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ST. LUCIE UNIT 1
NEW AND SPENT FUEL STORAGE CRITICALITY SAFETY EVALUATION
FOR NATURAL URANIUM AXIAL BLANKET FUEL

Prepared by

O. C. Brown

February 1986

EXXON NUCLEAR COMPANY, INC.

ST. LUCIE UNIT 1

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FOR NATURAL URANIUM AXIAL BLANKET FUEL

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1.0 INTRODUCTION

This report summarizes the results of three criticality safety analyses performed for the handling and storage of new and spent fuel at the St. Lucie Unit 1 Nuclear Generating Station. Specifically, the analyses addressed the following areas:

1) Spent Fuel Pool (Section 3.0)
2) New Fuel Storage Racks (Section 4.0)
3) Fuel Inspection Elevation, Upender, and Fuel Transfer Tube (Section 5.0)

An Exxon Nuclear Company (ENC) fuel assembly design which includes natural uranium axial blankets on the assembly ends and a central fuel region enriched to a maximum of 4.0 wt% U-235 was assumed for the analyses. Detailed descriptions of the handling and storage areas is provided in the criticality safety evaluation report summarized in Reference 1.

2.0 SUMMARY OF RESULTS

Calculations performed for the handling and storage of 4.0 wt% U-235 enriched fuel assemblies external to the St. Lucie Unit 1 reactor core indicate that the applicable criticality safety criteria are met.

Worst case reactivities calculated for each area are as follows:

Area	Calculated k_{eff} (95% CL)	Limiting k_{eff} (95% CL)	Criteria Reference
Spent Fuel Pool	0.918	0.95	2
New Fuel Storage Room	0.925	0.98	3
Fuel Elevator, Upender, Transfer Tube	0.924	0.95	2

3.0 SPENT FUEL STORAGE POOL ANALYSIS

The high capacity (HI-CAP™) spent fuel storage racks were designed to accommodate 728 fuel assemblies and are generally defined in Attachment A of Reference 1.

3.1 Design Base Fuel Assembly Description

The St. Lucie Unit 1 spent fuel storage racks are currently designed and licensed to accept fuel assemblies enriched to 3.7 wt% U-235. Table I summarizes fuel assembly parameters for this fuel design, as well as parameters for the Exxon Nuclear fuel design. Also listed in Table I are fuel assembly k_∞ calculation results obtained using the CCELL[4] computer code. (CCELL is a pin cell calculation code used to cell-average resonance corrected cross section data for input into other codes, i.e., KENO, etc.) These results indicate a +0.015 Δk_∞ change in fuel assembly reactivity between the two fuel types.

3.2 Storage Array Description (Spent Fuel Pool Analysis)

The spent fuel storage array design and dimensions are described in detail in Reference 1. The array consists of square stainless steel cans having an inside dimension of 8.4835 inches. The cans have a nominal wall thickness of 0.25 inches and are fixed in a square-pitched array on nominal 12.53-inch centers.

3.3 Storage Array Reactivity (k_{eff}) (Spent Fuel Pool Analysis)

For the nominal storage array geometry defined in Section 3.2, k_{eff} was calculated for the Exxon Nuclear fuel enriched to 4.0 wt% U-235 using the KENO IV[5] Monte Carlo computer code. (Cross section data from the XSDRN 123-energy group library were prepared for input into KENO IV using the

NITAWL[6] and XSDRNPM[6] codes. A detailed description of the calculation method is given in Section 6.0.) For a pool water temperature of 68°F, a k_{eff} of 0.874 $\pm$ 0.005 was calculated assuming an effectively infinite array. From results summarized in Reference 1, a Δk of +0.034 has been established between the nominal and worst case (minimum offset) storage array conditions. Applying this Δk to the KENO-calculated nominal k_{eff} value results in a worst case k_{eff} of 0.918(*) at the 95% confidence level.

For postulated accidents such as the dropping of a fuel assembly on top of the racks or having an assembly by accident achieve any other abnormal location in the pool, credit may be taken for realistic initial pool conditions[2]. These conditions include taking credit for soluble boron ($\geq$ 1720 ppm) present in the pool water. At about 1700 ppm, boron storage array reactivity (k_{eff}) is reduced approximately 20% Δk. Hence, postulated accidents defined above are of little concern from a criticality safety standpoint.

To better understand the reactivity sensitivity of the storage arrangement to stainless steel can wall thickness and fuel cell center-to-center spacing, several calculations were performed using the one-dimensional transport code XSDRNPM. A cylindricized-equivalent nominal storage cell was modeled with a reflective cell boundary to approximate an infinite array. Results of these calculations are summarized in Table II. For a stainless steel can wall thickness decrease of 0.05 inches, storage array reactivity increases about +0.005 Δk. For a storage cell center-to-center spacing decrease from 12.4375 inches to 12.0000 inches, results indicate a reactivity increase of about +0.020 Δk.

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* $k_{eff}(WC) = 0.874 + (2)(0.005) + 0.034 = 0.918$ (95% confidence level).

3.4 Conclusions (Spent Fuel Pool Analysis)

The results of this analysis demonstrate that Exxon Nuclear design fuel defined in Table I and enriched to 4.0 wt% U-235 can be stored in the St. Lucie Unit 1 storage pool and continue to meet NRC criticality safety criteria, i.e., the neutron multiplication factor in the pool shall be $\leq$ 0.95 under worst credible conditions. The adequacy of the calculational methods used in this analysis is discussed in Section 7.0.

4.0 NEW FUEL STORAGE ROOM ANALYSIS

The new fuel storage array was analyzed in 1979[1] for CE fuel assemblies enriched to 3.70 wt% U-235. Results of this analysis indicated the maximum storage array reactivity (k_{eff}) to be about 0.92 for all degrees of uniformly interspersed moderation.

4.1 Design Base Fuel Assembly Description

The Exxon Nuclear fuel assembly defined in Section 3.1 and Table I was used in reactivity calculations for the new fuel storage array.

4.2 Storage Array Description (New Fuel Storage Room)

The St. Lucie Unit 1 new fuel storage room consists of a 10x10 fuel assembly array arrangement with the two middle rows (running north and south) removed. The room has concrete walls around the entire array and cells are spaced on 21-inch centers. The fuel is normally stored in the dry condition. Additional details concerning the arrangement can be found in Attachment B of Reference 1.

4.3 Storage Array Reactivity (k_{eff}) (New Fuel Storage)

Since the new fuel is stored dry, reactivity calculations were performed for varying degrees of moderation in the event the array became moderated. For the case of full flooding, the array would remain subcritical due to neutron isolation between fuel assemblies. For the case of uniform moderation interspersed within and between fuel assemblies in the storage array, i.e., aqueous foam, etc., 123 energy group KENO-IV calculations were performed for water densities in the array ranging from 15% water (by volume) to 2.5%. Results of these calculations are summarized in Table III and indicate the maximum k_{eff} of the array to be 0.925 at the 95% confidence level.

4.4 Conclusions (New Fuel Storage)

The results summarized in Section 4.3 and Table III demonstrate that the maximum k_{eff} or the St. Lucie Unit 1 new fuel storage room will not exceed the limiting criterion of 0.98 for all degrees of uniform moderation in the array. Specifically, the maximum reactivity occurs for a moderator void fraction between 0.90 and 0.95 and is estimated to be about 0.925 at the 95% confidence level.

5.0 FUEL INSPECTION ELEVATOR, UPENDER AND TRANSFER TUBE

For the fuel inspection elevator, upender and transfer tube, k_{eff} calculations were performed for three worst case situations in the previous evaluation, see Attachment C of Reference 1. The case which produced the highest reactivity involved the fuel inspection elevator. For this case, it was assumed that one fuel assembly was in the elevator and one additional assembly was located four (4) inches edge-to-edge from the elevator assembly. Assuming this condition, a KENO calculation was performed for the Exxon Nuclear fuel assembly defined in Table I. This case was run with 18 energy group cross section data prepared using CCELL and gave a resulting k_{eff} of 0.914 ± 0.005 or 0.924 at the 95% confidence level. Hence, it is concluded that the fuel inspection elevator, upender and transfer tube will all meet the $k_{eff} \leq 0.95$ criterion for the 4.0 wt% U-235 enriched fuel.

6.0 CALCULATIONAL METHODS

The KENO-IV Monte Carlo code[5] was used to calculate the reactivities of the storage arrays and fuel elevator. Multigroup cross section data from the XSDRN 123 group library were generated for input into KENO-IV using the NITAWL[6] and XSDRNPM[6] codes. Specifically, the NITAWL code was utilized to obtain cross section data adjusted to account for resonance self-shielding by the Nordheim Integral Method. The XSDRNPM code, a discrete ordinates one-dimensional transport theory code, was then used to prepare spatially cell-weighted cross section data representative of the fuel assembly for input into KENO-IV. Cross section data for the fuel elevator k_{eff} calculation were prepared using the CCELL code. CCELL is a combination of the HRG[8] and THERMOS[9] codes, and produces multigroup cross section data treated in a similar fashion to the NITAWL/XSDRNPM methodology.

7.0 COMPUTER MODEL REVIEW AND VALIDATION

The calculated worst case k_{eff} (0.918) for the spent fuel storage array is about 3% less than the comparable value for 3.7 wt% fuel (0.947) reported in Reference 1. This is apparently due to a high degree of conservatism in the Reference 1 calculational model. The computer code models used for the calculations in this report have been benchmarked against experimental data and adequately reproduce the critical values. (On the average, the critical value is reproduced to within one standard deviation.) Detailed results of these benchmark calculations are given in Reference 7. In addition, the results and conclusions of this criticality safety evaluation have been second-party reviewed by an individual knowledgeable in the area of criticality safety. The findings of this review are given in Appendix I.

8.0 REFERENCES

1. Letter (with attachments) from R. E. Uhrig (FPL) to V. Stello (USNRC), "St. Lucie Unit 1 Docket No. 50-335 Proposed Amendment to Facilities Operating License DPR-67," dated October 4, 1979.

2. Letter to all Power Reactor Operators from B. K. Grimes (NRC), "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," dated April 14, 1978.

3. American Nuclear Society Standard, ANS-N18.2.

4. W. W. Porath, "CCELL Users Guide," BNW/JN-86, Pacific Northwest Laboratories, February 1972.

5. L. M. Petrie and N. F. Cross, "KENO IV: An Improved Monte Carlo Criticality Program," ORNL-4938, Oak Ridge National Laboratory, November 1975.

6. N. M. Green, et. al., "AMPX - A Modular Code System for Generating Coupled Multigroup Neutron-Gamma Libraries from ENDF/B," ORNL-TM-3706, Oak Ridge National Laboratory, March 1976.

7. C. O. Brown, "Criticality Safety Benchmark Calculations for Low-Enriched Uranium Metal and Uranium Oxide Rod-Water Lattices," XN-NF-499, Exxon Nuclear Company, Inc., April 1979.

8. J. L. Carter, Jr., "HRG-3: A Code for Calculating the Slowing-Down Spectrum in the P_1 or B_1 Approximation," BNWL-1432, Battelle-Pacific Northwest Laboratories, June 1970.

9. D. R. Skeen and L. J. Page, "THERMOS/BATTELLE: The Battelle Version of the THERMOS Code," BNWL-516, Battelle-Pacific Northwest Laboratories, September 1967.

Table I

St. Lucie Unit 1
Nominal Fuel Assembly Parameters

Type:	CE 14x14
Lattice Pitch:	0.580 inch
Clad O.D.:	0.440 inch
Clad Material:	Zr-4
Active Fuel Rods:	176
No. of Control Rod Guide Tubes:	5
Guide Tube Material:	Zr-4
Guide Tube O.D.:	1.115 inch
Guide Tube Tk:	0.040 inch
Eff. Array Dimension:	8.12 inch x 8.12 inch
Active Fuel Length:	136.7 inch

	CE 14x14	ENC 14x14
Enrichment, wt%	3.7	4.0 (assumed)
Pellet OD, inch	0.3765	0.3700
Stacked Fuel Density, g/cc	10.054	10.199
Clad Tk, inch	0.028	0.031
Moderator-to-Fuel Volume Ratio(1)	1.934	2.034
Axial U-235 Loading, g/cm	41.45	43.91(2)
k_∞ (CCELL)	1.447	1.462

1 This ratio takes into account the zirconium tubing and water associated with the instrument and control rod locations within the fuel assembly.

2 This value corresponds to an axial U-235 loading at 4.0 wt% and does not include the axial blankets of natural uranium.

Table II

St. Lucie Unit 1
Spent Fuel Storage Pool Reactivity Sensitivity Calculation Results
(NITAWL/XSDRNPM)

Case	Description	Nominal Storage Array Δk_∞	Comments
1 (base)	Enrichment - 4.0 wt% Temperature - 68°F Wall Thickness - 0.25" C-C Spacing - 12.4375"	--	Nominal Arrangement
2a	Wall Thickness - 0.20"	+ 0.0052	
2b	Wall Thickness - 0.10"	+ 0.0249	
3a	C-C Spacing - 12.0000"	+ 0.0199	Min. Offset Condition
3b	C-C Spacing - 11.5625"	+ 0.0462	

Table III

St. Lucie Unit 1
New Fuel Storage Room Reactivity Calculation Results

Fuel Assembly:	ENC 14x14 (4.0 wt% U-235)
Number of Assemblies in Room:	80
Array Geometry:	Per Ebasco Services Dwg No. 8770-G-832, Rev. 3
Calculation Model:	KENO IV with 123 group cross sections prepared using NITAWL/XSDRNPM

Case	Vol% Water in Array	$k_{eff} + \sigma$	Comments
1	2.5	0.825 ± 0.006	
2	5.0	0.905 ± 0.005 (1)	Max. Calculated k_{eff}
3	10.0	0.898 ± 0.006	
4	15.0	0.811 ± 0.006	

1 Peak k_{eff} estimated to be about 0.925 (at the 95% confidence level) and to occur between 2.5 and 5.0 vol% water in the storage array.

APPENDIX I

SECOND-PARTY REVIEW DOCUMENTATION

EXXON NUCLEAR COMPANY, INC.

Internal Correspondence

Distribution

Date: May 3, 1983

To: C. O. Brown

From: J. E. Pieper 5/3/83

Subject: CRITICALITY SAFETY ANALYSIS FOR ST. LUCIE UNIT 1 NEW AND SPENT FUEL STORAGE, XN-NF-83-36

Ref: A. S. Jameson (Combustion Engineering) to R. W. Winnard (Florida Power & Light), September 18, 1979, "Bases for Updating the St. Lucie-1 Technical Specification 5.3.1".

I have completed a review of the calculations and the analysis approach used in the study of the St. Lucie Unit 1 new, spent fuel storage and fuel inspection elevator, upender and transfer tube. I believe that in conjunction with the referenced letter, your analysis adequately demonstrates the criticality safety of Exxon Nuclear 4.0 wt.% ^{235}U fuel design when used in this equipment.

The cross-section library used for the analyses were traced through the NITAWL, XSDRNPM, NITAWL code stream. The KENO and CCELL runs were also reviewed.

The following specific runs were reviewed:

Code	Job Name	Date of Run	Time of Run
CCELL	XCEL 940	25/03/83	12.55.19
NITAWL	NITAWGE	28/03/83	13.34.40
XSDRNPM	XSDRNMP	28/03/83	13.57.27
NITAWL	NITAW81	28/03/83	15.26.40
KENO-IV	KSTLUGQ	28/03/83	19.20.06
XSDRNPM	XSDRNRU	05/04/83	20.42.17
XSDRNPM	XSDRNRX	05/04/83	20.24.10
XSDRNPM	XSDRNRO	05/04/83	20.46.15
XSDRNPM	XSDRNRZ	05/04/83	20.46.27
XSDRNPM	XSDRN7U	04/04/83	19.29.26
XSDRNPM	XSDRNDR	31/03/83	13.20.36
NITAWL	NITAWGB	31/03/83	12.57.20
CCELL	XCE19LG	14/04/83	16.04.53
KENO-2	KENO2OI	19/04/83	19.54.07
CCELL	XCEL9LF	08/21/83	12.58.58
CCELL	XCEL9FU	05/04/83	11.56.45
CCELL	XCEL9LC	05/04/83	12.26.17
CCELL	XCE19LD	05/04/83	12.26.18
NITAWL	NITAWST	08/04/83	13.24.21
NITAWL	NITAWSV	07/04/83	17.25.33

Code	Job Name	Date of Run	Time of Run
NITAWL	NITAWUO	06/04/83	15.46.14
NITAWL	NITAWSU	07/04/83	17.25.32
XSDRNPM	XSDRNV1	08/04/83	15.46.14
XSDRNPM	XSDRN9R	07/04/83	17.44.21
XSDRNPM	XSDRNXB	06/04/83	15.58.31
XSDRNPM	XSDRN9S	07/04/83	17.44.35
KENO-IV	KSTLU2D	08/04/83	17.59.02
KENO-IV	KSTLUG3	07/04/83	22.15.34
KENO-IV	KSTLUSP	07/04/83	20.11.34
KENO-IV	KSTLUHA	07/04/83	22.15.36

The following items were included in the review of the codes:

KENO:
- Options
- Correct Mixtures
- Geometry
- Adequate Histories
- K_{eff}
- Cross Sections (Data or Library tape)

CCELL:
- Geometry
- Atom Densities Verified
- Resonance Treatment
- Options
- K_∞

NITAWL:
- Input Tape
- Output Tape
- ID Numbers
- Resonance Parameters (where used)

XSDRNPM:
- Input Tape
- Output Tape
- Geometry
- Options
- Mixtures
- Convergence

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