks were:

1. Verify the adequacy of the convergence acceleration in GATT for the first static GATT case.
2. Check the control rod pattern for each GATT calculation.
3. Spot check regional macroscopic data in the core regions.
4. Spot check atom densities, in particular xenon, burnable poison, uranium, and control rods.
5. Compare the axial and radial power distribution against that expected.
6. Check the depletion data as a function of burnup for U-235, U-233 and burnable poison against that expected.

For this check the depletion data plotted in Fig. 7 were obtained from the reference design GAUGE analysis. The total core mass of U-233 and the fractional depletion $[N(t)/N(o)]$ of U-235 and lumped burnable poison were obtained as a function of Effective Full Power Days of reactor operation. During the GATT analysis these same quantities were computed for the total core and compared against the GAUGE data. In this manner, a variation in results could be investigated before the GATT depletion was completed.

Fig. 7. Depletion data comparison for GAUGE-GATT during 1 year GATT burn as built core

The comparison between the GATT calculated total core inventory and the reference design analysis inventory (Rodded GAUGE Results) is given in Fig. 7 and Table 16. For the isotopes of interest the results agree well. There are slight differences in U-235 and U-233 but these are considered not significant. These differences are probably due to three dimensional depletion effects.

4.5. CONTROL ROD PATTERNS

Knowing the operating excess reactivity as a function of burnup and the control rod worth for the control rod groups, the expected near critical configuration was determined for the GATT analysis and is summarized in Table 17. For each of these configurations the control rods were homogenized in the core assembly as described in Section 3.4.

During the depletion analysis, checks were made of both the burnup region atom density and macroscopic data to insure that the control rods were present in the appropriate regions. This was done by verifying that the burnup regions as shown in Table 9 had control rod data. The macroscopic value of Σa was checked against the value it should have to include control rods.

TABLE 16
COMPARISON OF GAUGE AND GATT DEPLETION FOR
URANIUM, THORIUM AND BURNABLE POISON

	Effective Full Power Days					
	BOC	100	150	200	255	310
Mass U-235, kg						
GATT (as built)	727.4	623.6	576.3	532.2	487.4	445.9
GAUGE (ref. design)	721.0	617.1	569.5	525.1	479.9	438.1
(Diff)	+6.4	+6.5	+6.8	+7.1	+7.5	+7.8
Mass U-233, kg						
GATT (as built)	0	40.4	67.8	93.7	119.5	142.7
GAUGE (ref. design)	0	40.6	68.2	94.2	120.4	143.8
(Diff)	0	-.2	-.4	-.5	-.9	-1.1
Mass Thorium, kg						
GATT (as built)	16050	15980	--	15910	15870	15830
GAUGE (ref. design)	15911	15841	15805	15770	15730	15690
(Diff)	139	139	--	140	140	140

TABLE 17
PREDICTED CONTROL ROD CONFIGURATION FOR AS BUILT GATT 3D BURNUP

Time Pt.	0	1	2	3	4	5	6	7	8	9
EFP days	0.0	2.0	5.0	10.0	50.0	100.0	150.0	200.0	255.0	310.0
k_{eff} (no rods)	1.074	1.039	1.037	1.032	1.022	1.022	1.023	1.023	1.020	1.015
C Rods for Critical Guess										
1	1(1/2)	1(1/2)	(1/2)	(1/2)	(1/2)			(1/2)	(1/2)	(1/2)
Δk	(.004)	(.004)	(.004)	(.004)	(.004)			(.004)	.004	(.004)
4C	22,28,34	F ^(a)	F	F	F			F	(2/3)	(1/3)
Δk	(.013)	(.013)	(.013)	(.013)	(.013)	(.013)	(.013)	(.013)	(.012)	(.008)
3D	11,15,19	F	F	(2/3)	(1/3)			(1/6)		
Δk	(.036)	(.036)	(.036)	(.033)	(.022)			(.017)		
4A	20,26,32	(1/3)	(1/3)							
Δk	(.046)	(.040)	(.040)							
4F	25,31,37									
Δk	(.056)									
3B	9,13,17 (1/2)									
Δk	(.071)									
k (exp)	1.003	.999	.997	.999	1.00	1.00	1.001	1.006	1.008	1.007

(a)_F = Fuel inserted.

5. CALCULATED GATT RESULTS

5.1. INTRODUCTION AND DESCRIPTION

The initial core assembly contains six axial layers with 247 fuel elements in each layer, making a total of 1482 fuel elements. By locating some fuel elements with the same metal loading in the same burnup region, it was possible to assemble a three-dimensional calculational model in which the majority of the fuel elements was explicitly represented. There were 1014 burnup regions total, and each of the six axial core layers was explicitly described. The top, bottom, and side reflectors were included so that a full core assembly was described.

A simulated core loading for the as-built initial core fuel elements was obtained using random loading pattern 1 as given in the previous section. By taking the actual fuel element loadings as obtained from Fuel Production, it was possible to locate a specific fuel element in each of the 1482 core locations and to compute as-built atom densities for each of the burnup regions. In addition, a C/U-235 ratio and an average lump thorium density for the fuel rods in each element were calculated and using these parameters, effective microscopic cross sections for each burnup region were obtained.

Each of the 37 control rod fuel element columns was also explicitly located so that the control rods could be homogenized in those columns to any size axial positions. In this manner, control rods could be represented from the full-in to the full-out position, including partially inserted control rods.

By combining the results as calculated from the GATT-2X three-dimensional nuclear diffusion theory code for a preselected control rod

configuration with a separate burnup code for each of the 1014 burnup regions, a depletion analysis was made for a simulated as-built initial core assembly.

5.2. CALCULATED CORE REACTIVITY DURING FIRST YEAR

From the depletion analysis with the calculational model, as discussed above, the excess reactivity during the first year of operation was calculated. Preselected critical control rod configurations were obtained using the control rod worths for the various groups of control rods as calculated with the two-dimensional XY model adjusted to take into account three-dimensional effects.⁽²⁰⁾ These values are given in Fig. 8 for all of the groups inserted during this analysis. For partially inserted control rods, the integral worth for the control rod bank fully inserted was adjusted by the rod withdrawal curve as a function of inserted position, also given in Fig. 8. These rod withdrawal curves were discussed in Reference 21 and were obtained with a one-dimensional model using effective macroscopic cross sections obtained from a two-dimensional XY calculation. These control rod worths, combined with the estimated core excess reactivity shown in Fig. 9 from design analyses, yielded predicted critical rod configurations at various times during the first year of operation.

A summary of the predicted critical control rod configuration, along with calculated k_{eff} as obtained from the analysis, is given in Table 1. As seen by the results, extremely good agreement between the predicted critical control rod positions and a critical k_{eff} (1.00) was obtained throughout the analysis. The agreement certainly generates confidence in the previously calculated data for the initial core. Through the first 50 days, the agreement is even better than indicated by the results in Table 1. At the beginning of the cycle, the k_{eff} expected for that control rod pattern was actually 1.003, but because the control rod bank

Fig. 8. Integral control rod worth as a function of position

Fig. 9. Calculated k_{eff} versus core burnup (EFPD)

had to be inserted by whole core layers, an increment of insertion was greater than the difference between criticality. At 2 days and 10 days, the equilibrated xenon distribution does not match the control rod pattern because a significant control rod motion was made at that time step. After 50 days, the loss of excess reactivity with burnup was greater with the three-dimensional depletion model than with the unrodded depletion calculation from the design analysis. This difference can best be seen by the results shown in Fig. 9. Consequently the predicted critical control rod configuration made before the depletion analysis began was modified at 200 days of full power burnup by withdrawing more control rods from that time until the end of the cycle. This difference was due mainly to the depletion of the core in six axial zones, whereas the normal depletion in the design analysis with the GAUGE depletion code was made with just one average axial zone. There is still a sufficient amount of excess reactivity at the end of the cycle, as well as at all times during the cycle.

5.3. RADIAL POWER DISTRIBUTION

The radial power distribution for the 37 refueling regions as calculated from the 3D-GATT analysis for the critical control rod configurations during the first year is summarized in Fig. 10. In addition, the calculated power distribution for each of the fuel columns is given in Tables 18 through 21. These results essentially show the power distribution at the beginning of cycle, after xenon has equilibrated, at 50 days with Sm and Pa near equilibrium, and at the end of cycle. This analysis was made using the control rod sequence as shown in Fig. 8. From these results, it can be seen that:

1. The regional power factors are below 1.83 for the entire cycle and are below 1.60 after the xenon has equilibrated.
2. The calculated intraregion peaking factor is larger than the allowable limit given in the Technical Specifications (Ref. 22) at the beginning of the cycle for some of the rodded regions. Refueling regions 20 and 26 have column/region tilts greater than 1.50.

Fig. 10. Calculated radial power distribution (GATT-2X)

TABLE 18
CALCULATED POWER DISTRIBUTION AT 0 DAYS

Region	Average Power Per Column							Avg. Region	Max. Col./ Avg. Region Power
	Col. 1	Col. 2	Col. 3	Col. 4	Col. 5	Col. 6	Col. 7		
1	.5678	1.1804	1.1461	1.1719	1.2246	1.2646	1.2416	1.1139	1.1353
2	1.0926	1.4582	1.6063	1.6846	1.7366	1.5406	1.4251	1.5050	1.1539
3	.9216	1.3185	1.3910	1.4530	1.5055	1.4010	1.3637	1.3364	1.1266
4	.7724	1.1740	1.2474	1.2632	1.2332	1.1459	1.1586	1.1421	1.1060
5	.8108	1.2189	1.2721	1.2888	1.2690	1.2039	1.2237	1.1839	1.0886
6	1.1411	1.6124	1.7525	1.7840	1.7469	1.6202	1.5791	1.6050	1.1115
7	1.1153	1.4872	1.6883	1.7948	1.8109	1.6623	1.5134	1.5817	1.1449
8	.4907	.6531	.8157	.9395	.8765	.6498	.5654	.7208	1.3728
9	.5728	.8888	1.1289	1.3502	1.4119	1.2386	.9875	1.0827	1.3040
10	.6905	1.1414	1.0903	1.1131	1.0544	.9216	1.0124	1.0042	1.1366
11	.4344	.9951	1.0529	1.2409	1.2785	1.2751	1.1993	1.0680	1.1971
12	.9151	1.3381	1.4670	1.5658	1.4217	1.2073	1.1392	1.2934	1.2105
13	.5196	.9441	1.0699	1.2053	1.2160	1.1585	1.0567	1.0243	1.1871
14	1.1268	1.8250	1.7523	1.7545	1.6294	1.4958	1.6168	1.5987	1.1416
15	.3223	.7843	.7875	.9464	1.0077	1.0090	.9725	.8329	1.2114
16	.9747	1.1293	1.3200	1.5661	1.4571	1.1919	1.0531	1.2273	1.2760
17	.3957	.6244	.7265	.9566	1.0668	.9585	.7554	.7834	1.3617
18	.6172	.8310	.9277	1.1610	1.1062	.8525	.7325	.8969	1.2945
19	.3563	.6631	.8336	1.1652	1.2930	1.0897	.8179	.8884	1.4554
20	.1000	.0000	.4859	.6178	.4874	.3324	.0000	.4047	1.5265
21	.3338	.5360	.5264	.6027	.6575	.5064	.4801	.5204	1.2635
22	.2532	.6120	.7879	.9388	.8363	.6657	.5700	.6663	1.4090
23	.6791	.0000	1.6810	1.6670	1.3515	1.2005	.0000	1.3158	1.2776
24	1.0643	1.5452	1.3562	1.4500	1.7107	1.7586	1.6539	1.5055	1.1681
25	.1859	.4838	.4077	.5717	.6339	.6623	.6651	.5158	1.2895
26	.1713	.0000	.8957	1.0006	.7562	.5085	.0000	.6665	1.5013
27	.9097	1.2766	1.2475	1.3885	1.5346	1.3417	1.2210	1.2742	1.2043
28	.2259	.6215	.6726	.7740	.7283	.6604	.6663	.6213	1.2456
29	.7256	.0000	1.7035	1.6288	1.3138	1.1621	.0000	1.3067	1.3036
30	1.2502	1.8105	1.6011	1.6360	1.9700	2.0025	1.9212	1.7488	1.1451
31	.3413	.7757	.6478	.6884	1.0551	1.1863	1.1169	.8588	1.3814
32	.1338	.0000	.4411	.5236	.4510	.3782	.0000	.3865	1.3547
33	.5406	.7578	.7558	.8892	1.0243	.8435	.7326	.7920	1.2934
34	.2477	.5416	.5899	.7499	.8247	.7495	.6380	.6202	1.3298
35	.3418	.0000	.5672	.5728	.5054	.5080	.0000	.4990	1.1479
36	.3286	.5300	.4743	.5342	.6352	.5815	.5489	.5190	1.2239
37	.1928	.3926	.3942	.6131	.6855	.6136	.4955	.4839	1.4166

TABLE 19
CALCULATED POWER DISTRIBUTION AT 5 DAYS

Region	Average Power Per Column							Avg. Region	Max. Col./ Avg. Region Power
	Col. 1	Col. 2	Col. 3	Col. 4	Col. 5	Col. 6	Col. 7		
1	.4095	.8803	.8501	.8701	.9068	.9399	.9226	.8256	1.1384
2	.9869	1.3577	1.3654	1.3093	1.3407	1.2535	1.2737	1.2553	1.0977
3	.7298	1.0429	1.0773	1.1041	1.1872	1.2175	1.1522	1.0730	1.1347
4	.6301	1.0747	1.0522	.9768	.9484	.9209	1.0097	.9447	1.1376
5	.6367	.9628	.9850	.9735	.9916	1.0319	1.0163	.9425	1.0948
6	.9235	1.4828	1.4703	1.3762	1.3417	1.3141	1.4020	1.3310	1.1141
7	.8825	1.2099	1.3190	1.3622	1.4156	1.4333	1.3027	1.2750	1.1241
8	.5683	.8266	.9389	1.0113	.9044	.8395	.8007	.8414	1.2019
9	.8774	1.1173	1.2496	1.4075	1.4855	1.4436	1.2738	1.2650	1.1744
10	.5539	.8726	.8566	.9015	.9331	.8009	.7869	.8151	1.1449
11	.3781	.9990	1.0542	1.0752	1.0209	1.0158	.9967	.9343	1.1508
12	.9462	1.4469	1.5303	1.5132	1.3588	1.3424	1.3473	1.3551	1.1297
13	.7847	1.0941	1.1358	1.2288	1.2647	1.2903	1.2383	1.1481	1.1238
14	.9968	1.4317	1.4070	1.4062	1.4046	1.2586	1.2627	1.2954	1.1053
15	.2989	.8761	.8682	.8540	.8182	.8200	.8605	.7708	1.1366
16	.9799	1.3538	1.4716	1.5704	1.4812	1.4824	1.4153	1.3935	1.1269
17	.6242	.8069	.8542	1.0123	1.1103	1.0957	.9582	.9231	1.2027
18	.5623	.7752	.8439	1.0082	1.0490	.8356	.7291	.8291	1.2653
19	.3428	.8359	.9486	1.0550	1.0663	.9436	.8204	.8590	1.2414
20	.3402	.0000	.8304	.9969	.9977	.8952	.0000	.8101	1.2317
21	.4035	.6301	.5886	.7132	.7952	.6669	.6333	.6330	1.2563
22	.2308	.5247	.6162	.7724	.8326	.7240	.5731	.6105	1.3637
23	.4948	.0000	1.2376	1.2330	1.0126	.8883	.0000	.9733	1.2716
24	.9715	1.3219	1.3244	1.2996	1.3535	1.3324	1.2795	1.2545	1.0789
25	.5227	.9014	.8111	.9892	.8649	.8497	.9380	.8253	1.1365
26	.4520	.0000	1.2101	1.3176	1.2140	1.0196	.0000	1.0425	1.2640
27	.9384	1.2861	1.2293	1.4502	1.6157	1.4780	1.3281	1.3323	1.2128
28	.2001	.5262	.5303	.6307	.6911	.6561	.6224	.5510	1.2543
29	.5507	.0000	1.3267	1.2473	1.0041	.8828	.0000	1.0023	1.3237
30	1.1156	1.7235	1.7547	1.6632	1.6515	1.6063	1.6063	1.5898	1.1044
31	1.0911	1.6219	1.4952	1.6115	1.6402	1.7144	1.7757	1.5643	1.1351
32	.4284	.0000	.7090	.7951	.8419	.8700	.0000	.7298	1.1936
33	.6355	.8813	.8539	1.0452	1.1973	1.0621	.9270	.9432	1.2695
34	.2574	.5496	.5677	.7285	.9010	.8497	.6906	.6492	1.3979
35	.3213	.0000	.5564	.5380	.4730	.4785	.0000	.4735	1.1751
36	.3762	.6804	.6923	.6485	.6407	.5794	.5933	.6017	1.1514
37	.7461	1.0781	1.0597	1.1974	1.1653	1.1042	1.0848	1.0622	1.1273

TABLE 20
CALCULATED POWER DISTRIBUTION AT 50 DAYS

Region	Average Power Per Column							Avg. Region	Max. Col./ Avg. Region Power
	Col. 1	Col. 2	Col. 3	Col. 4	Col. 5	Col. 6	Col. 7		
1	.4061	.8589	.8069	.8009	.8484	.9039	.9104	.7908	1.1513
2	.9982	1.3615	1.3372	1.2846	1.3595	1.3798	1.3753	1.2852	1.0736
3	.7162	1.0993	1.0955	1.0697	1.1543	1.1820	1.1344	1.0645	1.1104
4	.5839	.9699	.9289	.8831	.9046	.9391	.9826	.8846	1.1108
5	.5673	.9133	.9291	.8836	.8772	.8991	.8991	.8527	1.0897
6	.9566	1.3554	1.3558	1.2729	1.2398	1.2755	1.3337	1.2414	1.0922
7	.8720	1.3247	1.3949	1.3402	1.3405	1.3508	1.2997	1.2747	1.0943
8	.6236	.9098	.9640	1.0609	1.0525	1.0094	.9493	.9385	1.1305
9	.8304	1.0586	1.1769	1.3396	1.4350	1.4185	1.2359	1.2136	1.1825
10	.5352	.8539	.9024	.9239	.9008	.7538	.7416	.9017	1.1525
11	.7292	1.2405	1.2863	1.2852	1.2444	1.2499	1.2339	1.1813	1.0888
12	.9973	1.3723	1.3877	1.4030	1.3847	1.3778	1.3524	1.3108	1.0704
13	.6651	.9317	.9646	1.0535	1.0983	1.1403	1.0797	.9905	1.1513
14	.7779	1.2483	1.3093	1.2808	1.2148	1.0626	1.0663	1.1372	1.1517
15	.5106	.9660	.9756	.9600	.9121	.9009	.9365	.8802	1.1083
16	.9112	1.2797	1.3432	1.4547	1.4491	1.4367	1.3704	1.3207	1.1014
17	.5593	.7327	.7994	.9487	1.0164	.9940	.8631	.8448	1.2031
18	.5920	.8529	.9953	1.1014	1.0426	.8177	.7462	.8783	1.2540
19	.7478	1.2344	1.3400	1.4014	1.3908	1.2774	1.1866	1.2255	1.1435
20	.5496	.0000	1.1094	1.2830	1.3431	1.2530	.0000	1.1076	1.2126
21	.4135	.6464	.5782	.6978	.8184	.7520	.7112	.6596	1.2407
22	.2182	.4848	.5657	.7197	.7854	.6919	.5463	.5731	1.3703
23	.4576	.0000	1.1477	1.1543	.9340	.8058	.0000	.8999	1.2827
24	.8613	1.2634	1.3205	1.3912	1.3918	1.2832	1.2124	1.2463	1.1168
25	.5321	.9015	.8555	.9250	.9322	.8912	.9273	.8521	1.0940
26	.5943	.0000	1.3193	1.4184	1.3624	1.2015	.0000	1.1793	1.2027
27	.8284	1.1327	1.0586	1.2496	1.4517	1.4142	1.2500	1.1979	1.2119
28	.1679	.4317	.4359	.5275	.5848	.5604	.5231	.4616	1.2609
29	.4588	.0000	1.1007	1.0542	.8339	.7187	.0000	.8333	1.3209
30	.9720	1.4595	1.5517	1.5656	1.5005	1.3302	1.3527	1.3975	1.1203
31	1.0068	1.4675	1.4357	1.5465	1.5998	1.6021	1.5749	1.4619	1.0959
32	.5474	.0000	.7893	.8482	.8931	.9353	.0000	.8026	1.1653
33	.5896	.8225	.7836	.9496	1.1098	1.0409	.9098	.8865	1.2516
34	.2429	.5098	.5453	.7093	.8439	.7775	.6317	.6096	1.3965
35	.3312	.0000	.5897	.5663	.4738	.4672	.0000	.4856	1.2143
36	.4417	.7809	.8321	.8295	.7723	.6460	.6604	.7090	1.1737
37	.9199	1.3187	1.3489	1.4822	1.4923	1.3923	1.3104	1.3235	1.1275

TABLE 21
CALCULATED POWER DISTRIBUTION AT 310 DAYS

Region	Average Power Per Column							Avg. Region	Max. Col./ Avg. Region Power
	Col. 1	Col. 2	Col. 3	Col. 4	Col. 5	Col. 6	Col. 7		
1	.3902	.8608	.8061	.7831	.8309	.8880	.8978	.7796	1.1517
2	.7499	1.2250	1.2243	1.1536	1.2176	1.2835	1.2458	1.1572	1.1091
3	.6819	1.1697	1.1271	1.0471	1.1230	1.1757	1.1624	1.0696	1.0992
4	.5708	1.0056	.9331	.8750	.9375	1.0207	1.0427	.9122	1.1430
5	.5487	.9620	.9489	.8778	.8764	.9280	.9532	.8707	1.1048
6	.7194	1.2153	1.2166	1.1261	1.1050	1.1697	1.2038	1.1098	1.0972
7	.7471	1.2889	1.2926	1.1887	1.1963	1.2340	1.2452	1.1704	1.1044
8	.5711	.8680	.9475	1.0457	1.0609	.9941	.8921	.9099	1.1659
9	.7667	1.1358	1.2388	1.2977	1.3393	1.3018	1.1714	1.1789	1.1360
10	.5753	.9654	1.0287	1.0435	1.0142	.9109	.8923	.9196	1.1359
11	.8516	1.3102	1.3527	1.3556	1.3453	1.3662	1.3372	1.2741	1.0722
12	.7821	1.2628	1.2940	1.3113	1.3343	1.2924	1.2305	1.2140	1.0995
13	.6580	1.0343	1.0596	1.0761	1.0932	1.1513	1.1139	1.0266	1.1215
14	.7351	1.2075	1.2482	1.2337	1.2256	1.1353	1.1083	1.1277	1.1068
15	.6053	1.0223	1.0469	1.0412	1.0014	1.0118	1.0189	.9640	1.0861
16	.7621	1.1694	1.2343	1.3155	1.3087	1.2511	1.1973	1.1755	1.1191
17	.5788	.9020	.9568	1.0271	1.0467	1.0326	.9523	.9280	1.1279
18	.6422	.9640	1.0963	1.1859	1.1419	1.0041	.9253	.9943	1.1926
19	.8332	1.2519	1.3238	1.3835	1.4097	1.3696	1.2617	1.2619	1.1171
20	.4901	.0000	.9525	1.0862	1.1159	.9962	.0000	.9232	1.2022
21	.4220	.6909	.7092	.7941	.8350	.7444	.6974	.6990	1.1946
22	.5225	.8163	.8865	1.0082	1.0585	.9526	.8422	.8695	1.2173
23	.5319	.0000	1.1397	1.2063	1.1140	.9859	.0000	.9955	1.2117
24	.7695	1.1306	1.2059	1.3616	1.3718	1.2526	1.1430	1.1764	1.1661
25	.5038	.8068	.8208	.9284	.9920	.9381	.8751	.8378	1.1840
26	.5559	.0000	1.1451	1.2515	1.1792	1.0082	.0000	1.0280	1.2174
27	.7551	1.0878	1.1272	1.2565	1.3674	1.2894	1.1372	1.1457	1.1935
28	.3682	.6530	.6467	.7367	.8028	.7685	.7269	.6718	1.1950
29	.4890	.0000	1.0156	1.0646	.9446	.8214	.0000	.8672	1.2277
30	.7662	1.1313	1.2461	1.3472	1.3384	1.2216	1.1268	1.1683	1.1531
31	.7532	1.1016	1.1277	1.2652	1.3502	1.3167	1.2052	1.1600	1.1640
32	.4920	.0000	.7696	.8270	.8255	.7819	.0000	.7372	1.1218
33	.6299	.9542	1.0183	1.0951	1.1475	1.0405	.9428	.9755	1.1763
34	.5328	.9460	.9558	1.0846	1.1924	1.1324	1.0379	.9974	1.1955
35	.4168	.0000	.6582	.6984	.6779	.6751	.0000	.6253	1.1169
36	.4392	.7218	.8085	.8801	.8600	.7190	.6692	.7283	1.2096
37	.7295	1.0521	1.1129	1.2403	1.3305	1.2340	1.0927	1.1132	1.1953

3. For the remainder of the cycle the calculated peaking factors are all acceptable.

These results represent a summary of the calculated radial power distribution during the first year. More detail is shown in other sections of this report where comparisons are made with calculated results from GAUGE or BUGTRI.

5.4. CONTROL ROD SEQUENCE

The control rod sequence used in the multidimensional depletion analysis, as shown in Fig. 8, was selected to meet the following specific requirements as given in the Technical Specifications (Ref. 22):

1. Acceptable radial power distribution for all levels of power operation.

Regional power factor ≤ 1.83 (full power or $\bar{T}_{out} > 950^{\circ}\text{F}$)

Regional power factor ≤ 3.00 (power levels where $\bar{T}_{out} < 950^{\circ}\text{F}$)

2. Acceptable calculated maximum intraregion power peaking factor for all regions. This is often referred to as the column or block tilt factor and is defined as the ratio of the average power in a fuel element column to the average power in the refueling region. For full power operation the calculated value is less than 1.46 for fully rodded regions, 1.40 for partially rodded regions, and 1.34 for unrodded regions.
3. Calculated maximum control rod worth for all possible control rod configurations during full power operation is less than $0.016 \Delta k$.

As noted by the results in the previous section, the calculated peaking factors did not satisfy these criteria at the beginning of the operating cycle. An alternate acceptable control rod sequence is discussed later in this report.

5.5. AXIAL POWER DISTRIBUTION

The reactor core safety limit, which is discussed in some detail in the Technical Specifications, was established to assure the integrity of the fuel particles. This is done by limiting the reactor core power-to-flow ratio for some total integrated operating time. In establishing the core safety limit, some limits were required for the refueling region power fraction, the hot channel factors for that region, and the coolant flow in that region. A conservative estimate of the most unfavorable axial power distribution was used. That is, the ratio of power density in the bottom layer of fuel elements of any core region to the average power density of that region is less than or equal to 0.90 for regions with control rods fully inserted or withdrawn and 1.23 for regions with control rods partially inserted more than 2 feet. This distribution must be maintained throughout the lifetime of the fuel.

The calculated average axial power distributions for the core, specific refueling regions, and within the fuel element columns as obtained from the three-dimensional as-built analysis are shown in Figs. 11 through 20. Every axial distribution is normalized to one for that point, column, region, or core. The distributions are shown throughout the first year for regions with control rods fully inserted, partially inserted, and fully withdrawn. A review of the results summarized in these figures yields the following comments and conclusions:

1. The stability of the axial power distribution during burnup can be seen from the results in Fig. 11. The core average axial power distribution shows very little change from the BOC through the EOC. The tilt toward the top half of the core is maintained throughout the cycle.
2. The average axial power distribution in refueling regions 1, 7, 16, and 31 are shown in Figs. 12 through 15 for different times during the year. These regions are shown because they are

Fig. 11. As-built core averaged axial power

0 DAYS

ROD
INSERTION

FULL IN 11, 15, 19, 20, 22, 25,
26, 28, 31, 32, 34, 37

2/3 IN 1, 9, 13, 17

1/2 IN

1/3 IN

1/6 IN

Fig. 12. As-built region 1 averaged axial power, full blends 1 and 2

0 DAYS

ROD
INSERTION

FULL IN	11, 15, 19, 20, 22, 25, 26, 28, 31, 32, 34, 37
2/3 IN	
1/2 IN	1, 9, 13, 17
1/3 IN	
1/6 IN	

150 DAYS

**ROD
INSERTION**

FULL IN	22, 28, 34
2/3 IN	
1/2 IN	1
1/3 IN	11, 15, 19
1/6 IN	

Fig. 13. As-built region 7 averaged axial power, fuel blends 3 and 4

Fig. 14. As-built region 16 averaged axial power, fuel blends 5 and 6

310 DAYS

ROD
INSERTION

FULL IN
2/3 IN
1/2 IN
1/3 IN
1/6 IN

1
22, 28, 34

Fig. 15. As-built region 31 averaged axial power, fuel blends 7 and 8

0 DAYS

**ROD
INSERTION**

FULL IN	11, 15, 19, 20, 22, 25, 26, 28, 31, 32, 34, 37
2/3 IN	
1/2 IN	1, 9, 13, 17
1/3 IN	
1/6 IN	

Fig. 16. As-built region 11 and core averaged axial power at 5 days

5 DAYS

ROD
INSERTION

FULL IN	11, 15, 19, 22, 28, 34
2/3 IN	
1/2 IN	1
1/3 IN	20, 26, 32
1/6 IN	

ROD
INSERTION

FULL IN	22, 28, 34
2/3 IN	
1/2 IN	1
1/3 IN	11, 15, 19
1/6 IN	

Fig. 17. As-built region 11 and core averaged axial power at 50 days

Fig. 18. As-built region 34 and core averaged axial power at 255 days

255 DAYS

ROD
INSERTION

FULL IN	
2/3 IN	22, 28, 34
1/2 IN	1
1/3 IN	
1/6 IN	

Fig. 19. As-built region 34 and core averaged axial power at 310 days

310 DAYS

ROD
INSERTION

FULL IN

2/3 IN

1/2 IN

1/3 IN

1/6 IN

1

22, 28, 34

Fig. 20. As-built region 11; point powers at center of region and at inner and outer points of column 6 at 50 days

50 DAYS

ROD
INSERTION

FULL IN	22, 28, 34
2/3 IN	
1/2 IN	1
1/3 IN	11, 15, 19
1/6 IN	

refueling regions that contain fuel elements with assigned layer designations and some fuel loading adjustments for the lower layer. They include fuel compositions 1 through 8. In region 1, the regulating control rods are partially inserted during the whole cycle (half in); regions 6 and 7 are unrodded during the whole cycle; and in region 31, the control rods are fully inserted initially and then fully withdrawn after 10 days. For all of these regions, the ratio of power in the lower layer to the average power is below 0.8 and well within the design limits. The effect of partially inserted control rods on the axial power distribution for the refueling region is clearly shown by the results in Fig. 12.

3. The effect that partially inserted control rods have on the axial power distribution for that refueling region can also be seen in Figs. 16 through 20. In Fig. 17, the effect is shown for control rods inserted $1/3$ of the height and in Fig. 18 the control rods are inserted $2/3$ of the height; although the effect on the distribution is quite significant, the results are well below the design limits. In Fig. 20, for refueling region 11 with the control rods $1/3$ inserted, the axial power distribution is shown at various points in the refueling region. As expected, the largest perturbation to the power distribution occurs through the centerline of the fuel element column containing the control rods. At the outer edge of the adjacent fuel element columns, the perturbation is significantly reduced.

5.6. CORE THERMAL PERFORMANCE

An analysis to determine the in-core fuel temperatures and subsequent fuel migration was made using the calculated three dimensional power distribution from the above discussed analysis. The results of that thermal analysis are reported in detail in Ref. 23 and indicated that core performance was satisfactory with the exception of a few local points which

experienced high local power peaks for an extended period. Some fuel temperatures reported there are higher than the nominal temperature reported in the FSV-FSAR (Ref. 24), due to the inclusion of various conservative assumptions in the thermal design and the use of more detailed power distributions. Subsequent analysis has shown that a minor change in operating control rod patterns reduces these high local power peaks such that all points are within design limits.

From the GATT analysis, power distribution was available at 56 points in each fuel element, 7 radial locations, and 8 axial levels during the entire first year of operation. Temperature histories were calculated for each of the in-core point-wise locations using the TREVER code with POKE generated flow correlations (Ref. 25). Some important assumptions used which lead to conservatively high temperatures include:

1. Outlet coolant temperature mismatch 100°F (all time).
2. Local power density was multiplied by the maximum value of fuel homogeneity measured for the rod lots used in the element.
3. Gap between fuel rod and graphite was taken to be a function of temperature and fast neutron dose.
4. Length of the fuel rod was taken to be

93% of active core length for standard block
88% of active core length for control rod block

This accounts for spaces between fuel rods at block interfaces.

5. No axial conduction was allowed.
6. No radial conduction between fuel holes was allowed.

Kernel migration rates were calculated as a function of local temperature and temperature gradient in annular rings of each fuel rod considered. During the first year of operation of the as-built core, kernel migration was less than 1 μm in all but a small volume of the core. Details of the "Amoeba equivalent temperatures" with respect to the whole core, by axial layer, and by fuel type are shown in Ref. 23 but are not given here because refinements to the analysis were still being made. Maximum temperature and migration distance occurred in the bottom fuel layer of type "D" fuel for the fifth reload. Specific location in the core for the "H" and "A" types of fuel had succeeded in reducing the severity of their service compared to the other fuel types. Location in each layer where significant migration, greater than 1 μm , had taken place is briefly summarized and discussed below:

<u>Layer</u>	<u>Location of Significant Migration</u>	<u>Cause</u>	<u>Comment</u>
2	Buffer layer	High power peaks	Homogeneous buffer* in GATT
3	a. Buffer layer	High power peaks	Homogeneous buffer* in GATT
	b. One ring in from buffer	High power peaks from inserted rods	Regions 22, 28, 34. More detailed burn-up will reduce temperature.
	c. Partially rodded patches 11, 15, 19	High axial peaks in 3rd layer	
	d. Patches adjacent to partially rodded patches		
	e. Patches 30, 31, 32	High power peaks	Poison rod increase recommended for regions 30 and 31
4,5,6	a. Fully rodded patches 22, 28, 34	High power peaks due to rods in region	More detailed burn-up will reduce temperatures
	b. Patches 30, 31, 32	High power peaks	

*Use of a buffer layer with a fuel composition uniform throughout the block rather than representations of the two fuel compositions actually present overestimates of the reflector peak.

6. COMPARISON WITH OTHER ANALYSIS

6.1. BEST ESTIMATED REACTIVITY FOR AS BUILT CORE

Some additional calculations were performed for the as-built core assembly to determine the best estimated reactivity at the beginning of the cycle. In all of these analyses, the same random loading pattern for the as-built fuel elements was used to determine the core nuclear characteristics for the initial assembly. These results are summarized in Table 22 where they are compared with the values from the design calculations. Since the three-dimensional calculation was made primarily to investigate power distributions, that analysis was not made for the cold reactor condition. As seen by those calculated results, the excess reactivity at the beginning of the cycle for the as-built core is about $0.005 \Delta k$ greater than in the nuclear design; this is due mainly to the reduced burnable poison loading. All of these calculations were made with as-built mass data including impurities except for the residual hydrogen and chlorine present in the fuel rods. It has been previously estimated that the effect of these two materials on excess reactivity is nearly equal in value and opposite in sign so that inclusion in the analysis would have no effect.

6.2. GAUGE-GATT CORRELATION FOR AS BUILT CORE

In Sections 4.3 and 4.4, some analytical results are given in which comparisons are made between GAUGE calculations with the Reference 12 design loading and those made with as built fuel loading. Those analyses concentrated on effective differences that result from fuel loading or nuclear model variations. They were made prior to the three dimensional depletion calculation and in part were made to insure the adequacy of the model and its effective cross sections, to detect any errors in the loading, and to anticipate to some extent the multidimensional results.

TABLE 22
CALCULATED k_{eff} AT BEGINNING OF CYCLE FOR INITIAL CORE LOADING

Calculational Model	Cold Clean Core	Hot Clean Core
GATT, 3-dimensional, 4-group, as-built	--	1.074 ^(a)
BUGTRI (2 slice) RZ, 7-group (4 thermal), as-built	1.140 ^(b)	1.075 ^(b)
BUGTRI (2 slice) RZ, 7-group (4 thermal), design	1.133	1.068
Gauge, 2-dimensional, 4-group, as-built	1.135	1.073
Gauge, 2-dimensional, 4-group, design	1.130	1.069
Gauge, 2-dimensional, 4-group, design, as-built poison	--	1.073
Best estimated - design	1.133	1.069
Best estimated - as-built	1.139	1.074

(a) Obtained by adding control rod worth Δk to k_{eff} from rodded calculation.

(b) The radial buffer zone in the as-built analysis was homogenized into the outer row of fuel elements. In a multithermal analysis, this places too much uranium next to the side reflector and increases the calculated k_{eff} by about 0.003 Δk . These values were adjusted to take this into consideration.

Using that same as built GAUGE nuclear model, a comparison was made of the calculated radial power distributions as obtained with GATT throughout the one year of power operation. Partial control rod insertion was treated in the GAUGE analysis by reducing the density of the control rods in the appropriate burnup region. The correct control rod density to achieve the correct relative rod worth was first determined from a series of GAUGE calculations. These are summarized in Figs. 21 and 22. At the beginning of the cycle with 12 rod pairs fully inserted, the density of the rods in regions 1, 9, 13, and 17 were varied between 0 and full density. The relative change in rod worth versus density is shown in Fig. 21. The sensitivity of the radial power distribution to that density change is shown in Fig. 22. Using the data given in Fig. 21 for partially inserted rods, a one year GAUGE depletion calculation was then made identical to that performed with GATT. The power level, operating time, and control rod configurations were the same as those used in the multi-dimensional analysis.

A comparison between the GAUGE and GATT calculated results is given in Figs. 23 through 26. The k_{eff} and radial power distribution as well as the percent change in power distribution are shown for 0, 5, 100, and 310 days of power operation. Besides the general agreement being very good, two important features are noted. First, the percent change in calculated power distribution is almost directly proportional to the amount of control rods inserted. Initially there are as many as 16 rods fully or partially in and the percent change is $\pm 10\%$. At 5 days, there are only 10 rods inserted, and the percent difference is generally $\pm 5\%$. At the end of the cycle with only 4 control rods partially inserted, the difference is only $\pm 2\%$. Secondly, the difference is not random. There seems to be a tilt created in the core, in that all of the plus differences are on one side of the core and the negative on the opposite. This tilt persists until the end of the recycle cycle.

As noted earlier, the difference in k_{eff} between the GAUGE and GATT varies from the beginning to the end of the cycle. At the beginning it

Fig. 21. Relative worth of control rods 1, 9, 13, 17 versus C.R. density (beginning of cycle - as built GAUGE)

Fig. 22. GAUGE radial power distribution sensitivity to control rods partially inserted in regions 1, 9, 13, 17

Fig. 23. GAUGE and GATT radial power comparison (0 days)

Fig. 24. GAUGE and GATT radial power comparison (5 days)

Fig. 25. GAUGE and GATT radial power comparison (100 days)

Fig. 26. GAUGE and GATT radial power comparison (310 days)

is $+0.001 \Delta k$ and at the end $+0.006$. This is probably caused by three dimensional depletion effects.

Some additional GAUGE and GATT correlation studies were made and the results are given in Ref. 26. This discussed in some detail the radial power tilt just mentioned. It also pointed out that although the radial region peaking factors were only within $\pm 10\%$ accuracy, the intra region peaking factors compared more favorably, generally within a few percent. In addition, it pointed out that the synthesized axial power distribution using the core average axial power distribution, the region peaking factor, and the maximum local tilt factor leads to a conservative over-estimate of the calculated maximum fuel temperatures.

7. CONCLUSIONS

7.1. OVERALL AGREEMENT AND PRIMARY ACHIEVEMENTS

As pointed out in the initial summary, the three dimensional depletion study of the as-built initial fuel was very successful in demonstrating the adequacy of the initial core in meeting the performance requirements. It clearly demonstrated that the overall power distribution and core reactivity were within the design limits. It also clearly demonstrated that the present conventional methods used in analyzing the core behavior were generally adequate or conservative. This is true in terms of power distribution as well as core reactivity and control rod worth.

In addition, it did point out some design deficiencies, such as in the control rod sequence. Also, it became evident that by selectively adding some burnable poison rods before loading the initial core, the probability of having to add burnable poison shim later would be definitely improved.

One of the more definite bonuses that was obtained from the study was the basis that this data provided for correlation with other analytical methods. A significant number of correlation studies were made with the GAUGE nuclear model. More effort should have been made in correlating these results with the BUGTRI and BUGTRI-RZ nuclear models. And practically no comparison was made with the SCANAL nuclear model. Correlation studies are still recommended for these latter two analytical models.

Another bonus was the correlation that could be made between reflector flux and core power for the different control rod configurations. Axial thermal flux plots and axial thermal flux and power offsets have been generated for the FSV core and are given in Ref. 27. These have been used to study in-core and in-reflector detector response for large HTGRs.

7.2. CONTROL ROD SEQUENCE

It was pointed out in Sections 5.3 and 5.4 of this report that the calculated intra region peaking factors at the beginning of the cycle were not within the limits stated in the Technical Specifications.

Since the calculated radial power distribution was not acceptable for all of the expected control rod configurations during power operation, a review of the control rod sequence was made to determine a revised sequence that is acceptable. The alternative sequence, given in Table 23, is exactly the same as that used in the analysis except for the 13 and 16 rod-in patterns at full power; for these patterns a more uniform radial power distribution is obtained. A comparison of the calculated regional power factors and the intra region power peaking factors for each region, using both the alternative rod sequence and the rod sequence used in the 3D analysis, is given in Figs. 27 and 28 for both the 13 and 16 control-rod-in patterns. These were calculated for the as-built core with the average core GAUGE model and it is seen that the power distribution with the alternative rod sequence is more uniform and within the Technical Specifications for these control rod configurations.

With no Xe or Pa in the core, a condition that exists at the beginning of the cycle or after a shutdown of several months, it is anticipated that a control rod configuration of from 13 to 16 rod pairs inserted is quite likely for short periods of full power operation. In fact, for the configurations summarized in Table 23, at the BOC with the clean core, the control rod pattern was 12 rod pairs fully inserted and 4 rod pairs inserted halfway, and the calculated k_{eff} was 1.003. This is pointed out because the requirements for both power distribution and maximum control rod worth are more difficult to satisfy when more control rod pairs are inserted. Because of the stability of the Sm-149 isotope, once the equilibrium concentration is established it will always be present, and because some xenon is built up during a normal reactor startup before full power is achieved, it is never expected that all 16 rod pairs will be fully inserted at full power.

TABLE 23
REVISED CONTROL ROD WITHDRAWAL SEQUENCE AND ROD WORTH, INITIAL CORE^(a)

Control Rod Sequence	Group Number	Regions Comprising Group	Rod Group Worth When Pulled in Sequence, Δk	Accumulative Rod Pair Worth, Δk	Average Worth of Rod Pair in Group, Δk
1	3C	10, 14, 18	0.034	0.034	0.012
2	2A	2, 4, 6	0.029	0.063	0.010
3	4B	21, 27, 33	0.012	0.075	0.004
4	4F	25, 31, 37	0.015	0.090	0.005
5	2B	3, 5, 7	0.033	0.123	0.011
6	3A	8, 12, 16	0.021	0.144	0.007
7	4D	23, 29, 35	0.003	0.147	0.001
8	3B	9, 13, 17	0.025	0.172	0.008
9	4E	24, 30, 36	0.011	0.183	0.003
10	4A	20, 26, 32	0.010	0.193	0.003
11	3D	11, 15, 19	0.023	0.216	0.008
12	4C	22, 28, 34	0.009	0.225	0.003
13	1	1	0.006	0.231	0.006

(a) The calculated values for the control rod group or average rod pair worth (Δk) are all relative to the full power operating condition.

Fig. 27. Radial power distribution for 13 and 16 control rod patterns

Fig. 28. Radial distribution of column tilt factors for 13 and 16 control rod patterns

The alternative control rod sequence was not recommended initially because the calculated maximum worth control rod for the 16 control rod configuration was greater than $0.016 \Delta k$. However, this control rod configuration is not possible for the full power condition, and for that configuration, if the central control rod pair, which is the regulating rod for the assembly, is partially inserted the calculated maximum worth rod is about $0.016 \Delta k$ (see Table 24). Since the regulating rod pair will be kept partially inserted throughout the reactor operation, this requirement is also satisfied.

TABLE 24
CALCULATED MAXIMUM WORTH CONTROL ROD PAIR IN "AS-BUILT" CORE FOR
REVISED REFERENCE CONTROL ROD SEQUENCE

Control Rod Configuration	Calculated Maximum Worth Rod	
	Region No.	Worth, Δk
16 Rods in 1,4C,3D,4A,4E,3B	19	0.0176
16 Rods in Rod Group 1 out	19	0.0147
16 Rods in Rod Group 1 Part out	19	0.016
13 Rods in 1,4C,3D,4A,4E	19	0.013
10 Rods in 1,4C,3D,4A	15	0.015

Therefore, it was recommended that the sequence given in Table 23 be used as the reference control rod sequence for the initial core operation.

7.3. BURNABLE POISON SHIM PRIOR TO CORE LOADING

Earlier in this report it was shown that the excess reactivity as calculated for the as built core was different from the design values, both at the beginning and end of the cycle. The three dimensional GATT analysis showed the hot full power excess reactivity to be about $0.005 \Delta k$ higher than the design value at the beginning of the cycle and about $0.002 \Delta k$ lower at the end of the cycle. The increase at the beginning

of the cycle is due mainly to the reduced burnable poison loading in the as-built burnable poison rods. The decrease at the end of the cycle when the poison is essentially depleted was due mainly to three-dimensional depletion effects.

In addition to the slight change in the calculated excess reactivity, the calculated worth of the maximum worth control rods from the all-rods-in configuration increased by about $0.006 \Delta k$ for a single control rod pair and about $0.010 \Delta k$ for two control rod pairs. This was due primarily to the slightly greater uranium loading and reduced thorium loading for refueling regions 30 and 31 in the as-built core (see Table 6). Consequently, the core shutdown margin with one or two rod pairs stuck out was reduced by about $0.011 \Delta k$ and $0.015 \Delta k$, respectively, at the beginning of the cycle. A summary of some important nuclear parameters calculated for the initial as-built core is given in Table 2, and a comparison is made with those calculated for the reference design. It can be seen that sufficient shutdown margin and core excess reactivity still existed.

Since the nominal core shutdown margin at room or refueling temperature with all control rods inserted is typically larger than $0.08 \Delta k$, the shutdown margin of consequence is with one or two rod pairs stuck out. The rod withdrawal accident, considered to be the maximum credible reactivity accident, assumes that the rod being withdrawn cannot be reinserted. Therefore, as required in the Technical Specifications, the reactor cannot be operated at power unless a sufficient number of control rod pairs are operable to ensure that safe shutdown can be achieved with the maximum worth rod pair fully withdrawn. This means that to operate with one rod pair inoperable, assuming it to be the maximum worth rod pair, the reactor must be capable of being shut down with the two maximum worth control rod pairs fully withdrawn. Specifically, the reactor must be capable of being shut down at refueling temperature for a period of at least 1 week until appropriate action can be taken. The continual decay of Pa-233 will lead to a gradual increase in reactivity; however, over a period of 1 week, this increase is small.

A summary of the calculated core excess reactivity and shutdown margin for the as-built initial core for both the one and two stuck rod conditions is shown in Fig. 29 for the first year of operation. For the excess reactivity and the shutdown margin with the maximum worth rod stuck out, there is about $0.02 \Delta k$ margin during the first 6 months of operation. However, the calculated shutdown margin with two rods stuck (the maximum worth rod pair inoperable) is considerably reduced and is less than $0.02 \Delta k$ during the first 6 months.

Although sufficient margin did exist with the as-built core, it was considered desirable from an operational standpoint to increase the shutdown margin. The uncertainty on the calculated shutdown margin with one or two stuck rods out is certainly greater than that on the core excess reactivity, and a greater initial shutdown margin than that indicated in Fig. 29 would reduce the probability of having to add shim after the core loading was completed. An analysis was made to determine the effect of adding additional burnable poison rods to reduce the beginning-of-cycle excess reactivity and, at the same time, reduce the worth of the maximum worth control rod, thereby increasing the shutdown margin.

The change in excess reactivity and maximum control rod worth at room temperature for three different shim methods was calculated and is compared to the as-built core in Table 25. In all three cases, an additional poison rod was added to those fuel elements containing fuel blends 7 and 8, since the k_{∞} for these materials is the highest and consequently the maximum worth control rods are located in refueling regions with these materials. The results for all three cases are quite similar, but the most desirable location is in the buffer fuel elements for regions containing fuel blends 7 and 8.

The calculated worth of the burnable poison in the initial core is $0.056 \Delta k$. The loading consists of two type I poison rods in each of the fuel element columns as shown in Fig. 30. By adding one additional type I poison rod to the buffer fuel elements for fuel blends 7 and 8, as shown in Fig. 30, the initial poison loading would be increased by about 10%.

Fig. 29. Excess reactivity (hot) and shutdown margin (cold) for the as-built core and with shim poison added

TABLE 25
EFFECT OF ADDING SHIM POISON RODS TO INITIAL FSV CORE

	Initial As-Built Core	Add One Type I Poison Rod		Add One Type II Poison Rod Blends 7 and 8 Buffer Element
		Blends 7 and 8 Nonbuffer	Blends 7 and 8 Buffer Element	
k_{eff} - hot, unrodded, BOC (GAUGE)	1.0731	1.0692	1.0698	1.0688
Worth (Δk) of 144 added poison rods		-0.0039	-0.0033	-0.0043
k_{eff} - cold all rods in, BOC (GAUGE)	0.9128	0.9098	0.9105	0.9104
Rod 30 out	0.9640	0.9566	0.9547	
Maximum worth rod (Δk)	0.051	0.047	0.044	
Rods 31 and 30 out	1.0093	1.0008	0.9976	0.9972
Maximum worth two control rods (Δk)	0.098	0.091	0.087	0.087

Fig. 30. Lumped poison distribution in the initial core with additional poison rods added

The total worth of the burnable poison in the as built core would then be about the same as in the reference design. The actual boron loading for the type I poison rods had been 10% less than that requested (see Section 2.2), and by adding the shim poison rods, this difference was made up.

The effect this had on the initial core performance is shown in Table 2 and Figs. 29 and 31. The beginning of cycle excess reactivity (hot) was decreased by $0.003 \Delta k$, but the end of cycle reactivity was unchanged. The power distributions as given in Fig. 31 are essentially unchanged but the shut down margin was increased significantly (see Fig. 29).

Therefore, it was recommended that the initial burnable poison loading be modified as indicated prior to initial core loading. This was done in Ref. 28.

Fig. 31. Calculated radial power distribution with and without shim poison rods added

APPENDIX

COMPUTER TAPES FOR PSC AS BUILT CALCULATIONS

<u>Description</u>	<u>Tape No.</u>
Rods data tape	6272
Loading sequence data file	4987
WONDER PCF tape	2057
(Back-up)	4948
Cross sections	
INTERP data tape (TAPER)	3604
Degenerate GATT cross section (FINITO)	7455
Full core GATT cross section with reflector (GZINTS and TURKEY)	2599
Full core GATT cross section without reflector (GZINTS)	6707
GZINTS input data (C/U, NTH, Temp) for full core GATT (WONDER)	6181
Atom densities	
GATT	
Full core (WONDER)	6537
Degenerate (WONDER)	7147
BUGTRI	
Upper and lower core with correction to control rod blocks in lower core (WONDER)	1692
GAUGE	
Axial smear - no correction to central blocks	
Unformatted write (WONDER)	7386
NTRAN write (WONDAU)	5604
Axial smear - corrections to central blocks	
Unformatted write (WONDER)	7493
NTRAN write (WONGAU)	7505
Upper core - NTRAN write (WONGAU)	4176
Lower core - NTRAN write (WONGAU)	3907

	<u>Description</u>	<u>Tape No.</u>
BUGTRI results		
Upper core		
	Data file	3554
	Flux results (13)	5285
	Restart (21)	5299
Lower core		
	Data file	7540
	Flux results (13)	6370
	Restart (21)	6674

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