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San Onofre PWR Data for Code Validation of MOX Fuel Depletion Analyses

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Fissile Materials Disposition Program

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ABSTRACT

The isotopic composition of mixed-oxide fuel (fabricated with both uranium and plutonium isotopes) discharged from reactors is of interest to the Fissile Material Disposition Program. The validation of depletion codes used to predict isotopic compositions of MOX fuel, similar to studies concerning uranium-only fueled reactors, thus, is very important. The EEI-Westinghouse Plutonium Recycle Demonstration Program was conducted to examine the use of MOX fuel in the San Onofre PWR, Unit I, during cycles 2 and 3. The data, usually required as input to depletion codes, either one-dimensional or lattice codes, were taken from various sources and compiled into this report. Where data were either lacking or determined inadequate, the appropriate data were supplied from other references. The scope of the reactor operations and design data, in addition to the isotopic analyses, was considered to be of sufficient quality for depletion code validation.

1. INTRODUCTION

One goal of the U.S. Department of Energy's Fissile Materials Disposition Program[1] (FMDP) is to certify the capability to predict the characteristics of mixed-oxide (MOX) spent fuel. Further, any fuel-depletion code for this purpose should be evaluated to determine how the accuracies of the computations compare with those of the more common low-enriched-uranium (LEU)-fueled light-water reactors (LWRs). Validation studies have been performed for the SCALE code system fuel-depletion analyses of both pressurized-water reactors[2,3] (PWRs) and boiling-water reactors[4] (BWRs). During the operation of these reactors, the plutonium gradually increases in the fuel that initially contained uranium as the only actinide. However, there is a much more dominant influence on the flux and depletion characteristics from the plutonium in the MOX fuel because the plutonium concentration is several times greater than that of spent LEU fuel and it has a greater sensitivity to the thermal spectrum.

The EEI-Westinghouse Plutonium Recycle Demonstration Program, sponsored by Edison Electric Institute, Westinghouse Electric Corporation, and the Atomic Energy Commission, was conducted from 1968 to 1974. A significant part of this program involved the measurement of isotopic compositions of the uranium and plutonium in depleted MOX fuel withdrawn from the San Onofre PWR, a reactor having a Westinghouse design and operated by Southern California Edison and San Diego Gas & Electric companies. The operating and design data typically required by fuel depletion codes can be found in a large number of sources.[5–12] Also, the operating reports in the range from Docket 50206-79 through Docket 50206-201 contain similar data. The objective of this report is the compilation of the required data for fuel depletion codes into a single report. Data assumptions, which are considered to be reasonable, are included where the data are lacking. Even though input data for the SAS2H module of SCALE[13] are the primary objectives of the compilation, an effort was made to include data required by lattice codes and other depletion codes.

Four MOX fuel assemblies were loaded at the start of cycle 2 of the San Onofre Nuclear Generation Station, Unit I and irradiated for both cycles 2 and 3. Then, isotopic composition analyses were conducted by the Westinghouse Electric Corporation on six sample pellets from four fuel rods of test assembly D51X. The basic parameters describing these six sample cases are listed in Table 1. Note that in all cases natural uranium was used, and three different weight percent values for fissile plutonium ($^{239}Pu + {}^{241}Pu$) were applied in the initial MOX fuel. Also, one may refer to the appendix of this report for details of the determination of the case burnups listed in Table 1.

Table 1. Basic parameters of measured MOX spent fuel samples

Case No.	ID of test assembly	Pin ID	Cycle No.	Initial enrichment (atom % ^{235}U)[a]	Initial enrichment (wt % fissile Pu)	Location ht[b] (in.)	Burnup[c] (MWd/MTHM)[d]
1	D51X	067	2	0.72	3.31	53.0	8,167
2	D51X	141	2	0.72	2.84	95.5	6,808
3	D51X	079	2, 3	0.72	3.10	49.0	20,891
4	D51X	167	2, 3	0.72	2.84	16.5	17,447
5	D51X	167	2, 3	0.72	2.84	95.5	18,713
6	D51X	167	2, 3	0.72	2.84	114.0	11,065

[a]Taken from ref. 14.
[b]Height above bottom of active fuel.
[c]Determination from ^{148}Nd measurements described in the appendix.
[d]Megawatt days per metric ton heavy metal (U + Pu).
Source: ref. 12, unless otherwise specified.

2. MOX FUEL ASSEMBLY DESIGN DATA

The reactor fuel assembly design and fuel compositions for the four San Onofre MOX assemblies loaded in cycles 2 and 3 are presented in this section. The MOX fuel assembly design data, which would be significant to depletion codes, are listed in Table 2. Most of the data are taken from a Westinghouse report[9] (WCAP-4167-2) that was issued near the start of the demonstration program. Some of the data presented in Table 2 were calculated, as noted, from other data in the table. The average soluble boron, not found in the available sources, was estimated to be 500 ppm because the average for five PWRs in previous studies[2,3] was 490 ppm. The guide tube dimensions were taken from data for the uranium-only assemblies. Although the tube sizes and boron content could have been different, the change would probably not have a significant influence on results.

The initial isotopic compositions of the uranium and plutonium in the MOX fuel of the San Onofre PWR are given in Table 3. Two deficiencies are noted in the initial plutonium data of this table. First, it is known that the plutonium is recycled plutonium from spent fuel and $^{238}Pu/^{239}Pu$ ratios are usually from 0.02 to 0.04 in typical PWR spent fuel.[2,3] However, no ^{238}Pu is listed in the available data, and its absence needs to be considered in evaluating the uncertainty related to predicted analytic results. The other deficiency in the initial Pu specifications is that the decay of the 14.35-year half-life ^{241}Pu is not considered. In the fuel-assembly fabrication program, 17 analyses were made on ^{239}Pu and ^{241}Pu in revised analyses (as indicated in Table 3) for accountability. The change in ^{241}Pu composition, thus, can be taken into account.

The fractional composition of UO_2 and PuO_2 for the different values of wt % fissile plutonium are presented in Table 4. The weight fraction of UO_2 and PuO_2 in the MOX fuel is useful in determining fuel atomic densities or, more directly, as values input in the arbitrary material data of the fuel input to SAS2H. These weight fractions were calculated from the fuel stack density of Table 2, the initial isotopic atom compositions of Table 3 and the atomic weights[14] of the U and Pu isotopes. The initial Pu isotopic data were used because the revised data are incomplete. However, the decay time for the ^{241}Pu to change (5.2 to 4.8%) was calculated to be 1.66 year, or 605 days. The depletion code input should have a 605-day step of decay (or very low power) that precedes the irradiation cycles, accounting for the ^{241}Pu change. Although the revised data are incomplete, code input may either be revised to 4.8% for ^{241}Pu or include the 605-day step of decay.

The locations of the four fuel pins analyzed from the MOX fuel assembly D51X, listed in Table 1, are shown in Fig. 1. A complete description of the assembly, showing the fissile plutonium enrichment pattern, is presented in Fig. 2. This information would probably be required for depletion lattice codes. The locations of the four MOX fuel assemblies during cycles 2 and 3, are shown in Fig. 3. Lattice depletion code input may use the fact that assembly D51X is adjacent to the reflector region in cycle 2.

Table 2. San Onofre MOX fuel assembly design data

Parameter	Data
Assembly general data	
Designer	Westinghouse Electric
Rod lattice	14 × 14
Number of assemblies/core[a]	157
MOX assemblies in cycles 2 and 3	4
Total MOX loading, metric ton heavy metal, MTHM[b]	1.335
MOX fuel/assembly, kg U + Pu[c]	333.75
Number of MOX fuel rods	180
Number of instrument tubes[d]	1
Number of guide tubes[d]	15
Equivalent core diameter, cm (in.)[a]	282 (111)
Assembly pitch, cm (in.)[c]	19.941 (7.851)
H_2O moderator pressure, psia[a]	2100
Average moderator temperature, K (°F)[b]	576.5 (578)
Average moderator density, g/cm^3 [c]	0.7179
Average clad temperature, K (°F)[b]	615 (648)
Soluble boron (estimated), ppm (wt)[e]	500
Fuel rod data	
Type of fuel pellet	UO_2 plus PuO_2 (or MOX)
Stack density, g/cm^3 [c]	10.2235
Rod pitch, cm (in.)	1.41224 (0.556)
Clad OD, cm (in.)	1.07188 (0.422)
Diametrical gap, cm (in.)	0.01905 (0.0075)
Clad thickness, cm (in.)	0.06172 (0.0243)
Clad ID, cm (in.)[c]	0.94844 (0.3734)
Pellet OD, cm (in.)	0.92939 (0.3659)
Active fuel length, cm (in.)	303.28 (119.4)
Clad material	Zircaloy-4
Guide tubes	
Material	Stainless steel-304
Tube ID, cm (in.)[f]	1.29794 (0.511)
Tube OD, cm (in.)[f]	1.35890 (0.535)

[a]Taken from ref. 6.
[b]Taken from ref. 5.
[c]Calculated from other data in table.
[d]Taken from ref. 10 or 11.
[e]Assumed from similar reactors.
[f]Assumed similar to that of U assemblies, taken from ref. 7.
Source: ref. 9, unless otherwise specified.

Table 3. Initial compositions of the uranium and plutonium

Isotope	Atom % in U or Pu, ppm ^{241}Am		
Uranium[a]			
^{234}U	0.0055		
^{235}U	0.7200		
^{238}U	99.2745		
Plutonium[b]	Initial	Revised	Decay time, d
^{239}Pu	80.6	80.7	–[d]
^{240}Pu	13.4	–[c]	–[d]
^{241}Pu	5.2	4.8	605[e]
^{242}Pu	0.8	–[c]	–[d]
Other			
^{241}Am[f]	5000		

[a]Taken from ref. 14.
[b]Initial from ref. 5; revised from ref. 9.
[c]No revised measurement reported.
[d]Decay time not calculated.
[e]Calculated decay time required to change ^{241}Pu from the initial to the revised composition.
[f]In ppm (wt) of Pu, from ref. 9.

Table 4. Fractional composition[a] of UO_2 and PuO_2 in the MOX fuel

Case	Pin ID	No. of same type pins	Wt % fissile Pu	Weight fraction in MOX (UO_2 + PuO_2)	
				of UO_2	of PuO_2
1	067	24	3.31	0.961423	0.038577
2	141	64	2.84	0.966901	0.033099
3	079	92	3.10	0.963870	0.036130
4, 5, 6	167	64	2.84	0.966901	0.033099

[a]Computed from data in Tables 2 and 3 and atomic weights of ref. 14.

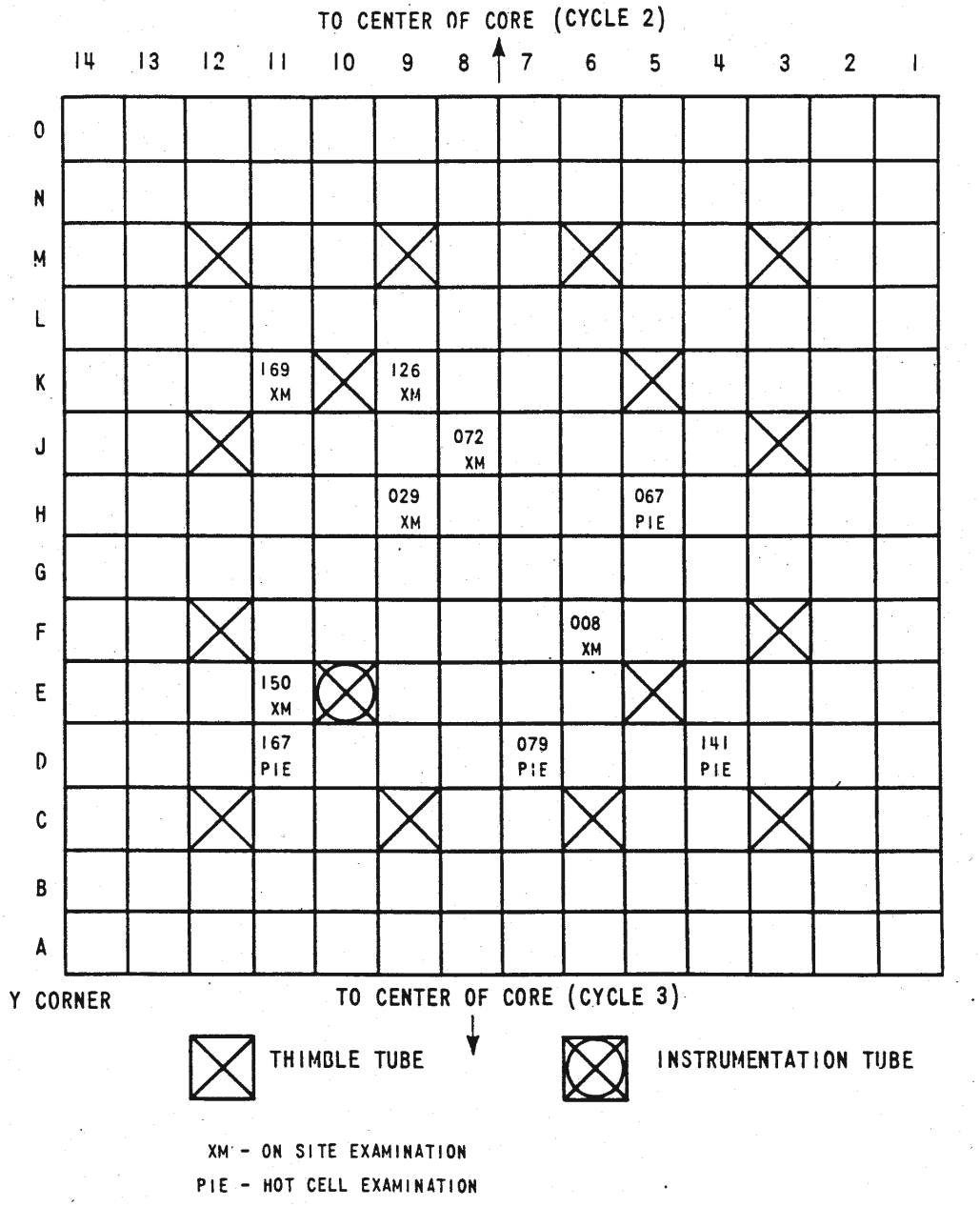

Fig. 1. Location of rods removed from San Onofre Plutonium Demonstration Assembly D51X for post-irradiation examination. *Source:* ref. 11.

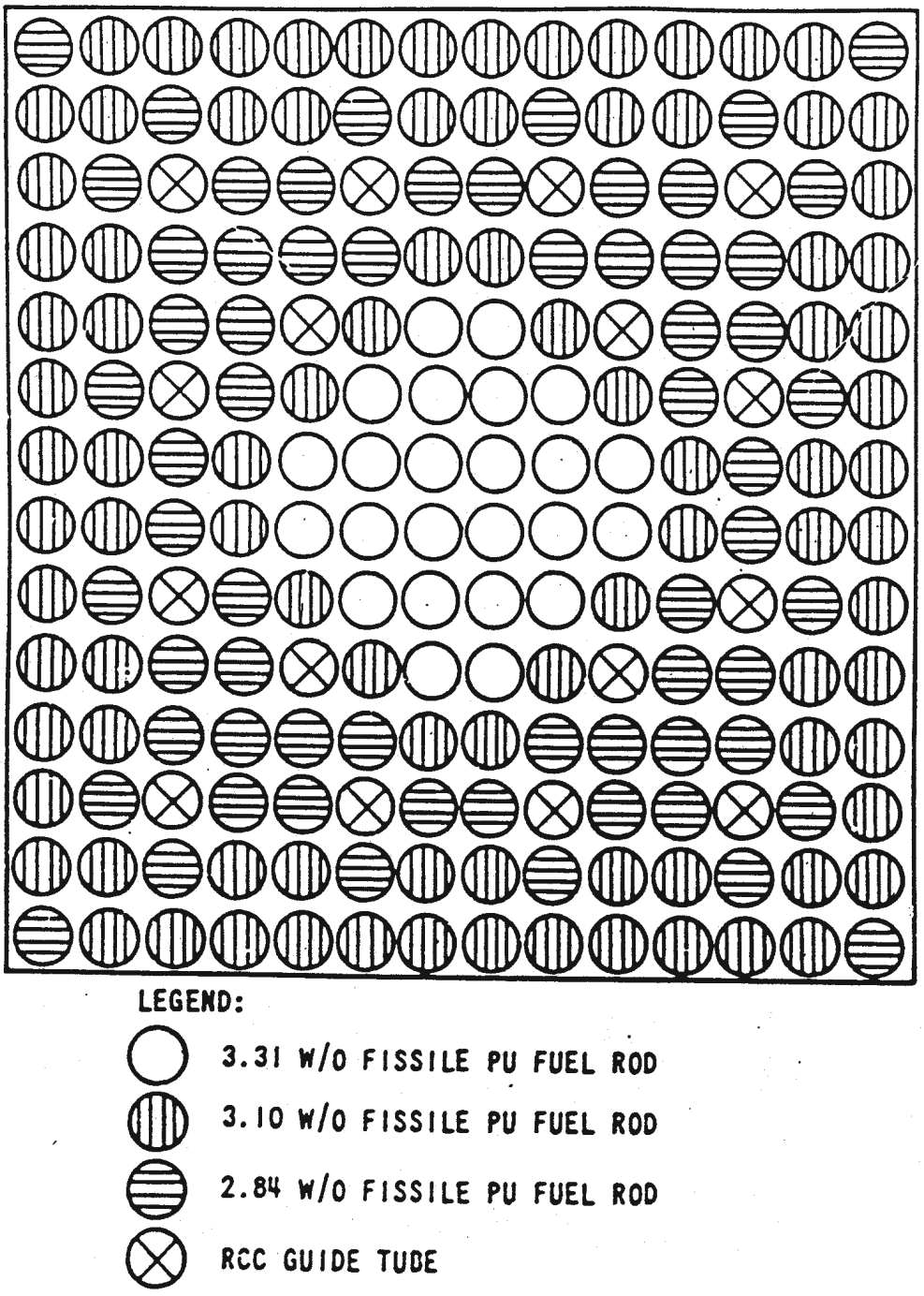

Fig. 2. Enrichment pattern for the four plutonium assemblies. *Source:* refs. 6 and 9.

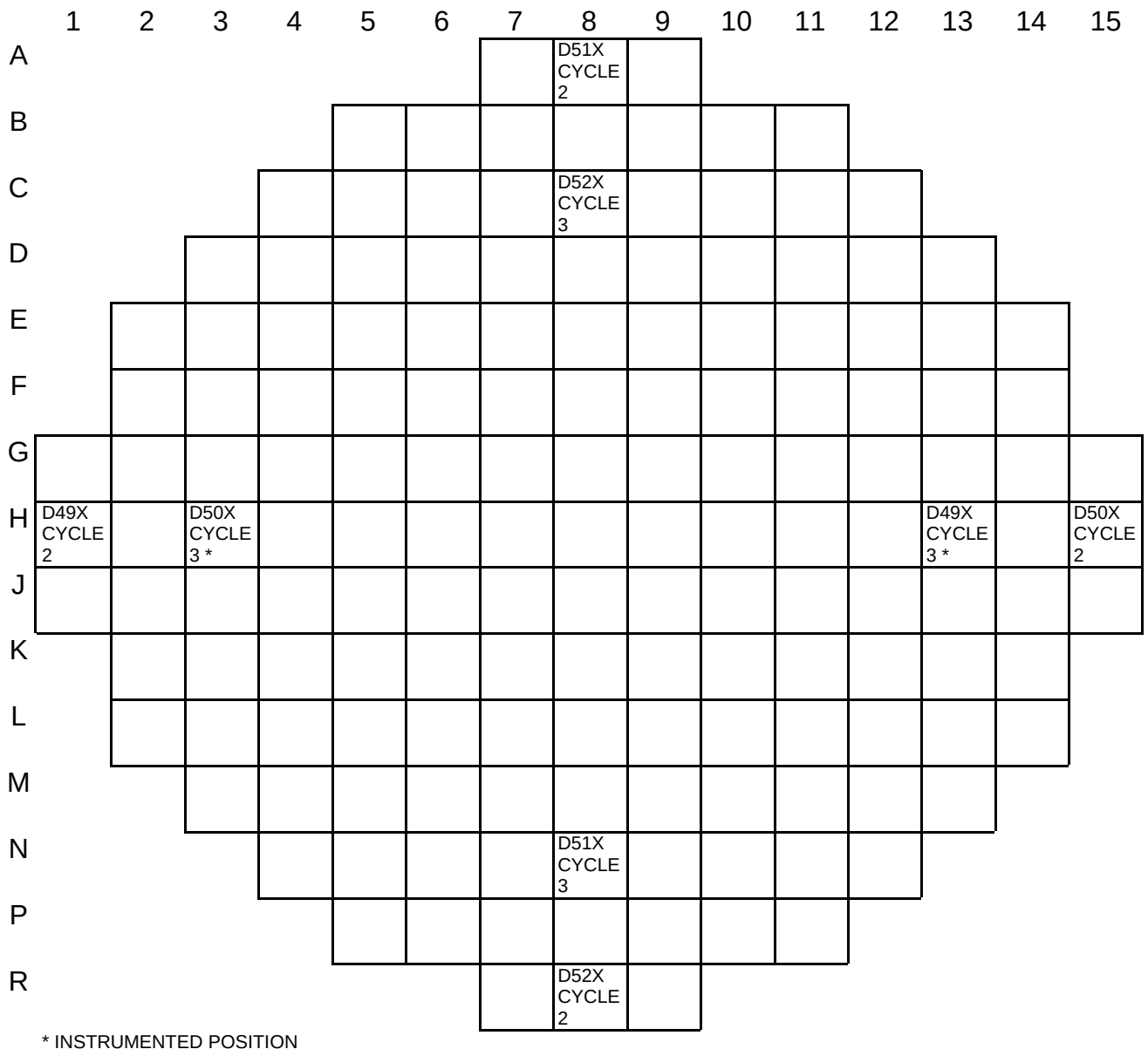

Fig. 3. Location of plutonium demonstration assemblies in San Onofre cycle 2 and cycle 3. *Source:* refs. 10 and 11.

3. THE PWR OPERATIONS DATA AND MODEL OF ASSEMBLIES

The San Onofre PWR, Unit I, operations data, pertaining to the MOX fuel assemblies in cycles 2 and 3 are presented in Table 5. The pellet sample burnups were derived from the ^{148}Nd measurements, as described in the appendix. The average cycle power experienced by each of the six pellet samples were calculated from the burnup and cycle times.

Portions of the operating conditions were listed in Table 2. The clad and water moderator temperatures[5] are in the "assembly general data" listed in the table. The moderator density was obtained by interpolation of data in the temperature-pressure-density table[13] at 578 °F and 2100 psia.[6]

The effective fuel temperatures, applied in the resonance treatment, were obtained from the fuel-temperature-vs-rod linear power curve[2] in Fig. 4. The curve was developed for the Obrigheim PWR. There are similarities in lattice (14 × 14), pellet OD (<1% difference) and moderator temperature (differ by 4.5 K) between the San Onofre and Obrigheim reactors. This method of estimating fuel temperature was applied in the validation of H. B. Robinson PWR analyses.[2] The resulting temperatures are given in Table 5.

The unit cell zone geometry data for Path-B of SAS2H is listed in Table 6.

It is recommended that cross sections should be determined during the neutronics calculation for all nuclides that are significant to the results, as recommended in the most current validation study[4] and a burnup-credit sensitivity study.[15]

Table 5. San Onofre operating data, including sample pellet powers and resonance-type fuel temperatures

Operation data type, Pin ID (height, in.)	Units	Cycle 2	Cycle 3	Total burnup
Cycle times[a]				
Startup date		11/18/70	3/1/72	
Shutdown date		12/26/71	6/2/73	
Uptime	Days	403	459	
Downtime	Days	66	–[b]	
Fuel pellet burnups[c]	MWd/MTHM			
067 (53.0)		8,167	–	8,167
141 (95.5)		6,808	–	6,808
079 (49.0)		7,015	13,877	20,891
167 (16.5)		5,999	11,448	17,447
167 (95.5)		6,434	12,279	18,713
167 (114.0)		3,843	7,222	11,065
Fuel pellet powers[d]	MW/MTHM			
067 (53.0)		20.266	–	
141 (95.5)		16.894	–	
079 (49.0)		17.406	30.232	
167 (16.5)		14.885	24.942	
167 (95.5)		15.965	26.751	
167 (114.0)		9.536	15.735	
Effective fuel temperatures[e]	K			
067 (53.0)		744	–	
141 (95.5)		713	–	
079 (49.0)		718	839	
167 (16.5)		695	787	
167 (95.5)		705	805	
167 (114.0)		650	703	
Shutdown to analysis times[b]	Days			
067 (53.0)		717	–	
141 (95.5)		719	–	
079 (49.0)			194	
167 (16.5)			187	
167 (95.5)			187	
167 (114.0)			192	

[a]Data taken from refs. 10 and 11.
[b]Time from cycle shutdown to sample analysis.
[c]Determined in the appendix from ^{148}Nd measurements.
[d]Calculated directly from the burnup- and cycle-time data.
[e]Effective fuel temperature for resonance treatment.

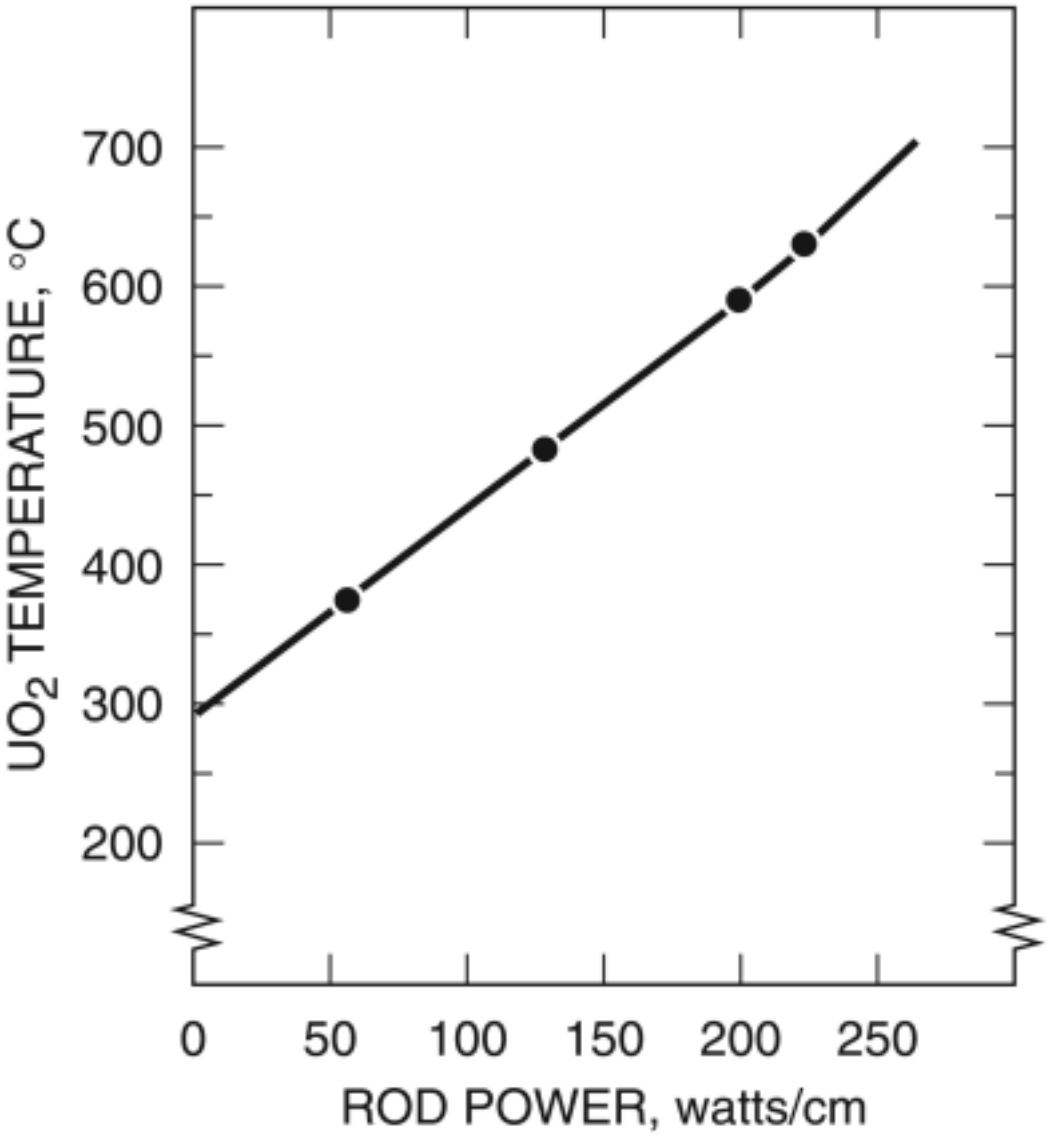

Fig. 4. Fuel-temperature-vs-rod power for Obrigheim. *Source:* ref. 2.

Table 6. Effective SAS2H geometry of the San Onofre MOX assembly model

Cycle	Radial zone	Mixture No.	Composition	Effective radius (cm)
2	1	3	Borated moderator	0.64897
2	2	5	Stainless steel-304	0.67945
2	3	3	Borated moderator[a]	0.79677
2	4	500	Homogenized fuel, clad and borated moderator	2.78870
2	5	3	Borated moderator between assemblies[b]	2.81266
3	1	3	Borated moderator	0.64897
3	2	5	Stainless steel-304	0.67945
3	3	3	Borated moderator[a]	0.84510
3	4	500	Homogenized fuel, clad and borated moderator	2.78870
3	5	3	Borated moderator between assemblies[b]	2.81266

[a]Radius different in cycles 2 and 3 to account for the removal of the two pins indicated in Tables 1 and 4.
[b]Calculated from assembly pitch and equivalent core diameter in Table 2.

4. ISOTOPIC MEASUREMENTS FOR MOX FUEL SAMPLES

Samples from the San Onofre MOX spent fuel were prepared at the Battelle Memorial Institute hot-cell facility (Columbus, Ohio). These samples were sent to the Westinghouse Waltz Mill Analytical Laboratory for the comprehensive spectrometric analysis of isotopic concentrations. The measured isotopic compositions, as atom ratios, of the six MOX sample pellets are given in Table 7. These results pertain to the listed date of analysis. The results for the uranium and plutonium isotopes adjusted to the end of the irradiation time of the sample are presented in Table 8.

The first 11 isotopic ratios (through $^{148}Nd/^{238}U$) that are listed in Table 7 were performed by mass spectrometric methods. The remaining five ratios in the table were measured by alpha spectrometry. Although these five cases were analyzed later, they were decay corrected in ref. 12 to the dates given in the table for consistency.

Table 7. Measured isotopic compositions of San Onofre MOX spent fuel in atom ratios at time of analysis

Pin ID (height, in.)	067 (53.0)	141 (95.5)	079 (49.0)	167 (16.5)	167 (95.5)	167 (114.0)
Date of analysis	12/11/73	12/13/73	12/13/73	12/06/73	12/06/73	12/11/73
Days after shutdown	717	719	194	187	187	192
Burnup, MWd,MTHM	8,167	6,808	20,891	17,447	18,713	11,065
$^{234}U/U \times 10^{-2}$	0.005	0.006	0.005	0.005	0.005	0.005
$^{235}U/U \times 10^{-2}$	0.628	0.641	0.470	0.483	0.479	0.569
$^{236}U/U \times 10^{-2}$	0.023	0.018	0.052	0.050	0.051	0.032
$^{238}U/U \times 10^{-2}$	99.344	99.335	99.473	99.462	99.465	99.394
$^{238}Pu/Pu \times 10^{-2}$	0.557	0.462	0.989	0.860	0.884	0.642
$^{239}Pu/Pu \times 10^{-2}$	71.886	73.218	56.998	57.626	57.130	66.193
$^{240}Pu/Pu \times 10^{-2}$	19.050	18.812	26.422	26.613	26.593	22.401
$^{241}Pu/Pu \times 10^{-2}$	7.210	6.384	12.530	12.047	12.444	9.088
$^{242}Pu/Pu \times 10^{-2}$	1.295	1.124	3.061	2.854	2.949	1.678
$^{239}Pu/^{238}U \times 10^{-2}$	2.619	2.293	1.741	1.594	1.601	1.965
$^{148}Nd/^{238}U \times 10^{-4}$	1.508	1.250	3.875	3.226	3.460	2.046
$^{241}Am/^{239}Pu \times 10^{-2}$	–[a]	–[a]	–[a]	6.51	6.83	1.59
$^{243}Am/^{239}Pu \times 10^{-2}$	–[a]	–[a]	–[a]	1.41	1.55	0.27
$^{236}Pu/^{239}Pu \times 10^{-9}$	4.04	4.60	17.7	12.4	13.4	6.50
$^{238}Pu/^{239}Pu \times 10^{-3}$	7.65	6.16	17.0	14.7	15.2	9.43
$^{237}Np/U \times 10^{-5}$ [b]	–[a]	–[a]	–[a]	9.7	11.1	5.7

[a]No measurement.
[b]In place of atom ratio, units are grams $^{237}Np/g$ U.
Source: ref. 12.

Although there were no indications that uncertainties or standard deviations were developed for the measured data, the reanalysis of three samples "showed excellent agreement between initial and repeat results."

Table 8. Measured isotopic compositions of San Onofre MOX spent fuel in atom ratios adjusted to shutdown time

Pin ID (height, in.)	067 (53.0)	141 (95.5)	079 (49.0)	167 (16.5)	167 (95.5)	167 (114.0)
Burnup, MWd/MTHM	8,167	6,808	20,891	17,447	18,713	11,065
$^{234}U/U \times 10^{-2}$	0.005	0.006	0.005	0.005	0.005	0.005
$^{235}U/U \times 10^{-2}$	0.628	0.641	0.470	0.483	0.479	0.569
$^{236}U/U \times 10^{-2}$	0.023	0.018	0.052	0.050	0.051	0.032
$^{238}U/U \times 10^{-2}$	99.344	99.335	99.473	99.462	99.465	99.394
$^{238}Pu/Pu \times 10^{-2}$	0.560	0.465	0.986	0.857	0.880	0.639
$^{239}Pu/Pu \times 10^{-2}$	71.345	72.729	56.744	57.379	56.877	65.979
$^{240}Pu/Pu \times 10^{-2}$	18.907	18.686	26.304	26.499	26.495	22.328
$^{241}Pu/Pu \times 10^{-2}$	7.903	7.003	12.919	12.423	12.831	9.382
$^{242}Pu/Pu \times 10^{-2}$	1.285	1.116	3.047	2.842	2.936	1.673
$^{239}Pu/^{238}U \times 10^{-2}$	2.619	2.293	1.741	1.594	1.601	1.965

Source: ref. 12.

5. SUMMARY

One of the important functions of the Fissile Materials Disposition Program is to calculate the isotopic composition of mixed-oxide fuel withdrawn from reactors. The validation of depletion codes for this purpose have been conducted[2–4] for uranium-only fueled light-water reactors.

During the period from 1968 to 1974 the EEI-Westinghouse Plutonium Recycle Demonstration Program was established to examine the use of MOX fuel. One of the reactors used in the program was the San Onofre PWR Unit I. Many reports of the preliminary plans, progress and summaries were written during the program. The main purpose of this report is to compile the assembly design and cycle operating data of the MOX fuel assemblies into a single report. It is intended that the latest and required data for most depletion codes is included in this compilation. One of the codes in which the data may be applied is the SAS2H[13] control module of the SCALE system. Also, data which may be required for lattice codes are included (Figs. 1, 2 and 3). The results of both mass and alpha spectrometer analyses, producing16 atom ratios of significant isotopes, are presented in Table 7.

An effort was made to have the correct reference given for each item of data. When the item was calculated from other data, a reference was given to the proper tables. In a few cases, where data were lacking, the method of estimation was explained. The burnup data reported,[12] having been calculated in 1974, was not considered accurate. A more universally applied method, as covered in the appendix, was used for converting ^{148}Nd data to burnup.

It was concluded that the comprehensive design and operating data and the isotopic MOX spent fuel measurements for the San Onofre PWR were of adequate quality for the proper validation of depletion codes. Calculations that were performed with SAS2H for the six samples using the data of this report indicated that the data were adequate.

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APPENDIX A

SAMPLE BURNUP DETERMINATIONS

This appendix provides an expanded explanation regarding the sample burnup data listed in Table 5. First, the reasons are given why the available burnup data[A.1] are considered inadequate, probably due to the use of methods that have become obsolete. Then, the method applied for converting the ^{148}Nd measurement to burnup is referenced and discussed. Finally, the total burnup results and the separate cycle data are shown.

There is a high degree of confidence in the ANSI/ASTM standards for converting ^{148}Nd measurements to burnup in nuclear research concerning uranium-only fueled LWRs. A total of 68 spent fuel samples from eight different reactors have been used in comparisons of measured and calculated isotopic results in previous validation studies.[A.2–A.4] The referenced burnups were, except for one reactor, derived from the ^{148}Nd analysis. In 40 of the sample cases, from five reactors, the ^{148}Nd measured result was given and compared with the SAS2H calculation that applied the ^{148}Nd-derived burnup. The computed/measured ^{148}Nd differences in the 40 cases ranged form −2.4 to 1.7%, with an average of −0.32%. The averages for the five reactors were −1.4, 1.7, 0.4, −2.0 and 0.3%, with an average of 0.20%. Or, the average of the absolute values of the reactor averages was 1.16%.

The preceding comparisons of analyses and SAS2H results validate that the methods of converting ^{148}Nd measurements to burnups are adequate. However, preliminary computations, using SAS2H, for the MOX fuel cases gave ^{148}Nd differences as large as 4% for 1-cycle cases and 12% for 2-cycle cases. Although the reports giving data that were used in the validation study did not usually give the conversion factors used in determining burnup, the data[A.5] for the Turkey Point PWR did contain the required conversion factors. Thus, the decision was made to apply the conversion factors used with the Turkey Point data in deriving burnups for the MOX fuel for the following reasons. First, applying these factors showed that the referenced burnups[A.1] were too high by approximately the same differences indicated by the ^{148}Nd code-to-measured-results comparisons. Using the same factors from the Turkey Point PWR on data for the Calvert Cliffs PWR, the Cooper BWR and the JPDR BWR of 18 cases gave values within about 1% of the reported burnups. The discussion on the Turkey Point PWR referred to the ANSI/ASTM Standards E321-75 and E267, and that on the Calvert Cliffs PWR and Cooper BWR referred to the ANSI/ASTM Standard E321-79 and E219.

One more question should be considered before using the above ^{148}Nd-burnup conversion method (i.e., factors applied in Turkey Point PWR cases) for MOX spent fuel. Higher concentrations of Pu isotopes and lower concentrations of U isotopes are noted in the average MOX fuel than in the average uranium-only fuel during irradiation. There are two factors that this can change: the average fission fraction of ^{148}Nd production, and the average recoverable energy per fission. The percent fission fractions (based on thermal reactors) for ^{148}Nd (or mass 148) listed in ENDF/B-V[A.6] are 1.670 for ^{235}U, 2.081 for ^{238}U (fast), 1.635 for ^{239}Pu, and 1.990 for ^{241}Pu. An approximate average fission fraction for a given case may be calculated by weighting these four fission fractions by the average isotopic density times the average microscopic fission cross section

of the corresponding isotope (or, rather, the isotopic average macroscopic fission cross section). In two SAS2H trial cases, average macroscopic fission cross sections of the isotopes were estimated from atomic densities and cross sections listed in the case outputs. An estimate of the sum of the weighted fission fractions indicated a 1 to 2.5% increase in the average fission fraction from MOX fuel compared with that from typical uranium-only fuel. The ORIGEN-S code computes and writes the total recoverable energy per fission for each time step. The values of the average energy per fission of MOX fueled reactors compared with uranium-only reactors indicate an increase of approximately 1 to 1.5%. The fission fraction is in the denominator, and the energy per fission is in the numerator in converting the ^{148}Nd/U atom ratio to burnup. Thus, it is estimated that the burnup could be no greater than 1.5% less for MOX fuel than that produced by the conversion method used for uranium-only fuel cases.

The detailed procedure for converting the measured atom ratio of ^{148}Nd/^{238}U to burnup by using the same conversion factors, C and F_{148}, applied in deriving the Turkey Point data,[A.5] is the following:

$$B = RCAF_u(1 - D_{238}) / F_{148} \ , \tag{A.1}$$

where

- B = calculated burnup, MWd/MTHM,
- R = atom ratio of (final ^{148}Nd atoms)/(final ^{238}U atoms),
- F_{148} = fission fraction for mass 148 = 0.0168,[A.5]
- C = 9600 (MWd/MTHM)/at. % of fuel which fissions,[A.5]
- A = ^{238}U at. % in natural U, or 99.2745,
- F_u = weight fraction of UO_2 in the MOX from Table 4,
- D_{238} = ^{238}U fractional depletion, 0.007 per cycle estimated.

F_u should be the atom fraction of U in the U + Pu of the MOX, because each term of the equation is always the ratio of atoms. For the sample compositions, the atom ratio of U to U + Pu is not significantly different than the weight ratio of UO_2 to UO_2 + PuO_2. The estimate of D_{238} is 0.007 ± 0.002, which should cause an error in B no greater than 0.2%. The method in ref. A.5 applied the same factors C and F_{148} in converting R to B, but equivalent measurements were used instead of AF_u (1 – D_{238}).

As an example of using Eq. (A.1), consider the 2-cycle case for pin ID 079, where the value of F_u from Table 4 is 0.963870 and R = 3.875 × 10^{-4}:

$$\begin{aligned} B &= 3.875 \times 10^{-4} \times 9600 \times 99.2745 \times 0.96387 \times 0.986 / 0.0168 \\ &= 20{,}891.4 \text{ MWd / MTHM.} \end{aligned} \tag{A.2}$$

The reported burnup of 23,500 MWd/MTHM is 12.5% greater than that determined in Eq. (A.2).

Burnups of the six cases computed by Eq. (A.1) are listed in Table A.1. The revised values are given in Table 5. It is seen that the old, or reported, burnups[A.1] are significantly greater than the revised burnups.

The calculation of the burnup for each cycle of the samples irradiated in both cycles 2 and 3 are given in Table A.2. The use of the reported[A.1] linear power for both cycles of the samples permitted the calculation of linear burnups, the fraction of total burnup per cycle, and the final burnup by cycle. These resulting burnups are listed in Table 5, in addition to the computed cycle powers required in depletion code input.

Table A.1 Data applied in determining recommended burnup[a]

Case	R	No. of cycles	$1-D_{238}$	% fissile Pu	F_u	B, MWd/MTHM revised	B, MWd/MTHM old	% diff.[b]
1	1.508×10^{-4}	1	0.993	3.31	0.961423	8,167	8,700	6.5
2	1.250×10^{-4}	1	0.993	2.84	0.966901	6,808	7,200	5.8
3	3.875×10^{-4}	2	0.986	3.10	0.963870	20,891	23,500	12.5
4	3.226×10^{-4}	2	0.986	2.84	0.966901	17,447	19,800	13.5
5	3.460×10^{-4}	2	0.986	2.84	0.966901	18,713	21,200	13.3
6	2.046×10^{-4}	2	0.986	2.84	0.966901	11,065	12,500	13.0

[a]Derived B by applying Eq. (A.1), the constants F_{148}, C and A, in addition to data in this table.
[b]$(B_{old}/B_{revised} - 1)$ 100%.

Table A.2 Calculation of burnup for each cycle for 2-cycle cases

	P_{linear}[a] kw/ft		Uptime, d		B_{linear}, kwd/ft		Fraction B_{total}		B_{cycle}, MWd/MTHM	
	Cycle		Cycle		Cycle		Cycle		Cycle	
Case	2	3	2	3	2	3	2	3	2	3
3	3.8	6.6	403	459	1,531.4	3,029.4	0.335774	0.664226	7,015	13,877
4	3.7	6.2	403	459	1,491.1	2,845.8	0.343817	0.656183	5,999	11,448
5	3.7	6.2	403	459	1,491.1	2,845.8	0.343817	0.656183	6,434	12,279
6	2.0	3.3	403	459	806.0	1,514.7	0.347309	0.652691	3,843	7,222

[a]P_{linear} is linear power at the sample height in ref. A.1.

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