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— Reload Report —

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1. INTRODUCTION AND SUMMARY

This report justifies the operation of the second cycle of Arkansas Nuclear One (ANO), Unit 1 at the rated core power of 2568 MWt. Included are the required analyses outlined in the USNRC document, "Guidance for Proposed License Amendments Relating to Refueling," dated June 1975. Supporting Cycle 2 operation of ANO Unit 1, this report employs analytical techniques and design bases established in reports that have been accepted by the USNRC (see references).

Cycle 1 and 2 reactor parameters related to power capability are summarized briefly in section 5. All accidents analyzed in the FSAR have been reviewed for Cycle 2 operation. In cases where Cycle 2 characteristics were conservative as compared to those analyzed for Cycle 1 operation, no new accident analyses were performed.

The Technical Specifications have been reviewed, and the modifications required for Cycle 2 operation are justified in this report. Based on the analyses and considering the postulated effects of fuel densification and the Final Acceptance Criteria for Emergency Core Cooling Systems, ANO Unit 1 can be operated safely during Cycle 2 at the rated core power level of 2568 MWt.

2. OPERATING HISTORY

Unit 1 of the Arkansas Nuclear One (ANO) station achieved initial criticality on August 6, 1974, and power escalation commenced on August 13, 1974. The 100% power level of 2568 MWt was reached on December 8, 1974. A control rod interchange was performed at 242 effective full-power days (EFPD). The design fuel cycle is scheduled for completion in January 1977, after 490 ± 10 EFPD. The first cycle involved no operating anomalies that would adversely affect fuel performance during the second cycle.

Operation of Cycle 2 is scheduled to begin in March 1977. The design cycle length is 272 ± 10 EFPD; no control rod interchanges are planned.

3. GENERAL DESCRIPTION

The Arkansas Nuclear One (ANO) — Unit 1 reactor core is described in detail in Chapter 3 of the FSAR.[1] The Cycle 2 core consists of 177 fuel assemblies (FA), each of which is a 15 by 15 array containing 208 fuel rods, 16 control rod guide tubes, and one incore instrument guide tube. The fuel rod cladding is cold-worked Zircaloy-4 with an OD of 0.430 inch and a wall thickness of 0.0265 inch. The fuel consists of dished end, chamfered, cylindrical pellets of uranium dioxide which are 0.700 inch in length for batches 2 and 3, 0.600 inch long for batch 4 and 0.370 inch in diameter. (See Tables 4-1 and 4-2 for additional data.) All FAs in Cycle 2 maintain a constant nominal fuel loading of 463.6 kg of uranium. The undensified nominal active fuel lengths and theoretical densities vary slightly between batches. Specific values are given in Tables 4-1 and 4-2.

Figure 3-1 is the core loading diagram for ANO-1, Cycle 2. The initial enrichments of batches 2 and 3 were 2.72 and 3.05 wt % uranium-235, respectively. Batch 4 is enriched to 2.64 wt % uranium-235. All the batch 1 assemblies will be discharged at the end of Cycle 1. Some of the batch 2 and 3 assemblies will be shuffled to new locations. The batch 4 assemblies will occupy primarily the periphery of the core and eight locations in its interior. Figure 3-2 is an eighth-core map showing the assembly burnup and enrichment distribution at the beginning of Cycle 2.

Reactivity control is supplied by 61 full-length Ag-In-Cd control rods and soluble boron shim. Besides the full-length control rods, eight partial-length axial power shaping rods (APSRs) are provided for additional control of axial power distribution. The Cycle 2 locations of the 69 control rods and the group designations are shown in Figure 3-3. The core locations of the total pattern (69 controls rods) for Cycle 2 are identical to those of the reference cycle indicated in Chapter 3 of the FSAR.[1] However, the group designations differ between Cycle 2 and the reference cycle to minimize power peaking. Neither control rod interchanges nor burnable poison rods are required for the designed operation in Cycle 2.

The nominal system pressure is 2200 psia, and the core average densified nominal heat rate is 5.81 kW/ft at the rated core power of 2568 MWt. There were no relevant changes in core design between the reference and reload cycles. The same calculational methods and design information were used to obtain the important nuclear design parameters. In addition, no significant operational procedure changes exist from the reference cycle with regard to axial or radial power shape control, xenon control, or tilt control. The operational limits (Technical Specifications changes) for the reload cycle are shown in section 8.

A fuel melt limit of 19.4 kW/ft has been employed in calculating the reactor protection system (RPS) setpoints. Thirteen ANO-1 batch 3 FAs have centerline melt limits between 19.2 and 19.6 kW/ft (class 3 fuel). However, only two assemblies, 1C56 and 1C59, need selective loading based on 19.4 kW/ft. Thus, a sufficient fuel melt margin will be maintained through Cycle 2.

Figure 3-1. Core Loading Diagram for ANO Unit 1, Cycle 2

Fuel Transfer Canal →

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A						4	4	4	4	4					
B				4	4	4	2 F-7	3 A-8	2 F-9	4	4	4			
C			4	3 B-4	3 A-6	2 C-6	2 D-5	3 B-8	2 D-11	2 C-10	3 A-10	3 B-12	4		
D		4	3 D-2	2 E-8	3 A-7	3 B-5	4	3 C-4	4	3 B-11	3 A-9	2 H-11	3 D-14	4	
E		4	3 F-1	3 G-1	3 C-3	2 E-6	2 B-7	2 G-4	2 E-9	2 E-10	3 C-13	3 G-15	3 F-15	4	
F	4	4	2 F-3	3 E-2	2 F-5	2 D-7	3 B-6	2 C-8	3 B-10	2 G-12	2 F-11	3 E-14	2 F-13	4	4
G	4	2 G-6	2 E-4	4	2 G-2	3 F-2	3 D-3	2 G-8	3 C-12	3 F-14	2 G-14	4	2 E-12	2 G-10	4
H	4	3 H-1	3 H-2	3 N-[illegible]	2 N-7	2 H-3	2 H-7	2 H-8	2 H-9	2 H-13	2 D-9	3 D-13	3 H-14	3 H-15	4
K	4	2 K-6	2 M-[illegible]	4	2 K-2	3 L-2	3 O-4	2 K-8	3 N-13	3 L-14	2 K-14	4	2 M-12	2 K-10	4
L	4	4	[illegible] L-3	3 M-2	2 L-5	2 K-4	3 P-6	2 O-8	3 P-10	2 N-9	2 L-11	3 M-14	2 L-13	4	4
M		4	3 L-1	3 K-1	3 O-3	2 M-6	2 P-7	2 K-12	2 P-9	2 M-10	3 O-13	3 K-15	3 L-15	4	
N		4	3 N-2	2 M-5	3 R-7	3 P-5	4	3 O-12	4	3 P-11	3 R-9	2 M-8	3 N-14	4	
O			4	3 P-4	3 R-6	2 O-6	2 N-5	3 P-8	2 N-11	2 O-10	3 R-10	3 P-12	4		
P				4	4	4	2 L-7	3 R-8	2 L-9	4	4	4			
R						4	4	4	4	4					

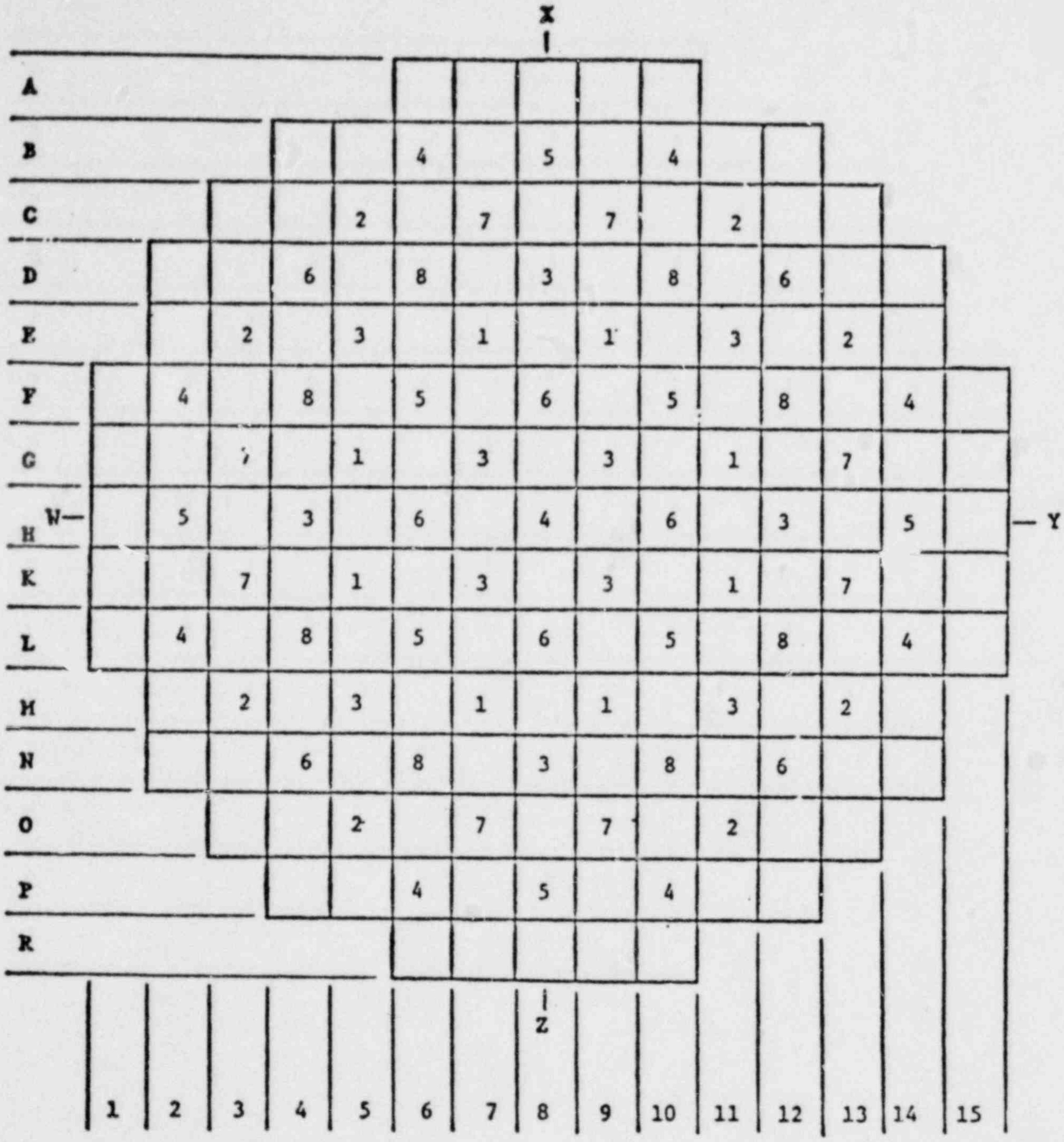
Batch / Previous Core Location

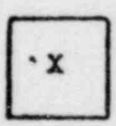
Figure 3-2. ANO Unit 1 Enrichment and Burnup Distribution for Cycle 2

	8	9	10	11	12	13	14	15
H	2.72 14,316	2.72 18,613	2.72 18,165	2.72 18,814	3.05 14,083	3.05 16,160	3.05 12,140	2.64 0
K		3.05 13,170	3.05 13,872	2.72 15,368	2.64 0	2.72 16,618	2.72 19,742	2.64 0
L			2.72 18,628	2.72 18,835	3.05 12,608	2.72 16,493	2.64 0	2.64 0
M				3.05 9,351	3.05 11,616	3.05 8,944	2.64 0	
N					2.72 19,794	3.05 8,743	2.64 0	
O						2.64 0		
P								
R								

2.72	Initial enrichment
14,316	BOC burnup; MWd/mtU

Figure 3-3. Control Rod Locations for ANO Unit 1, Cycle 2





Group Number

Group	Number of Rods	Function
1	8	Safety
2	8	Safety
3	12	Safety
4	9	Safety
5	8	Control
6	8	Control
7	8	Control
8	8	APSRs
TOTAL	69	

4. FUEL SYSTEM DESIGN

4.1. Fuel Assembly Mechanical Design

Pertinent fuel design parameters are listed in Table 4-2. All fuel assemblies are identical in concept and are mechanically interchangeable. The new FAs have modified end fittings, primarily to reduce FA pressure drop and to increase holddown margin. All other results presented in the FSAR fuel assembly mechanical discussion are applicable to the reload fuel assemblies.

4.2. Fuel Rod Design

Pertinent fuel rod dimensions for residual and new fuel are listed in Table 4-3. The mechanical evaluation of the fuel rod is discussed below.

4.2.1. Cladding Collapse

Creep collapse analyses were performed for three cycle assembly power histories for ANO-1 and yielded acceptable results. A three-cycle design power history for batch 4 fuel has not been finalized; thus, no collapse analysis was performed for this batch. However, batch 2 and 3 fuel is the worst case for Cycle 2 operation due to its previous incore exposure time. Since the analytical parameters of the fresh batch 4 fuel are similar to those of the limiting batch 3 fuel, and batch 4 will have a shorter three-cycle exposure, there is no collapse concern with batch 4 fuel. Batch 2 and 3 fuel pellets were resintered, and the lowest (worst-case) density was determined from as-fabricated data. This was coupled with the projected worst-case assembly power history to determine the most limiting collapse time as described in BAW-10084P-A.[2] Measured power distribution data obtained during Cycle 1 operation confirmed the accuracy of the Cycle 1 design calculations used for the collapse analysis. The conservatisms of the analytical procedure are summarized below.

1. The CROV computer code was used to predict the time to collapse. CROV conservatively predicts collapse times, as demonstrated in reference 2.
2. No credit is taken for fission gas release. Therefore, the net differential pressures used in the analysis are conservatively high.

3. The cladding thickness used was the lower tolerance limit (LTL) of the as-built measurements. The initial ovality of the cladding was the upper tolerance limit (UTL) of the as-built measurements. These values were taken from a statistical sampling of the cladding.

4. Batch 2 and 3 fuel pellets underwent additional sintering, producing a range of densities. For this collapse analysis, the rods that had the lowest as-fabricated pellet densities were conservatively assumed to be located in the worst-case power region of the core.

The most limiting assembly was found to have a collapse time greater than the maximum projected three-cycle exposure life of 25,344 hours (see Table 4-2). Input to the analysis is shown in Table 4-4. This analysis used the assumptions on densification described in reference 2.

4.2.2. Cladding Stress

The ANO Unit 1 fuel densification report[3] analysis indicated that the batch 1 fuel was the most limiting from a stress point of view. Batch 4 fuel has a higher nominal theoretical density and a greater pin prepressurization than the batch 1 fuel. Therefore, the analysis performed in the fuel densification report[3] conservatively envelopes the worst-case conditions for Cycle 2 fuel.

4.2.3. Fuel Pellet Irradiation Swelling

The fuel design criteria sepcify a limit of 1.0% on cladding circumferential plastic strain. The pellet design is such that the plastic cladding strain is less than 1% at 55,000 MWd/mtU. The conservatisms in this analysis are listed below.

1. The maximum specification value for the fuel pellet diameter was used.
2. The maximum specification value for the fuel pellet density was used.
3. The cladding ID used was the lowest permitted specification tolerance.
4. The maximum expected three-cycle local pellet burnup is less than 55,000 MWd/mtU.

4.3. Thermal Design

The core loading for Cycle 2 operation is shown in Figure 3-1. There are 56 fresh (batch 4) FAs and 121 once-burned (batches 2 and 3) FAs, which are thermally and geometrically similar. However, batch 4 fuel has a lower nominal initial density, slightly different dimensions, and a higher linear heat rate capability (Table 4-5). The heat rate capability of batches 2 and 3 varied on an assembly basis because the fuel pellets were resintered to increase the initial nominal density to 96.0% TD.[4] This resintering resulted in pellet diameter and density variations. As described in reference 4, some of the variations were large enough to effectively reduce the allowable heat rate to which the resintered fuel may be exposed while maintaining the desired safety margins. To minimize selective fuel loading, a design linear heat rate limit of 19.4 kW/ft was selected for the batch 2 and 3 fuel in the Cycle 2 core. The linear heat rate capability for batch 4 fuel is 20.15 kW/ft. Linear heat rate capabilities are based on centerline fuel melt and were established utilizing the TAFY-3[5] code with full fuel densification penalties.

4.3.1. Power Spike Model

The power spike model used in this analysis is the same as that presented in BAW-10055[6] with modifications applied to F_g and F_k as described in reference 7. These probabilities have been changed to reflect additional data from operating reactors that support a somewhat different approach and yield less severe penalties due to power spikes. F_g was changed from 1.0 to 0.5. F_k was changed from a Gaussian to a linear distribution reflecting a decreasing frequency with increasing gap size. The maximum gap size versus axial position is shown in Figure 4-1, while the power spike factor versus axial position is shown in Figure 4-2. These calculated values are based on an initial fuel density of 93.5% TD and an enrichment of 3.0 wt % uranium-235. The gap size and power factor for the fuel in the Cycle 2 core would be smaller because of the lower enrichment of the batch 4 fuel and the higher density of batch 2 and 3 fuel. Although the initial enrichment of batch 3 fuel in Cycle 1 was slightly greater (3.05%), applying these factors to the Cycle 2 thermal-hydraulic design (section 6) yields conservative results.

4.3.2. Fuel Temperature Analysis

Thermal analysis of the fuel rods assumed in-reactor fuel densification to 96.5% TD. The basis for the analysis is given in references 5 and 6 with the following modifications:

1. The code option for no restructuring of fuel has been used in this analysis in accordance with the NRC's interim evaluation of TAFY.
2. The calculated gap conductance was reduced by 25% in accordance with the NRC's interim evaluation of TAFY.

During Cycle 2 operation the highest relative assembly power levels occurred in batch 3 fuel (Figures 3-1 and 5-1). The fuel temperature analysis documented in the ANO-1 fuel densification report[3] is based on batch 1 fuel, (the limiting fuel for Cycle 1) and is conservative with respect to the fuel in the Cycle 2 core. The results of this analysis predict an average fuel temperature of 1330F at 5.79 kW/ft, which is shown in Table 4-5 as the average fuel temperature at nominal heat rate for batch 2 and 3 fuel. A re-analysis, based on batch 4 fuel specification parameters, results in a slightly lower predicted average fuel temperature, 1315F at 5.8 kW/ft, shown in Table 4-5 for batch 4 fuel. Both analyses are based on BOC (beginning-of-cycle, zero burnup) conditions.

4.4. Material Design

The chemical compatibility of all possible fuel-cladding-coolant-assembly interactions for the batch 4 fuel assemblies is identical to that of the present fuel.

4.5. Operating Experience

The Mark B-4 FA does not constitute a departure from past design philosophy. Its adequacy has been verified by the operating experience of the six B&W 177-FA plants. As of September 1, 1976, the operating experience shown in Table 4-1 has been amassed for the six 177-FA plants using the Mark B fuel assembly.

Table 4-1. Operating Experience

Reactor	Current cycle	Maximum assembly burnup, MWd/mtU	Cumulative net electrical output, MWh
Oconee 1	3	21,304	13,826,959
Oconee 2	2	19,020	8,687,947
Oconee 3	1	19,163	8,745,066
Three Mile Is. 1	2	20,777	10,105,588
Arkansas One	1	15,450	6,922,082
Rancho Seco	1	7,553	2,042,758

Table 4-2. Fuel Design Parameters

	Residual fuel assembly		New fuel assembly
	Batch 2	Batch 3	Batch 4
Fuel assembly type	Mark B-3	Mark B-3	Mark B-4
Number of assemblies	61	60	56
Initial fuel enrichment	2.72	3.05	2.64
Initial fuel density, % TD	>93.11[a]	>93.11[a]	93.5
Batch burnup, BOC, MWd/mtU	17,805	11,704	0
Design life, EFPH	18,288	25,344	20,640
Cladding collapse time, EFPH	>30,000	>30,000	∿30,000[b]

[a] Variable due to additional sintering of batch 2 and 3 pellets.

[b] A detailed three-cycle collapse analysis for batch 4 will be performed for the Cycle 3 reload report. A cladding collapse time of ∿30,000 hours is a preliminary estimate based on a comparison of batch 4 design parameters with those of batches 2 and 3.

Table 4-3. Fuel Rod Nominal Dimensions

Component	Residual fuel assembly, batches 2 and 3	New fuel assembly, batch 4
Fuel rods		
OD, in.	0.430	0.430
ID, in.	0.377	0.377
Fuel Pellet		
Diameter, in.	0.367	0.370
Density, % TD	96.0	93.5
Undensified Active Fuel		
Length, in.	141.0	142.6
Flexible spacers, type	corrugated/spring	spring
Solid spacers, material	ZrO_2	none
Tubular spacers, material	ZrO_2	Zr-4

Table 4-4. Input Summary for Cladding Creep Collapse Calculations

		Residual fuel assembly, batches 2 and 3
1.	Pellet diameter, in.	0.366
2.	Pellet density, % TD	93.11[(a)]
3.	Densified pellet diameter, in.	0.364
4.	Cladding ID, in.	0.377
5.	Reactor system pressure, psia	2200
6.	Stack height (undensified), in.	144.0

[(a)] Represents a conservative value for collapse analysis.

Table 4-5. Fuel Thermal Analysis Parameters

	Batches 2 and 3	Batch 4
Fuel pellet, nominal		
Initial density, % TD	96.0[a]	93.5
Initial stack length, in.	141.00[a]	142.6
Initial diameter, in.	0.367[a]	0.370
Densified density, % TD	96.5	96.5
Densified stack length, in.	140.65	140.49
Densified diameter, in.	0.3646	0.3645
Hot channel factor on linear heat rate	1.014	1.014
Nominal linear heat rate, kW/ft	5.79[b]	5.80[b]
Average fuel temperature at nominal linear heat rate, F	1330[c]	1315
Linear heat rate to centerline fuel melt, kW/ft	19.40[d]	20.15[e]

(a) Nominal values after resintering.

(b) Based on densified length.

(c) Based on batch 1 fuel, which is more limiting than either batch 2 or 3.

(d) Design limit. The linear heat rate capability of each fuel assembly in batches 2 and 3 has been determined from resinter data.[4]

(e) Minimum capability based on batch 4 fuel specification.

Figure 4-1. Maximum Gap Size Vs Axial Position — ANO Unit 1, Cycle 2