1561

CEND-377(Vol.2)

MASTER

CONCEPTUAL DESIGN OF A LARGE SPECTRAL
SHIFT CONTROLLED REACTOR

By
R. A. Matzie
G. P. Menzel

August 1979

Work Performed Under Contract No. EN-77-C-01-5070

C-E Power Systems
Combustion Engineering, Inc.
Windsor, Connecticut

TECHNICAL INFORMATION CENTER
UNITED STATES DEPARTMENT OF ENERGY

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

DISCLAIMER

"This book was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise, does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof."

This report has been reproduced directly from the best available copy.

Available from the National Technical Information Service, U. S. Department of Commerce, Springfield, Virginia 22161.

Price: Paper Copy \$31.00
Microfiche \$3.50

CONCEPTUAL DESIGN of a LARGE SPECTRAL SHIFT CONTROLLED REACTOR

VOLUME II

**prepared for the
UNITED STATES DEPARTMENT OF ENERGY
under
CONTRACT No. EN-77-C-01-5070**

August 1979

CONCEPTUAL DESIGN OF A LARGE
SPECTRAL SHIFT CONTROLLED REACTOR

By

R. A. MATZIE and G. P. MENZEL

August 1979

The material contained in this report represents the cooperative effort of many individuals. In particular, the contribution by the Stone & Webster Engineering Corporation in establishing the PWR plant modifications to realize spectral shift control, and the contribution by the C-E Lummus Company in developing the D₂O upgrader design are gladly acknowledged. In addition, the advice and counsel of Drs. R. L. Hellens and N. L. Shapiro of Combustion Engineering, Inc. are deeply appreciated. The contributions of the following individuals whose efforts made this report possible are hereby acknowledged.

Combustion Engineering, Inc. Stone & Webster Engineering Corporation

R. W. Beekmann
R. F. Brown
A. T. Chen
H. R. Freeburn
M. C. Jacob
H. F. Jones
M. A. Krammen
Y. Liu
E. S. Markowski
J. H. Mason
J. C. Muller
K. Paduch
S. R. Pati
J. R. Rec
T. A. Schultz
F. M. Sider
G. U. Singco
J. A. Zelez

G. D. Gowdy
J. A. Voelxen

F. K. Giaccio
R. C. Johnson
R. K. Barchas

ABSTRACT

Within the framework of the Nonproliferation Alternative Systems Assessment Program (NASAP), the U.S. Department of Energy (DOE) has sponsored the development of a conceptual design of a large Spectral Shift Controlled Reactor (SSCR). This report describes the results of the development program and assesses the performance of the conceptual SSCR on the basis of fuel resource utilization and total power costs. The point of departure of the design study was a 1270 MW(e) PWR using Combustion Engineering's System 80TM reactor and Stone & Webster's Reference Plant Design. The initial phase of the study consisted of establishing an optimal core design for both the once-through uranium cycle and the denatured U-235/thorium cycle with uranium recycle. The performance of the SSCR was then also assessed for the denatured U-233/thorium cycle with uranium recycle and for the plutonium/thorium cycle with plutonium recycle. After the optimal core design was established, the design of the NSSS and balance of plant was developed.

Because of the elimination of neutron losses in control poisons and the associated improvement in fuel utilization efficiency, the fuel resource requirements of the SSCR are 11% less than those of the LWR for the UO₂ once-through and the denatured U-235/thorium cycles. The net fissile requirement of the denatured U-235/thorium cycle is 27% lower than that of the similarly fueled PWR when both U-233 and fissile plutonium are considered. The improvement in fissile consumption is considerably less for the plutonium/thorium SSCR; a reduction of only 8% in the net fissile requirement is obtained for this cycle. The resulting savings in fuel

cost are balanced by additional plant capital costs (\$26 million) and heavy water inventory costs (\$54 million), so that the power costs of the SSCR are comparable to those of the conventional LWR for the UO_2 once-through cycle when the price of uranium reaches \$100/lb U_3O_8 . For the denatured U-235/thorium SSCR, the power costs are competitive with those of the UO_2 once-through cycle only at higher uranium prices (in excess of \$100/lb).

Since the SSCR is basically a PWR with an alternate depletion-reactivity control system, its response to accident conditions is, in general, similar to that of the PWR. It does not appear, therefore, that the SSCR is confronted with any significant licensing issues which arise from spectral shift operation. Given the necessary economic incentive, the prospect of early deployment of the SSCR should be significantly better than that of other alternative systems because it is based on proven LWR technology and because the risk associated with deployment is limited to the cost of additional equipment, estimated to be only about 4% of the total cost of the plant.

CONTENTS

VOLUME I

<u>Section</u>	<u>Title</u>	<u>Page</u>
	Abstract	
1.	Executive Summary	1-1
2.	Introduction	
2.1	Study Objectives	2-1
2.2	The Spectral Shift Controlled Reactor Concept	2-4
2.3	Role of the SSCR in NASAP	2-7
2.4	Previous SSCR Development	2-9
3.	Summary	
3.1	Summary Description of SSCR	3-1
3.1.1	Design Approach	3-1
3.1.2	Overall Plant Layout	3-3
3.1.3	Core Design	3-5
3.1.4	Fuel Design	3-6
3.1.5	Primary Coolant System	3-9
3.1.6	Adjustment of Coolant Concentration	3-12
3.1.7	D ₂ O Upgrader	3-20
3.1.8	Refueling	3-26
3.1.9	D ₂ O Inventory	3-28
3.1.10	D ₂ O Leakage	3-29
3.1.11	Load Change Capability	3-32
3.2	Resource Utilization and Economic Analysis	3-34
3.2.1	Resource Requirements	3-34
3.2.2	Total Power Generation Costs	3-42
3.3	Fuel Management in the Spectral Shift Controlled Reactor	3-49
3.3.1	Difference Between Fuel Management in the SSCR and in the PWR	3-49
3.3.2	UO ₂ Once-Through SSCR Core	3-52
3.3.3	ThO ₂ SSCR Cores	3-53
3.4	Safety and Licensing	3-57
3.4.1	Safety-Related Plant Parameters	3-57
3.4.2	Accident Analysis	3-58

<u>Section</u>	<u>Title</u>	<u>Page</u>
3.4.3	Fluence at Reactor Vessel	3-61
3.4.4	Tritium Release During Normal Operation	3-61
3.4.5	Licensability	3-63
3.5	Research and Development	3-64
3.6	Commercial Introduction of the SSCR	3-67
3.7	Conclusions and Recommendations	3-71
3.7.1	Conclusions	3-71
3.7.2	Recommendations	3-75

VOLUME II

4.	Optimization of the Spectral Shift Controlled Reactor Plant Design	
4.1	Purpose of Optimization	4-1
4.2	Optimization Procedure	4-4
4.3	Core Parameters	4-7
4.4	Impact of Core Optimization on Overall Reactor Plant Design	4-10
4.5	Optimization of the UO ₂ Once-Through Fuel Cycle	4-12
4.5.1	Effect of Reducing the Fuel Stack Density	4-13
4.5.2	Effect of Increasing the Fuel Discharge Exposure	4-16
4.5.3	Effect of Varying the Number of Fuel Batches	4-18
4.5.4	Effect of Modifying the Fuel Rod Pitch	4-20
4.5.5	Effect of Modifying the Fuel Rod Diameter	4-23
4.5.6	Effect of Combining Increased Fuel Discharge Exposure with More In-Core Batches	4-26
4.6	Optimization of the Denatured U-235/Thorium Fuel Cycle	4-28
4.6.1	Effect of Reducing the Fuel Stack Density	4-30
4.6.2	Effect of Increasing the Fuel Discharge Exposure	4-32
4.6.3	Effect of Varying the Number of Fuel Batches	4-34
4.6.4	Effect of Modifying the Fuel Rod Pitch	4-36
4.6.5	Effect of Modifying the Fuel Rod Diameter	4-38
4.6.6	Effect of Employing Higher Heavy Water Concentrations	4-38
4.7	Other Fuel Cycles	4-41
4.8	Selection of the SSCR Reference Core Design	4-43

<u>Section</u>	<u>Title</u>	<u>Page</u>
5.	SSCR Reference Design	
5.1	Description of SSCR Reference Design	5-1
5.1.1	Design Approach	5-1
5.1.2	Description of Plant Layout	5-4
5.1.3	Core Design	5-37
5.1.4	Fuel Design	5-39
5.1.5	Reactor Coolant System	5-66
5.1.6	Plant Systems for the Management of Coolant Concentrations	5-75
5.1.7	Modifications to Balance-of-Plant Systems	5-91
5.1.8	D ₂ O Upgrader Design	5-101
5.1.9	D ₂ O Leakage and Losses	5-126
5.1.10	D ₂ O Inventory	5-142
5.1.11	Refueling	5-151
5.1.12	Coolant Management Operations	5-157
5.1.13	Load Change Capability	5-216
5.1.14	Heat Balance	5-231
5.2	Resource Utilization and Economics of SSCR Reference Design	5-236
5.2.1	General	5-236
5.2.2	Description of Fuel Cycles	5-236
5.2.3	Resource Utilization	5-243
5.2.4	Economic Evaluation	5-267
5.2.5	Comparison of the SSCR and PWR Performance	5-292
5.3	Fuel Management in the SSCR	5-301
5.3.1	Introduction	5-301
5.3.2	UO ₂ Once-Through Fuel Cycles	5-303
5.3.3	Thorium Fuel Cycles	5-324
5.3.4	Comparison of SSCR and PWR Fuel Management	5-357
5.4	Safety Assessment	5-361
5.4.1	Determination of Core Physics Parameters	5-361
5.4.2	Loss-of-Coolant Accident	5-381
5.4.3	Accident Analysis Excluding Loss-of-Coolant Accident	5-393

<u>Section</u>	<u>Title</u>	<u>Page</u>
5.4.4	Tritium Exposure During Normal Operation	5-419
5.4.5	Neutron FLuence on Reactor Vessel	5-434
5.4.6	Long-Term Cooling	5-436
5.5	Environmental Impact of SSCR	5-439
5.5.1	Effect of Tritium Release on Allowable Site Boundary Dose	5-439
5.5.2	D ₂ O Upgrader Facility	5-447
6.	Commercial Introduction of the SSCR	
6.1	Possible Scenarios for Introduction of the SSCR	6-1
6.1.1	General	6-1
6.1.2	Routes for Introducing the SSCR	6-2
6.1.3	Potential Introduction Dates for the SSCR	6-5
6.2	Technical Problem Areas	6-10
6.2.1	General	6-10
6.2.2	SSCR-Features Requiring Special Attention	6-14
6.3	Licensability	6-20
6.4	Commercial Viability of the SSCR	6-23
6.4.1	Utility Perception of the Need for an Advanced Converter	6-23
6.4.2	Potential Benefits of the SSCR-Are They Sufficient?	6-24
6.4.3	The Costs of Spectral Shift Control - Who Will Pay?	6-25
6.4.4	Prospect of Success	6-27
6.4.5	Summary of Commercial Viability	6-30
6.5	Required Research and Development	6-35
6.5.1	Refueling Coolant Exchange	6-35
6.5.2	D ₂ O Concentration Measurements	6-36
6.5.3	Critical Heat Flux Data	6-37
6.5.4	Physics Verification	6-38
6.5.5	Tritium Releases	6-39
6.5.6	Tritium Separation	6-39
6.5.7	Cost of D ₂ O	6-40
6.5.8	D ₂ O Upgrader Development	6-40

<u>Section</u>	<u>Title</u>	<u>Page</u>
6.5.9	Development of a Preliminary Safety Analysis Report (SAR) for the SSCR	6-41
6.6	Backfitting Potential	6-43
6.6.1	General	6-43
6.6.2	Backfitting Existing PWR for Spectral Shift Operation	6-44
6.6.3	Backfitting PWR for Spectral Shift Control Operation During Construction of Plant	6-49
7.	Appendices	
7.1	Calculation of Tritium Concentration in Containment Building	7-1
7.2	Calculation of Dilution Steam Requirements	7-5
7.3	Calculation of Evaporation from Refueling Cavity	7-9
7.4	Specifications for D ₂ O Upgrader Plant	7-12
7.4.1	Function, Processing Scheme, and Operating Conditions	7-12
7.4.2	Charge Stocks and their Specifications	7-12
7.4.3	Charge Rate	7-13
7.4.4	Products and their Specifications	7-13
7.4.5	Yields of Products	7-14
7.4.6	Battery Limit Conditions	7-15
7.4.7	Utilities and Their Specifications	7-16
7.4.8	Estimated Utilities Consumptions	7-18
7.5	D ₂ O Upgrader Equipment List	7-19
7.6	D ₂ O Upgrader Heat Balance	7-21
7.6.1	Steam and Condensate System	7-21
7.6.2	Cooling Water System	7-23
7.7	Physics Analysis Methods	7-25
7.7.1	General	7-25
7.7.2	CEPAK Lattice Cell Code	7-25
7.7.3	Development of Spatial Diffusion Model	7-27
7.7.4	Procedure for Analyzing SSCR Fuel Cycles	7-29
7.8	Plant Cost Estimates	7-33
7.9	Reactor Fuel Cycle Charge and Discharge Inventory Data	7-47

VOLUME II

FIGURES

<u>Figure</u>	<u>Title</u>	<u>Page</u>
4.5.1-1	Resource Requirements vs. Fuel Density for UO_2 Once-Through SSCR	4-15
4.5.1-2	Busbar Power Costs vs. Fuel Density for UO_2 Once-Through SSCR	4-15
4.5.2-1	Resource Requirements vs. Fuel Discharge Exposure for UO_2 Once-Through SSCR	4-17
4.5.2-2	Busbar Power Costs vs. Fuel Discharge Exposure for UO_2 Once-Through SSCR	4-17
4.5.3-1	Resource Requirements vs. In-Core Batches for UO_2 Once-Through SSCR	4-21
4.5.3-2	Busbar Power Costs vs. In-Core Batches for UO_2 Once-Through SSCR	4-21
4.5.4-1	Resource Requirements vs. Water-to-Fuel Ratio (variable rod pitch, constant rod diameter) for UO_2 Once-Through SSCR	4-22
4.5.4-2	Busbar Power Costs vs. Water-to-Fuel Ratio (variable rod pitch, constant rod diameter) for UO_2 Once-Through SSCR	4-22
4.5.5-1	Resource Requirements vs. Water-to-Fuel Ratio (variable rod diameter, constant rod pitch) for UO_2 Once-Through SSCR	4-25
4.5.5-2	Busbar Power Costs vs. Water-to-Fuel Ratio (variable rod diameter, constant rod pitch) for UO_2 Once-Through SSCR	4-25
4.5.6-1	Savings in Resource Requirements vs. Number of In-Core Batches for UO_2 Once-Through SSCR	4-27
4.5.6-2	Savings in Busbar Power Costs vs. Number of In-Core Batches for UO_2 Once-Through SSCR	4-27
4.6.1-1	Fuel Resource Requirements vs. Fuel Density for Denatured U-235/ ThO_2 SSCR	4-31

FIGURES

<u>Figure</u>	<u>Title</u>	<u>Page</u>
4.6.1-2	Busbar Power Costs vs. Fuel Density for Denatured U-235/ThO ₂ SSCR	4-31
4.6.2-1	Resource Requirements vs. Discharge Exposure for Denatured U-235/ThO ₂ SSCR	4-33
4.6.2-2	Busbar Power Costs vs. Discharge Exposure for Denatured U-235/ThO ₂ SSCR	4-33
4.6.3-1	Resource Requirements vs. In-Core Batches for Denatured U-235/ThO ₂ SSCR	4-35
4.6.3-2	Busbar Power Costs vs. In-Core Batches for Denatured U-235/ThO ₂ SSCR	4-35
4.6.4-1	Resource Requirements vs. Water-to-Fuel Ratio (variable rod pitch, constant rod diameter) for Denatured U-235/ThO ₂ SSCR	4-37
4.6.4-2	Busbar Power Costs vs. Water-to-Fuel Ratio (variable rod pitch, constant rod diameter) for Denatured U-235/ThO ₂ SSCR	4-37
4.6.5-1	Resource Requirements vs. Water-to-Fuel Ratio (variable rod diameter, constant rod pitch) for Denatured U-235/ThO ₂ SSCR	4-39
4.6.5-2	Busbar Power Costs vs. Water-to-Fuel Ratio (variable rod diameter, constant rod pitch) for Denatured U-235/ThO ₂ SSCR	4-39
4.6.6-1	Resource Requirements vs. Heavy Water Concentrations for Denatured U-235/ThO ₂ SSCR	4-40
4.6.6-2	Busbar Power Costs vs. Heavy Water Concentrations for Denatured U-235/ThO ₂ SSCR	4-40
5.1.2-1(a)	Plot Plan, Spectral Shift Controlled Reactor	5-5
5.1.2-1(b)	Annulus Building Plan, SSCR	5-8
5.1.2-1(c)	Solid Waste and Decontamination Building, SSCR	5-27
5.1.2-2	Heavy Water Upgrader, Plot Plan SSCR	5-33
5.1.2-3	Heavy Water Upgrader, Plan View SSCR	5-34

FIGURES

<u>Figure</u>	<u>Title</u>	<u>Page</u>
5.1.3-1	Reactor Vertical Arrangement, SSCR	5-38
5.1.4-1	SSCR Fuel Design, Standard Burnup	5-41
5.1.4-2	Fission Gas Release Behavior	5-46
5.1.4-3	Internal Rod Pressure vs. Rod Length	5-56
5.1.5-1	Reactor Coolant System Block Diagram	5-67
5.1.5-2	Reactor Coolant System Arrangement	5-68
5.1.6-1	Composite of the Boron Recovery, Primary Grade Water, Chemical and Volume Control Systems. SWESSAR/CESSAR PWR	5-78
5.1.6-2	Composite of the Boron Recovery, Primary Grade Water, Chemical and Volume Control Systems and the D ₂ O Upgrader Facility. SSCR	5-79
5.1.6-3	SWESSAR - CESSAR PWR, Primary Grade Water System	5-81
5.1.6-4	SSCR, Primary Grade Water System	5-82
5.1.6-5	SWESSAR - CESSAR PWR, Boron Recovery System	5-84
5.1.6-6	SSCR, Boron Recovery System	5-86
5.1.8-1	D ₂ O Upgrader, Distillation Section, SSCR	5-109
5.1.8-2	D ₂ O Upgrader, Product Recovery Section, SSCR	5-110
5.1.8-3	Relative Upgrader Feed, Overhead, and Bottoms Flow Rates as a Function of Upgrader Feed Concentration	5-123
5.1.9-1	Typical Leakoff Collection Arrangement	5-128
5.1.11-1	SSCR Refueling - Coolant Exchange	5-153
5.1.12-1	Dilution Feed, Diagram, SSCR	5-160
5.1.12-2	Dilution Bleed, Diagram, SSCR	5-161

FIGURES

<u>Figure</u>	<u>Title</u>	<u>Page</u>
5.1.12-3	Boration Diagram, SSCR	5-164
5.1.12-4	Boration Diagram, SSCR	5-165
5.1.12-5	Cooldown Diagram, SSCR	5-166
5.1.12-6	Cooldown Diagram with Contraction Volume Injection Path, SSCR	5-167
5.1.12-7	Heatup Diagram, SSCR	5-170
5.1.12-8	Deborate Diagram, SSCR	5-171
5.1.12-9	Boration Diagram with Light Water and Boric Acid Additions, SSCR	5-178
5.1.12-10	Cooldown Diagram with Light Water and Boric Acid Additions, SSCR	5-179
5.1.12-11	Basic Drain Diagram	5-180
5.1.12-12	Drain Diagram, SSCR	5-181
5.1.12-13	Reactor Vessel Flush Diagram	5-182
5.1.12-14	Refill Diagram, SSCR	5-183
5.1.12-15	Reactor Vessel Flush Tank Processing	5-184
5.1.12-16	Reactor Vessel Flush Diagram, SSCR	5-189
5.1.12-17	RCS Loop Drain Tank Processing, SSCR	5-190
5.1.12-18	Establishing D ₂ O Concentration Diagram with Letdown and Heavy Water Additions, SSCR	5-191
5.1.12-19	Establishing D ₂ O Concentration Diagram with Letdown Addition, SSCR	5-192
5.1.12-20	Boric Acid - D ₂ O Separation Unit Processing Diagram, SSCR	5-195

FIGURES

<u>Figure</u>	<u>Title</u>	<u>Page</u>
5.1.12-21	Relative Upgrader Feed, Overhead, and Bottoms Flow Rates as a Function of Upgrader Feed Concentration	5-198
5.1.12-22	Flush Water Processing Diagram, SSCR	5-199
5.1.12-23	Legend for Figure 5.1.12-24	5-210
5.1.12-24	Water Management and D ₂ O Inventory for Equilibrium Fuel Cycle	5-211
5.1.13-1	Reactor Power Responses Upon 10% Step Load Decrease, No CEA Motion	5-219
5.1.13-2	System 80 Daily Load Cycle Transients	5-223
5.1.13-3	SSCR Daily Load Cycle Transients	5-226
5.2.2-1(a)	UO ₂ Once-Through Cycle (5 Batch Extended Burnup)	5-237
5.2.2-1(b)	Denatured U-235/Thorium Cycle (Plutonium Stored)	5-237
5.2.2-1(c)	Denatured U-233/Thorium Cycle (Plutonium Stored)	5-238
5.2.2-1(d)	Plutonium/Thorium Cycle (U-233 Sold)	5-238
5.2.4-1	Denatured U-235/ThO ₂ SSCR, U-233 Indifference Values	5-278
5.2.4-2	Denatured U-233/ThO ₂ SSCR, Fuel Cycle Costs	5-282
5.2.4-3	Plutonium/ThO ₂ SSCR, Fuel Cycle Costs	5-285
5.2.4-4	Functional Relationship Between Fissile Plutonium Value and U-233 Value from Denatured U-233/ThO ₂ and Plutonium/ThO ₂ Fuel Cycles	5-286
5.3.2-1	D ₂ O Concentration vs. Cycle Exposure, UO ₂ Once-Through SSCR, 3 Batch	5-306

FIGURES

<u>Figure</u>	<u>Title</u>	<u>Page</u>
5.3.2-2	Unrodded Planar Power Distribution First Cycle, UO ₂ Once-Through SSCR, 3 Batch	5-307
5.3.2-3	Unrodded Planar Power Distribution Second Cycle, UO ₂ Once-Through SSCR, 3 Batch	5-308
5.3.2-4	Unrodded Planar Power Distribution Third Cycle, UO ₂ Once-Through SSCR, 3 Batch	5-309
5.3.2-5	D ₂ O Concentration vs. Cycle Exposure, UO ₂ Once-Through SSCR, 5 Batch	5-314
5.3.2-6	Unrodded Planar Power Distribution First Cycle, UO ₂ Once-Through SSCR, 5 Batch	5-317
5.3.2-7	Unrodded Planar Power Distribution Second Cycle, UO ₂ Once-Through SSCR, 5 Batch	5-318
5.3.2-8	Unrodded Planar Power Distribution Third Cycle, UO ₂ Once-Through SSCR, 5 Batch	5-319
5.3.2-9	Unrodded Planar Power Distribution Fourth Cycle, UO ₂ Once-Through SSCR, 5 Batch	5-320
5.3.2-10	Unrodded Planar Power Distribution Fifth Cycle, UO ₂ Once-Through SSCR, 5 Batch	5-321
5.3.3-1	Unrodded Planar Power Distribution First Cycle, Denatured U-235/ThO ₂ SSCR	5-328
5.3.3-2	Unrodded Planar Power Distribution Second Cycle, Denatured U-235/ThO ₂ SSCR	5-329
5.3.3-3	Unrodded Planar Power Distribution Third Cycle, Denatured U-235/ThO ₂ SSCR	5-330
5.3.3-4	Unrodded Planar Power Distribution Fourth Cycle, Denatured U-235/ThO ₂ SSCR	5-331
5.3.3-5	D ₂ O Concentration vs. Cycle Exposure, Denatured U-235/ThO ₂ SSCR	5-332
5.3.3-6	Unrodded Planar Power Distribution Eighth Cycle, Denatured U-235/ThO ₂ SSCR	5-335

FIGURES

<u>Figure</u>	<u>Title</u>	<u>Page</u>
5.3.3-7	D ₂ O Concentration vs. Core Exposure, Denatured U-233/ThO ₂ SSCR	5-341
5.3.3-8	Unrodded Planar Power Distribution First Cycle, Denatured U-233/ThO ₂ SSCR	5-342
5.3.3-9	Unrodded Planar Power Distribution Second Cycle, Denatured U-233/ThO ₂ SSCR	5-343
5.3.3-10	Unrodded Planar Power Distribution Third Cycle, Denatured U-233/ThO ₂ SSCR	5-344
5.3.3-11	D ₂ O Concentration vs. Cycle Exposure, Plutonium/ThO ₂ SSCR	5-349
5.3.3-12	Unrodded Planar Power Distribution First Cycle, Plutonium/ThO ₂ SSCR	5-351
5.3.3-13	Unrodded Planar Power Distribution Second Cycle, Plutonium/ThO ₂ SSCR	5-352
5.3.3-14	Unrodded Planar Power Distribution Third Cycle, Plutonium/ThO ₂ SSCR	5-353
5.4.1-1	Moderator Temperature Coefficient vs. Moderator Temperature, PWR and SSCR UO ₂ Once-Through Fuel Cycle	5-365
5.4.1-2	Fuel Temperature Coefficient vs. Average Fuel Temperature, PWR and SSCR UO ₂ Once-Through Fuel Cycle	5-367
5.4.1-3	Control Rod Worth vs. Moderator Temperature, UO ₂ Once-Through SSCR	5-369
5.4.1-4	Typical Control Rod Worth vs. Moderator Temperature, UO ₂ Once-Through PWR	5-370
5.4.1-5	Inverse D ₂ O Worth vs. D ₂ O Concentration, UO ₂ Once-Through SSCR	5-372
5.4.1-6	Moderator Temperature Coefficient vs. Moderator Temperature, Denatured U-235/ThO ₂ SSCR	5-374
5.4.1-7	Fuel Temperature Coefficient vs. Average Fuel Temperature, Denatured U-235/ThO ₂ SSCR	5-376

FIGURES

<u>Figure</u>	<u>Title</u>	<u>Page</u>
5.4.1-8	Control Rod Worth vs. Moderator Temperature, Denatured U-235/ThO ₂ SSCR	5-378
5.4.1-9	Inverse D ₂ O Worth vs. D ₂ O Concentration, Denatured U-235/ThO ₂ SSCR	5-379
5.4.2-1	Reactivity vs. Moderator Density, U/ThO ₂ Fueled SSCR	5-383
5.4.2-2	Cladding Temperature for Double-Ended Guillotine Discharge Break	5-385
5.4.2-3	System 80 Core Pressure for Double-Ended Guillotine Discharge Break	5-386
5.4.2-4	Perforation Temperature vs. Clad Differential Pressure for System 80	5-387
5.4.2-5	Circumferential Strain of Zircaloy as a Function of Clad Differential Pressure at Rupture for System 80	5-389
5.4.2-6	High Burnup Fuel, Comparison of Fuel Rod Array Dimensions	5-390
5.4.4-1	Tritium Concentration in Containment Building	5-428
5.4.4-2	Maximum Allowable D ₂ O Concentration in Refueling Cavity without Exceeding Tritium MPC in Containment	5-432
6.1.3-1	Estimated Schedule for Commercial Availability of the SSCR	6-6
7.6-1	D ₂ O Upgrader Plant - Steam Balance Diagram	7-22

VOLUME II

TABLES

<u>Section</u>	<u>Title</u>	<u>Page</u>
4.3-1	SSCR Core Optimization Study Parameter List	4-8
4.5-1	Economic and Resource Parameters	4-14
4.8-1	System 80 Core Parameters	4-44
5.1.4-1	Enrichment and Burnup Requirements for UO_2 Fuel Pellets	5-53
5.1.4-2	Fuel Rod Internal Pressures	5-59
5.1.4-3	Comparison of Fuel Rod Internal Pressures	5-60
5.1.4-4	UO_2 Design Options with Standard Rod Length	5-63
5.1.4-5	UO_2 Design Options with Increased Fuel Rod Length	5-64
5.1.9-1	Total Annual D_2O Losses	5-141
5.1.10-1	D_2O Inventory Requirements	5-144
5.1.10-2	Upgrader D_2O Inventory	5-147
5.1.10-3	D_2O Inventory in Distillation Towers	5-148
5.1.10-4	D_2O Inventory for SSCR	5-150
5.1.12-1	D_2O Inventory Equilibrium Cycle	5-207
5.1.12-2	Equilibrium Cycle Feed and Bleed Operation	5-208
5.1.12-3	Equilibrium Upgrader Operation	5-209
5.1.14-1	Possible Turbine Cycle Improvements for SWESSAR/CESSAR	5-235

VOLUME II Continued
TABLES

<u>Section</u>	<u>Title</u>	<u>Page</u>
5.2.3-1	Thirty Year Fissile Requirements for the SSCR	5-244
5.2.3-2(a)	U ₃ O ₈ Ore Requirements, UO ₂ Once-Through Cycles	5-247
5.2.3-2(b)	Separative Work Requirements, UO ₂ Once- Through Cycles	5-247
5.2.3-3	UO ₂ Once-Through SSCR, Three and Five Batch Fuel Inventories	5-248
5.2.3-4	Apparent Fissile Consumption	5-249
5.2.3-5	Equilibrium Cycle Core Average Integral Conversion Ratios	5-251
5.2.3-6(a)	U ₃ O ₈ Ore Requirements, Denatured U-235/ Thorium Cycle	5-252
5.2.3-6(b)	Separative Work Requirements, Denatured U-235/Thorium Cycle	5-252
5.2.3-7(a)	U ₃ O ₈ Ore Requirements, Denatured U-235/ Thorium Cycle	5-254
5.2.3-7(b)	Separative Work Requirements, Denatured U-235/Thorium Cycle	5-254
5.2.3-8(a)	Inventories for Denatured U-235/ThO ₂ Makeup Fuel	5-256
5.2.3-8(b)	Inventories for Denatured U-235/ThO ₂ Recycle Fuel	5-257
5.2.3-9	Fuel Type Fractions for the Denatured U-235/ThO ₂ Cycle	5-258
5.2.3-10	Average Annual Fissile Requirements, Denatured U-233/ThO ₂ Fuel Cycle	5-260
5.2.3-11	Inventories for Denatured U-233/ThO ₂ Fuel Cycle	5-262
5.2.3-12	Average Annual Fissile Requirements, Pu/ThO ₂ Fuel Cycle	5-264

VOLUME II Continued
TABLES

<u>Section</u>	<u>Title</u>	<u>Page</u>
5.2.3-13	Inventories for Plutonium/ThO ₂ Fuel Cycle	5-265
5.2.4-1	SSCR Capital Cost Estimate Summary	5-269
5.2.4-2	SSCR Capital Cost Increase Relative to PWR	5-270
5.2.4-3	Economic and Resource Parameters	5-275
5.2.4-4	SSCR Thirty Year Levelized Fuel Cycle Costs	5-279
5.2.4-5	Denatured U-233/ThO ₂ SSCR 30-Year Levelized Fuel Cycle Costs	5-281
5.2.4-6	Plutonium/ThO ₂ SSCR 30-Year Levelized Fuel Cycle Costs	5-284
5.2.4-7	Fuel Cycle Cost Sensitivity	5-288
5.2.4-8	Total Busbar Power Costs for a 1260 MW(e) SSCR	5-291
5.2.5-1	SSCR and PWR Comparison, 30-Year Ore Requirement	5-293
5.2.5-2	SSCR and PWR Comparison, 30-Year Separative Work Requirements	5-294
5.2.5-3	Denatured U-233/ThO ₂ Fuel Cycle, SSCR and PWR Comparison of Fissile Requirements	5-296
5.2.5-4	Plutonium/ThO ₂ Fuel Cycle, SSCR and PWR Comparison of Fissile Requirements	5-297
5.2.5-5	Comparison of Total Power Costs for PWR and SSCR	5-299
5.3.2-1	New Core Fuel Management Characteristics UO ₂ Once-Through 3 Batch	5-310
5.3.2-2	Material Inventories, Initial Core, UO ₂ Once-Through 3 Batch	5-311
5.3.2-3	New Core Fuel Management Characteristics, UO ₂ Once-Through 5 Batch	5-316

VOLUME II Continued
TABLES

<u>Section</u>	<u>Title</u>	<u>Page</u>
5.3.2-4	Material Inventories, Initial Core, UO ₂ Once-Through 5 Batch	5-333
5.3.3-1	Fuel Management Characteristics, Denatured U-235/ThO ₂ SSCR	5-337
5.3.3-2	Material Inventories, Initial Core - Denatured U-235/ThO ₂ SSCR	5-338
5.3.3-3	New Core Fuel Management Characteristics, Denatured U-233/ThO ₂ SSCR	5-346
5.3.3-4	Material Inventories, Initial Core - Denatured U-233/ThO ₂ SSCR	5-347
5.3.3-5	New Core Fuel Management Characteristics, Pu/ThO ₂ SSCR	5-354
5.3.3-6	Material Inventories, Initial Core - Pu/ThO ₂ SSCR	5-356
5.4.1-1	PWR and SSCR Core Physics Parameters	5-364
5.4.3-1	Category/Frequency Matrix	5-394
5.4.3-2	Categorization vs. Acceptance Guideline Scheme	5-395
5.4.4-1	Tritium Build-up in SSCR	5-422
5.4.6-1	Containment Data Comparisons	5-439
5.5.1-1	Estimated Annual Losses of D ₂ O to the Environment	5-444
6.2.1-1	SSCR-Survey of Technology Requirements	6-11
7.6.2-1	Cooling Water Balance	7-24
7.8.1-1	SSCR-Cost Increase Relative to PWR	7-34
7.8.1-2	SSCR-Cost Increase Relative to PWR	7-45
7.9-1	Fuel Management Information, UO ₂ Once-Through 3 Batch SSCR	7-48

VOLUME II Continued
TABLES

<u>Section</u>	<u>Title</u>	<u>Page</u>
7.9-2	Charge Data, UO_2 Once-Through 3 Batch SSCR	7-49
7.9-3	Discharge Data, UO_2 Once-Through 3 Batch SSCR	7-50
7.9-4	Fuel Inventory BOC, UO_2 Once-Through 3 Batch SSCR	7-51
7.9-5	Fuel Inventory EOC, UO_2 Once-Through 3 Batch SSCR	7-52
7.9-6	Fuel Management Information, UO_2 Once-Through 5 Batch SSCR	7-53
7.9-7	Charge Data, UO_2 Once-Through 5 Batch SSCR	7-54
7.9-8	Discharge Data, UO_2 Once-Through 5 Batch SSCR	7-55
7.9-9	Fuel Inventory BOC, UO_2 Once-Through 5 Batch SSCR	7-56
7.9-10	Fuel Inventory EOC, UO_2 Once-Through 5 SSCR	7-57
7.9-11	Fuel Management Information, Denatured U-235/ ThO_2 SSCR	7-58
7.9-12	Makeup Fuel Charge Data, Denatured U-235/ ThO_2 SSCR	7-59
7.9-13	Recycle Fuel Charge Data, Denatured U-235/ ThO_2 SSCR	7-60
7.9-14	Makeup Fuel Discharge Data, Denatured U-235/ ThO_2 SSCR	7-61
7.9-15	Recycle Fuel Discharge Data, Denatured U-235/ ThO_2 SSCR	7-62
7.9-16	Re-enrichment Component of Recycle Fuel Charge Data, Denatured U-235/ ThO_2 SSCR	7-63

VOLUME II Continued
TABLES

<u>Section</u>	<u>Title</u>	<u>Page</u>
7.9-17	Makeup Fuel Inventory BOC, Denatured U-235/ThO ₂ SSCR	7-64
7.9-18	Recycle Fuel Inventory BOC, Denatured U-235/ThO ₂ SSCR	7-65
7.9-19	Makeup Fuel Inventory EOC, Denatured U-235/ThO ₂ SSCR	7-66
7.9-20	Recycle Fuel Inventory EOC, Denatured U-235/ThO ₂ SSCR	7-67
7.9-21	Fuel Management Information, Denatured U-233/ThO ₂ SSCR	7-68
7.9-22	Charge Data, Denatured U-233/ThO ₂ SSCR	7-69
7.9-23	Discharge Data, Denatured U-233/ThO ₂ SSCR	7-70
7.9-24	Fuel Inventory BOC, Denatured U-233/ThO ₂ SSCR	7-71
7.9-25	Fuel Inventory EOC, Denatured U-233/ThO ₂ SSCR	7-72
7.9-26	Fuel Management Information, Plutonium/ThO ₂ SSCR	7-73
7.9-27	Charge Data, Plutonium/ThO ₂ SSCR	7-74
7.9-28	Discharge Data, Plutonium/ThO ₂ SSCR	7-75
7.9-29	Fuel Inventory BOC, Plutonium/ThO ₂ SSCR	7-76
7.9-30	Fuel Inventory EOC, Plutonium/ThO ₂ SSCR	7-77

SECTION 4

OPTIMIZATION OF THE SPECTRAL SHIFT CONTROLLED REACTOR DESIGN

4.1

PURPOSE OF OPTIMIZATION

The elimination of soluble boron in the reactor coolant suggests that certain modifications from the current PWR design/fuel management may result in greater benefits in the SSCR than would occur in the PWR if the same modifications were undertaken there. For example, softening the neutron spectrum by increasing the water-to-fuel (W/F) ratio reduces the fissile inventory requirements and thus the fuel resource requirements in once-through cycles. However, in a PWR, increasing the W/F ratio often reduces the magnitude of the moderator temperature coefficient, making the consequences of certain postulated accidents more severe. Thus, the degree to which the W/F ratio can be increased is limited in the PWR. This restriction is not applicable to the SSCR because of the elimination of soluble boron.

A further example of how the elimination of control poison can suggest practices different from those in the conventional PWR is with respect to longer cycle lengths. Increasing the cycle length reduces the annual fissile makeup requirements for once-through cycles. For the PWR, however, this fuel management strategy requires the use of higher soluble boron concentrations to control the excess reactivity of the fuel. The higher poison concentrations result in more parasitic capture of neutrons and a lower integrated conversion ratio. In the SSCR, on the other hand, longer

cycle lengths result in higher heavy water concentrations to control the excess reactivity. Fertile conversion is enhanced by the use of more heavy water, thus resulting in additional savings in fuel resources.

Because of this increased potential for enhanced performance, an optimization study of the SSCR was undertaken to identify core design changes (from the conventional PWR design) and/or fuel management strategies which would result in improved resource utilization (primarily U_3O_8 requirements) and lower power generation costs for the SSCR. It was intended that the core design established from the results of this optimization study would then be used for the development of the conceptual design of the SSCR reactor plant. The optimization study consisted of single variable core parameter studies for two fuel cycles: the UO_2 once-through cycle and the denatured $U-235/ThO_2$ cycle with uranium recycle. In undertaking this design optimization of the SSCR, it was recognized that major changes from current PWR design could raise questions of design feasibility or of new licensing issues, which could seriously hamper the prospects of early and rapid introduction. Consequently, these parameter studies first explored changes in fuel management and fuel design which could be employed within the confines of the current PWR lattice. Changes, such as the use of reduced density (e.g., hollow pellets) fuel, extended fuel burnup, and a larger number of in-core batches fall within this category. Other more major modifications, such as changes in the lattice pitch or fuel rod diameter, which affect the mechanical design of the reactor, the thermal hydraulic performance, and/or the safety evaluation were also explored. The optimization study was evaluated both

from a fuel resource utilization standpoint and from an economic standpoint, with the balance between the two criteria used to select the optimum core design. For those modifications which affect the core lattice design, the impact of such modifications on the overall plant design was also assessed and included in the economic evaluation.

Before discussing the details of the optimization study, it is beneficial to recall some important differences between the two fuel cycles analyzed. In the case of the once-through uranium cycle, maximum fuel cycle performance is achieved by minimizing the initial fissile loadings required to produce a given amount of energy. The conversion ratio is important only to the extent that it reduces the initial fuel inventory; thus, optimal once-through cycles may have relatively low conversion ratios. On the other hand, when recycle is employed, a high conversion ratio becomes more important because fissile material produced during the cycle is recovered from the spent fuel and utilized in subsequent cycles. The initial loading is important only because increased carrying charges result as the initial loading increases. An economic optimum occurs because of the balance between the fuel resource savings which result from increased conversion and the higher carrying charges on the initial inventories which increase as the lattice design is modified to improve conversion. The use of thorium fuel also results in a different optimum than when uranium fuel is employed. Consequently, it was anticipated that the optimal design would be different for the once-through uranium cycle and the denatured U-235/ThO₂ cycle.

Optimization was therefore pursued independently for each of the fuel cycle options. An overall design which preserved as much as possible the performance of both fuel cycles was the target for the study.

4.2 OPTIMIZATION PROCEDURE

In order to establish a basis for judging the benefits of core optimization in the SSCR, fuel management calculations were performed for the initial few cycles of the UO_2 once-through fuel cycle and the denatured $U-235/ThO_2$ fuel cycle using the reference PWR core design and conventional three-batch fuel management strategy. A two-dimensional, coarse mesh (16 nodes per assembly), PDQ model¹ was employed in these calculations which provided not only accurate estimates of the fuel resource requirements for the reference cycles, but also the operating history, in terms of D_2O rundown and interbatch neutron leakages, required to perform the optimization studies in a less detailed point reactor model. The use of point reactor calculations was necessary since a relatively large number of cases had to be analyzed to investigate the core parameters of interest; a complete analysis using core spatial calculations would have been prohibitive.

The analysis of fuel performance in the SSCR using a point reactor model is significantly more difficult than in the conventional PWR. Depletion calculations to determine the fuel loading requirements of any reactor concept must be performed in a critical spectrum; failure to do so will

result in either a surplus or a deficiency of neutrons during the depletion, and will lead to an erroneous conversion ratio and associated mass flow data. In the conventional PWR, excess neutrons are lost either through leakage or through capture in the soluble boron poison introduced for this purpose; in either case, the effect is much the same, since neutrons are lost from the system. Consequently, conventional PWR point reactor calculations can employ approximate representations of core leakage and utilize a search for critical boron concentration, or alternately use approximate boron concentrations and search for critical neutron leakage in order to achieve a critical spectrum during depletion. The situation is different, however, for the SSCR, since the leakage represents a loss of neutrons from the system while changes in D_2O concentration adjust the capture in fertile material and hence the amount of fissile material bred. Thus, an accurate knowledge of either the neutron leakage or of the evolution of D_2O concentration with depletion is necessary in order to perform accurate point reactor calculations for the SSCR.

Optimization of the SSCR core was performed using the CEPAC point reactor model.² In this model, a single enrichment fuel batch is analyzed with its exposure followed through its full incore residency time. In order to simulate in CEPAC the environment the fuel batch would experience in a critical core, the D_2O concentrations required for overall core criticality are input to CEPAC as a function of fuel exposure. This approach requires a foreknowledge of the core operating history that the fuel batch experiences

through several depletion cycles and fuel shuffles. Although this foreknowledge does not exist for other than the reference fuel cycles analyzed with the spatial core model, the beginning of cycle (BOC) D_2O concentration can be estimated reasonably well from the knowledge of the excess reactivity of the particular fuel being analyzed and from the interbatch neutron leakages, while the end of cycle (EOC) D_2O concentration was taken to be 2% for all cases. Furthermore, since the general shape of the D_2O rundown curve is similar for all fuel cycles, the D_2O rundown can be established for any particular optimization case by using the shape of the D_2O rundown curves obtained from the reference PDQ calculations together with the BOC D_2O concentration determined for the particular case of interest.

The BOC D_2O concentrations for the various optimization cases were determined from an empirical relationship derived from calculations using the core spatial (PDQ) models. This empirical relationship was obtained by varying the BOC D_2O concentration and plotting the change in interbatch neutron leakage versus the D_2O concentration. The resulting relationship between D_2O concentration was then used to determine what BOC D_2O concentration was required to control the excess reactivity in CEPAC for the individual optimization cases.

The specific procedure followed in the point reactor model consisted of first estimating the fuel loading required to achieve the desired fuel exposure. A guess was then made as to the BOC D_2O concentration and associated neutron leakage (in the form of fast and thermal buckling values) which when input to CEPAC would result in a critical system. A BOC reactivity

calculation was then performed and, depending on the results of this calculation, the input D_2O concentration and neutron leakage were varied to achieve criticality. From the BOC D_2O concentration so defined, a D_2O rundown was constructed for the depletion calculation. If the fuel was to reside in core for multiple cycles, for example three cycles, then three sequential D_2O rundowns were input to CEPAC. Since this D_2O rundown would not maintain the reactor exactly critical, a buckling search which adjusted the interbatch neutron leakages was employed at each depletion step (about 1000 MWd/t fuel exposure).

The correct fuel loading in the CEPAC point reactor model was determined when the EOC average infinite multiplication factor ($\bar{k}_\infty$) equaled a reference value obtained from the core spatial calculations. This definition of criticality has been used extensively in PWR fuel cycle analyses and has been shown to give reasonably good agreement with more detailed methods². When more than one fuel type was resident in the same core, then this definition of reactivity lifetime must be expanded. For this study, it was required that each fuel satisfy the same reactivity lifetime condition.

4.3 CORE PARAMETERS

The core parameters investigated in the SSCR Optimization Study are listed in Table 4.3-1. Also shown in this table are the assumptions which accompanied each parameter variation. In general, only one parameter was allowed to

TABLE 4.3-1

SSCR CORE OPTIMIZATION STUDY PARAMETER LIST^(a)

<u>Parameter Varied</u>	<u>Assumptions</u>
Fuel Density	Base case lattice Constant cycle length Base case fuel management
Discharge Exposure	Base case lattice Base case fuel management
Number of Fuel Batches	Base case lattice Constant discharge exposure
Fuel Rod Pitch	Constant rod diameter Base case fuel management Constant cycle length
Fuel Rod Diameter	Constant rod pitch Base case fuel management Constant cycle length
Range of Heavy Water Concentrations ^(b)	Base case lattice Base case fuel management Constant cycle length

Notes: (a) Base case parameters are those of C-E System 80TM PWR

(b) For denatured U-235/ThO₂ cycle only

change while the rest were held constant at their base case values. For this study, the C-E System 80TM PWR design³ was used as the reference core configuration. The reference fuel management scheme employed annual refueling of one-third of the core and a 75% plant capacity factor.

For the denatured U-235/ThO₂ cycle, the range of D₂O concentration utilized during the operating cycle was treated as an additional parameter to be varied. The rationale behind this choice was that the use of higher D₂O concentrations throughout the cycle may result in a significantly higher integrated conversion ratio and thus reduce fissile makeup requirements. It was hoped that the savings in resource requirements would also outweigh the increased fuel and D₂O inventory charges which accompany this strategy.

In deciding to perform a single variable parameter study, it was recognized that certain combinations of parameter variations could result in additive benefits. The judgment as to which combinations should be pursued was reserved until the initial parameter studies were completed so that insights gained from these studies could guide the choice of such combinations. Upon completion of the initial studies, it was therefore decided to investigate the combination of multiple in-core batches and extended fuel discharge exposure. Other combinations may also result in additive savings; however, time did not permit the investigation of these.

All core parameter variations effect the reactor plant design to some extent. Even the relatively minor variation of changing the fuel discharge exposure may require modification of the D_2O upgrader and/or various storage tanks because this alters the time available to process the reactor coolant and changes the D_2O concentration needed during the operating cycle. However, only those plant modifications which are associated with core lattice changes were explicitly considered in this study. Such modifications require major redesign at a considerable expense and are therefore not undertaken without the incentive of substantial benefits. Other modifications were considered minor and could most likely be achieved at little or no additional cost once a firm design was established.

The major components affected by lattice changes are the reactor vessel and internals, the pressurizer, the safety injection system, the reactor coolant pumps, the reactor containment shell, the reactor vessel support structure, and the primary shield cavity. The cost of the reactor vessel, the internals, the vessel support, and the cavity are primarily a function of the reactor vessel diameter and as such, change when the overall core dimensions change. Variations of the lattice pitch will therefore affect the cost of these components. The cost of the pressurizer, the safety injection system, and the containment are functions of the coolant volume in the core and/or the coolant volume in the entire reactor vessel; these coolant volumes are dependent on the water-to-fuel ratio of the core and the overall core

dimensions. The cost of these components can therefore change with either the fuel rod pitch or the fuel rod diameter, although fuel rod diameter changes will result in less of a change in total coolant volume in the vessel. Finally, the cost of the reactor coolant pumps is a function of the primary flow, which is set to maintain a given thermal margin. The cost of the reactor coolant pumps and also the power required to operate these pumps is therefore dependent on both the fuel rod pitch and the fuel rod diameter.

One of the largest contributors to the cost variation which results from core parameter optimization is the change in D_2O inventory requirements. The various cases in the optimization study require a broad range of D_2O concentrations to control the beginning of cycle excess reactivity; a factor of two difference in D_2O concentrations was observed for the cases investigated in this study. This broad range of D_2O requirements would obviously affect the design of the D_2O upgrader and would also affect the design of other auxiliary systems. However, for the more attractive cases (from both a resource and economic standpoint) investigated, the range of D_2O concentrations was much narrower and therefore had less of an impact on plant design than was initially indicated.

Analyses were performed to assess the impact of fuel lattice modifications on the various components listed in the above discussion; cost changes

resulting from these lattice modifications were then included in the individual optimization cases as part of the capital cost of the SSCR plant. In general, the overall impact of these plant changes was small, representing less than 1% of the total power costs.

4.5 OPTIMIZATION OF THE UO_2 ONCE-THROUGH FUEL CYCLE

Optimization of the SSCR with the UO_2 once-through cycle consists of minimizing the fissile feed requirements needed to obtain a given cycle length (energy production). Fissile requirements can be reduced by either increasing the conversion ratio, so that more fissile material is produced and available for in situ energy production, and/or by more fully consuming the fissile material which is initially loaded. Optimum once-through cycles typically display more complete consumption of loaded fissile material at the expense of somewhat lower conversion ratio. Examples of methods of minimizing feed requirements are:

1. Reducing the core average exposure at end of cycle so that fission products present less reactivity burden (e.g., by use of more in-core fuel batches),
2. Softening the neutron spectrum so that the fuel is more reactive (e.g., increasing the water-to-fuel ratio), and
3. Increasing the fuel discharge burnup so that the average annual feed requirement is less (e.g., by the use of 18-month refueling with the standard three batch fuel management).

The parameter optimization studies for the UO_2 SSCR core concentrated on these three mechanisms as a means of improving resource utilization and power costs. The results of these studies are summarized in Figures 4.5.1-1 through 4.5.6-2. In these figures are shown the variation of U_3O_8 and separative work requirements and total busbar power costs as a function of the individual core parameter listed in Table 4.3-1. The economic and resource assumptions used in the analyses are given in Table 4.5-1.

4.5.1 EFFECT OF REDUCING THE FUEL STACK DENSITY

The effect on fuel resource requirements of reducing the stack fuel density is shown in Figure 4.5.1-1. As the density decreases, the total resonance capture in U-238 decreases, resulting in fuel with higher reactivity. The fissile inventory requirements to support criticality are thereby reduced so that at end of cycle the fissile content of the spent fuel can be less. The lower EOC fissile requirements allow the loading of less fertile material at BOC for a given energy extraction, thus giving lower U_3O_8 requirements. At the same time that removal of fertile material increases fuel reactivity, it also reduces the conversion ratio since less fertile material is available for capture. Eventually, the loss of conversion outweighs the reduction in fissile inventory requirements which accompany the higher fuel reactivity, so that the required U_3O_8 increases. A fuel density of about 60% appears to be optimum from the standpoint of U_3O_8 requirements. At this point, the U_3O_8 savings over the base case is about 7%. However, since the fissile loading requirements do not decrease as fast as the fuel density, the enrichment requirements increase monotonically as the density decreases.

TABLE 4.5-1
ECONOMIC AND RESOURCE PARAMETERS

	Near Term ¹	Long Term ²
Net Power Rating, Mwe		
SSCR		1260
PWR		1270
Nominal Capacity Factor, %		75
U ₃ O ₈ Costs, \$/lb	40	100
Thorium, \$/lb		15
Conversion, \$/kg		4
Separative Work, \$/kg SWU		100
Fabrication Costs, \$/kg HM		
LE U5-U8		106
DE U5-U8-Th	139	123
DE U3-U8-Th	664	574
Pu-Th	438	379
Reprocessing Costs, \$/kg HM		
ThO ₂	241	154
Shipping and Waste Storage, \$/kg HM		85
Throwaway Cost, \$/kg HM		150
D ₂ O Price, \$/lb		80
Enrichment Tails Assay, wt%		0.2
Reactor Plant Life, years		30
Discount Rate, %		4.525
Fixed Charge Rate, %		10
Carrying Charge Rate, %		8

-
1. Assumes a developed industry
 2. Assumes high capacity industry

Figure 4.5.1-1

Figure 4.5.1-2

The effect on power costs of the competing trends of lower U_3O_8 requirements and higher separative work requirements as the fuel density decreases can be seen in Figure 4.5.1-2. In the near term, when U_3O_8 is only \$40/lb, the higher separative work costs result in an economic optimum around 80% fuel density; the power cost savings, however, is only about 0.1 mills/kWh. In the long term, when U_3O_8 is assumed at \$100/lb, the economic optimum shifts toward lower fuel densities which show better utilization of U_3O_8 . In the long term, the economic optimum occurs at about 70% fuel density and consists of a 0.3 mills/kWh savings.

4.5.2 EFFECT OF INCREASING THE FUEL DISCHARGE EXPOSURE

The variation of fuel resource requirements as the discharge exposure is increased is shown in Figure 4.5.2-1. Optimization on discharge exposure involves two competing processes. First of all, as the discharge exposure is increased, it becomes necessary to discharge and replace spent fuel less frequently. As a result, the end-of-cycle fissile fuel inventory is discharged less frequently, i.e., the quantity of fissile material which resides unproductively in discharged spent fuel per MWd of energy produced is reduced, thus improving U_3O_8 utilization. On the other hand, the required end-of-cycle fissile inventory increases as discharge exposure increases because of the higher fission product burden. This higher fission product burden tends to reduce the cycle average conversion ratio. Consequently, an optimum discharge exposure exists due to the competition of these two opposing factors. From the shape of the U_3O_8 curve in Figure 4.5.2-1, it

appears that the optimum discharge exposure from the standpoint of uranium utilization occurs around 60,000 MWd/t and is about 8%. This value of exposure is about 10,000 MWd/t higher than a similar value for the conventional PWR. In the PWR, the higher soluble boron concentration required for the extended burnup cycles has a detrimental effect on the conversion ratio causing the optimum discharge exposure to be lower than that of the SSCR. Since higher enriched fuel is needed for extended discharge exposure and since separative work requirements change rapidly in the range of enrichments used for these extended burnup cases, an increase in separative work requirements accompanies the decrease in U_3O_8 requirements as a function of discharge exposure.

The power cost for the extended fuel discharge exposure cases are shown in Figure 4.5.2-2 and reflect not only the competition between higher separative work and lower U_3O_8 , but also the reduced throughput of fabrication and disposal services. These latter two factors cause the optimum power cost to occur around 50,000 MWd/t exposure for both near term and long term cost assumptions. The total power cost savings is about 0.5 mills/kWh for both sets of economic assumptions.

4.5.3 EFFECT OF VARYING THE NUMBER OF FUEL BATCHES

One well-known method for improving uranium resource utilization is to increase the number of fuel batches resident in the core; the reduced uranium requirements realized with multiple in-core batches is one of the

reasons why the standard PWR fuel management employs a three-batch fuel management scheme rather than a single in-core fuel batch. A larger number of in-core batches reduces the core average end-of-cycle exposure for a given fuel discharge exposure. Since the fission product burden is proportional to this core average exposure, smaller quantities of fissile material will be required at end of cycle to maintain criticality. Since the end-of-cycle fissile inventories are reduced, the initial inventory can also be reduced as the number of batches is increased, thereby reducing U_3O_8 requirements.

Use of spectral shift control suggests that the penalty of employing a smaller number of in-core batches would not be as costly as it is in the conventional PWR. In the PWR, the use of a single in-core fuel batch would require very high soluble boron concentrations to control the excess reactivity. These high poison concentrations cause a reduction of the conversion ratio which further increases the U_3O_8 requirements (an increase above that caused by the higher fission product burden). Since the SSCR employs higher D_2O concentrations instead of higher boron concentrations, it does not suffer this additional degradation of conversion ratio and therefore it does not experience the added U_3O_8 requirements. In fact, higher D_2O concentrations improve the conversion ratio by preferentially directing neutrons to fertile capture. An advantage that the one- and two-batch cores have over cores employing a larger number of in-core batches is its potential for reduced outage time for refueling. This potential advantage was reflected in the power cost calculations to give the maximum benefit possible to this strategy.

The variation of fuel resource requirements as the number of in-core batches changes (while holding the fuel discharge exposure constant) is shown in Figure 4.5.3-1. As the refueling strategy goes from a three-batch to a single batch scheme, the U_3O_8 requirements increase about 39%; a similar trend is observed for separative work requirements. This U_3O_8 penalty can be compared with that for a conventional PWR where the single batch requirement is about 44% higher than the three-batch requirement. The 5% lower requirement for the SSCR results from its improved method of utilizing the excess neutrons generated during the cycle.

The change in power costs with the number of in-core batches is shown in Figure 4.5.3-2. It can be seen from this figure that the minimum for \$40/lb U_3O_8 occurs at three batches while for \$100/lb U_3O_8 it occurs at a higher number of in-core batches (about four). At \$100/lb U_3O_8 , the 5% savings in U_3O_8 and 8% savings in separative work, realized in going from three to four batch refueling, more than balance the costs associated with more frequent refuelings and lower reactor availability.

4.5.4 EFFECT OF MODIFYING THE FUEL ROD PITCH

The effect on fuel resource requirements of changing the fuel rod pitch (while keeping the fuel rod diameter constant) is shown in Figure 4.5.4-1. As the rod pitch increases, so does the water-to-fuel ratio, resulting in a softer neutron spectrum. A softer spectrum causes the reactivity of the fuel (for a given fissile loading) to be higher and consequently a lower fissile inventory is required to sustain criticality. However, the fertile conversion decreases as the neutron spectrum softens because of the reduced

epithermal (resonant) capture which results from the lower epithermal flux. As the water-to-fuel ratio continues to increase, the competition between these two opposing effects results in minima for both U_3O_8 and separative work requirements. It appears that the minimum in U_3O_8 requirements for the UO_2 once-through SSCR occurs between a W/F ratio of 2.5 and 3.0; the minimum for separative work occurs at a W/F ratio of about 3.0.

The variation of power costs with changing W/F ratio is shown in Figure 4.5.4-2. Because of the larger heavy water inventory required to control the excess reactivity at high W/F ratios, the minimum for both the \$40/lb U_3O_8 and \$100/lb U_3O_8 cost assumptions occurs at a lower W/F ratio than that for the minimum fuel resource requirements. As the price of uranium increases so does the minimum power cost water-to-fuel ratio. At \$100/lb U_3O_8 , the optimum W/F ratio is about 2.4 and the power cost savings is 0.2 mills/kWh.

4.5.5 EFFECT OF MODIFYING THE FUEL ROD DIAMETER

The trends observed for the variable rod diameter cases are much the same as those observed for the variable rod pitch cases; that is, a reduction in fuel resource requirements and power costs as the water-to-fuel ratio increases above that of the current PWR lattice. The reduced fuel rod diameter cases, however, result in higher resource savings than the increased rod pitch cases having the same water-to-fuel ratio. While thicker rods have reduced resonance absorption, the thinner rods have an increased discharge burnup. This increased burnup is the result of constant energy extraction from a smaller mass of fuel. For the case in point, this burnup

increase results in a more optimized discharge burnup and subsequently an increased resource utilization. The additional resource savings due to the use of a reduced fuel rod diameter lattice is therefore dependent primarily upon the potential resource savings achievable with increased discharge exposure.

The effect on fuel resource requirements of varying the fuel rod diameter is shown in Figure 4.5.5-1. A maximum U_3O_8 savings of about 8% occurs at a W/F ratio of 3.5. Because of the lower total fuel loadings that accompany smaller fuel rod diameters, the uranium enrichment must increase. The separative work increases faster than the fuel loading decreases resulting in a minimum separative work requirement at a lower W/F ratio. The separative work requirements attain a minimum of 3% savings at a W/F ratio of 2.5.

The variation of power costs with changing fuel rod diameter is shown in Figure 4.5.5-2. For the near term cost assumptions (e.g., \$40/lb U_3O_8), a maximum savings of 0.2 mills/kWh is obtained at a W/F ratio of 2.5, while for the long term cost assumptions (e.g., \$100/lb U_3O_8), a power cost savings of 0.4 mills/kWh is obtained at a W/F ratio of about 2.7.

The effects of varying the discharged fuel exposure and the number of in-core fuel batches individually have been discussed in previous sections. It was found that these two mechanisms can be combined with their fuel resource savings being largely additive. By the proper choice of discharge exposure with a given number of fuel batches, a broad range of refueling intervals can be chosen. Annual refueling was selected for the present study since the majority of commercial reactors are operated on this schedule.

The variation of fuel resource requirements with the extended burnup/multibatch fuel management scheme is shown in Figure 4.5.6-1. The reduction in U_3O_8 requirements for six batches and 60,800 MWD/T is over twice that observed for any of the single parameter variations. The reduction in separative work requirements is negligible, owing to the rapid increase in separative work required for the higher fuel enrichments of this strategy. For six fuel batches, a separative work penalty of about 1.5% occurs. It should be noted that further reduction in U_3O_8 requirements is possible if the discharge exposure were extended to still higher values and the number of fuel batches increased further. A maximum reduction of between 25-30% is possible in the limit of on-line refueling with about 70,000 MWD/T discharge exposure.

The total power cost savings which results from this improved fuel management scheme is shown in Figure 4.5.6-2. The maximum savings occurs between the five- and six-batch strategies. At higher batch numbers (higher fuel exposures), the enrichment costs and carrying charges on the fuel outweigh the savings in U_3O_8 , resulting in increasing power costs. The maximum savings for \$40/lb U_3O_8 is about 0.5 mills/kWh and for \$100/lb U_3O_8 , about 0.8 mills/kWh.

4.6

OPTIMIZATION OF THE DENATURED U-235/THORIUM FUEL CYCLE

For fuel cycles which employ recycle, optimum performance from the standpoint of uranium utilization is governed primarily by the conversion ratio, while the fissile inventory requirements are important only from an economic standpoint. In the conventional PWR, decreasing the W/F ratio hardens the neutron spectrum and reduces the excess reactivity of the fuel and the soluble boron requirements of the core. The lower neutron capture in control poison and structural materials generally results in improved conversion. In the SSCR, however, decreasing the W/F ratio reduces the concentration of D_2O required to control the excess reactivity; the spectrum hardness is much the same as for the standard lattice except near end-of-cycle when the moderator is essentially pure light water.

A well-known method for increasing the conversion ratio for recycle cases is to reduce absorptions in fission products. This can be accomplished by

two main methods: by increasing the number of in-core batches and by increasing the frequency of refuelings while holding the other fuel management parameters constant (e.g., discharge exposure). Both of these mechanisms reduce the core average exposure and thus the average fission product concentration during the cycle.

For denatured U-235/thorium cycle, an additional factor plays an important role in determining which core parameter variations can contribute to improved performance. Typically, higher fissile inventories are associated with improvements in fuel performance of advanced converter reactors. For the denatured thorium cycle, those variations which result in higher fissile inventories require higher loadings of uranium which are composed primarily of U-238 because of the restriction on the allowable enrichment of the uranium component. Since the additional uranium displaces thorium more plutonium and less of the more desirable U-233 is bred. The bred plutonium was not utilized in the present optimization study; neither a fuel resource credit nor an economic credit was given for the fissile plutonium contained in the spent fuel. If these credits were included in the optimization study, then the conclusions reached as a result of the present study may be altered. Further analyses would be required to fully assess the impact of plutonium credits on fuel resource requirements and power costs.

The results of the parameter optimization studies are summarized in Figures 4.6.1-1 through 4.6.6-2. In these figures are shown the variation of U_3O_8

and separative work requirements and total busbar power costs as a function of the individual core parameters. The economic and resource assumptions used in the analyses are given in Table 4.5-1.

4.6.1 EFFECT OF REDUCING THE FUEL STACK DENSITY

The effect on fuel resource requirements of reducing the stack fuel density is shown in Figure 4.6.1-1. As the fuel density is reduced, the amount of U-233 bred through fertile captures is also reduced, requiring that more U-235 makeup be supplied at each reload. The resource requirements do not increase as fast as the density decreases because the fertile components become less self-shielded and therefore exhibit higher epithermal capture per atom of fertile material. This increased epithermal capture does not compensate, however, for the overall reduction in the concentration of fertile material.

The variation of power costs as the fuel density decreases is shown in Figure 4.6.1-2. Several factors are competing to cause the power costs to decrease with fuel density. As shown in Figure 4.6.1-1, the fissile makeup requirements are greater for the lower density fuel. These higher makeup requirements are offset by lower fabrication costs, lower reprocessing costs, and lower carrying charges on the fissile inventory. Lower fabrication costs are obtained because the recycle fuel fraction (with its high fabrication unit costs) is smaller for the lower fuel density cases. The reprocessing

FIGURE 4.6.1-1
DENATURED U-235/ThO₂ SSCR

FIGURE 4.6.1-2

costs are lower because the heavy metal throughput is lower for the reduced density cases. Finally, the carrying charges are lower because less total fissile material is required to maintain criticality. For both the near term and long term cost assumptions, the power costs decrease monotonically with fuel density.

4.6.2 EFFECT OF INCREASING THE FUEL DISCHARGE EXPOSURE

The effect on fuel resource requirements of increasing the fuel discharge exposure is shown in Figure 4.6.2-1. Although it was recognized that increasing the fuel discharge exposure would result in higher fuel resource requirements for fuel cycles employing recycle, it was hoped that spectral shift control operation would minimize the increase in requirements, while at the same time resulting in significantly lower power generation costs. The U_3O_8 requirements increase about 8% when the discharge exposure is extended to 50,000 MWd/t; the separative work requirements exhibit an increase of about 5% for the same discharge exposure.

The effect on total power costs of increasing the discharge exposure is shown in Figure 4.6.2-2. A significant reduction in power costs is obtained for both sets of cost assumptions. This reduction is primarily the result of a smaller annual throughput of fuel to the fabrication and the reprocessing plants. The higher fuel exposure causes the cycle lengths to be longer (for the three-batch fuel management scheme assumed for these cases) and therefore refueling is less frequent. In addition, the longer cycle lengths result in somewhat higher capacity factors which also reduce the power

FIGURE 4.6.2-1

FIGURE 4.6.2-2

generation costs. The 1.3 mills/kWh lower power costs for the near term cost assumptions represent nearly a 7% reduction relative to the base case.

4.6.3 EFFECT OF VARYING THE NUMBER OF FUEL BATCHES

The variation of fuel resource requirements with the number of in-core fuel batches is shown in 4.6.3-1 (constant burnup). As discussed previously, increasing the number of fuel batches increases the conversion ratio by reducing the fission product absorptions. The increased number of fuel batches also reduces the excess reactivity to be controlled which means lower D_2O concentrations in the SSCR. Increasing the number of batches from three to four reduces the U_3O_8 and separative work requirements only about 1% and 1.5%, respectively.

The change in power costs with the number of fuel batches is shown in Figure 4.6.3-2. The trends observed in the figure are primarily a result of the change in capacity factor as the refueling interval changes. With four in-core fuel batches, refueling takes place every nine months. The more frequent refuelings cause the capacity factor to be lower (under the assumptions made in the core parameter studies) which results in higher power costs. If the capacity factor is held constant, then the trend of increasing power costs with the number of fuel batches is reversed. The magnitude of the total variation in power costs is about the same for the constant capacity factor cases as for the variable capacity cases. For this situation, the four-batch refueling scheme would have power costs about 0.1 and 0.2 mills/kWh lower than the base case for the near term and

FIGURE 4.6.3-1
DENATURED U-235/ThO₂ SSCR

FIGURE 4.6.3-2
DENATURED U-235/ThO₂ SSCR
VARIATION OF TOTAL BUSBAR POWER COSTS vs NUMBER OF IN-CORE BATCHES

long term cost assumptions, respectively.

4.6.4 EFFECT OF MODIFYING THE FUEL ROD PITCH

The effect on fuel resource requirements of varying the fuel rod pitch (while keeping the fuel rod diameter constant) is shown in Figure 4.6.4-1. It appears that the conventional PWR lattice configuration is essentially optimized for the denatured U-235/thorium fueled SSCR. This result should probably have been expected since, to a large extent, the PWR lattice has been designed with recycle in mind. As mentioned previously, for tighter lattices (lower water-to-fuel ratio), thorium is replaced by uranium because of the higher fissile loading requirements and therefore conversion to U-233 is reduced. For looser lattices, structural materials and other parasitic absorbers compete more favorably for neutrons.

The variation of power cost with changing W/F ratio (as a result of changing the lattice pitch) is shown in Figure 4.6.4-2. Slightly lower power costs are obtained at the higher W/F ratios because of the lower carrying charges on the fissile inventory. The higher resource requirements are offset by the lower fabrication charges which occur because the recycle fraction is smaller for the looser lattice and the unit costs for makeup fuel fabrication are much smaller than those for recycle fuel fabrication.

FIGURE 4.6.4-1
DENATURED U-235/ThO₂ SSCR

FIGURE 4.6.4-2
DENATURED U-235/ThO₂ SSCR

4.6.5

EFFECT OF MODIFYING THE FUEL ROD DIAMETER

The trends observed for the variable rod diameter cases are similar to those observed for the variable rod pitch cases. The effect on fuel resource requirements of varying the fuel rod diameter is shown in Figure 4.6.5-1. Again, the standard PWR lattice appears nearly optimal. The sensitivity to lattice design is reduced considerably from that observed for the variable rod pitch cases. This is a result of the variable fuel loading that accompanies the changing fuel rod diameter. As the W/F ratio decreases, both the uranium and thorium loadings increase. This increased thorium loading helps compensate for the additional captures in U-238, which competes with Th-232 for the excess neutrons. Since the residual plutonium in the spent fuel is not utilized in the present strategy, a large fraction of U-238 captures can be considered parasitic and diminish thorium conversions.

The variation of total power costs with changing fuel rod diameters is shown in Figure 4.6.5-2. The trends observed in this figure are almost identical to those of Figure 4.6.4-2. Somewhat larger cost savings can be obtained by employing higher W/F ratios. Power cost savings of about 0.2 and 0.3 mills/kWh can be obtained for the near term and long term cost assumptions, respectively, by use of a lattice design with a water-to-fuel ratio of 2.5.

4.6.6

EFFECT OF EMPLOYING HIGHER HEAVY WATER CONCENTRATIONS

The variation of fuel resource requirements with range of D_2O concentrations is shown in Figure 4.6.6-1. As mentioned in Section 4.3, the range of heavy water concentrations was selected for investigation in anticipation

FIGURE 4.6.5-1
DENATURED U-235/ThO₂ SSCR

FIGURE 4.6.5-2
DENATURED U-235/ThO₂ SSCR

FIGURE 4.6.6-1
DENATURED U-235/ThO₂ SSCR

FIGURE 4.6.6-2
DENATURED U-235/ThO₂ SSCR

that the conversion ratio would be enhanced by the higher epithermal flux. It turned out, however, that the reduction in η for the fissile isotopes far outweighs the effect of increased epithermal flux, resulting in a lower integrated conversion ratio. The fissile isotope U-235 suffers the most from the harder neutron spectrum. Employing heavy water concentrations starting with 70 wt% D_2O at BOC and ending with 59 wt% D_2O at EOC (compared to 50 wt% and 2 wt%) results in an average η value for U-235 which is about 10% lower than the base case value; the other fissile isotopes exhibited much smaller degradation of η than U-235. Also, as the fissile loadings increased to maintain criticality in the harder neutron spectrum, the amount of thorium loaded has to be decreased to make room for the uranium. Thus, less thorium was available for fertile conversion. The net result of the poorer performance of U-235 and the smaller loadings of thorium was to substantially reduce the conversion ratio and therefore raise the fuel resource requirements of the SSCR with increasing average D_2O concentration.

The effect on total power costs of varying the range of D_2O concentrations is shown in Figure 4.6.6-2. The trend follows closely to that of the fuel resource requirements. The additional economic penalty of higher D_2O inventory costs also contributes to the unattractiveness of this strategy.

4.7 OTHER FUEL CYCLES

Two additional fuel cycles were analyzed as part of the SSCR study: the denatured U-233/ThO₂ fuel cycle and the plutonium/ThO₂ fuel cycle.

Optimization studies were not conducted for these cycles; however, it is felt that they would behave much the same as the denatured U-235/ThO₂ fuel cycle. The denatured U-233/ThO₂ cycle, in particular, has similar properties: a considerable amount of U-238 denaturant which will displace thorium when the fissile loading requirement increases and the conversion of U-238 to plutonium which is assumed not to be recycled. One important feature of this fuel cycle which distinguishes it from the denatured U-235/ThO₂ cycle and which may result in somewhat different behavior in an optimization study is the superior performance of U-233 over a wide range of neutron spectra. The eta (η) value of U-233 is relatively constant over the neutron spectra present in the SSCR and may therefore lead to improved fuel performance when the spectrum is intentionally hardened (e.g., when the range of D₂O concentrations is raised).

Since both the denatured U-233/ThO₂ cycle and the plutonium/ThO₂ cycle require an exogenous supply of fissile material and since they both breed a fissile material which is not utilized in the parent cycle, a different objective may be desirable if optimization studies were conducted on these cycles. The objective of the present two parameter optimization studies was to obtain good fuel resource utilization and economics for the individual cycles being analyzed. A more system oriented objective may be appropriate for the denatured U-233/ThO₂ and plutonium/ThO₂ cycles. For example, it may be desirable to maximize U-233 production in the plutonium/ThO₂ cycle at the expense of plutonium consumption. Total fissile consumption may be higher in the plutonium/ThO₂ cycle, but more U-233 may be available for use in an alternate system (e.g., in the LWBR).

The core parameter studies have shown that there is little incentive to change the SSCR lattice design from that of the conventional PWR. In general, savings of less than 8% in fuel resource requirements and less than 1% in power generation costs can be obtained by modifying the lattice configuration. In addition, the potential of backfitting spectral shift control into existing plants or plants under construction would be severely limited if modification to the core geometry were made. In view of these considerations, the SSCR core configuration was chosen identical to that of the reference PWR. The reactor design and operating characteristics of this reference are summarized in Table 4.8-1.

TABLE 4.8-1
SYSTEM 80TM CORE PARAMETERS

Core Power, MW(th)	3800
Nominal Electrical Output, MWe	1270
Fuel Assemblies	241
Fuel Rods Per Assembly	236
Control Rod Channels Per Assembly	4
Instrumentation Channels Per Assembly	1
Active Fuel Length (in.)	150
Specific Power (W/cm ³)	94.9
Average Linear Heat Rate (kw/ft)	5.34

OPERATING PARAMETERS

Primary System Pressure, psia	2250
Reactor Coolant Average Temperature (°F)	592
Nominal Fuel Temperature (°F)	1270

FUEL ASSEMBLIES (Dimensions at 68°F)

Fuel Rod Array	16x16
Assembly Pitch (in.)	8.18
Fuel Cell Pitch (in.)	0.5063
Fuel Pellet Diameter (in.)	0.325
Zr Clad Thickness (in.)	0.025
Gap Thickness (in.)	0.00350
Water Channel between Assemblies (half width) (in.)	0.0394
CEA/Instrumentation Channel Zr Guide-Tube: ID/OD (in.)	0.901/0.972
Coolant to Fuel Pellet Volume Ratio	
Fuel Cell Ratio	1.714
Assembly Ratio	2.041

FUEL STACK DENSITY (g/cm³)

UO ₂	10.054
THO ₂	9.173

REFERENCES FOR SECTION 4

"Optimization of the Spectral Shift Controlled Reactor Design"

1. W. R. Cadwell, "PDQ-7 Reference Manual," WAPD-TM-678, Bettis Atomic Power Laboratory, January 1967.
2. "Assessment of Thorium Fuel Cycles in Pressurized Water Reactors," EPRI NP-359, Electric Power Research Institute, February 1977.
3. "Combustion Engineering Standard Safety Analysis Report (CESSAR)," Docket Number STN-50-470.

SECTION 5 - SSCR REFERENCE

DESIGN

5.1 DESCRIPTION OF SSCR REFERENCE DESIGN

5.1.1 DESIGN APPROACH

The conceptual design of the SSCR was developed by modifying the existing design of a large PWR nuclear power plant for spectral shift control operation. In developing the SSCR design the capability of the plant was retained to resume operation with conventional poison control essentially without hardware changes.

The nuclear power plant which is the basis for the SSCR design consists of Combustion Engineering's System 80TM Pressurized Water Reactor with a nominal thermal output of 3800 MW(th) and Stone & Webster's Reference Nuclear Power Plant with a nominal electrical output of 1300 MW(e). A description of Combustion Engineering's System 80TM is provided in CESSAR⁽¹⁾ - Combustion Engineering's Standard Safety Analysis Report. The companion description of Stone & Webster's Reference Nuclear Power Plant incorporating the System 80TM reactor is provided in SWESSAR⁽²⁾ - Stone & Webster's Standard Safety Analysis Report. In the following description of the SSCR reference is made frequently to CESSAR and SWESSAR or SWESSAR/CESSAR. CESSAR refers to the System 80TM PWR using conventional poison control while SWESSAR or SWESSAR/CESSAR refers to the plant systems of a nuclear power station incorporating the System 80TM conventional PWR.

The modifications for spectral shift control were made essentially all in the plant systems of the PWR. As described in the previous chapter, the System 80TM core lattice turned out to be a good choice for the SSCR. Thus, no changes to the reactor core and vessel became necessary and, consequently, the primary coolant loops including steam generators remain the same for the SSCR as for the conventional PWR.

REFERENCES FOR SECTION 5.1.1

"Design Approach"

1. CESSAR, Combustion Engineering Standard Safety Analysis Report; Docket Number STN-50-470.
2. SWESSAR-P1, Pressurized Water Reactor Reference Nuclear Power Plant Safety Analysis Report; Docket Number STN-50-495.

5.1.2

DESCRIPTION OF PLANT LAYOUT

5.1.2.1

Overall Plant

The overall plant layout is based on the SWESSAR/CESSAR arrangement for a single unit and is shown in Figure 5.1.2.1-a. The cylindrical Containment Structure is surrounded by the Annulus Building. South of the Annulus Building is the Turbine Building, while the Control Building is located to the west of the Annulus Building. The Fuel Building is placed east of the Annulus Building and the Solid Waste and Decontamination Building is north of it. The heat sink for the site consists of mechanical draft cooling towers. The site is suitable for two units.

The major modifications for spectral shift control operation to the SWESSAR/CESSAR arrangement are also shown on the plot plan of Figure 5.1.2.1-a. They are the addition of a D_2O Upgrader Facility to process heavy water in the north-east corner of the site, expansion of the Solid Waste and Decontamination Building to accommodate the Boron Recovery and Radioactive Liquid Waste System equipment formerly located within the Annulus Building, additional tankage in the Annulus Building, and a reduction of the yard tankage.

NOTES

1. SCALE: 1"=40'-0"

LEGEND

INDICATES SEISMIC CAT I STRUCTURES
INDICATES SEISMIC CAT I FOUNDATIONS

INDICATES SEISMIC CAT. I STRUCTURES

FIG. 5.1.2.1-a (SH 1 OF 1)

PLOT PLAN

SPECTRAL SHIFT CONTROLLED REACTOR

CONCEPTUAL DESIGN
PWR STATION-1300MW UNIT-MODEL C

CONCRETE & STEELWORK REINFORCEMENTS CORPORATION

REPORTS & WHITEPAPER: **RECOMMENDING COMMISSION**
 REPORT, 2002.

13062C-PWR-EM-20

DATE					
DESIGNED BY	DRIVEN BY	REVIEWED BY	DATE	REVISION	DATE

5.1.2.1.1 Annulus Building

The equipment arrangement in the Annulus Building is shown on Figure 5.1.2.1-b (19 sheets) which is based on SWESSAR/CESSAR.

The basic approach used in accommodating the SSCR design in SWESSAR/CESSAR was to minimize changes to existing safety-related equipment layout (e.g., engineered safeguards areas), thereby, minimizing the impact on licensing the SSCR design. Furthermore, an effort was made to arrange D_2O carrying tanks or equipment in earthquake- and tornado-protected structures or in structures with earthquake-protected foundations. Arrangement of additional storage tanks for spectral shift control operation in the Annulus Building (a Seismic Category I, tornado-protected structure) required shifting equipment which was originally in this space to the Solid Waste and Decontamination Building, which, in turn, had to be expanded. For these additional storage tanks a cubicle tank design with stainless steel liner is chosen to more efficiently use the space available. The technology currently used for the fuel pool and reactor cavity liner installation will be applied for these atmospheric tanks. Drainage from the cubicle tanks is accomplished by sloping the tank bottoms or by incorporating a sump design into the tank bottoms similar to the cubicle sumps shown in the Auxiliary Feedwater Storage Tank cubicle in Figure 5.1.2.1-b (sheet 14).

5.1.2.1.2 Solid Waste and Decontamination Building

The equipment arrangement in the Solid Waste and Decontamination Building is shown on Figure 5.1.2.1-c (4 sheets) which is based on SWESSAR/CESSAR.

The Solid Waste and Decontamination Building is expanded to accommodate the increased boron evaporator capacities required for SSCR and to house the Radioactive Liquid Waste equipment (waste concentrator(s), high and low level waste drain tanks, etc) which was moved from the Annulus Building.

5.1.2.1.3 Yard Tankage

The function of the two Boron Recovery Tanks of the SWESSAR/CESSAR design are performed by cubicle hold-up tanks in the Annulus Building. Thus, these tanks could be eliminated from the yard tank farm. The Primary Grade Water Tanks are renamed the Light Water Storage Tanks to more accurately reflect their uses in the SSCR design.

In general, yard piping is buried to a depth which will preclude freezing of the process fluid. Protective coating systems are provided on external surfaces of the pipes to minimize corrosion. Guard pipes and monitoring systems are provided for all potentially radioactive process systems such that leakage can be detected and contained.

5.1.2.1.4 D₂O Upgrader Facility

The D₂O Upgrader Facility is located northeast of the Solid Waste and Decontamination Building to allow for future expansion of additional units on the site with a maximum utilization of common facilities (e.g., service building, warehouse, etc.) as provided with the SWESSAR/CESSAR design. The distance of the Upgrader Facility from the Solid Waste and Decontamination Building prevents the Upgrader Towers (225 ft) from impacting on any Seismic Category I foundations in the event of failure. The arrangement of the Upgrader Facility is described in more detail in the following section.

FIG. 5.1.2.1-b (SH 2 OF 19)

GENERAL ARRANGEMENT
ANNULUS BLDG - PLAN EL-152'-6"
SPECTRAL SHIFT CONTROLLED REACTOR

CONCEPTUAL DESIGN
PWR STATION-1300 MW UNIT-MODEL C
SPECTRAL SHIFT CONTROLLED REACTOR

DRAWING NUMBER 13062C-PWR-EM-249B-1

Designed by C. MIDDENDORF
Checked by S. TEDESCHI

EQUIPMENT LIST:

4. REACTOR PLANT COMPONENT COOLING WATER PUMP
5. REACTOR PLANT COMPONENT COOLING WATER HEAT EXCHANGER
26. BORIC ACID D₂O SEPARATING UNIT AUXILIARY EVAPORATOR
27. BORIC ACID D₂O SEPARATING UNIT DISTILLATE TANK
34. LOW LEVEL WASTE EFFLUENT FILTER
43. BORON RECOVERY FILTER
44. BORON EVAPORATOR BOTTOMS FILTER
45. CESIUM REMOVAL ION EXCHANGER
46. DEGASIFIER & CONDENSER
51. PROCESS GAS CHARCOAL BED ADSORBER
53. PROCESS GAS COMPRESSOR PREFILTER AFTERCOOLER
54. PROCESS GAS RECEIVER TANK
56. BORIC ACID TANK
57. BORIC ACID MIXING TANK
59. BORIC ACID TRANSFER PUMP
60. BORIC ACID FILTER
61. DEBORATING ION EXCHANGER
62. PURIFICATION ION EXCHANGER
63. PURIFICATION FILTER
65. CHEMICAL MIXING TANK

PLAN EL(-)25'-0"
(EXCEPT AS NOTED)

NOTES:
1. SCALE 1"=1'-0"
2. GENERAL NOTES & REFERENCE DWGS EM-249A

FIG. 5.1.2.1-b (SH 3 OF 19)

GENERAL ARRANGEMENT
ANNULUS BLDG - PLAN EL(-)25'-0"
SPECTRAL SHIFT CONTROLLED REACTOR

CONCEPTUAL DESIGN
PWR STATION-1300 MW UNIT-MODEL C
GENERAL & SPECIAL REQUIREMENTS CORPORATION
DESIGNED, MADE

DRAWING NUMBER 13062C-PWR-EM-249C-1

Designed by: C. MCDONNELL
Drawn by: S. F. TEDESCHI

FIG. 5.1.2.1-b (SH 4 OF 19)

GENERAL ARRANGEMENT
ANNULUS BLDG PLAN EL(-)25'-0"
SPECTRAL SHIFT CONTROLLED REACTOR

CONCEPTUAL DESIGN
PWR STATION-1300 MW UNIT-MODEL C
SPONS & ENGINEER: BOSTONIAN CORPORATION
BOSTON, MASS.

DRAWING NUMBER 13062C-PWR-EM-249D-1

Designed by: C. MIDDENDORF
Rev. 0001 by:

Drawn by: M. JOHNSON
Rev. 0001 by: S. TEDESCO

FIG. 51.21-b (SH 5 OF 19)

GENERAL ARRANGEMENT
ANNULUS BLDG - PLAN EL 0'-6"
SPECTRAL SHIFT CONTROLLED REACTOR

CONCEPTUAL DESIGN
PWR STATION-1300-MW UNIT-MODEL C
BROWN & WHEELER ENGINEERING CORPORATION
BOSTON, MASS.

DRAWING NUMBER 13062C-PWR-EM-249E-1

Designed by C. HEDENDORF
Checked by S.F. TEDESCO

FIG.5.1. 2.1-b (SH 6 OF 19)

GENERAL ARRANGEMENT
ANNULUS BLDG - PLAN EL 0'-6"
SPECTRAL SHIFT CONTROLLED REACTOR

CONCEPTUAL DESIGN
PWR STATION-1300 MW UNIT-MODEL C
SPECTRAL SHIFT CONTROLLED REACTOR

13062C-PWR-EM-249F-1

Revised by C. LINDENBERG
Rev. 0001 by S. T. DESCO

FIG. 5.1.2.1-b (SH 7 OF 19)

GENERAL ARRANGEMENT
ANNULUS BLDG - PLAN EL 29'-3"
SPECTRAL SHIFT CONTROLLED REACTOR

CONCEPTUAL DESIGN
PWR STATION-1500 MW UNIT-MODEL C
SPECTRAL SHIFT CONTROLLED REACTOR
GENERAL NOTES

DRAWING NUMBER 13062C-PWR-EM-249G-1

Designed by: C. H. DENDORF
Checked by: S. T. DESSO

NOTES: 1. SCALE: 1/8" = 1'-0"
2. GENERAL NOTES & REFERENCE DWGS EM-249A

FIG. 5.1.21-b (SH 8 OF 19)

GENERAL ARRANGEMENT
ANNULUS BLDG - PLAN EL 29'-3"
SPECTRAL SHIFT CONTROLLED REACTOR

CONCEPTUAL DESIGN
PWR STATION 1300 MW UNIT-MODEL C
GENERAL & SPECIAL REQUIREMENTS ORGANIZATION

DRAWING NUMBER 13062C-PWR-EM-249H-1

Designed by C. MIDDENDORF Drawn by J. L. JOHNSON
Rev. 0000 by S. TEDESCO

FIG. 5.1. 2.1-b (SH9 OF 19)

GENERAL ARRANGEMENT
ANNULUS BLDG - PLAN EL 45'-0"
SPECTRAL SHIFT CONTROLLED REACTOR

CONCEPTUAL DESIGN
PWR STATION-1300MW UNIT-MODEL C
BROWN & WHEELER ENGINEERING CORPORATION

DRAWING NUMBER 13062C-PWR-EM-249J-1

Designed by C. MIDDENDORF
Drawn by S. TEDESCO

EQUIPMENT LIST:
6. REACTOR PLANT COMPONENT COOLING WATER SURGE TANK

NOTES:
1. SCALE: 1/8" = 1'-0"
2. GENERAL NOTES & REFERENCE: DWGS EM-249A

FIG. 5.1.2.1-b (SH 10 OF 19)

GENERAL ARRANGEMENT
ANNULUS BLDG - PLAN EL 45'-0"
SPECTRAL SHIFT CONTROLLED REACTOR

CONCEPTUAL DESIGN
PWR STATION 1300MW UNIT-MODEL-C
ANNULUS & VENTILATION COMPARTMENT

DRAWING NUMBER 13062C-PWR-EM-249K-1

Designed by CAMDENORF
Drawn by EML PERRO
Checked by S. TEDESCO

EQUIPMENT LIST
14. LETDOWN TANK
20. HOLD-UP TANK TRANSFER PUMP
21. HOLD-UP TANK
76. HYDROGEN RECOMBINER

NOTES
1. SCALE: 1/4" = 1'-0"
2. GENERAL NOTES & REFERENCE DWGS EM-249A

FIG. 5.1.21-b(SH 11 OF 19)

GENERAL ARRANGEMENT
ANNULUS BLDG - SECT 1-1 & 2-2
SPECTRAL SHIFT CONTROLLED REACTOR

CONCEPTUAL DESIGN
PWR STATION-SCOMM UNIT-MODEL C
DESIGN & FABRICATION
DRAWING NUMBER 13062C-PWR-EM-249L-1

Designed by C. M. D. B. D. P. O. R. F. Date 10/5/75 TEL 2530

3-3

(EM-249B K-7)
(EM-249D K-7)
(EM-249F K-7)
(EM-249H K-7)
(EM-249K K-7)

EQUIPMENT LIST

14. LETDOWN TANK
17. LETDOWN TANK TRANSFER PUMP
18. HEAVY WATER STORAGE TANK
19. HEAVY WATER TRANSFER PUMP
22. REACTOR COOLANT SYSTEM LOOP DRAIN TANK PUMP
23. REACTOR COOLANT SYSTEM LOOP DRAIN TANK
69. CHILLED WATER REFRIGERATION UNIT
71. CHILLED WATER CIRCULATING PUMP
77. PROCESS VENT FILTER
79. PROCESS VENT FILTER FAN

4-4

(EM-249B I-8)
(EM-249D I-8)
(EM-249F I-8)
(EM-249H I-8)
(EM-249K I-8)

NOTES:

1. SCALE: 1/8" = 1'-0"
2. GENERAL NOTES & REFERENCE DWGS. EM-249A

FIG. 5.1.2.1-b (SH12 OF 19)

GENERAL ARRANGEMENT
ANNULUS BLDG - SECT 3-3 & 4-4
SPECTRAL SHIFT CONTROLLED REACTOR

CONCEPTUAL DESIGN

PWR STATION-1300MW UNIT-MODEL C

DESIGNED & DRAWN BY: [Signature]

CHECKED BY: [Signature]

DATE: 10/1/77

DRAWING NUMBER: 13062C-PWR-EM-249M-1

SCALE: 1/8" = 1'-0"

BY: [Signature]

DATE: 10/1/77

5-5
(EM-249B H-8)
(EM-249D H-8)
(EM-249F H-8)
(EM-249H H-8)

EQUIPMENT LIST
9. AUXILIARY FEEDWATER PUMP, TURBINE DRIVEN
15. REACTOR VESSEL FLUSH TANK
16. REACTOR VESSEL FLUSH TANK TRANSFER PUMP
69. CHILL WATER REFRIGERATION UNIT
71. CHILL WATER CIRCULATING PUMP
75. STEAM GENERATOR BLOWDOWN FLASH TANK
77. PROCESS VENT FILTER
78. PROCESS VENT FILTER FAN

6-6
(EM-249B G-8)
(EM-249D G-8)
(EM-249F G-8)
(EM-249H G-8)

NOTES
1. SCALE: 1/8" = 1'-0"
2. GENERAL NOTES & REFERENCE DWGS EM-249A

FIG. 5.1.2.1-b (SH 13 OF 19)

GENERAL ARRANGEMENT
ANNULUS BLDG - SECT 5-5 & 6-6
SPECTRAL SHIFT CONTROLLED REACTOR

CONCEPTUAL DESIGN
PWR STATION 1300MW UNIT MODEL C
BURNS & MCDONNELL CORPORATION
BOSTON, MASS.

DRAWING NUMBER 13062C-PWR-EM-249N-1

Designed by C. MCDONNELL
Drawn by S. TEDESCHI

EQUIPMENT LIST
 6 REACTOR PLANT COMPONENT COOLING WATER SURGE TANK
 10 AUXILIARY FEEDWATER STORAGE TANK
 11 REFUELING WATER STORAGE TANK
 12 CHEMICAL ADDITION TANK
 13 CHEMICAL ADDITION TANK RECIRCULATING PUMP

NOTES
 1. SCALE 1/8"=1'-0"
 2. GENERAL NOTES & REFERENCE DWGS EM-249A

FIG. 5.1.2.1-b (SH 14 OF 19)

GENERAL ARRANGEMENT
 ANNULUS BLDG - SECT 7-7 & 8-8
 SPECTRAL SHIFT CONTROLLED REACTOR

CONCEPTUAL DESIGN
 PWR STATION 1300MW UNIT - MODEL C
 DESIGN & CONSTRUCTION ORGANIZATION

13062C-PWR-EM-249P-1

Designed by C. MOEDORF
 Drawn by S. TEDESCO

9-9

(EM-249A A-7)
(EM-249C A-7)
(EM-249E A-7)
(EM-249G A-7)
(EM-249J A-7)

EQUIPMENT LIST

- 1 HIGH HEAD SAFETY INJECTION PUMP
- 2 LOW HEAD SAFETY INJECTION PUMP
- 3 CONTAINMENT SPRAY PUMP
- 4 REACTOR PLANT COMPONENT COOLING WATER PUMP
- 5 REACTOR PLANT COMPONENT COOLING WATER HEAT EXCHANGER
- 6 AUXILIARY FEEDWATER PUMP, MOTOR DRIVEN

10-10

(EM-249A B-5)
(EM-249C B-5)
(EM-249E B-5)
(EM-249G B-5)
(EM-249J B-5)

NOTES

1. SCALE: 1/4" = 1'-0"
2. GENERAL NOTES & REFERENCE DINGS EM-249A

FIG. 51. 21-b (SH15OF19)

GENERAL ARRANGEMENT
ANNULUS BLDG - SECT 9-9 & 10-10
SPECTRAL SHIFT CONTROLLED REACTOR

CONCEPTUAL DESIGN
PWR STATION-1300MW UNIT-MODEL C
REACTOR & VESSEL CONTAINMENT SYSTEM

13062C-PWR-EM-249Q-1

FOR SSCR STUDY REPORT
Drawn by E. F. FERRARO
Checked by S. TEDESCHI

- EQUIPMENT LIST
46 DEGASIFIER	(EM-249C, C-4)
47 DEGASIFIER RECIRCULATING PUMP	(EM-249E, C-4)
48 DEGASIFIER RECOVERY EXCHANGER	(EM-249G, C-4)
49 DEGASIFIER FEED PREHEATER	(EM-249J, C-4)
50 TRIM COOLER	
52 PROCESS GAS REFRIGERANT DRYER	
53 PROCESS GAS COMPRESSOR PREFILTER AFTER COOLER	
54 PROCESS GAS RECEIVER TANK	
55 PROCESS GAS GLYCOL COOLER	

- 12-12
(EM-249A, C-3)
(EM-249C, C-3)
(EM-249E, C-3)
(EM-249G, C-3)
(EM-249J, C-3)

NOTES:
1. SCALE: 1/4"=1'-0"
2. GENERAL NOTES & REFERENCE DWGS EM-249A

FIG. 5.1.2.1-b (SH 16 OF 19)

GENERAL ARRANGEMENT
ANNULUS BLDG - SECT 11-11 & 12-12
SPECTRAL SHIFT CONTROLLED REACTOR

CONCEPTUAL DESIGN
PWR STATION-1300MW UNIT-MODEL C
DESIGN & WIREMENT SUPERVISOR'S COMPLETION
DATE: 10/1/80

13062C-PWR-EM-249R-1

FOR SSCR STUDY REPORT

DESIGNED BY: C. MODONORI
CHECKED BY: S. TEBESKO

13-13
(EM-249A D-2)
(EM-249C D-2)
(EM-249E D-2)
(EM-249G D-2)
(EM-249J D-2)

EQUIPMENT LIST:
24. BORIC ACID D₂O SEPARATING UNIT BOTTOMS PUMP
25. DISTILLATE PUMP
26. AUXILIARY EVAPORATOR
27. DISTILLATE TANK
28. CONDENSER
35. DEBORATING SURGE TANK
36. BORON EVAPORATOR FEED PUMP
51. PROCESS GAS CHARCOAL BED ADSORBER
56. BORIC ACID TANK
79. CHARGING PUMP
80. ANNULUS BUILDING RECOVERABLE AERATED DRAINS COLLECTION TANK TRANSFER PUMP

14-14
(EM-249A F-1)
(EM-249C F-1)
(EM-249E F-1)
(EM-249G F-1)
(EM-249J F-1)

NOTES:
1. SCALE 1"=1'-0"
2. GENERAL NOTES & REFERENCE DWGS EM-249A

FIG. 5.1.2.1-b (SH17 OF 19)
GENERAL ARRANGEMENT
ANNULUS BLDG - SECT 13-13 & 14-14
SPECTRAL SHIFT CONTROLLED REACTOR

CONCEPTUAL DESIGN
PWR STATION 1300MW UNIT MODEL C
GENERAL & SPECIAL REQUIREMENTS CORPORATION

DRAWING NUMBER 13062C-PWR-EM-249S-1

Designed by: C. MIDDENDORF
Drawn by: S. TEDESCO

15-15
(EM-249A H-1)
(EM-249C H-1)
(EM-249E H-1)
(EM-249G H-1)

- EQUIPMENT LIST
- 34 LOW LEVEL WASTE EFFLUENT FILTER
 - 57 BORIC ACID MIXING TANK
 - 58 BORIC ACID BATCHING TANK
 - 59 BORIC ACID TRANSFER PUMP
 - 60 BORIC ACID FILTER
 - 63 PURIFICATION FILTER
 - 64 VOLUME CONTROL TANK
 - 65 CHEMICAL MIXING TANK
 - 66 CHARGING PUMP
 - 67 ANNULUS BUILDING BACKWASH HOLD TANK
 - 68 ANNULUS BUILDING BACKWASH HOLD TANK PUMP
- NOTE 4

16-16
(EM-249A K2)
(EM-249C K2)
(EM-249E K2)
(EM-249G K2)

- NOTES
1. SCALE: 1/8" = 1'-0"
 2. GENERAL NOTES & REFERENCE DWGS EM-249A

FIG. 5.1.2.1-b (SH 18 OF 19)

GENERAL ARRANGEMENT
ANNULUS BLDG - SECT 15-15 & 16-16
SPECTRAL SHIFT CONTROLLED REACTOR
CONCEPTUAL DESIGN
PWR STATION 13062C UNIT MODEL C
DESIGN & CONSTRUCTION ORGANIZATION
GENERAL, 1988
13062C-PWR-EM-249T-1

17-17

(EM-249A L-2)
(EM-249C L-2)
(EM-249E L-2)
(EM-249G L-2)

EQUIPMENT LIST:
43 BORON RECOVERY FILTER
44 BORON EVAPORATOR BOTTOMS FILTER
61 DEIONIZING ION EXCHANGER
62 PURIFICATION ION EXCHANGER

18-18

(EM-249A M-3)
(EM-249C M-3)
(EM-249E M-3)
(EM-249G M-3)

NOTES
1. SCALE: 1/8" = 1'-0"
2. GENERAL NOTES & REFERENCE DWGS EM-249A

FIG. 5.1.2.1-b (SH 19 OF 19)

GENERAL ARRANGEMENT
ANNULUS BLDG - SECT 17-17 & 18-18
SPECTRAL SHIFT CONTROLLED REACTOR

CONCEPTUAL DESIGN
PWR STATION 13062C UNIT MODEL C
DESIGN & CONSTRUCTION ORGANIZATION

13062C-PWR-EM-249U-1

FOR SSCR STUDY REPORT

DESIGNED BY: W. H. HODGSON
CHECKED BY: J. E. HODGSON
DATE: 11/15/78

EQUIPMENT LIST

- 300 WASTE SLUDGE TANK PUMP
 301 WASTE SLUDGE TANK PUMP
 302 RESIN HOLD TANK
 303 UREA FORMALDEHYDE TANK PUMP
 304 UREA FORMALDEHYDE TANK PUMP
 305 UREA FORMALDEHYDE TANK UNLOADING PUMP
 306 UREA FORMALDEHYDE CATALYST TANK PUMP
 307 UREA FORMALDEHYDE CATALYST TANK PUMP
 308 WASTE DEMINERALIZER
 309 WASTE DEMINERALIZER FILTER
 310 BORDO DEMINERALIZER
 311 BORDO DEMINERALIZER FILTER
 312 CONTAMINATED DRAINS TANK
 313 CONTAMINATED DRAINS TANK PUMPS
 314 LOW LEVEL WASTE DRAIN TANK
 315 LOW LEVEL WASTE DRAIN PUMP
 316 EVAPORATOR BOTTOMS TANK
 317 EVAPORATOR BOTTOMS TANK PUMP
 318 RESIN HOLD TANK
 319 SPENT RESIN TRANSFER PUMP
 320 SPENT RESIN TRANSFER PUMP FILTER
 321 RESIN RECYCLE PUMP
 322 LAUNDRY EVAPORATOR DISTILLATE TANK PUMP
 323 LAUNDRY EVAPORATOR BOTTOMS TANK PUMP
 324 LAUNDRY EVAPORATOR BOTTOMS TANK PUMP
 325 LAUNDRY EVAPORATOR FEED TANK PUMP
 326 LAUNDRY EVAPORATOR FEED TANK PUMP
 327 AUXILIARY CONDENSATE RECEIVER
 328 AUXILIARY CONDENSATE RECEIVER PUMP
 329 SOLID WASTE & DECONTAMINATION RECOVERABLE
 330 SOLID WASTE & DECONTAMINATION RECOVERABLE
 331 AERATED DRAINS COLLECTION TANK
 332 AERATED DRAINS COLLECTION TANK TRANSFER PUMPS
 333

OPTION
NOTE 3

NOTES:

1. SCALE: 1"=1'-0"
2. ALL DOOR OPENINGS TO BE 7'-0" HIGH & ALL LABYRINTH WALLS TO BE 8'-0" HIGH UNLESS OTHERWISE NOTED
3. ARRANGEMENT ENCLOSED AS INDICATED - G - IS AN OPTION ONLY IF LAUNDRY EVAPORATOR AND ASSOCIATED EQUIPMENT ARE TO BE USED

REFERENCE DRAWINGS:

GENERAL ARRANGEMENT SOLID WASTE & DECONTN
 BUILDING SHEET 2,3&4

EM-209B,C&D

FIG. 5.1.2.1-c (SH 1 OF 4)

GENERAL ARRANGEMENT
 SOLID WASTE & DECONTN BLDG SH 1
 SPECTRAL SHIFT CONTROLLED REACTOR

CONCEPTUAL DESIGN
 PWR STATION-1300 MW UNIT-MODEL C
 DESIGN & CONSTRUCTION CONSULTANTS
 CONSULTING ENGINEERS

13062C-PWR-EM-209A-1

FOR SSCR STUDY REPORT

Designed by S.F. TEDESCO

Checked by T. CONROY

EQUIPMENT LIST:

- 300 WASTE SLUDGE TANK
 301 WASTE SLUDGE TANK PUMP
 302 RESIN SURGE TANK
 303 UREA FORMALDEHYDE TANK
 304 UREA FORMALDEHYDE TANK UNLOADING PUMP
 305 UREA FORMALDEHYDE CATALYST TANK
 306 UREA FORMALDEHYDE CATALYST TANK PUMP
 307 UREA FORMALDEHYDE CATALYST TANK PUMP
 308 WASTE DEMINERALIZER
 309 WASTE DEMINERALIZER FILTER
 310 BORON DEMINERALIZER
 311 BORON DEMINERALIZER FILTER
 312 CONTAMINATED DRAINS TANK PUMP
 313 CONTAMINATED DRAINS TANK PUMP
 314 EVAPORATOR BOTTOMS TANK PUMP
 315 EVAPORATOR BOTTOMS TANK PUMP
 316 RESIN HOLD TANK
 317 SPENT RESIN TRANSFER PUMP
 318 SPENT RESIN TRANSFER PUMP FILTER
 319 RESIN RECYCLE PUMP
 320 LAUNDRY EVAPORATOR DISTILLATE TANK
 321 LAUNDRY EVAPORATOR DISTILLATE TANK PUMP
 322 LAUNDRY EVAPORATOR BOTTOMS TANK PUMP
 323 LAUNDRY EVAPORATOR BOTTOMS TANK PUMP
 324 LAUNDRY EVAPORATOR FEED TANK PUMP
 325 LAUNDRY EVAPORATOR FEED TANK PUMP
 326 LAUNDRY EVAPORATOR FEED TANK PUMP
 327 LAUNDRY EVAPORATOR FEED TANK PUMP
 328 LAUNDRY EVAPORATOR FEED TANK PUMP
 329 LAUNDRY EVAPORATOR FEED TANK PUMP
 330 AUXILIARY CONDENSATE RECEIVER
 331 AUXILIARY CONDENSATE RECEIVER PUMP
 332 SOLID WASTE & DECONTAMINATION RECOVERABLE
 333 SOLID WASTE & DECONTAMINATION RECOVERABLE
 AERATED DRAINS COLLECTION TANK
 AERATED DRAINS COLLECTION TANK TRANSFER PUMPS

- NOTES:
 1. SCALE: 1/4" = 1'-0"
 2. GENERAL NOTES & REFERENCE DRAWINGS EM-209A

FIG. 5.1.2.1-c (SH 2 OF 4)

GENERAL ARRANGEMENT
 SOLID WASTE & DECONTN BLDG SH 2
 SPECTRAL SHIFT CONTROLLED REACTOR

CONCEPTUAL DESIGN
 PWR STATION-1300MW UNIT-MODEL C
 BROWN & WHEELER ENGINEERING CORPORATION
 BOSTON, MASS.

DRAWING NUMBER 13062C-PWR-EM-209B-1

Designed by S.F. TEDESCO Drawn by T. CONROY
 Rev. 000 by

13062C-PWR-EM-209C-1

EQUIPMENT LIST

- 1 BORON EVAPORATOR
- 2 BORON EVAPORATOR REBOILER
- 3 BORON EVAPORATOR REBOILER PUMPS
- 4 BORON EVAPORATOR BOTTOMS PUMPS
- 5 BORON DISTILLATE COOLER
- 6 BORON DISTILLATE PUMP
- 7 BORON EVAPORATOR CONDENSER
- 8 BORON DISTILLATE TANK
- 9 WASTE EVAPORATOR
- 10 WASTE EVAPORATOR REBOILER
- 11 WASTE EVAPORATOR REBOILER PUMP
- 12 WASTE EVAPORATOR BOTTOMS PUMP
- 13 REGENERANT CHEMICAL EVAPORATOR
- 14 REGENERANT CHEMICAL EVAPORATOR REBOILER
- 15 REGENERANT CHEMICAL EVAPORATOR REBOILER PUMP
- 16 REGENERANT CHEMICAL EVAPORATOR BOTTOMS PUMP
- 17 REGENERANT DISTILLATE COOLER
- 18 REGENERANT DISTILLATE PUMP
- 19 REGENERANT CHEMICAL CONDENSER
- 20 REGENERANT DISTILLATE TANK
- 21 HIGH LEVEL WASTE DRAIN TANKS
- 22 WASTE EVAPORATOR FEED PUMPS
- 23 BORON TEST TANK PUMPS
- 24 WASTE TEST TANK PUMPS
- 25 LIGHT WATER TRANSFER PUMPS

- NOTES
- 1 SCALE: 1/4"=1'-0"
 - 2 GENERAL NOTES & REFERENCE DRAWINGS EM-209A

FIG. 51.21-c (SH 3 OF 4)

GENERAL ARRANGEMENT
SOLID WASTE & DECONTN BLDG SH 3
SPECTRAL SHIFT CONTROLLED REACTOR

CONCEPTUAL DESIGN
PWR STATION-1300MM UNIT-MODEL C
SPONE & WEBSTER ENGINEERING CORPORATION
BOSTON, MASS.

DRAWING NUMBER 13062C-PWR-EM-209C-1

FOR SSCR STUDY REPORT

EQUIPMENT LIST

- 1 BORON EVAPORATOR
- 2 BORON EVAPORATOR REBOILER
- 3 BORON DISTILLATE COOLER
- 4 BORON DISTILLATE PUMP
- 5 BORON EVAPORATOR CONDENSER
- 6 BORON DISTILLATE TANK
- 7 WASTE EVAPORATOR
- 8 WASTE EVAPORATOR REBOILER
- 9 WASTE DISTILLATE COOLER
- 10 WASTE DISTILLATE PUMP
- 11 WASTE EVAPORATOR CONDENSER
- 12 WASTE DISTILLATE TANK
- 13 REGENERANT CHEMICAL EVAPORATOR
- 14 REGENERANT CHEMICAL EVAPORATOR REBOILER
- 15 REGENERANT DISTILLATE COOLER
- 16 REGENERANT DISTILLATE PUMP
- 17 REGENERANT CHEMICAL CONDENSER
- 18 REGENERANT DISTILLATE TANK
- 19 HIGH LEVEL WASTE DRAIN TANK

NOTES

- 1 SCALE: 1/8"=1'-0"
- 2 GENERAL NOTES & REFERENCE DRAWINGS EM-209A

FIG. 5.1.21-c (SH 4 OF 4)

GENERAL ARRANGEMENT
SOLID WASTE & DECONTN BLDG SH 4
SPECTRAL SHIFT CONTROLLED REACTOR

CONCEPTUAL DESIGN
PWR STATION-1300MW UNIT-MODEL C
STONE & WEBSTER ENGINEERING CORPORATION
BOSTON, MASS.

DRAWING NUMBER 13062C-PWR-EM-209D-1

DESIGNED BY S.F. TEDESCHI CHECKED BY T. CONROY

5.1.2.2 D₂O Upgrader Facility

5.1.2.2.1 Structural Design Criteria

Since the Upgrader serves no safety-related function, the structural design of the upgrader building and the equipment in the upgrader facility was considered not to be designed to criteria appropriate to non-safety-related system. Thus, the upgrader structure does not fall under Seismic Category I. Instead, because of the tritium activity contained in its liquid systems, the upgrader structure was considered to perform a similar function as a structure containing liquid radioactive waste management systems. Guidance by the Nuclear Regulatory Commission (NRC) for the design of structures containing radioactive waste management systems is provided in Sections Ia and Ib of Ref. (1). These guidelines recommend that foundations and adjacent walls of structures housing liquid radwaste systems be designed for the Operating Base Earthquake (0.15 g maximum ground acceleration for a SWESSAR design) to a height sufficient to contain the liquid inventory in the building. These guidelines were applied to the design of the upgrader building. The foundations of the building and of the distillation towers are designed in a "bath-tub" like manner capable of containing the liquid inventory of the upgrader facility. To ensure that a possible release of liquid radioactive material through cracks in the "bath-tub" foundation to the groundwater is not a problem, at some plant sites the liquid retention area may require installation of a metal liner. Because the allowable liquid release to ground water depends on the particular situation at a specific site, the need for a metal liner can be determined only after a site-specific analysis.

With respect to airborne releases of tritium activity, the liquid inventory in the upgrader facility is less than the amount which could be released with respect to site-boundary dose limits (for more details see Section 5.5).

The distillation towers of the upgrader facility are 225 ft high and in the case of their failure, damage to Seismic Category I structures is to be avoided. Consequently, the upgrader facility including the distillation towers is located far enough away from the Annulus Building and Solid Waste and Decontamination Building to preclude any interaction with them.

With respect to wind load, the upgrader building and the distillation towers are designed to withstand a wind speed of 120 miles/hr.

In general, the design of the upgrader facility and the distillation towers is quite similar to the design of a D_2O upgrader facility which is in operation at the Canadian Heavy Water Reactor power station at Pickering, Ontario.

5.1.2.2.2 Building Arrangement

Details of the upgrader building are shown on Figure 5.1.2-2 and 5.1.2-3.

The building is L-shaped and covers an area of about 135 ft by 105 ft. One portion of the building is about 80 ft high and contains the process equipment which is arranged on four floors (for equipment designations on Figures 5.1.2-2 and 5.1.2-3 refer to Section 5.1.8). Adjacent to this building are the three distillation towers. The second portion of the building contains

PRIMARY
BATH-TUB AREA

<p align="center">CE POWER SYSTEMS SSCR HEAVY WATER UPGRADER REFERENCE DESIGN STUDY</p>											
△											
△											
△											
△											
△											
△	1/8" 1/16" 1/32"					L.R		11/11			
REV	DATE	DESCRIPTION	DATE	BY	CHKD	DATE	BY	CHKD	DATE	BY	CHKD
LUMMUS			<p align="center">C.E. LUMMUS COMBUSTION ENGINEERING, INC. BLOOMFIELD, NEW JERSEY 07003</p>								
<p align="center">PLOT PLAN</p>											
<p align="center">DISTILLATION AND PRODUCT RECOVERY</p>											
<p align="center">SECTIONS</p>						<p align="center">AREA - 100</p>					
<p>SCALE 1/8"=1'-0" DWG. NO. E 8358 - 00010 - 0</p>

Figure 5.1.2-3
D₂O Upgrader Facility
Plan View

CE POWER SYSTEMS SSCR HEAVY WATER UPGRADER REFERENCE DESIGN STUDY									
1	2	3	4	5	6	7	8	9	10
11	12	13	14	15	16	17	18	19	20
21	22	23	24	25	26	27	28	29	30
31	32	33	34	35	36	37	38	39	40
41	42	43	44	45	46	47	48	49	50
51	52	53	54	55	56	57	58	59	60
61	62	63	64	65	66	67	68	69	70
71	72	73	74	75	76	77	78	79	80
81	82	83	84	85	86	87	88	89	90
91	92	93	94	95	96	97	98	99	100
101	102	103	104	105	106	107	108	109	110
111	112	113	114	115	116	117	118	119	120
121	122	123	124	125	126	127	128	129	130
131	132	133	134	135	136	137	138	139	140
141	142	143	144	145	146	147	148	149	150
151	152	153	154	155	156	157	158	159	160
161	162	163	164	165	166	167	168	169	170
171	172	173	174	175	176	177	178	179	180
181	182	183	184	185	186	187	188	189	190
191	192	193	194	195	196	197	198	199	200
201	202	203	204	205	206	207	208	209	210
211	212	213	214	215	216	217	218	219	220
221	222	223	224	225	226	227	228	229	230
231	232	233	234	235	236	237	238	239	240
241	242	243	244	245	246	247	248	249	250
251	252	253	254	255	256	257	258	259	260
261	262	263	264	265	266	267	268	269	270
271	272	273	274	275	276	277	278	279	280
281	282	283	284	285	286	287	288	289	290
291	292	293	294	295	296	297	298	299	300
301	302	303	304	305	306	307	308	309	310
311	312	313	314	315	316	317	318	319	320
321	322	323	324	325	326	327	328	329	330
331	332	333	334	335	336	337	338	339	340
341	342	343	344	345	346	347	348	349	350
351	352	353	354	355	356	357	358	359	360
361	362	363	364	365	366	367	368	369	370
371	372	373	374	375	376	377	378	379	380
381	382	383	384	385	386	387	388	389	390
391	392	393	394	395	396	397	398	399	400
401	402	403	404	405	406	407	408	409	410
411	412	413	414	415	416	417	418	419	420
421	422	423	424	425	426	427	428	429	430
431	432	433	434	435	436	437	438	439	440
441	442	443	444	445	446	447	448	449	450
451	452	453	454	455	456	457	458	459	460
461	462	463	464	465	466	467	468	469	470
471	472	473	474	475	476	477	478	479	480
481	482	483	484	485	486	487	488	489	490
491	492	493	494	495	496	497	498	499	500
501	502	503	504	505	506	507	508	509	510
511	512	513	514	515	516	517	518	519	520
521	522	523	524	525	526	527	528	529	530
531	532	533	534	535	536	537	538	539	540
541	542	543	544	545	546	547	548	549	550
551	552	553	554	555	556	557	558	559	560
561	562	563	564	565	566	567	568	569	570
571	572	573	574	575	576	577	578	579	580
581	582	583	584	585	586	587	588	589	590
591	592	593	594	595	596	597	598	599	600
601	602	603	604	605	606	607	608	609	610
611	612	613	614	615	616	617	618	619	620
621	622	623	624	625	626	627	628	629	630
631	632	633	634	635	636	637	638	639	640
641	642	643	644	645	646	647	648	649	650
651	652	653	654	655	656	657	658	659	660
661	662	663	664	665	666	667	668	669	670
671	672	673	674	675	676	677	678	679	680
681	682	683	684	685	686	687	688	689	690
691	692	693	694	695	696	697	698	699	700
701	702	703	704	705	706	707	708	709	710
711	712	713	714	715	716	717	718	719	720
721	722	723	724	725	726	727	728	729	730
731	732	733	734	735	736	737	738	739	740
741	742	743	744	745	746	747	748	749	750
751	752	753	754	755	756	757	758	759	760
761	762	763	764	765	766	767	768	769	770
771	772	773	774	775	776	777	778	779	780
781	782	783	784	785	786	787	788	789	790
791	792	793	794	795	796	797	798	799	800
801	802	803	804	805	806	807	808	809	810
811	812	813	814	815	816	817	818	819	820
821	822	823	824	825	826	827	828	829	830
831	832	833	834	835	836	837	838	839	840
841	842	843	844	845	846	847	848	849	850
851	852	853	854	855	856	857	858	859	860
861	862	863	864	865	866	867	868	869	870
871	872	873	874	875	876	877	878	879	880
881	882	883	884	885	886	887	888	889	890
891	892	893	894	895	896	897	898	899	900
901	902	903	904	905	906	907	908	909	910
911	912	913	914	915	916	917	918	919	920
921	922	923	924	925	926	927	928	929	930
931	932	933	934	935	936	937	938	939	940
941	942	943	944	945	946	947	948	949	950
951	952	953	954	955	956	957	958	959	960
961	962	963	964	965	966	967	968	969	970
971	972	973	974	975	976	977	978	979	980
981	982	983	984	985	986	987	988	989	990
991	992	993	994	995	996	997	998	999	1000
1001	1002	1003	1004	1005	1006	1007	1008	1009	1010
1011	1012	1013	1014	1015	1016	1017	1018	1019	1020
1021	1022	1023	1024	1025	1026	1027	1028	1029	1030
1031	1032	1033	1034	1035	1036	1037	1038	1039	1040
1041	1042	1043	1044	1045	1046	1047	1048	1049	1050
1051	1052	1053	1054	1055	1056	1057	1058	1059	1060
1061	1062	1063	1064	1065	1066	1067	1068	1069	1070
1071	1072	1073	1074	1075	1076	1077	1078	1079	1080
1081	1082	1083	1084	1085	1086	1087	1088	1089	1090
1091	1092	1093	1094	1095	1096	1097	1098	1099	1100
1101	1102	1103	1104	1105	1106	1107	1108	1109	1110
1111	1112	1113	1114	1115	1116	1117	1118	1119	1120
1121	1122	1123	1124	1125	1126	1127	1128	1129	1130
1131	1132	1133	1134	1135	1136	1137	1138	1139	1140
1141	1142	1143	1144	1145	1146	1147	1148	1149	1150
1151	1152	1153	1154	1155	1156	1157	1158	1159	1160
1161	1162	1163	1164	1165	1166	1167	1168	1169	1170
1171	1172	1173	1174	1175	1176	1177	1178	1179	1180
1181	1182	1183	1184	1185	1186	1187	1188	1189	1190
1191	1192	1193	1194	1195	1196	1197	1198	1199	1200
1201	1202	1203	1204	1205	1206	1207	1208	1209	1210
1211	1212	1213	1214	1215	1216	1217	1218	1219	1220
1221	1222	1223	1224	1225	1226	1227	1228	1229	

the control room, chemical laboratory, switch gear room, various maintenance areas, and locker and change rooms on a single floor. The foundations for the process building wing, the distillation towers and a drain tank incorporate the "bath-tub" design so that they can retain the liquid inventory of the equipment. The required utility services are assumed to come into the upgrader building on the second floor of the process equipment wing.

5.1.2.2.3 Ventilation and Drain System

Ventilation. Air conditioning is provided for the control room only. The other service areas have local ventilation and heating units. These units will provide four air changes per hour. Building heating will be accomplished by steam heat (about 6000 lb/hr steam). The floors of the process equipment wing are equipped with exhaust ventilators, which can be activated from the control room.

Drain System. Controlled drainage of equipment as well as leakage will be collected via a drain piping system and directed to the drain tank.

REFERENCES FOR SECTION 5.1.2

"Description of Plant Layout"

1. Design Guidance For Radioactive Waste Management Systems Installed In Light-Water-Cooled Nuclear Power Reactor Plants, Branch Technical Position - ETSB No. 11-1 (Rev. 1); November 1975.

The optimization study of the core configuration (which is described in more detail in Section 4) showed that the System 80TM core geometry is a good choice for spectral shift operation with either uranium or thorium fueling. Thus, the unmodified System 80TM core geometry, reactor vessel and vessel internals were retained for the SSCR.

The reactor core consists of 241 fuel assemblies and 89 control element assemblies. The fuel assemblies are arranged to approximate a right circular cylinder with an equivalent diameter of 143 inches and an active length of 150 inches. The core is surrounded by a core shroud and is supported by a core support barrel of 162.25 in. O.D. and 157 in. I.D.

The reactor vessel of 182.25 in. I.D. contains the core as well as the core internals which support and orient the fuel and control element assemblies. The core internals also guide the reactor coolant through the reactor vessel. A plan view of the vessel and its internals is shown in Figure 5.1.3-1.

The Control Element Assemblies (CEA's) have either 4, 8 or 12 "fingers" which contain the neutron absorber material. The fingers are individually guided and protected from hydraulic forces by the Upper Guide Structure. A standard magnetic jack control element drive mechanism is used with all three CEA types.

Figure 5.1.3-1
REACTOR VERTICAL ARRANGEMENT

5.1.4 FUEL DESIGN

5.1.4.1 General

The SSCR can be fueled with either UO_2 or ThO_2 based fuel. The SSCR core lattice is compatible without modifications with either fuel type. The present PWR fuel management strategy employs three fuel batches which remain in the core for a batch-average burnup of about 30,000 MWd/t. If this strategy is retained for the SSCR, then the fuel rod and assembly design remains identical to that of System 80TM. A brief description of this fuel design is provided in the following subsection 5.1.4.2.-Standard Burnup Fuel Design.

Savings in uranium resources as well as in fuel cycle cost can be made for the UO_2 once-through fuel cycle by extending the burnup and increasing the number of fuel batches. These savings can be realized for the SSCR as well as for the conventional PWR. The batch-average fuel burnup would be extended into the 50,000 MWd/t range while the refueling batches would be increased to five. A batch-average burnup of about 50,000 MWd/t results in a maximum fuel rod burnup of about 62,000 MWd/t and a fuel rod design for this burnup must address several technical problems. These problems are outlined and fuel rod design options are described in subsections 5.1.4.3-Design Considerations For High Burnup Fuel and 5.1.4.4-High Burnup Fuel Design Study.

As mentioned above the fuel design for the SSCR for standard burnup (30,400 MWd/t, batch average) is identical to the System 80TM design and is shown in Figure 5.1.4-1. The basic structure of the fuel assembly consists of five large zircaloy guide tubes which are connected by common lower and upper end fittings. Spacer grids which retain 236 fuel rods arranged in a square 16 x 16 array are welded to the guide tubes.

The fuel rods consist of hermetically sealed, prepressurized fuel tubes which contain the ceramic fuel pellets. The fuel pellet diameter is 0.325 in. and pellets are stacked within the fuel tubes to an active core height of 150 in. The fuel tube outside diameter is 0.385 in. and its wall thickness is 0.025 in.

Each of the five guide tubes displaces four fuel rods from the 16 x 16 array. The four guide tubes closest to the assembly perimeter provide channels to guide the control elements over their reactive length of travel. The fifth guide tube is located in the center of the fuel assembly and contains the in-core instrumentation in the instrumented assemblies.

Figure 5.1.4-1
SSCR FUEL DESIGN-STANDARD BURNUP

5.1.4.3

Design Considerations For High Burnup Fuel

In this section the main technical considerations which affect the design of high burnup fuel are discussed. In order to establish a frame of reference, high burnup target values (based on solid pellet geometry) are compared below with values for the standard System 80TM burnups:

	Batch-Average Burnup, GWd/t	Lead Rod Axial- Average Burnup GWd/t	Lead Pellet Burnup GWd/t
UO ₂ -Standard System 80 TM , Solid Fuel Pellets	30.4	38	46.5
UO ₂ -High Burnup, Solid Fuel Pellets	50	62	76

In general, high burnup fuel designs should account for the possibly more severe effects of:

- higher fission product release and fuel rod internal pressure,
- increased oxide and/or crud layer thickness at the cladding outside surface

- additional pellet-cladding interaction duty cycles, and
- increased fuel rod and assembly dimensional changes.

These individual effects are described in more detail in the following subsections.

5.1.4.3.1 Fission Product Release and Rod Internal Pressures

The release of fission products from fuel pellets to the available void volumes within the rod (e.g., plenum, gap and dish volumes) has several detrimental effects on the performance of operating fuel rods which may possibly worsen at higher burnup. These effects are:

- increased rod internal pressure,
- reduced gap conductance, and
- increased availability of chemical ingredients (iodine, cesium) which may cause stress corrosion cracking.

The release of the gaseous fission products xenon and krypton increases the internal pressure established initially by the helium fill gas. In sufficient quantities, these fission products can reduce the heat conductance through the gap between fuel and cladding in even the highly prepressurized PWR fuel rods. A reduced gap conductance could cause additional gas release through an increase in fuel temperatures. The effects can be more severe at high burnup due to:

- a larger fission gas inventory within the fuel,
- a fission gas release rate which might be burnup enhanced, and
- a decrease in available void space for accommodating the mixture of helium and fission gas.

The void space decrease is a direct result of clad creepdown and fission-product swelling. Fuel rod designs having initially larger plena reduce the concern over high internal pressures and degrading gap conductance at high burnups. Gaseous fission product swelling is dependent on the retained fission gas inventory and local temperature in the fuel. More data is needed to properly quantify high burnup fuel swelling and its effect on fuel thermal conductivity and the internal void volume calculation.

A model for the prediction of fission product release enhancement was first developed by the NRC based on high burnup mixed oxide fuel⁽¹⁾. The model could not be verified against high burnup UO_2 data because of the lack of such data. C-E believes the NRC model to be conservative when applied to UO_2 fuel operating at typical PWR heat ratings, and to be excessively conservative for low-temperature operation of non-restructured fuel, for which little open-literature data exist at burnups greater than 20 GWd/t.

A version of C-E's gas release model has been modified for high burnup UO_2 and applied in this study to the calculation of the EOL rod internal pressure. This developmental model has separate release rates and burnup dependencies for fuel operating in five temperature regimes:

- above columnar grain growth temperatures,
- equiaxed to columnar grain growth temperatures, and
- three temperature regimes below the temperature for equiaxed grain growth.

Total gas release and release enhancement rates at high burnups are significantly lower below 1250°C than above this temperature.

The prediction of fission gas release is considered to be the factor which most strongly affects a high burnup fuel rod design. The primary reason for this is the current uncertainties in the industry as to fission gas release enhancement at high burnups. Fission gas release predicted by the developmental model and presented in Figure 5.1.42 shows a marked increase after 20 GWd/t burnup. The end-of-life (EOL) gas release (65 GWd/t average burnup) predicted with this model is about eight times higher than that at 20 GWd/t, whereas the EOL burnup is only 3.25 times higher. As an illustration of the wide difference in gas release models, it may be noted here that with the previously mentioned NRC-model, a EOL gas release about fifteen times higher than the 20 GWd/t value would be predicted.

Figure 5.1.4-2

FISSION GAS RELEASE BEHAVIOR OF 65 GWd/t BURNUP FUEL BASED ON THE
FISSION GAS RELEASE MODEL DEVELOPED FOR THIS STUDY

5.1.4.3.2 Waterside Corrosion of Fuel Rod Cladding

Under normal steady-state operation, current PWRs have not experienced excessive corrosion on the outside surface of fuel rods. The heat fluxes and residence times have been accommodated without serious consequence. However, if exposures and residence times are increased significantly, it is likely that a thicker layer of oxide and crud deposits will develop. The increased insulation may raise the oxide-cladding interface temperature sufficiently to accelerate the corrosion of the Zircaloy cladding.

The in-reactor oxidation of Zircaloy has been described as a function of time and temperature. Separate rate equations have been developed for three distinct phases of corrosion, designated as pretransition, transition, and post-transition⁽²⁾. Kraftwerk Union (KWU) has developed this further by considering the insulating effect of the oxide on enhancement of the corrosion rate⁽³⁾. Some data are available from extended exposure of Zircaloy tubing at the Shippingport PWR⁽⁴⁾ and from the Saxton plutonium experiments⁽⁵⁾.

A joint EPRI/KWU/C-E program has been initiated to develop further the data base for in-reactor corrosion of Zircaloy at typical power reactor conditions. Analytic methods will be developed as part of the joint program to predict oxide thickness and its effective thermal conductivity at higher burnups.

5.1.4.3.3 Pellet-Clad Interaction

Evidence exists through two cycles (~20,000 Mwd/t) that PWR fuel rods of current design are more susceptible, at progressively higher burnups, to failure by a Pellet-Clad-Interaction (PCI) mechanism that is possibly assisted by Stress Corrosion Cracking (SCC). Contributing factors could be deterministic or probabilistic in nature and may include the following:

- higher cladding stresses during power ramps due to the closure of the as-fabricated gap between fuel and cladding by clad creepdown and by pellet swelling and relocation,
- a reduced cladding stress required for failure due to the increased availability of the iodine or cesium corrosive species,
- increased irradiation damage to the cladding,
- increased number of handling events, and
- increased number of fuel reconditioning* events.

Beyond the second irradiation cycle, firm contact generally exists between fuel and cladding at steady-state operating conditions. On a deterministic basis, the propensity for PCI/SCC failure of the cladding adjacent to a local

* Conditioning fuel to higher powers after significant previous operation at lower powers.

pellet-pellet interface site does not appear to increase significantly with additional burnup in conditioned fuel. Further consideration must be given, however, to the larger fission gas inventory available for release during power ramps and to the fission gas swelling at high burnups. A possible increase in the probability of failure sites in each fuel rod at progressively higher burnups must be weighed against the lighter duty experienced on the average by the population of high-burnup fuel beyond two cycles when appropriate fuel management schemes are employed. Power ramp experiments involving single rods and representative fuel assemblies are expected to aid in the establishment of acceptable high-burnup power envelopes and operating guidelines for the safe operation of current and future design fuels relative to PCI/SCC.

5.1.4.3.4 Fuel Assembly Dimensional Changes

Irradiation can cause both stress-free and stress-induced permanent dimensional changes in the components of the fuel assembly. In the SSCR design, these include: axial growth of the fuel rods and control rod guide tubes, fuel rod bowing, and fuel assembly bowing. The fuel assembly design will provide allowances for the growth of fuel rods and guide tubes.

Fuel rod bowing is not expected to be a concern if exposures and cycle lengths are increased. C-E has experienced considerably less bow than has been reported by another PWR vendor⁽⁶⁾. Analyses of second and third cycle rod bow data from one C-E plant of 14 x 14 design indicates that little additional bow would be expected for five cycles of exposure in either the 14 x 14 or 16 x 16 designs. An extensive fuel characterization program is

underway as a joint C-E/EPRI effort (RP-386-1) on the 16 x 16 design in Arkansas Nuclear One, Unit 2, the precursor of System 80TM. Fuel rod and assembly bow data obtained from this program will help qualify the extension of System 80TM to higher burnups.

5.1.4.4 High Burnup Fuel Design Study

5.1.4.4.1 General

A short study was performed to investigate fission gas release and resulting fuel rod internal gas pressures. Using the System 80TM fuel rod design as the starting point, several design options were investigated parametrically which were aimed at reducing internal fuel rod pressures. The two main approaches to reducing fuel rod gas pressure are to provide more fission gas plenum space (longer fuel tubes, shorter stack heights, hollow pellets) or to reduce fission gas release by lowering fuel temperatures (hollow pellets, duplex pellets). These two approaches can be utilized singly or in combination.

The two design approaches are not without inherent problems. An increase in fuel tube length is limited to about 13 inches and would require various modifications of the fuel assembly design and/or of core internals. It also would probably eliminate the possibility of using this fuel in existing light water reactors. Reduction of stack height, hollow and duplex pellets results in even higher rod-average burnups and volumetric heat rates

to maintain the energy yield per fuel rod. In addition, the fabrication and irradiation experience with hollow and duplex pellets is limited relative to that for solid pellets.

5.1.4.4.2 Ground Rules Of Study

Fuel performance calculations are based on the peak power rod for System 80TM and using a single-node calculation at an axial-average Linear Heat Generation Rate (LHGR). Fuel temperature profiles used in the calculation of fission gas release are based on a peak LHGR envelope of 13.4 kW/ft (total heat rating produced in the fuel, cladding, and local coolant) in the lead rod and consider normal operating transients. Fuel depletion effects and the fuel management scheme act to reduce the peak LHGR envelope to about 11.9 kW/ft between 68 and 91 percent of the EOL burnup, and to about 11.3 kW/ft between 91 and 100 percent of the EOL burnup.

Fuel temperatures at high burnups are based on a fixed gap conductance of 2000 Btu/hr-ft²-°F to reflect a saturated fuel-cladding contact situation. The effects of flux depression and a thermal conductivity change due to porosity added by swelling are accounted for in calculating the temperature profile through the fuel.

The rod internal pressure calculation accounts for the initial prepressurized helium fill gas, cladding creepdown and axial growth, fuel swelling induced by fission-product buildup, helium absorption by the fuel, and the differential thermal expansion of the fuel and cladding. Available void volume

regions within the rod for the accommodation of the helium, xenon and krypton gas mixture include the plenum volume, pellet dish and chamfer volume, pellet open porosity, the annular gap volume that is transformed into pellet crack volume by relocated pellet fragments, and the center hole volume present in hollow pellets. The plenum is assumed to contain a standard System 80TM fuel column spring which occupies 0.0244 cu. in. per inch over the 9.68-inch working length. Parametric studies which consider longer fuel rods and/or shorter fuel columns, include the standard spring and a stainless steel support tube in series in the plenum. The support tube has an ID of 0.310 inches and an OD of 0.325 inches, and occupies 0.0075 cu. in. per inch of length.

Solid, hollow, and duplex pellet types are examined. Hollow pellet ID holes sizes of 0.09 in. (8 vol%), 0.13 in. (16 vol%), and 0.16 in. (24 vol%) were investigated. A single duplex geometry was studied having a center region of natural UO_2 (0.711 wt% U-235) and an outer region of highly enriched UO_2 (5.3 wt% U-235). The center region of this duplex design contains 25 percent of the fuel volume. Table 5.1.4.1 compares equivalent burnups associated with the use of the three pellet types, assuming equivalent fissile material inventories in the core.

TABLE 5.1.4-1

COMPARISON OF ENRICHMENT AND BURNUP REQUIREMENTS FOR
VARIOUS UO₂ FUEL PELLET DESIGNS STUDIED

<u>Pellet Type</u>	<u>Central Hole ID, in.</u>	<u>Fuel Stack Density, fraction of theoretical^a</u>	<u>Fuel Length, in.</u>	<u>Peak Rod Axial-Average LHGR, kW/ft</u>	<u>Peak Rod Axial-Average EOC Burnup, Gwd/t</u>
Solid	0.0	0.918	150	13.4	62.0
			145	13.9	64.1
			140	14.4	66.4
Hollow	0.09	0.849	150	13.4	67.1
			145	13.9	69.4
			140	14.4	71.9
Hollow	0.13	0.780	150	13.4	73.0
			145	13.9	75.5
			140	14.4	78.2
Hollow	0.16	0.708	150	13.4	80.4
			145	13.9	83.2
			140	14.4	86.1
Duplex	0.0	0.918	150	13.4	62.0 (37.8/70.0) ^b
			145	13.9	64.1 (39.2/72.4) ^b
			140	14.4	66.4 (40.8/74.9) ^b

^a Accounts for fuel fabrication porosity, pellet dishing and chamfers, and central hole if present.

^b (inner zone/outer zone)

5.1.4.4.3 Fuel Rod Internal Pressure Study

The results of parametric analyses are presented in this section for solid, hollow and duplex pellets as well as for variable rod lengths and/or fuel column lengths which provide additional plenum volume for accommodation of the internal pressure in the System 80TM design.

Several design changes to System 80TM may be considered which would enable the use of longer fuel rods. These are listed below:

- eliminate the extension tubes which protrude from the core alignment plate, and add about 6-7 inches to the fuel assembly and fuel rods;
- substitute a stainless steel guide tube at the center of the assembly, to provide assembly holddown, and add about 3-4 inches to the fuel rods; and
- redesign the bottom plate for bottom lockdown, and add about 5-6 inches to the fuel rods.

Any of these three options could be used singly, or option one could be combined with either option two (adding 9-11 inches to the fuel rod) or option three (adding 11-13 inches to the fuel rod). Options two and three would not be used together. Option two is the only option which is expected to change the core physics and could result in the need for slightly higher enrichments.

Figure 5.1.4-3 shows the sensitivity of the calculated EOL rod internal pressure to increases in the rod length and/or decreases in the fuel column length for the use of solid, duplex or hollow pellet types. Hollow pellets are shown to provide the lowest pressures due to their lower fuel temperatures and added plenum volume provided by the central hole. As more plenum volume is added, the pressures tend to converge for all designs to values between 1500 and 2000 psia.

For the current C-E design objective that EOL internal pressures not exceed system pressure, only those designs which fall in the shaded area below 2200 psia in Figure 5.1.4-3 would be acceptable, without considering secondary design changes (discussed later) which could reduce the internal pressure to a lesser extent. If a no-liftoff* criterion is applied, then designs which exceed 2200 psia may be acceptable.

Table 5.1.4-2 shows the sensitivity of the rod internal pressure calculation to a ± 5 Gwd/t perturbation in the lead rod burnup which would result from a ± 8 percent uncertainty in the capacity factor assumption. The pressures

* A no-liftoff criterion generally states that the internal pressure of the peak power rod can exceed system pressure, but will be limited to a value below that which could cause the diametral gap to increase due to outward cladding creep during steady state operation. Such a criterion has been developed by Westinghouse (7) and has been accepted by the NRC (8) for use in licensing applications.

Figure 5.1.4-3

FUEL ROD EOL INTERNAL PRESSURE SENSITIVITIES TO PELLET DESIGN, ROD LENGTH, AND FUEL COLUMN LENGTH
USING A GAS RELEASE MODEL FOR HIGH BURNUP IN UO_2 FUEL

listed in the table for the base burnup are taken from the curves of Figure 5.1.4-3. The sensitivity values listed are roughly proportional to the EOL absolute pressures. The larger the plenum volume, the lower the EOL pressure and its sensitivity to the burnup perturbation.

The rod internal pressures from Figure 5.1.4-3 and Table 5.1.4-2, derived with the high burnup release model developed for this study, are compared in Table 5.1.4-3 with internal pressures obtained using the NRC gas release enhancement model. The study model values are about 50 to 65% of the NRC-model values. With the use of the NRC model, a design utilizing hollow pellets with both the shortened fuel length and increased rod length would be the only reasonable option, and application of a no-liftoff criterion would be necessary. Operation at reduced LHGRs would not change this scenario markedly, since the NRC model does not consider reduced fission gas release for operation at fuel temperatures lower than 1200°C.

Additional design measures are available for consideration which can help to provide secondary reductions in the rod internal pressure at EOL. These include a reduction in initial fill pressure, the use of a lighter-duty plenum spring, or the insertion of a burst chamber in the fuel rod plenum. A reduced fill pressure will result in greater creepdown, a greater propensity for clad collapse, and a larger ratio of fission gas to helium atoms which can degrade the gap conductance. Clad collapse may not be a significant problem for current fuels since such fuels are stable to in-reactor densification and are less likely to produce the large axial gaps in the fuel column observed with early densifying fuels. The burst chamber has

- the single advantage of permitting higher prepressurization at BOL while providing added volume for the accommodation of higher pressures if needed in the peak power rods.

The results of the study clearly favor the use of hollow pellet fuel over either solid or duplex pellet fuel in high-burnup designs, based on lower fuel temperatures and EOL rod internal pressure in the peak power rod. The experience with hollow pellet fuel relative to that achieved with solid pellets, however, is very limited. More data also is needed to assure that a local filling of the central hole with pellet fragments from higher in the fuel stack does not occur. This event would result in higher peak-pellet LHGRs and possibly more severe local fuel temperatures and local PCI performance. A licensing penalty for this situation could be imposed which might make the solid or duplex designs more attractive.

TABLE 5.1.4-2

LEAD ROD INTERNAL PRESSURES AT EOL USING A
GAS RELEASE MODEL FOR HIGH BURNUP UO_2

<u>Fuel Type</u>	<u>Fuel Rod</u>	<u>Fuel Column</u>	<u>Rod Internal Pressure, psia^b</u>	
	<u>Length Change,</u> <u>in.</u>	<u>Length Change,</u> <u>in.</u>	<u>EOL^a</u>	<u>+ 5 Gwd/t</u>
Solid	Standard	Standard	5600	+850
		-10	3100	+350
	+13	Standard	2400	+250
		-10	2100	+170
Hollow (0.16 in. ID)	Standard	Standard	2100	+120
		-10	1900	+100
	+13	Standard	1700	+80
		-10	1600	+70
Duplex	Standard	Standard	4600	+650
		-10	2900	+300
	+13	Standard	2100	+180
		-10	1900	+150

^a62 Gwd/t in Standard Design with Solid Pellet

^bResults should not be extrapolated due to nonlinearities in the pressure calculation.

TABLE 5.1.4-3

COMPARISON OF FUEL ROD INTERNAL PRESSURES FOR THE USE OF THE C-E
GAS RELEASE MODELS FOR UO₂

<u>Pellet Type</u>	<u>Fuel Rod Length Change, in.</u>	<u>Fuel Column Length Change, in.</u>	<u>Lead Rod Internal Pressure, psia</u>	
			<u>Study Model Gas Release</u>	<u>NRC Gas Release</u>
Solid	Standard	Standard	5600	10,000
		-10	3100	5700
	+13	Standard	2400	4000
		-10	2100	3500
Hollow (0.16 in. ID)	Standard	Standard	2100	4300
		-10	1900	3800
	+13	Standard	1700	3200
		-10	1600	3000
Duplex	Standard	Standard	4600	9700
		-10	2900	5500
	+13	Standard	2100	3900
		-10	1900	3400

Standard Fuel Rod Length. If it is desirable to maintain the standard fuel rod length and fuel column length, then the best available design option, using the design sensitivity curves of Figure 5.1.4-3, is to use hollow pellet fuel and to size the central hole ID to meet the desired maximum pressure criterion. A hole size of 0.11 inches (11 vol%) or greater likely would satisfy a no-liftoff criterion, and a hole size of 0.15 inches (21 vol%) or greater would satisfy the criterion that the internal pressure not exceed the system pressure. If it is acceptable to reduce the fuel stack length by about 12 inches (8 percent), with the accompanying increases in LHGR, EOL burnups and starting enrichment, then it likely would be possible to meet a no-liftoff criterion with the use of either the solid or duplex pellet designs. The design options for a standard length fuel rod are summarized in Table 5.1.4-4.

Increased Fuel Rod Length. Increases in fuel rod length provide considerably more flexibility in meeting rod internal pressure criteria without resorting to significant reductions in the fuel column length. Table 5.1.4-5 presents a few of the options available with the use of solid, hollow or duplex fuel designs. A comparison of these design options to those available for the use of the standard rod length in the previous table results in the following factors which favor the longer rod:

- A no-liftoff criterion likely can be met with little or no decrease in the standard fuel column length if solid or duplex pellets are utilized.

- The system pressure criterion can be met easily using hollow pellets with minimum hole sizes.
- For the maximum rod length change possible (about 13 inches), the system pressure criterion can be met with a minimal change in fuel column length (about three inches), if solid pellets are utilized.

However, as was pointed out before, longer fuel rods will require modifications which are not inconsequential to the fuel assembly and core internals configuration.

The fuel rod internal pressure study indicates that a number of design approaches are available for developing fuel rod designs to accommodate the fission gas release at high burnups. However, the study also points out that the present uncertainties associated with accurately predicting the amount of gas release significantly affects the evaluation of how drastic a change from the standard System 80TM fuel rod design would be required for high burnup. Because of these uncertainties it is considered premature to settle at the present time on a specific design. Rather, the results of additional in-pile testing are required before a more substantive choice for a high burnup fuel rod design can be made.

TABLE 5.1.4-4

UO₂ DESIGN OPTIONS WITH STANDARD ROD LENGTH

<u>Pellet Type</u>	<u>Hole ID, in.</u>	<u>Fuel Column Length Change, in.</u>	<u>Rod Internal Pressure, psia</u>	<u>Rod Internal Pressure Design Criterion</u>
Hollow	> 0.11 (11 vol%)	Standard	< 2900	No liftoff.
Hollow	> 0.15 (21 vol%)	Standard	< 2200	System pressure.
Solid	0.0	-12 (8%)	~ 2900	No liftoff.
Duplex	0.0	-12 (8%)	~ 2700	No liftoff.
Hollow	> 0.11 (11 vvol%)	-12 (8%)	< 2200	System pressure.

TABLE 5.1.4-5

UO₂ DESIGN OPTIONS WITH INCREASED FUEL ROD LENGTH

<u>Pellet Type</u>	<u>Hole ID, in.</u>	<u>Fuel Column Length Change, in.</u>	<u>Fuel Rod Length Change, in.</u>	<u>Rod Length Change Design Option Employed</u>	<u>Rod Internal Pressure, psia</u>	<u>Rod Internal Pressure Design Criterion</u>
Hollow	> 0.12 (14 vol%)	None	6-7	(1)	< 2200	System Pressure.
Duplex	0.0	None	6-7	(1)	~ 2700	No liftoff.
Solid	0.0	-3 (2%)	6-7	(1)	~ 2900	No liftoff.
Solid	0.0	-12 (8%)	6-7	(1)	~ 2200	System Pressure.
<hr/>						
Hollow	0.08 (6 vol%)	None	10-12	(1) and (2) or (1) and (3)	< 2200	System Pressure.
Duplex	0.0	None	10-12	"	< 2200	System Pressure.
Solid	0.0	None	10-12	"	~ 2600	No liftoff.
Solid	0.0	-3 (2%)	10-12	"	~ 2200	System Pressure.

Option (1) - Eliminate extension tube on alignment plate.

Option (2) - Holddown achieved by center s-s guide tube.

Option (3) - Assembly lockdown through bottom plate redesign.

REFERENCES FOR SECTION 5.1.4

"Fuel Design"

1. R. O. Meyer et al, "Fission Gas Release From Fuel at High Burnup", NUREG-0418, March 1978.
2. E. Hillner, "The Corrosion of Zirconium-Base Alloys, An Overview", Symposium on Zirconium in the Nuclear Industry, ASTM, Quebec City, Canada, August 1976.
3. H. Stehle et al, "External Corrosion in PWRs", Nuclear Engineering and Design, Volume 33, Pages 155-160, 1976.
4. E. Hillner, "Corrosion and Hydriding Performance of Zircaloy Tubing After Extended Exposure in the Shippingport Pressurized Water Reactor", Zirconium in Nuclear Applications, ASTM-STP-551, Pages 449-462, 1974.
5. R. C. Goodspeed, "Saxton Plutonium Project, Quarterly Progress Report for the Period Ending June 30, 1973", WCAP-3385-36, July 1973.
6. M. G. Andrews, "Combustion Engineering's Fuel Performance Experience in Operating PWRs", Paper Presented at the ANS Topical Meeting on Water Reactor Fuel Performance, May 9-11, 1977, St. Charles, Illinois.
7. D. H. Risher, et. al., "Safety Analysis for the Revised Fuel Rod Internal Pressure Design Basis", Westinghouse Report WCAP-8963 (Proprietary), November 1976.
8. Core Performance Branch, US-NRC, "Safety Evaluation of the Westinghouse Topical Report - Safety Analysis for the Revised Fuel Rod Internal Pressure Design Basis (WCAP-8963)", May 1978.

The Reactor Coolant System of the SSCR is essentially the same as for a System 80TM PWR, with the exception of an additional shaft seal for the reactor coolant pumps which will be described below.

In the following sections a brief description of the reactor coolant system is given. A more detailed description of the reactor coolant system is available in CESSAR.

5.1.5.1 Summary System Description

The Reactor Coolant System block diagram is shown in Figure 5.1.5-1. The major components of the system are a reactor vessel, two parallel heat transfer loops, each containing one steam generator and two reactor coolant pumps, and a pressurizer connected to one of the reactor vessel outlet pipes. All components are located inside the Containment building. Their physical arrangement is shown in Figure 5.1.5-2 Sheet 1 and 2. The two steam generators and the four reactor coolant pumps are symmetrically located on opposite sides of the reactor vessel.

During normal operation, the reactor coolant is circulated through the reactor vessel and steam generators by the reactor coolant pumps. The reactor coolant is heated as it passes through the reactor vessel by energy produced by the fissioning fuel in the core and is cooled in the steam generators as it gives up heat to the secondary system. Feedwater entering the shell side of the steam generators absorbs the heat from the primary system forming steam. The reactor coolant also serves as a neutron moderator

Figure 5.1.5-1
REACTOR COOLANT SYSTEM BLOCK DIAGRAM

Figure 5.1.5-2
REACTOR COOLANT SYSTEM ARRANGEMENT
(SHEET 1 - PLAN)

Figure 5.1.5-2
 REACTOR COOLANT SYSTEM ARRANGEMENT
 (SHEET 2 - ELEVATION)

in the core, containing D_2O-H_2O mixture. Except for some local boiling in the hottest channels in the core, the reactor coolant is maintained in a subcooled condition by maintaining a high system pressure.

The Reactor Coolant System pressure is maintained and controlled through the use of a pressurizer where steam and water are maintained in thermal equilibrium. During full load operation, the pressurizer volume is almost evenly divided between saturated water and saturated steam. Steam is formed by energizing immersion heaters in the pressurizer, or is condensed by a subcooled pressurizer spray as necessary to maintain operating pressure and limit pressure variations due to plant transients.

Overpressure protection for the Reactor Coolant System is provided by four spring-loaded ASME Code Safety Valves. These valves discharge to the reactor coolant drain tank where the steam is released under water to be condensed and cooled. If the steam discharge exceeds the capacity of the tank, the tank is relieved to the Containment via a rupture disc installed in the tank.

5.1.5.2 Summary Component Description

5.1.5.2.1 Reactor Vessel

The reactor vessel contains the fuel assemblies, the control rods and other internals necessary for support and flow direction (for a more detailed description of the reactor vessel see Section 5.1.3 - CORE DESIGN).

5.1.5.2.2 Steam Generators

Each of the two steam generators is a vertical U-tube heat exchanger with an integral economizer which operates with the reactor coolant on the tube side and secondary coolant on the shell side. Hot reactor coolant from the reactor vessel enters the steam generator through the inlet nozzle in the primary head. From here it flows through the U-tubes where it gives up heat to the secondary coolant to the outlet side of the primary head where the flow splits and leaves through the two outlet nozzles. A vertical divider plate separates the inlet and outlet plenums of the primary head.

5.1.5.2.3 Reactor Coolant Pumps

The four identical reactor coolant pumps are vertical single-stage, motor-driven centrifugal pumps designed to circulate the reactor coolant at the rate required for power operation. They are also used during plant startup to heat the reactor coolant.

The pump drive shafts which penetrate the pump housing pressure boundary are sealed by hydrodynamic shaft sealing systems. For the System 80TM pumps, the sealing system consists of three identical hydrodynamic seals of the mechanical face type. The three seals are arranged in series with the outermost seal acting as back-up. Controlled by-pass leakage flow provides equal pressure differential across the seals as well as cooling for the seal faces. The face leakage from the outermost seal is collected and piped to the Reactor Drain Tank.

For SSCR application, an additional vapor seal assembly is provided outboard of the third seal to eliminate or greatly reduce escape of any gaseous leakage into the containment atmosphere. The seal assembly consists of a chrome-plated stainless steel shaft runner which mates with a stationary carbon steel seal ring. This seal ring is attached to the conventional seal housing which requires some minor modification. Both the shaft runner and the stationary ring are wearing elements which require periodic inspection and eventual replacement. Supply of these vapor seals is within current technology and vapor seals of essentially the same design have been provided for D₂O-pumps in Canadian Heavy Water Reactor power stations.

Because the D₂O concentration in the coolant of the SSCR changes from about 60 wt% at the beginning-of-cycle (BOC) to 2 wt% at the end-of-cycle (EOC), the fluid density changes also and is about 6% higher at BOC and about 0.2% higher at EOC than the fluid density of System 80TM. The motors of the reactor coolant pumps operate at constant speed; thus, their volumetric coolant flow remains constant. Due to the higher fluid density of the SSCR, the mass flow rate will be 6% higher at BOC than for System 80TM and 0.2% at EOC. The higher mass flow at BOC is beneficial for the thermal-hydraulic design of the SSCR and can be tolerated with respect to fuel assembly uplift. At EOC the SSCR mass flow is essentially the same as System 80TM. The higher coolant mass flow requires proportionately higher pumping power. The size of the System 80TM motors is large enough to accommodate the 6% pumping power increase at BOC for the SSCR.

5.1.5.2.4 Reactor Coolant System Piping

Each of the heat transfer loops contains five pipe assemblies; one 42 inch internal diameter pipe assembly between the reactor vessel outlet nozzle and the steam generator inlet nozzle, two 30-inch internal diameter pipe assemblies from the two steam generator outlet nozzles to the reactor coolant pump suction nozzles, and two 30-inch internal diameter pipe assemblies from the pump discharge nozzles to the reactor vessel inlet nozzles. These pipe assemblies are referred to as the hot leg, the suction leg, and the cold leg, respectively. The other major section of reactor coolant piping, the surge line, is a 12-inch schedule 160 pipe assembly between the pressurizer and the hot leg in Loop 1. A typical arrangement of piping is shown in Figure 5.1.5-2.

5.1.5.2.5 Pressurizer

The pressurizer is designed to maintain Reactor Coolant System operating pressure and compensate for changes in reactor coolant volume during load changes. The pressurizer controls the Reactor Coolant System pressure by maintaining the temperature of the pressurizer liquid at the saturation temperature corresponding to the desired system pressure. Pressurizer temperature control is maintained by heaters and spray. The pressurizer heaters supply energy to heat the pressurizer liquid to the required temperature and to offset heat losses to ambient. The pressurizer spray functions to cool the pressurizer by injecting cold leg water into the steam space if the pressurizer pressure should, during transients, increase above its desired pressure.

5.1.5.2.6 Primary Safety Valves

Four primary safety valves, located on the top head of the pressurizer, provide overpressure protection for the Reactor Coolant System. They are totally enclosed, backpressure compensated, spring-loaded safety valves meeting ASME Code requirements. The primary safety valves discharge through the relief line piping into the reactor coolant drain tank.

5.1.5.2.7 Pressurizer Spray Valves (Actuator-Operated Throttling Valves)

The position of these two air operated throttling spray valves on loss of actuating signal (failure position) is failed closed. This position was selected to ensure safe and efficient operation of the system and plant. The auxiliary spray of the Chemical and Volume Control System provides a backup to the pressurizer spray. Valve position indication is provided at the main control panel.

5.1.6 PLANT SYSTEMS FOR THE MANAGEMENT OF COOLANT CONCENTRATIONS

5.1.6.1 Introduction

The major difference in Plant Systems between a PWR of the SWESSAR/CESSAR design and the SSCR is in the systems used for management of the D_2O and boron concentration of the reactor coolant. These systems are:

- a) The Chemical and Volume Control System (CVCS)
- b) The Boron Recovery System (BRS)
- c) The Primary Grade Water System (PGWS).

Furthermore, a new system has been added to the SSCR: The D_2O Upgrader Facility.

The difference in plant systems stems from additional functional requirements for spectral shift control operation. In general terms, spectral shift control requires that the D_2O concentration in the coolant at the beginning of cycle be about 60 wt% and be gradually reduced throughout the cycle to an end-of-cycle concentration of 2 wt%. Also, for normal power operation, boron is not required in the coolant and its concentration is kept below 10 ppm. Boration is required, however, for shutdown to cold condition or for refueling. The plant systems have been modified for spectral shift operation in such a way that the plant retains the capability to operate in the conventional boron poison control mode.

5.1.6.3.2 SSCR Design

In the SSCR design, processing of the RCS letdown water in the D₂O Upgrader Facility is required to provide water of 85 wt% D₂O concentration; this is sufficient to achieve a 60 wt% D₂O concentration in the RCS at the beginning of a core cycle. Additional tankage has been provided to facilitate the processing and storage of reactor coolant. Figure 5.1.6-4 provides a simplified diagram of the SSCR PGWS.

Eight Letdown Tanks (LDT), two Light Water Storage Tanks (LWST), and one Heavy Water Storage Tank (HWST) are provided in the system. The LDT's store deborated letdown from the BRS at varying D₂O concentrations. The letdown stored in these tanks may be used as: 1) feed material for heavy water dilution of the RCS if the D₂O concentration in the tanks is low (i.e. letdown from the previous cycle with a D₂O concentration less than approximately 6 wt%), 2) feed material for processing by the Upgrader Facility if the D₂O concentration is relatively high (i.e. letdown greater than approximately 6 wt%); or 3) feed material for boration and to compensate for liquid contraction during a cold shutdown (i.e. letdown with a D₂O concentration approximately equal to the existing RCS concentration at time of shutdown). The LWST's and the HWST store the Upgrader overheads (0.5 wt% D₂O), and bottoms (85 wt% D₂O), respectively, for reuse as feed to the RCS. The approximate capacities of the tanks are:

Letdown Tanks (8):	35,000 gallons each
Light Water Storage Tanks (2):	100,000 gallons each
Heavy Water Storage Tank (1):	80,000 gallons

The following sections provide summary descriptions of the systems included in the SWESSAR-CESSAR design and of the fundamental differences in the SSCR. Section 5.1.12 (COOLANT MANAGEMENT OPERATIONS) provides a detailed description of the integrated operation of the coolant management systems of the SSCR to accomplish typical plant evolutions. Simplified composite flow diagrams of the subject plant systems of the SWESSAR/CESSAR PWR and of the SSCR and shown in Figures 5.1.6-1 and 5.1.6-2.

5.1.6.2 Chemical and Volume Control System (CVCS)

The CVCS in the SWESSAR-CESSAR PWR and in the SSCR remain functionally identical except that a Boric Acid Tank and a Mixing Tank are provided in the SSCR design where only a Boric Acid Tank is provided in the SWESSAR-CESSAR design.

In the SWESSAR-CESSAR design the Boric Acid Tank stores 4 wt percent boric acid. During boration operations, this boric acid is used directly from the Boric Acid Tank or is blended with Primary Grade Water during feed and bleed or makeup operations.

In the SSCR design, shutdown boration and volume contraction makeup must be accomplished without changing the D_2O concentration in the Reactor Coolant System (RCS) existing at the time of shutdown. Because two parameters (D_2O concentration and boron concentration) must be established in the RCS, the CVCS design has been modified for the SSCR so that the volume of coolant

Figure 5.1.6-1
COMPOSITE OF THE BORON
RECOVERY, PRIMARY GRADE
WATER AND CHEMICAL AND
VOLUME CONTROL SYSTEMS

SWESSAR/CESSAR PWR

necessary to borate the RCS and make up for volume contraction during cooldown can be premixed in the CVCS. This mixing capability simplifies the operations required to establish two parameters concurrently in the RCS. The Boric Acid Tank stores the 12 wt percent boric acid bottoms flow from the Boric Acid/D₂O Separation Unit (BDSU) (see Section 5.1.6.4.2). The boric acid concentration has been increased from 4 to 12 wt percent to minimize the volume of boric acid solution needed for boration and hence the D₂O inventory required. Boric acid from the Boric Acid Tank is mixed in the Mixing Tank with primary grade water from the Letdown Tanks, a Light Water Storage Tank and/or the Heavy Water Storage Tank, as necessary to achieve the required volume of coolant at the proper boron and D₂O concentration for boration and cooldown. The capacity of the Boric Acid Tank and of the Mixing Tank is about 35,000 gallons each.

5.1.6.3 Primary Grade Water System (PGWS)

5.1.6.3.1 SWESSAR-CESSAR PWR Design

Figure 5.1.6-3 provides a simplified representation of the SWESSAR-CESSAR PGWS. In this design, the Primary Grade Water Storage Tanks (PGWST's) via the Primary Grade Water Transfer Pumps provide makeup to the RCS when required for feed and bleed during boration, deboration, or makeup for normal leakage. The PGWST's are normally replenished from the distillate produced by the Boron Recovery System (BRS).

Figure 5.1.6-3
SWESSAR-CESSAR PWR PRIMARY GRADE WATER SYSTEM

Figure 5.1.6-4
SSCR PRIMARY GRADE WATER SYSTEM

5.1.6.3.2 SSCR Design

Figure 5.1.6-4 provides a simplified representation of the SSCR PGWS. In this design, additional processing of the letdown by the D_2O upgrader is required to separate D_2O from H_2O to obtain high D_2O concentration coolant for the start of the next operating cycle. As a result, additional tankage has been provided in the SSCR PGWS design.

5.1.6.4 Boron Recovery System (BRS)

5.1.6.4.1 SWESSAR-CESSAR PWR DESIGN

In the SWESSAR-CESSAR design, the BRS processes reactor coolant letdown to recover primary grade water and boric acid for reuse or disposal. During normal operation the letdown entering the BRS is produced by the feed and bleed operations necessary to establish and maintain the boron concentration for chemical shim in the reactor coolant at the desired level.

Figure 5.1.6-5 provides a simplified representation of the SWESSAR-CESSAR BRS. Letdown entering the BRS is stored in one of two Boron Recovery Tanks. The letdown is then processed by the Boron Evaporator where primary grade water (distillate) is separated from the boric acid (bottoms). The distillate is collected in a Boron Test Tank where it is transferred to the Primary Grade Water System for reuse as RCS feed. The Boron Evaporator bottoms (boric acid) are pumped to the Boric Acid Tank in the CVCS where it is stored for subsequent boration operations, or transferred to the Radioactive Waste Management System for disposal.

Figure 5.1.6-5
SWESSAR-CESSAR PWR BORON RECOVERY SYSTEM

In the SSCR design deboration of the RCS is required to maintain adequate shutdown margins during cold shutdown and refueling; however, boron is not used as a chemical shim during normal operation. In addition, because the heavy water concentration of the RCS is decreased over the fuel cycle, an Upgrader Facility is provided in the SSCR design to separate heavy water from light water for reuse in the next fuel cycle (see Reference Section 5.1.6.5). The Upgrader requires boron free feed. Therefore, the BRS has been designed: 1) to provide the means to remove boron from the RCS to achieve criticality after cold shutdowns; 2) to remove any trace amounts of boron from the RCS letdown during normal operation so that the letdown can be processed by the Upgrader Facility; and 3) to salvage the heavy water-light water mixture from the boric acid solution in the evaporator bottoms. Figure 5.1.6-6 provides a simplified representation of the SSCR BRS.

Deboration Following Cold Shutdown. In the SWESSAR-CESSAR design, the capacity of the boron evaporator is smaller than the capacity of the charging pumps in the Chemical and Volume Control System (CVCS) and deboration of the RCS is accomplished by a feed and bleed process with sufficient makeup inventory of primary grade water for feed and sufficient storage volume for letdown in the BRS. Owing to the high cost of D_2O and the cost of processing D_2O by the Upgrader Facility, this method is economically impractical for the SSCR design. As a result, the BRS has been modified by increasing the capacity of the boron separators to the capacity of the charging pumps so that "on line", high capacity deboration is possible. To accomplish "on line" deboration following plant heatup, reactor coolant is letdown from

Figure 5.1.6-6

SSCR BORON RECOVERY SYSTEM

the CVCS at a maximum letdown rate of 132 gpm. This letdown is pumped via the Deborating Surge Tank to three Boron Evaporators (capacity of 45 gpm each) where the boric acid is separated from the letdown. The distillate from the Boron Evaporators is then pumped back to the RCS from a Boron Test Tank via the Volume Control Tank and Charging Pumps in the CVCS. Recycling the RCS letdown as makeup feed minimizes the amount of excess D_2O inventory required as well as minimizes the amount of letdown to be processed by the D_2O Upgrader.

Deborating During Normal Operation. During normal operation, the D_2O concentration of the RCS is decreased by the normal reactor coolant system feed and bleed process. Letdown of relatively high D_2O concentration which occurs during the early portion of the fuel cycle is processed by the Upgrader Facility to separate heavy water from light water for use as fill material for the next cycle start-up. As stated above, this letdown must be free of boron prior to processing by the Upgrader Facility; therefore, the BRS has been designed to remove the boron remaining in the RCS following deboration after cold shutdown: Letdown, is directed from the CVCS to one of the two Holdup Tanks (about 35,000 gallons each). This letdown is then directed to a Boron Evaporator, and/or Boron Demineralizers, (depending on the boron concentration) where the remaining boron is removed from the letdown. The letdown is then transferred to a Letdown Tank in the PGWS for use as either RCS dilution feed (letdown D_2O concentration less than approximately 6 wt%) or feed to the Upgrader Facility (letdown D_2O concentration more than approximately 6 wt%). The Boron Evaporators produce a bottoms flow with 21,000 ppm boron (12 wt% boric acid) and an overhead (distillate) flow with 10 ppm boron maximum. The Boron Demineralizers reduce the boron concentration to less than 1 ppm.

Deboriation of RCS Flush Water Generated During Refueling. As described in more detail in Section 5.1.11, to reduce the tritium concentration in the RCS prior to refueling operations, the RCS is partially drained down to the hot leg centerlines and is then flushed with water (less than approximately .06 wt% D_2O) from the Refueling Water Storage Tank (RWST). The BRS has been modified to include two tanks for these operations: 1) RCS Loop Drain Tank (RCSLDT) and 2) Reactor Vessel Flush Tank (RVFT). The RCSLDT (approximately 80,000 gallons) stores the coolant drained from the RCS; the RVFT (approximately 90,000 gallons) stores the water volume produced by the reactor vessel flushing operations. Following refueling, these volumes are processed by the Boron Evaporators, and the resulting distillate is transferred to the PGWS for storage.

Extraction of Heavy Water-Light Water Mixture From Boric Acid Solution.

The Boron Evaporators in the BRS are designed to produce bottoms at 12 wt% boric acid. In order to salvage the heavy water-light water mixture contained in these bottoms, a Boric Acid/ D_2O Separation Unit (BDSU) has been added to the BRS. The primary component in the BDSU is an atmospheric evaporator which receives the bottoms from the Boron Evaporators and slowly boils them to dryness. The BDSU is designed for a capacity of 1150 gallons/day. The BDSU distillate is transferred to a PGWS Letdown Tank for reuse as RCS dilution feed or processing by the Upgrader. The BDSU bottoms are then mixed with light water from a PGWS Light Water Storage Tank (0.5 wt% D_2O max) until a 12 wt% boric acid solution is achieved. This solution is then transferred to the Boric Acid Tank in the CVCS for reuse during boration operations. This system has also been designed to allow the boric acid bottoms to be transferred to the Radioactive Waste Management Systems for ultimate disposal, if necessary (e.g., radiologically unacceptable for reuse).

5.1.6.5 D₂O Upgrader Facility

The D₂O Upgrader Facility is a new system which has been added to the SSCR design. The facility uses a vacuum distillation process to separate heavy water from light water. A detailed description of the facility is provided in Sections 5.1.2.2 (D₂O Upgrader Facility) and 5.1.8 (D₂O UPGRADER DESIGN).

5.1.6.5.1 Upgrader Processing of RCS Letdown During Normal Operation

During normal operation, the Upgrader receives feed from the LDT's of the PGWS. The overhead (0.5 wt% D₂O) and bottoms (85 wt% D₂O) from the Upgrader are transferred to the Light and Heavy Water Storage Tanks, respectively, where it is stored for reuse as RCS feed.

5.1.6.5.2 Upgrader Processing of Reactor Vessel Flush Volumes

Prior to refueling, the reactor vessel is flushed to the Reactor Vessel Flush Tank (RVFT) using water from the Refueling Water Storage Tank to reduce the tritium levels in the RCS. Prior to the next refueling the flush volume taken from the RWST must be replenished with light water containing a lower than normal concentration of D₂O. To accomplish this, the volume of the RVFT (at approximately 0.5 wt% D₂O) is deborated by the BRS and transferred to a Light Water Storage Tank (LWST). Feed from the LWST is processed by the upgrader in an operational mode which is different from normal operation yielding an overhead concentration of 0.1 wt% D₂O, and a bottoms concentration of approximately 17 wt%. The Upgrader overheads are transferred to the RWST until the reactor vessel flush volume in the RWST is replenished. The bottoms are transferred to a Letdown Tank before being

reused as RCS feed or being processed by the upgrader again to achieve 0.5 wt% D₂O overhead and 85 wt% D₂O bottoms for storage in the Light Water and Heavy Water Storage Tanks, respectively.

5.1.7 MODIFICATIONS TO BALANCE-OF-PLANT SYSTEMS

The following Balance-Of-Plant (BOP) Systems which are described in SWESSAR/CESSAR require varying degrees of modifications for spectral shift operation:

1. Auxiliary Steam and Condensate System
2. Annulus Building Ventilation System
3. Demineralized Water Make-Up System
4. Fire Protection System
5. Radioactive Liquid Waste System
6. Reactor Plant Chilled Water System
7. Reactor Plant Component Cooling Water System
8. Reactor Plant Sampling System
10. Turbine Plant Component Cooling Washer System
11. Turbine Plant Service Water System
12. Vents and Drains (gaseous and activated) Systems
13. Spent Fuel Pool

The modifications to these systems are described in the following subsections 5.1.7.1 through 5.1.7.13.

The Auxiliary Steam and Condensate System is essentially the same as for a conventional PWR (SWESSAR/CESSAR Section 10.4.12) with the following addition:

An Upgrader Auxiliary Condensate Receiver Tank will be installed in the Upgrader Facility with two full capacity (250 gpm) pumps to return the auxiliary condensate to the cycle.

The auxiliary boiler capacity has been increased a total of 100,000 lb/hr, from about 230,000 lb/hr to 330,000 lb/hr. This increase in capacity is necessary, because the capacity of the Boron Evaporators for the SSCR design has been significantly increased (from about 25 gpm to 135 gpm) to allow rapid on-line deboration during plant start-up. Because nuclear steam is not available at this time, increased capacity is required from the auxiliary boilers. The rapid deboration of the RCS is only required during plant start-up. Therefore, the additional auxiliary boiler capacity provided for the Boron Evaporators is also available during the remaining portions of a power cycle. The steam requirements of the D₂O upgrader are also about 100,000 lb/hr. Thus, the increased auxiliary boiler capacity can be used to allow Upgrader operation during a plant outage when the Boron Evaporators are not in use. In the event that it is desirable to operate both the Upgrader and the Boron Evaporators simultaneously with the plant out of service (i.e., during start-up) other auxiliary steam system loads (e.g., radwaste evaporators, building heating) must be reduced to provide the required capacity.

5.1.7.2 Annulus Building Ventilation System

The Annulus Building Ventilation System is the same as a conventional PWR (see SWESSAR/CESSAR, Section 9.4.2) with the following modification: Cubicles containing components containing heavy water are isolated during normal operation and are provided with unit coolers consisting of fan, cooling coil, and ductwork, to recirculate air within the cubicle to maintain component temperatures within design limits.

5.1.7.3 Demineralized Water Makeup System

The Demineralized Water Makeup System remains the same as a conventional PWR (see SWESSAR/CESSAR Section 9.2.2) except that the demineralized water supply header has been provided with a new connection to the Radioactive Liquid Waste System. This connection is used as a backup supply to the seal and flush water header (See Section 5.1.7.5).

5.1.7.4 Fire Protection System

The Fire Protection System is the same as a conventional PWR (see SWESSAR/CESSAR, Section 9.5.1) with the addition of a branch line containing a post indication valve. This addition supplies water for the D₂O Upgrader Fire Protection Water System. In addition, hose stations would be provided in the Solid Waste and Decontamination Building Extension.

5.1.7.5 Radioactive Liquid Waste System

The Radioactive Liquid Waste System (RLWS) remains essentially the same as a conventional PWR (see SWESSAR, Section 11.2), except the Waste Test Tanks (containing the waste evaporator distillate) and their associated pumps have been modified to supply seal and/or flush water for the waste management system components formerly supplied by the Primary Grade Water System in the SWESSAR/CESSAR design. This modification has been made to minimize heavy water losses because the Primary Grade Water System in the SSCR contains heavy water which would be lost to the RLWS if used for providing seal and flush water to the below listed components. The specific equipment affected is:

- a. Waste Evaporator Reboiler Pump
- b. Waste Sample Cooler
- c. Waste Evaporator Feed Pumps
- d. Waste Evaporator Bottoms Pump Flush Connection
- e. Regenerant Waste Sample Cooler
- f. Regenerant Waste Evaporator Reboiler Pump
- g. Regenerant Waste Evaporator Bottoms Pump Suction
- h. Waste Sludge Tank
- i. Spent Resin Surge Tank
- j. Spent Resin Hold Tank

In addition, a connection from the demineralized water supply header to the seal and flush water header has been provided so that seal and flush water can be provided by the Demineralized Water Makeup System in the event that inadequate quantities are derived from liquid waste processing.

5.1.7.6 Reactor Plant Chilled Water System

The Reactor Plant Chilled Water System is the same as conventional PWR (see SWESSAR/CESSAR, Section 9.2.8.1) with chilled water supplied to the following additional equipment:

1. Reactor Plant Gaseous Vents and Drains Vent Header Condenser
2. Reactor Plant Aerated Vents and Drains Annulus Building Vent Header Condenser
3. Solid Waste and Decontamination Building Vent Header Condenser
4. Boron/D₂O Separation Unit (BDSU) Distillate Cooler

5.1.7.7 Reactor Plant Component Cooling Water System

The Reactor Plant Component Cooling Water (RPCCW) System is the same as a conventional PWR (see SWESSAR/CESSAR, Section 9.2.2) except that component cooling water is provided to the Distillate Cooler, Evaporator Condenser, and Sample Cooler for three 45 gpm Boron Evaporators, instead of one 25 gpm evaporator in the SWESSAR/ CESSAR design. The existing SWESSAR/CESSAR System design has sufficient capacity to accommodate this additional heat load without modification.

5.1.7.8 Reactor Plant Sampling System

The Reactor Plant Sampling System is the same as a conventional PWR (see SWESSAR/CESSAR, Section 9.3.2) with the following modifications:

5.1.7.8.1 Annulus Building Reactor Plant Sampling System

The inputs from the High Level Waste Drain Tanks and the Low Level Waste Drain Tanks have been deleted because these tanks are located in the Solid Waste and Decontamination Building in the SSCR Design, and are sampled by the sampling system in the Solid Waste and Decontamination Building.

The following inputs have been added:

1. Holdup Tanks
2. Deborating Surge Tank
3. BDSU Distillate Tank
4. Letdown Tanks
5. Heavy Water Storage Tank
6. RCS Loop Drain Tank
7. Reactor Vessel Flush Tank
8. Boric Acid Tank
9. Mixing Tank

To minimize the amount of sample water to be processed, the sample recirculation lines are directed back to the tank being sampled for the above input sources.

To minimize losses from sampling, the sample sink drain is directed to the Annulus Building Recoverable Aerated Drain Tank.

5.1.7.8.2 Solid Waste and Decontamination Building (SW&DB) Reactor Plant Sample System

The input from the Boron Recovery Tanks has been deleted because these tanks are not included in the SSCR design. The Holdup Tanks (the SSCR analogy to the Boron Recovery Tanks) are located in the Annulus Building and are sampled by the Annulus Building Reactor Plant Sampling System.

The following inputs have been added:

- a. High Level Waste Drain Tanks
- b. Low Level Waste Drain Tanks
- c. Light Water Storage Tanks

The recirculation lines for the High and Low Level Waste Drain Tanks are directed to the Radioactive Liquid Waste System. The recirculation line for the Light Water Storage Tanks is directed back to the tank being sampled.

The input from Primary Grade Water System (PGWS) is deleted. Each tank in the PGWS has its own sample connection in either the Annulus Building or SW&DB Reactor Plant Sampling System for SSCR design.

Two sinks are provided for sampling; one sink drains to the SW&DB sump; the other drains to the SW&DB Recoverable Aerated Drain Tank (for use with samples containing D_2O).

5.1.7.9 Solid Waste and Decontamination Building
Ventilation System

The Solid Waste and Decontamination Building Ventilation System concept is the same as a conventional PWR (see SWESSAR/ CESSAR, Section 9.4.3) with the following additions:

The Solid Waste and Decontamination Building Ventilation supply and exhaust units would be increased in size to accommodate the building extension.

5.1.7.10 Turbine Plant Component Cooling Water System (TPCCWS)

The additional SSCR flow for a conventional PWR Turbine Plant Component Cooling Water System (TPCCWS) (SWESSAR/CESSAR Section 10.4.9) would result in doubling the system flow from approximately 16,000 gpm to 32,000 gpm. This increase in flow is required to provide cooling water for the Upgrader Facility needs. The system modification for SSCR could be accomplished in one of two ways:

- a. Install a completely new system called Upgrader Facility Component Cooling Water System or
- b. Add to the capacity of the existing system.

For purposes of this study, it was decided to increase the capacity of the existing system assuming that sufficient space would be available for the equipment. The TPCCWS would be the same as the conventional PWR system with the addition of a supply line to and return line from the Upgrader Facility.

5.1.7.11 Turbine Plant Service Water System

The Turbine Plant Service Water System for SSCR is the same as a conventional PWR (SWESSAR/CESSAR Section 10.4.11) except that its capacity has been incrementally increased from approximately 16,000 gpm to 32,000 gpm. This additional service water is necessary for cooling the increased Turbine Plant Component Cooling Water System flows for SSCR. An alternate method for expanding the Turbine Plant Service Water System would be to install a separate (and duplicate) system coupled to a separate (and duplicate) Turbine Plant Component Cooling Water System (see Alternative a. above in TPCCWS).

The alternative would have to be used in the event that it was decided to install the SSCR design after construction of the plant had begun.

For purposes of this study, equipment component sizes have been incrementally increased (as was done for TPCCWS).

5.1.7.12 Vents and Drains System (Gaseous and Aerated)

The Vents and Drains System which is comprised of the Reactor Plant Gaseous Vents and Drains and Aerated Vents and Drains remains the same as a conventional PWR (see SWESSAR/CESSAR Section 9.3.3) with the following modifications:

5.1.7.12.1 Gaseous Vents and Drains System

A chilled water condenser to condense vapor containing D_2O has been added to the existing vent header prior to discharge to the Radioactive Gaseous

Waste System. This condensate is drained to the Primary Drains Transfer Tank and is processed by the degasifier, in the same manner as the SWESSAR/CESSAR design.

5.1.7.12.2 Aerated Vents and Drains System

The Aerated Vents and Drains System has been divided into two subsystems:

- 1) a subsystem to collect and process vents and drains from those components potentially containing heavy water (Recoverable Aerated Vents & Drains);
- and 2) a subsystem to collect and process vents and drains from those components containing essentially light water only (Non-Recoverable Aerated Vents and Drains).

The Recoverable Aerated Vents and Drains subsystem is designed in an analogous manner to the SWESSAR/CESSAR Gaseous Vents and Drains Systems as modified above (i.e., a totally closed system to prevent leakage to the atmosphere). A nitrogen blanket, however, is not maintained as in the Gaseous Vents and Drains System because the collected fluid is not hydrogenated. This subsystem adds a collection tank, and two transfer pumps in both the Annulus Building and the Solid Waste and Decontamination Building. In addition, a common vent header is provided to collect the vents from all components containing D_2O in each building. A chilled water condenser is provided on each vent header to condense the vapor (and hence D_2O) from the vents prior to discharge. This fluid is drained to the respective Recoverable Aerated Drain Tank in its respective building.

The fluids collected in the Recoverable Vents and Drains Tanks are transferred to the Boron Recovery System upstream of the Cesium Removal Ion Exchangers for processing in an analogous manner to normal reactor coolant system letdown.

The components providing input to the Recoverable Aerated Vents and Drains subsystem are those components in the SSCR which contain aerated water with recoverable concentrations of D_2O (approximately greater than 0.5 wt%). The Non-Recoverable Aerated Vents and Drains Subsystem collects and processes the remaining reactor plant aerated vents and drains in the same manner as described in SWESSAR/CESSAR.

5.1.7.13 Spent Fuel Pool

The Spent Fuel Pool is the same as for a conventional PWR (See SWESSAR/CESSAR Section 9.12) except that to minimize evaporation losses of D_2O and to minimize the radiological hazard to the site boundary associated with tritium a suitable covering might have to be placed over the spent fuel pool. This cover should be designed for both partial and total removal so that normal fuel handling activities in the pool can proceed in the same manner as a conventional PWR without interference from the cover. The eventual decision whether a pool covering is needed depends on the meteorological conditions at a specific plant site.

5.1.8 D₂O UPGRADER DESIGN

5.1.8.1 Introduction

The purpose of the D₂O Upgrader is to process about 340,000 gallons/cycle of diluted reactor system coolant of the SSCR in order to produce a D₂O-rich stream of 85% concentration and a D₂O-lean stream of 0.5% concentration. The 85% D₂O material is used to establish the beginning-of-cycle D₂O concentration in the reactor coolant of 60%, while the 0.5% material is used to dilute the reactor coolant during the cycle to the end-of-cycle D₂O concentration in the coolant of 2%. The separation of the heavy water from the light water is accomplished in a vacuum distillation process. The upgrader facility consists of a Distillation Section with 3 distillation towers and a Product Recovery Section which establishes the vacuum conditions in the distillation equipment and recovers D₂O from the vacuum system. The D₂O upgrader operates normally on extraction steam from the power plant turbine but it can also be supplied with steam from the plant auxiliary boilers. The feed flow rate of the upgrader varies with the D₂O concentration in the feed stream; however, the design flow rate of the facility is 1725 gallons/day at a 5% D₂O concentration in the feed flow. Assuming a 300 day power generation cycle of the power station, the upgrader remains idle approximately for the first 70 days, then operates continuously during the remainder of the cycle and is shut down during the refueling outage. The operation of the upgrader together with the utilization of the various storage tanks is described in more detail in Section 5.1.12, COOLANT MANAGEMENT OPERATIONS.

The design of the D₂O Upgrader for the SSCR is similar to the design of units which are in operation at Canadian Heavy Water Reactors, e.g., at the Pickering Nuclear Power Station. Also, the same distillation technology is used in the final concentration steps in heavy water production facilities, e.g., Savannah River and Dana Plants in the USA as well as plants in Canada.

In the following sections the equipment of the upgrader as well as design and sizing considerations are described in more detail. A description of the upgrader building and its incorporation in the over all plot plan of the power plant is provided in Section 5.1.2, DESCRIPTION OF PLANT LAYOUT.

5.1.8.2 Description of Distillation Section

The designations of various pieces of equipment in the following description of the Distillation Section refer to the flow diagram shown on Figure 5.1.8-1.

Upgrader Feed Tank FB-101 and Upgrader Feed Pumps GA-105&S provide a continuous feed to the distillation columns. FB-101 will be filled a maximum of once per shift by batch feeding rate of about 20 gpm.

To prevent the release of tritium to the environment, and to prevent carbon dioxide from the atmosphere from contaminating the feed, the Upgrader Feed Tank has been designed as a pressurized vessel blanketed with inert gas.

Nitrogen is introduced into FB-101 to maintain the tank under an inert gas blanket at 10 psig. FB-101 is vented to the vacuum system where light water/heavy water vapors are recovered by chilling.

The feed is routed to the location within the tower system where the composition of liquid on the feed tray most closely matches the feed composition. An on-line infra red D_2O analyzer is used to monitor the feed composition. The table in note 1 on Figure 5.1.8-1 indicates the feed tray locations for various feed compositions.

The feed pumps are low volume, high head pumps. The flow rate through these pumps is more than the process requires, and the balance is recirculated to the tank. This is achieved through a back pressure control valve which maintains a constant back pressure on the pump and hence a constant pump flow. The recirculating feature can be used for mixing the storage tank contents to ensure a uniform composition.

The upgrading of heavy water is a distillation process. Due to the extremely close boiling points of the two molecules, the process is operated under a vacuum to achieve the separation in a practical number of contact stages.

In order to create the lowest possible mean pressure in the system, a vacuum is drawn on each column in parallel. Each column, therefore, has the same top pressure, while the bottom pressure is determined by the tray pressure drops.

In order to prevent all of the column overhead vapor from being drawn into the vacuum generating equipment, it is condensed and returned to the upstream column base in liquid form. There it is revaporized via the reboiler to provide the vapor load for that column. Thus, although there are three columns, the liquid and vapor flow is, in fact, in series as if the system consisted of only one column.

The overhead vapors of Column DA-101 are condensed in Water Cooler Condenser EA-102 and the condensate flows to Reflux Drum FA-101. Reflux is returned to tray 1 by either of two Reflux Pumps GA-101 & S. The reflux is under flow control, which is reset by the level controller on the Reflux Drum.

The overhead (dilute heavy water) product is taken off via flow control. The concentration controller will override the flow controller and will stop the overhead product flow should the overhead composition exceed a preselected value. The overhead product is revaporized in Kettle Vaporizer EA-107, and condensed in Water Cooler Exchanger EA-108. The steam flow to the vaporizer is throttled to control the kettle level.

The purpose of the overhead product revaporization step is to remove traces of iron and other impurities and meet the specifications of 2 micromhos/cm. The parts of the kettle vaporizer, condenser, and associated piping which are in direct contact with the overhead product are constructed of stainless steel.

The overhead product is stored in Overhead Product Storage Tank FB-102. Overhead product storage is provided to monitor product quality before transferring the product to the Light Water Storage Tanks and to provide surge volume for periods where product delivery is restricted.

Nitrogen is introduced into FB-102 to maintain EA-107, EA-108, and FB-102 under an inert gas blanket at 10 psig.

The vapor for DA-101 is generated by Vertical Thermosyphon Reboiler EA-101. The heating medium is steam and is admitted to the shell side of this exchanger via flow control. The condensate is returned to a tank (FA-120) via a steam trap.

The bulk supply of liquid to DA-101 is condensate from condenser EA-104. This condensate flows by gravity onto the heat transfer tray 153 located below the last mass transfer tray 152. The bottoms from DA-101 forms the reflux flow to the downstream column DA-102, and is pumped there by Bottoms Pumps, GA-102 & S. The DA-101 column level is controlled by a level controller which resets the bottoms flow to DA-102.

Since there normally is a constant volumetric flow through the pumps, the balance not needed by the process is recycled back to the base of the column by means of a back pressure controller.

The spare pump for both the DA-101 reflux and bottoms starts up automatically when the on line pump shuts down for any reason.

Operation of Column DA-102 is similar to that of DA-101. The overhead vapors are condensed in EA-104 and the condensed liquid flows by gravity to the heat transfer tray of column DA-101. Column vapors are generated by

Vertical Thermosyphon Reboiler EA-103 in the same way as for the previous column. The level in the column is maintained by a level controller which resets the flow to Column DA-103. The steam to the reboiler is under flow control.

The Bottoms Pumps GA-103 & S are interconnected electrically so that one starts if the other fails. The flow through these pumps is constant and that not required by the process is recycled to the base of the column by a back pressure controller.

Operation of Column DA-103 is similar to that of DA-101 and DA-102. The overheads are condensed in EA-106 and gravitate back to Column DA-102. The Reboiler EA-105 generates the vapor. Steam is admitted to the shell side under flow control.

The pumps GA-101 & S are electrically interconnected so that one starts up when the other fails. The back pressure on these pumps is controlled via recycle of excess to the base of the column.

The bottoms (concentrated heavy water) product is taken off via flow control reset by the column level controller. If the bottoms composition falls below a preselected value, a concentration controller on the bottoms pump recycle line will stop the bottoms product flow.

The bottoms product is revaporized in Kettle Vaporizer EA-109, and condensed in Water Cooler Exchanger EA-110. The steam flow to the vaporizer is throttled to control the kettle level.

The purpose of the bottoms product vaporization step is to remove traces of iron and other impurities picked up by the heavy water during processing in the Upgrader. The product specification of 2 micromhos/cm is thus obtained.

The parts of the kettle vaporizer, condenser, and associated piping which are in direct contact with the overhead product are constructed of stainless steel.

The bottoms product is stored in Bottoms Product Storage Tank FB-103, and pumped to the Heavy Water Storage Tank via GA-107&S. Bottoms product storage is provided to monitor product quality before transferring to the Heavy Water Storage Tank, and to provide surge capacity for periods where product delivery is restricted.

Nitrogen is introduced into FB-103 to maintain EA-109, EA-110, and FB-103 under an inert gas blanket at 10 psig.

The designations of various pieces of equipment in the following description of the Product Recovery Section refer to the flow diagram shown on Figure 5.1.8-2.

A vacuum is maintained on all columns in order to facilitate the distillation process. Condensers prevent the majority of overhead vapors from being withdrawn into the vacuum equipment. Nevertheless, there is some air leakage into the system which must be withdrawn and this air carries water and heavy water with it. Losses of even small quantities of heavy water are costly and thus a recovery system utilizing chilled water refrigeration has been provided.

Vacuum Pump GB-101 & S extracts air from the system via the Column Overhead Condensers (EA-102, EA-104 and EA-106). This air is compressed in two stages and rejected to the battery limit.

The majority of the heavy water recovery process takes place in the Primary Recovery Chiller (EA-111X) which operates at the system pressure (65 mm Hg). The vapors are cooled to 50°F by chilled water. The liquid-vapor is separated in the Suction Drum FA-111X. Vapor/Liquid separation takes place in FA-112X. The separated vapors are compressed to 85 psig and chilled to 45°F in Secondary Recovery Chiller (EA-112X). Vapor/liquid separation takes place in FA-113X and the compressed air is vented via a back pressure

- NOTES:**
- FEED TRAY LOCATION IS FOR DESIGN CONCENTRATION OF 53.4% D₂O IN FEED. FOR FEED CONCENTRATIONS OTHER THAN DESIGN CASE REFER TO TABLE.
 - AIR LEAKAGE INDICATED IS ALLOWANCE FOR THE APPROPRIATE TOWER SYSTEM AND IS INDICATED SCHEMATICALLY FOR MATERIAL BALANCE PURPOSES.
 - EACH COLUMN TO HAVE SINGLE CROSS FLOW SIEVE TRAYS WITH THE FOLLOWING TRAY SPACINGS:
 - OVERHEAD AND BOTTOMS PRODUCT FLOW RATES TO BE CONTROLLED AT 0.5 B.L.
FEED CONC. % H ₂ O	TOWER NO.	TRAY NO.
2	DA-101	95
5	DA-102	60
10	DA-102	119
20	DA-103	14
40	DA-103	54
60	DA-103	91
TRAY	TRAY SPACING INCHES
1 - 3	24
4 - 7	21
8 - 13	18
14 - 17	16
18 - 23	15
24 - 31	14
32 - 41	13
42 - 54	12
55	24 (MANUAL)
56	23
57 - 65	16
66 - 75	15
76 - 88	14
89 - 108	13
109	24 (MANUAL)
110 - 152	12
- LEGEND**
- FC - FLOW CONTROL
 - LC - LEVEL CONTROL
 - PC - PRESSURE CONTROL
 - TC - TEMPERATURE CONTROL
 - XC - CONCENTRATION CONTROL
 - WT PERCENT VAPORIZED
 - ◇ - STREAM REFERENCE NUMBER FOR MATERIAL BALANCE
 - CW - COOLING WATER

Figure 5.1.8-1
D₂O Upgrader,
Distillation Section

COMPONENT		STREAM																											
		1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	43	44	45	46
H ₂ O LB MOLES/HR		31.80	1181.84	1125.07	2422.82		1180.79	1222.67	2087.79		1064.39	1064.12	391.05		0.268	31.530				5.236	5.123	4.524	14.883	1074.47	1217.55	14.877			
D ₂ O LB MOLES/HR		1.51	5.25	5.08	69.02		33.69	33.99	352.80		185.21	183.85	1994.31		1.384	0.142				0.022	0.135	0.734	0.891	184.01	33.86	0.8907			
INERTS LB MOLES/HR																													
TOTAL LB MOLES/HR		33.31	1187.09	1130.15	2491.89		1214.48	1256.66	2440.59		1249.60	1247.98	2385.36		1.632	31.672	1.52	1.52	1.52	6.778	6.778	6.778	20.334	1258.48	1251.41	15.767			
TOTAL LB/HR		602.5	21019.3	20353.0	44898.9		21920.0	22687.7	44654.0		22863.3	22863.6	46924.8		32.09	570.4	44.03	44.03	44.03	138.7	138.9	140.2	417.8	23017.6	22588.0	285.5			
TOTAL KG/HR		279.3	9534.1	9231.9	20411.1		9946.3	10290.9	20254.6		10370.5	10356.0	21284.7		14.56	250.7	19.87	19.87	19.87	62.8	63.0	63.6	189.5	10440.6	10245.7	129.5			
MOL % D ₂ O (AIR FREE)		4.52	0.45	0.45	2.77		2.78	2.70	14.82		14.82	14.73	83.61		83.61	0.45				0.43	2.56	13.95	5.61	14.62	2.71	5.64			
M.W. (AIR FREE)		18.09	18.01	18.01	18.06		18.06	18.05	18.30		18.30	18.30	19.67		19.67	18.01				18.01	18.05	18.28	18.11	18.29	18.05	18.11			
SP. GR. @ 60°F		1.005		1.000	1.002		1.002		1.017		1.017		1.090		1.090	1.000							1.017	1.002	1.005				
SCFM @ 14.7 PSIA @ 60°F																	9.61	9.61	9.61	42.9	42.9	42.9	128.7						
SP. GR. @ T				0.990	0.968		0.968		0.990		0.990		0.951		1.051	0.990							1.005	0.980	1.00				
US GPM @ T		1.198		41.1	92.9		45.3		20.2		46.2		89.2		0.061	1.15							45.8	46.1	0.37				
Z @ TAP		1.0						1.0				1.0			1.0								1.0	1.0	1.0				
TEMPERATURE °F		AMB.	197	118	197	197	197	125	197	197	197	125	198	198	198	117							118	118		45			
PRESSURE PSIG		125			560																								
PRESSURE mmHG			100					100					100																

CE POWER SYSTEMS
SSCR HEAVY WATER UPGRADER
REFERENCE DESIGN STUDY

NO.	DATE	DESCRIPTION	BY	CHKD	APP'D	REV
1						

FOR STUDY TGF

LUMMUS COMBUSTION ENGINEERING, INC.
BLOOMFIELD, NEW JERSEY 07003

PROCESS FLOW DIAGRAM
DISTILLATION SECTION
AREA-100

SCALE DWG. NO. E 8358-01001A-0

Figure 5.1.8-2
D₂O Upgrader,
Product Recovery Section

MATERIAL BALANCE														
COMPONENT	STREAM	22	40	41	42	43	44	45	46	47	48	49	50	51
H ₂ O	MOL/S/HR	14.883	0.7128	0.7084	0.0375	14.877	0.0064	9681.0	3340.0	121.0	17.0	358.0	17.85	0.85
D ₂ O	MOL/S/HR	0.891	0.0383	0.0390	0.0020	0.8907	0.0003							
INERTS	MOL/S/HR	4.58	4.58				4.58							
TOTAL	MOL/S/HR	20.354	5.3121	0.7454	0.0395	15.767	4.5667	9681.0	3340.0	121.0	17.0	358.0	17.85	0.85
TOTAL	LB/HR	417.8	145.7	13.49	0.7156	285.5	132.2	86259.0	80120.0	2180.0	312.0	8450.0	321.3	15.3
TOTAL	KG/HR	189.5	66.1	6.11	0.3246	129.5	60.0	30054.0	28667.0	989.0	142.0	2926.0	145.7	6.9
MOL % D ₂ O	(AIR FREE)	5.61	5.19	5.23	5.30	5.64	4.48							
M	(AIR FREE)	18.11	18.10	18.10	18.11	18.11	18.08	18.0	18.0	18.0	18.0	18.0	18.0	18.0
SP. GR. AT 60°F				1.00	1.00	1.00		1.000	1.000	1.000	1.000	1.000	1.000	1.000
SCFM @ 14.7 PSIA @ 60°F		128.7	33.6				28.9			101.7				
SP. GR. @ T				1.004	1.005	1.005		1.000	1.000	1.000	1.000	1.000	1.000	1.000
US. GPM @ T				0.03		0.57		132.3	120.0	4.35	0.82	12.8	0.64	0.03
Z @ TAP		1.0	1.0				0.98				1.0			1.0
TEMPERATURE	°F	110	50	45	45	40		35	40	40	35	40		35
PRESSURE	PSIG				85	85								
PRESSURE	mmHg	85	65	760		85					5.2			

GB-101&S
VACUUM PUMP PACKAGE
85 mmHg - 25 PSIG
33.8 SCFM (STAGE 1)
29.1 SCFM (STAGE 2)

CE POWER SYSTEMS SSCR HEAVY WATER UPGRADER REFERENCE DESIGN STUDY			
FOR STUDY	DATE	BY	CHKD
DESCRIPTION	DATE	BY	CHKD
C.E. LUMMUS COMBUSTION ENGINEERING, INC. BLOOMFIELD, NEW JERSEY 07003			
PROCESS FLOW DIAGRAM PRODUCT RECOVERY SECTION AREA 100			
SCALE	DWG. NO.	E-8358-01001B-0	

controller. The liquid from FA-113X is cascaded to FA-111X via FA-112X and from there it gravitates to the base of Column DA-102.

The suction pressure of the first stage of the vacuum pump controls the column operating pressures. Steady control is achieved by a pressure controller which recycles some of discharge to the suction. The interstage pressure is controlled in a similar manner by another pressure controller.

Pumps GA-110 & S circulate chilled water at 35°F to the recovery chillers. The design water outlet temperature from the recovery chillers is 40°F and this water is returned to FA-105X.

Vacuum is drawn on this tank by a three stage ejector system with primary and secondary barometric condensers. The pressure in the flash tank is indirectly controlled by a temperature controller situated in the suction line to the circulating pumps. This controller admits steam to the flash tank and is set at 35°F to eliminate any risk of ice formation. The level is maintained in the flash tank by a level controller which admits condensate to compensate for water losses in flashing the blowdown.

5.1.8.4 Design Considerations

5.1.8.4.1 Storage Tanks

The Upgrader Feed Tank (800 gallons capacity) is sized for one shift (8 hours) storage based on a maximum Upgrader feed rate of approximately 2400 gallons per day.

The Overhead Product Storage Tank (800 gallons capacity) and the Bottoms Product Storage Tank (500 gallons capacity) are based on providing one shift (8 hours) of surge capacity for each product at the maximum product rates of 2400 gallons per day (overhead) and 1500 gallons per day (bottoms).

5.1.8.4.2 Distillation Section

The number of equilibrium stages required to achieve the required product fractionation cannot be built into a single column owing to mechanical limitations. For this reason, more than one column is required.

The highest mean relative volatility is obtained when the system operates at the lowest mean pressure; therefore, vacuum is drawn on all columns in parallel. The relative volatility ranges from approximately 1.03 at the bottom of each column to approximately 1.05 at the top. Based on an average relative volatility of 1.04, a minimum of 180 theoretical stages (at total reflux) are required to produce an overhead product of 0.5 wt percent

D₂O and a bottoms product of 85 wt. percent D₂O. Studies indicate that approximately 296 theoretical stages in a three tower system result in an optimal design. A design sieve tray efficiency of 65 percent has been used. This results in a requirement of 456 actual trays, or a three tower system with 152 trays in each tower.

The operating reflux ratio is a function of the feed concentration and varies from approximately 25:1 or 1.2 times the minimum reflux ratio at a feed concentration of 2 wt percent D₂O to a maximum of approximately 95:1 or 1.4 times the minimum reflux at a feed concentration of approximately 60 wt percent D₂O. (The ratio of actual reflux ratio to minimum reflux ratio reaches a maximum of approximately 1.6 at a feed concentration of approximately 20 wt percent D₂O.)

Each column is equipped with sieve trays which were chosen on the basis of low pressure drop, moderate operating flexibility, and low cost. The sieve trays each contain a large number of 3/16 in. diameter holes, through which the ascending vapor passes. The liquid flows across the trays and descends to the trays below through a number of "downcomer" pipes. Each column has a number of zones, each with a slightly different design (hole area). This arrangement is necessary to achieve a turndown to 80 percent of design flows on all trays. Hydraulic considerations dictate that the distillation towers must have a minimum of 80 percent of the design vapor flow passing upward through the holes in the sieve trays in order to prevent excessive quantities of liquid from "weeping" through the holes. The feed to the columns can vary from 0 to 100 percent of the design flow, so long as 80 percent of the design vapor loading is maintained.

To economize on shell height, the tray spacing is generally reduced going down the tower, since the vapor volume is progressively shrinking. A two-diameter design has been chosen to avoid a large number of low-load trays at the minimum (12 in.) tray spacing. By reducing the diameter and increasing tray spacing after tray #55, the tray loading is improved.

Sufficient space is provided at the base of each column to accommodate the entire column liquid inventory. Thus, in the event of an emergency shutdown, each column can be restarted under total reflux without requiring extra tankage, operations, or external feed supply.

When starting up the distillation section, the proper concentration must be established on each tray. To facilitate building up the proper light and heavy water inventories throughout the system, inventories of approximately 3000 gallons per tower should be established in the bottom section of each column. This is especially important if the distillation section is to be started up on dilute feed, since it would take several weeks to establish the proper concentration gradient if the system were not initially inventoried. By initially inventorying the system, startup can be achieved in 1-2 days. The approximate D_2O inventories in each column are a function of the feed concentration and are discussed in Section 5.1.9.

5.1.8.4.3 Product Recovery Section

There is inevitable leakage of air into vacuum equipment. The evacuated air leaves the system saturated with light and heavy water. The heavy

water content in this air stream is recovered by three stages of chilling at various pressures. In excess of 99.9 percent of the D_2O entering the chilling system is recovered. For purposes of designing the vacuum pump system, the design air leakage into the distillation system (indicated on the process flow diagram) has been conservatively estimated as 132 lbs/hr. It is anticipated that the expected air leakage during normal operation will be an order of magnitude lower than this figure. Consequently, the material balance reflects the peak load, and not the normal load from air leakage into the system. (For a detailed discussion of D_2O vent losses, refer to Section 5.1.10).

5.1.8.4.4 Product Vaporization

Vaporization steps are provided to remove traces of iron picked up in the primarily carbon steel distillation system by the overhead and bottoms products. The product vaporizers and downstream equipment are constructed of stainless steel to prevent possible contamination of clean product streams.

5.1.8.4.5 Relief System

The heavy water feed is contaminated with tritium. This is a hazardous material and should not be released to the atmosphere. For this reason the columns do not have any relief valves. The following alternative feature, while requiring code waiver, is an acceptable design practice which has

been used in currently operating upgraders.

The worst condition of pressure that the column is likely to experience is due to unrestricted energy input via the steam reboiler. Under these circumstances the column can eventually reach the pressure of the steam system. The maximum pressure that the steam system can reach is its safety valve set pressure. For this reason the column design pressure has been chosen equal to the steam relief valve set pressure of 75 psig. Thus, under emergency conditions, the steam relief valve protects the columns from overpressuring.

5.1.8.4.6 Heat Exchangers

All heat exchangers handling light water/heavy water mixtures have been designed with double tube sheets and bleeds so that there is minimal possibility of contaminating the product with cooling water or steam.

Should there be a leak due to tube failure, light water from the cooling water or steam would flow into the process since the process is at a lower pressure than either the steam or cooling water systems.

5.1.8.4.7 Pumps

All pumps associated with the columns are of the "canned" type, i.e., the motor is enclosed with the pump and they do not have any seals. This

feature was incorporated to avoid external leakage which must be minimized due to the presence of tritium in the heavy water.

In addition, the range of capacities called for dictates the need for pump recirculation when the net flow is small. This is achieved by a back pressure control valve which maintains a constant volumetric flow through the pump. The flow not required by the process is recirculated to the column bottoms.

5.1.8.4.8 Valves and Fittings

All shutoff valves in positive-pressure D_2O service will be diaphragm-type or an acceptable alternative type having minimum stem leakage. Control valves in positive-pressure D_2O service will be doubly packed, with a bleed-off collection system to recover leakage from between the two packings.

Leakage from flanges is kept to a minimum by minimizing the number of flanged joints in D_2O service and using special torquing procedures for making up the flanged joints (See Section 5.1.10).

5.1.8.4.9 Heat Transfer Trays

The vapor from the overhead of each column is condensed. This liquid is subcooled with respect to the column bottom pressure and has to be elevated

to the bubble point before vaporization can take place. Heating is achieved on a heat transfer tray which produces rapid elevation in temperature. Potential pressure fluctuations at the column base arising from the imbalance between the liquid vapor pressure at the column bottoms temperature and the liquid vapor pressure at the condenser temperature are thus avoided.

5.1.8.4.10 Column Interconnecting Piping

Column overhead vapors are condensed and the liquid is returned to the upstream column onto the heat transfer tray. This flow is by gravity via a barometric leg. The static head of liquid required to overcome the pressure differences between the top and bottom column pressures (plus the flow friction losses) dictates the elevation of the overhead condensers.

5.1.8.4.11 Steam System

Saturated steam is supplied to the D₂O Upgrading Plant battery limit at a pressure of 150 psig. Part of this steam provides the motive fluid for the chilled water ejectors. The remainder of the steam is reduced to 45 psig and desuperheated to provide steam for the various reboilers and vaporizers.

5.1.8.4.12 Condensate System

Steam condensate from the various exchangers flows into Condensate Tank FA-120. This tank feeds Condensate Pumps GA-120 & S. A portion of the condensate

is used for desuperheating the 150 psig steam. Another portion is used to provide makeup water for the water chilling system. The balance is delivered to the D₂O Upgrading Plant battery limit.

A small water-cooled Vent Condenser, EA-114, recovers flash steam as condensate and returns the condensate to FA-120 by gravity flow.

5.1.8.4.13 Cooling Water System

Cooling water to the D₂O Upgrading Plant is mainly supplied from a recirculating closed loop system (secondary loop cooling water). The supply temperature varies from 75°F to 105°F. The design temperature rise has been chosen as 10°F to limit the maximum return temperature to 115°F.

A backpressure station is provided at the common outlet of the elevated tower condensers to prevent boiling at the exchanger outlets.

A small amount of cooling tower water (primary loop cooling water) is used in the vacuum chilling system barometric condensers and serves as blow-down from the primary loop.

5.1.8.4.14 Drain and Spillage Recovery System

The bottoms of each column can be drained to a separate tank (FB-104) should maintenance on a column be required. This drain tank is located in

an underground concrete pit. Pumps are provided to transfer material from the tank back to the towers (GA-108) and for draining the pit sump (GA-109).

Spills are recovered by draining through a network of trenches and conduits to Drain Tank FB-104. The inventory in FB-104 is periodically recycled to the process for recovery of D_2O .

5.1.8.4.15 D_2O Measurements

The concentration of D_2O in the feed and product streams is monitored by on-line infrared analyzers which should be checked periodically in the laboratory using a mass spectrograph.

Monitoring the feed stream is of interest in optimizing plant operations, since the Upgrader will operate most efficiently if the feed is routed to the tray which is closest in composition to the feed.

Monitoring the product streams is necessary to ensure that both product specifications are being met. Should either product go off-spec, product flows are automatically stopped until specification products can be re-established.

D_2O solution samples for checking instruments are taken from piped-up connections into either vacuum or nitrogen-purged sample bombs. Sample coolers are installed on hot sample lines to prevent flashing. After

laboratory analysis, the samples are collected and manually transferred either to the process drain tank or to specific equipment for recovery of D_2O .

5.1.8.6 Upgrader Capacity

For given D_2O concentrations in the overhead and bottoms streams, the Upgrader capacity (feed, overhead or bottoms flow rates) is a function of the D_2O concentration in the feed stream. For the design D_2O concentrations of 0.5 wt% in the overhead and 85 wt% in the bottoms, the Upgrader capacity is shown in Figure 5.1.8-3. The Upgrader feed flow rate goes through a minimum at a feed concentration of approximately 20 wt% D_2O . The feed flow rate curve asymptotically approaches infinity as feed concentrations approach the overhead and bottoms specification of 0.5 and 85 wt%, since there is no separative work required. For the purpose of describing the feed capacity of the Upgrader in a simplified manner, the feed flow rate is quoted in general as 1725 gallons/day at 5 wt% D_2O concentration in the feed stream.

In general, the Upgrader can be operated at overhead and bottoms concentrations different from the design values as long as the ratio of bottoms to overhead concentration remains 170:1 as is the case for the design values. It was mentioned in Section 5.1.6 that the inventory of the Refueling Water Storage Tank (RWST) is replenished with overhead from the Upgrader containing only 0.1 wt% D_2O . For this mode of operation, the Upgrader feed D_2O concentration is 0.5 wt% while the overhead concentration is 0.1 wt% and the bottoms concentration is 17 wt%. For these operating parameters the feed flow rate is 2190 gallon/day, and the bottoms flowrate is 53 gallon/day.

5.1.8.6 Upgrader Optimization Studies

In developing the D₂O Upgrader design, a preliminary design was established which was later refined to a reference design. The preliminary design was based on an initial review of SSCR requirements as well as on general experience with Upgrader designs. The preliminary design was conceived as processing approximately 1500 gallons per day of solutions ranging in concentration from 2 to 60 wt% D₂O, to produce a 0.5 wt% D₂O overhead and an 85 wt% D₂O bottoms. In developing the reference design, it was considered important to more firmly establish the values of some fundamental parameters of the Upgrader; namely the design product specifications and general configuration (number of distillation columns and trays per column).

5.1.8.6.1 Upgrader Bottoms Concentration

As described in Section 5.1.6 and 5.1.11, a residual water volume of low D₂O concentration remains in the reactor vessel after refueling. By filling up the reactor vessel and the remainder of the Reactor Coolant System (RCS) with coolant of about 85 wt% D₂O concentration the required D₂O concentration in the RCS at beginning of cycle can be established fairly easily.

Since the bottoms specification in the region of 85 wt% D₂O is considerably less sensitive than the overhead specification from a separative work standpoint, the 85 wt% value was maintained for the reference design. A change in the bottoms specification within the narrow range of interest does not significantly impact on the Upgrader configuration cost.

5.1.8.6.2 Upgrader Overhead Concentration and Tower Configuration

A study was performed to investigate the effect of the overhead D_2O concentration on number of required distillation towers and, consequently, on the total cost of the D_2O upgrader facility. The results of this study indicate that within the range of 0.3 - 1.5 wt% D_2O for the overhead product, the best Upgrader configuration consists of either 2 or 3 towers with the practical maximum number of theoretical stages in each tower (98 theoretical stages, or approximately 152 actual stages using a design tray efficiency of 65 percent). As the overhead specification is increased within the above range, the configuration shifts from 3 to 2 towers. The breakpoint occurs at about 1.2 wt%.

Comparison of these best tower configurations for various overhead specifications indicates an optimum overhead specification in the 0.5 - 0.75 wt% D_2O range. The estimation and valuation of D_2O inventory (which is a function of the Upgrader capacity and operating mode as well as the overhead specification) could influence this optimum point somewhat. However, since lower levels of tritium activity exist in the overhead stream at lower D_2O concentrations in the overhead, it was concluded that 0.5 wt% D_2O be used as the basis for the Reference Design. Also, three distillation towers were chosen for the Reference Design.

5.1.9 D₂O LEAKAGE AND LOSSES

5.1.9.1 General

D₂O leakage and losses are of concern because of the high cost of D₂O and the radiological hazard due to tritium. For this reason, components in all systems containing D₂O are designed for zero leakage. For the purpose of the following discussion, leakage is defined as the unintentional effluents from the pressure boundary of D₂O-containing fluid systems, e.g., from valve stems or via the tubes in steam generators. Leakage can be broken into a recoverable (e.g., collection of valve stem leakage) and irrecoverable (e.g., tube leakage in steam generators) category. In addition to leakage there are unavoidable effluents, usually called losses, e.g. through vents, evaporation from the refueling cavity during refueling. The unrecoverable leakage together with the losses constitute the total losses of fluid which have to be made up to conserve the fluid inventory.

Where possible, components with a potential for leakage are equipped to enable collection of leakage for eventual processing and reuse in the system. It should be noted that this recoverable leakage is less of a problem for the SSCR than for a typical Heavy Water Reactor (HWR). In a HWR this leakage must be made up with highly concentrated reactor grade D₂O. For the SSCR, this leakage constitutes a portion of the normal coolant system bleed for D₂O concentration reduction and is made up with light water.

Even with a leak-tight design, some irrecoverable leakage, i.e., leakage lost to the environment, is expected. This irrecoverable leakage along with operational, refueling and maintenance losses result in the total D_2O losses. Here, also, the SSCR is faced with less of a problem than a HWR. On the average, the D_2O concentration in the reactor coolant system is about 30% and about 20% in the entirety of all D_2O -containing systems as compared to the essentially 100% D_2O concentration in a HWR. Thus, equal liquid volumes of irrecoverable leakage and other losses are less expensive and less of a radiological hazard for the SSCR than for the HWR.

5.1.9.2 Design Aspects of D_2O Systems To Minimize Leakage

With the exception of the reactor coolant system, D_2O -containing fluid systems operate at low pressure (up to about 150 psig) and temperature (less than 212°F). This in general, facilitates the task of making the SSCR system more leak-tight. General methods for accomplishing zero leakage in D_2O -containing fluid systems are as follows:

1. Packless (diaphragm or bellows sealed) valves are used where possible. All non-packless large diameter (in general, with nominal sizes larger than two inches) valves and non-packless control valves (regardless of size) shall be provided with double packing with lantern ring and a leakoff connection which drains to closed collection systems (the Reactor Plant Gaseous Vents & Drains, and the Reactor Plant Recoverable Vents and Drains System; see Section 5.1.7). Figure 5.1.9-1, showing the internals of the pressurizer spray control valve leakage collection design, is representative for all such equipped valves.

Figure 5.1.9-1
TYPICAL LEAKOFF COLLECTION ARRANGEMENT
(PRESSURIZER SPRAY CONTROL VALVE)

2. Welded connections are used in piping systems to the maximum extent possible. Where flanges cannot be eliminated, seal welds and special torquing procedures for making up flanged joints which minimize the potential for leakage are utilized.
3. All centrifugal pumps with the exception of the reactor coolant pumps are of the canned-rotor type, eliminating pump shaft seals. The reactor coolant pumps are equipped with an additional shaft seal (see Section 5.1.5 REACTOR COOLANT SYSTEM) as well as with the conventional leakage collection system. Seal leakage from positive displacement pumps, i.e., the charging pumps, is contained in a seal cooling system with an overflow connection to the closed collection systems (the Reactor Plant Gaseous Vents and Drains, and the Reactor Plant Recoverable Vents and Drains System; see Section 5.1.7).
4. Double valving or seal welded blank flanges are provided at the interface between systems containing heavy water and those which contain light water.
5. Heat exchangers are provided with a leak free design (e.g., tube to tube sheet seal welds).
6. All relief valve discharge, vent and drain lines are directed to a closed collection system (the Reactor Plant Gaseous Vents and Drains, and the Reactor Plant Recoverable Vents and Drains System; see Section 5.1.7).

7. During fabrication and/or repair, stringent leakage tests are performed to assure that systems are leak free (e.g., halogen or inert gas leakage tests with sensitivities approaching 10^{-9} scc/sec).
8. There are no relief valves on the distillation towers of the D_2O upgrader. Instead, the towers are designed for the set pressure (75 psig) of the process steam system safety valves. The process steam system safety valves thus protect the towers from over-pressurization.
9. A multistage chilling system is incorporated in the D_2O upgrader to recover D_2O from the vacuum system vent stream.

5.1.9.3 D_2O Losses From Reactor Coolant System

The D_2O losses from the reactor coolant system (RCS) are essentially those due to irrecoverable leakage. It is difficult to estimate the magnitude of this leakage. Potential sources of the leakage are: a) a small number of flanged connections in the RCS, b) stem packing leakage from a number of small size valves and c) steam generator leakage. Leakage from the flanged connections and small valves will contribute to the moisture content of the containment building atmosphere, and will be removed by the containment ventilation and air conditioning system.

5.1.9.3.1 Irrecoverable Leakage From RCS Piping System

Flanged Connections. All piping connections are welded except for flanged connections on the pressurizer safety valves which are necessary to facilitate

bench testing and maintenance. In the CESSAR/SWESSAR radiological considerations for a standard PWR, a maximum expected leakage from flanges of 30 cc/hr is used for analysis. This would result in flange leakage from the Reactor Coolant System of about 0.37 lbs/hr. Considering that this is a maximum value, the actual leakage is expected to be considerably less.

Small Valves. There are approximately 40 small size valves in the Reactor Coolant System which do not utilize the leakage collection method described in 5.1.9.2. These valves are designed to zero leakage, but there is a potential for stem packing leakage. In the CESSAR/SWESSAR radiological considerations for a standard PWR, maximum expected leakage of 10 cc/hr/in stem diameter is used for analysis. This would result in leakage from these valves of about 0.54 lbs/hr. Considering that this is a maximum value (and that stem packings are tested for zero leakage and will be well maintained in the SSCR design), the actual leakage is expected to be considerably less for these valves.

Using the leakage values described above, the maximum expected leakage from the RCS piping system to the containment atmosphere is approximately 0.91 lbs/hr or about 22 lbs/day. However, using the methods for reducing leakage for the SSCR design described before, the leakage is expected to be much less. It is estimated that the leakage can be reduced by at least a factor of ten resulting in a leakage of about 2.2 lbs/day. Using an average Reactor Coolant System D_2O concentration of 31 wt% and 300 days of operation a year, the total D_2O losses from the RCS piping system is expected to be about 205 lbs/yr.

5.1.9.3.2 Irrecoverable Leakage in Steam Generators

Irrecoverable leakage to the secondary system in the steam generators is a potential source of D_2O loss. In general, leakage through the tubes of the steam generators is avoided by avoiding tube perforations. This is accomplished by preventative water chemistry control. If excessive thinning of tube walls is detected during routine inspections, the affected tubes are plugged before perforations occur. Field experience indicates that the majority of CE-built steam generators operate without detectable leakage. Thus, in the present study, zero leakage in the steam generators is assumed when estimating the inventory losses which have to be made up.

For safety analyses or evaluations of radiological hazards, conservative leakage values are assumed traditionally which are upper limit rather than experience values. This practice was followed in the SSCR study and for the evaluation of the tritium radiological hazard of the site boundary (see Section 5.5.1-EFFECT OF TRITIUM RELEASE ON ALLOWABLE SITE BOUNDARY DOSE). A leakage of 30 lb/day from the RCS through the steam generators to the Secondary System was assumed. Using an average D_2O concentration of 31 wt% in the RCS and 300 days of plant operation per year, the D_2O loss value for radiological hazards evaluation is 2790 lb/year of 100% D_2O .

5.1.9.4 D_2O Losses From Auxiliary System

5.1.9.4.1 D_2O Losses Due to Irrecoverable Leakage In Piping Systems

To estimate the annual loss of D_2O due to irrecoverable leakage, it has been assumed that the irrecoverable leakage from auxiliary systems is at a

level equivalent to the RCS (i.e., 2.2 lbs/day for an RCS of about 560,000 lbs inventory or 3.93×10^{-4} percent per day). Even though the auxiliary systems have much more piping connections for potential leakage, they operate for the most part at low pressure and temperature (i.e., less than 212°F and 150 psig) and, therefore, leakages are expected to be comparable to that of the RCS. Taking the inventory of the plant at the beginning of the fuel cycle less the RCS inventory, yields a total inventory in the auxiliary systems of 330,376 gallons containing 28,068 gal of D_2O (i.e., 8.5 wt% average D_2O concentration). Using this as a representative value for the entire fuel cycle and assuming the leakage rate of 3.93×10^{-4} percent per day, yields a total leakage over a 300 day fuel cycle of 390 gallons at 8.5 wt% D_2O . This converts to 33 gallons/year of D_2O , or 300 lbs/year of 100% D_2O .

5.1.9.4.2 D_2O Losses Resulting from the Disposal of Ion Exchanger Resins

To minimize the loss of D_2O during the disposal of ion exchanger resins, the SSCR has been designed so that the fluid in each ion exchanger can be drained to the Reactor Plant Vents and Drains Systems for collection and recovery of D_2O . In addition, the ion exchangers have been designed so that they can be flushed to the vents and drains systems with water from a Light Water Storage Tank (0.5 wt% D_2O maximum) to reduce the D_2O concentration of the fluid contained with the resin. After flushing and draining the ion exchanger, the water remaining with the resin is lost

during disposal by the Radioactive Waste Management System. To estimate the annual loss of D_2O during resin disposal, it was assumed that 245 ft³ of resins are disposed of annually (an annual recharge during refueling of each resin bed in systems containing D_2O). For each ft³ of resin, it was assumed that 137 lbs of fluid remained with the resin after draining. Assuming a D_2O concentration of 0.5 wt% for this fluid (D_2O concentration of the flush water), the annual estimated loss of D_2O from these operations is 168 lb.

5.1.9.4.3 D_2O Losses from Degasifier

The degasifier processes 15.66 lbs of water per hour during continuous degasification operation of which 15.6 lbs per hour is recovered by the process gas dryer. Assuming 31 wt% average heavy water concentration, this results in an annual loss of 135 lbs of heavy water, based on a 300 day yearly cycle.

5.1.9.5 D_2O Losses During Refueling

D_2O losses during refueling occur by the following mechanisms:

1. Evaporation from the refueling cavity during refueling operations,
2. Transfer of D_2O to the Refueling Water Storage Tank by the water used during refueling,
3. Loss of D_2O to the Spent Fuel Pool via the fuel transfer tube.

5.1.9.5.1 Evaporation During Refueling Operations

As described in Section 5.4.4.5, it has been conservatively calculated that water from the refueling cavity will evaporate to the containment atmosphere at a rate of approximately 500 lb/hr; however, the D_2O concentration of this water vapor is extremely low (approximately 0.0032 wt percent at the first refueling increasing asymptotically to approximately 0.064 wt percent at the 40th refueling of the end of plant life). Using an average D_2O concentration of the refueling cavity (i.e., 0.053 wt percent at the 20th refueling) and assuming an annual refueling period of seven days, the annual loss of D_2O by evaporation to the containment atmosphere during refueling is estimated to be 45 lb. Since the containment is continuously purged during refueling, this D_2O is lost to the environment (see Section 5.5.1 for the effect of the resulting tritium release on the offsite dose).

5.1.9.5.2 Losses to the Refueling Water Storage Tank (RWST) During Each Refueling

During refueling, water from the RWST is used to flush the reactor vessel, and to fill the RCS and refueling cavity (see Section 5.1.12 for detailed operations during refueling). When this water is returned to the RWST after each refueling some heavy water is present. Also, after each refueling, the RWST is replenished with approximately 90,000 gallons of overheads from the Upgrader containing 0.1 wt percent of D_2O . After 40 refuelings, the D_2O concentration in the RWST is approximately 0.062 wt percent or 4000 lb of D_2O , yielding a levelized estimated annual loss of 100 lb D_2O .

5.1.9.5.3 D_2O Losses to Spent Fuel Pool

During refueling, water in the refueling cavity (containing a small D_2O concentration) is in contact with the water in the spent fuel pool at the fuel transfer tube. As a result, some D_2O is transferred to the spent fuel pool during each refueling. It was assumed that 80 fuel assemblies are transferred and it was conservatively estimated that a volume of refueling cavity water equal to the volume of the fuel transfer tube is transferred to the spent fuel pool during each fuel assembly transfer. The average D_2O concentration of the refueling cavity water was assumed to be the same value as used to determine evaporative losses from the refueling cavity (.053 wt% D_2O). This yields an annual loss of 316 lbs of D_2O to the fuel pool.

5.1.9.6 D_2O Losses In D_2O Upgrader Facility

5.1.9.6.1 D_2O Losses Due to Irrecoverable Leakage

The sources of potential irrecoverable leakage from the upgrader facility are valves and flanges. Leakages from valves and flanges is kept to a minimum by minimizing the number of flanged joints within the plant and utilizing the methods described in Section 5.1.9.2 which minimize the potential for leakage. Since the bulk of the process operates under vacuum, the direction of leakage will generally be into the process. To ensure

that the portion of the process that operates under a vacuum is tight during normal operation, a vacuum tightness test is performed by drawing a vacuum on the process system, blocking the system in and determining the relative loss of vacuum with time. A tight system will also ensure that the vent stream and corresponding D_2O loss will be small and that the vacuum system will not be overloaded during normal operation.

The feed and product vaporization and storage facilities and the pump discharge circuits operate under a positive pressure. Valves in positive-pressure D_2O service will have double packing, with a bleed-off collection system to recover leakage between the two packings. The anticipated loss of D_2O -containing material to the environment assuming that the plant is well-maintained is estimated to be in the order of 0.01 lbs/hr during normal operation, based on an estimated 25 small-size valves in positive-pressure service and a low-leakage rate of 0.01 lbs/day per valve. Assuming that this small amount is vapor or liquid that totally evaporates, and that 20 to 30 percent of this material is D_2O , the unrecoverable loss due to leakage would be in the order of 0.002 to 0.003 lbs/hr D_2O .

From the above, using 0.003 lbs/hr D_2O loss for 300 days of operation a year, the irrecoverable D_2O leakage from the upgrader facility is about 22 lbs/year.

5.1.9.6.2 D₂O Losses from the Upgrader Vent

The vent steam exhaust from the vacuum pump system consists of an average flow of approximately 3 SCFM of air saturated with light and heavy water at 85 psig and 45°F. The heavy water content of the vent stream is approximately 0.01 lbs/hr after D₂O recovery. The peak vent stream flow (shown on process flow diagrams and used for the vacuum system design) which represents higher-than-normal air leakage into the system and maximum storage tank fill rates is conservatively estimated to be 30 SCFM. The D₂O content of the vent stream at the 30 SCFM peak flow rate is estimated to be a maximum of 0.1 lbs/hr. The vent stream, after D₂O recovery, is sent from the Upgrading plant to the radwaste system.

The normal loss from the vent stream of 0.01 lbs/hr, with 300 days of operation, results in a D₂O loss of about 80 lbs/yr.

5.1.9.7 D₂O Losses Due to Water Sampling

It is estimated that on the average one liter of D₂O-containing fluid is sampled per eight hour shift, or approximately 7 lbs per day. After analysis, the fluid is returned to the Reactor Plant Recoverable Aerated Vents and Drains System where it is transferred to the Boron Recovery System for purification, and reuse for makeup. Water samples in the D₂O Upgrader Facility are returned to the facility drain trunk. To estimate the losses of D₂O from sampling, it was assumed that ten percent (approximately 0.7

lb/day) of the sample volume is irrecoverably lost due to the analytical techniques. This volume is sent to the radwaste system for processing and ultimate disposal. Assuming a 300 day fuel cycle with an average D_2O concentration of 31 wt%, it is estimated that approximately 65 lb of D_2O are lost annually due to sampling.

5.1.9.8 D_2O Losses Due To Maintenance

Loss of D_2O during maintenance operations is a concern. The loss of D_2O , however, is minimized by system design where all maintenance vents and drains are directed to either the Reactor Plant Gaseous Vents and Drains or the Reactor Plant Recoverable Aerated Vents and Drains Systems for reuse in the plant.

A further reduction of D_2O losses during maintenance is attained by utilizing good maintenance practices such as insuring adequate component drainage and using drip pans to recover spills. In addition, in systems which contain high concentrations of D_2O and have the potential to contain relatively large volumes of non-drainable fluid, flushes can be provided using Primary Grade Water (0.5 wt% D_2O) to reduce the D_2O concentration in the system prior to draining and disassembly during maintenance. Using the above techniques, it is considered that the losses of D_2O during maintenance will be insignificant when compared to other losses.

5.1.9.9 Summary

5.1.9.9.1 Leakage

It is difficult to estimate the amount of leakage from D_2O -containing fluid systems under actual operating conditions. Assuming leak-tight systems design as described above, a total leakage of liquid (recoverable and irrecoverable) from fluid systems of about 25 gal/day of liquid appears feasible. This leakage rate is about ten percent of the rate used for sizing leakage collection equipment for a conventional PWR. It is also in the same range as experienced during actual plant operation at the Argentinian 340 MW(e) pressurized heavy water reactor at Atucha (about ten gal/day total leakage).

5.1.9.9.2 Total D_2O Losses

Most of the 25 gal/day leakage mentioned above is recovered. It is estimated that less than 3 gal/day of liquid are lost in the form of irrecoverable leakage or unavoidable effluents. This amount is comparable to the operational values observed at the Atucha Plant (about one gal/day). The D_2O loss associated with the three gal/day total loss of liquid is estimated as 1586 lb/year of 100 percent D_2O . The contributing sources for this total D_2O loss which were described in Sections 5.1.9.3 through 8 are summarized in Table 5.1.9-1.

TABLE 5.1.9-1
TOTAL ANNUAL D₂O LOSSES

<u>Source Of Loss</u>	<u>Lbs of 100 Percent D₂O/Year</u>
1. Irrecoverable Leakage from Reactor Coolant System	205
2. Irrecoverable Leakage from Auxiliary Systems	300
3. Evaporation During Refueling Operations	45
4. Losses to the RWST During Refueling Operations	100
5. Losses to the Spent Fuel Pool	316
6. Disposal of Ion Exchanger Resins	168
7. Degasifier Operation Losses	135
8. Sampling Losses	65
9. Irrecoverable Leakage from Upgrader	22
10. Upgrader Venting Losses	80
11. Miscellaneous Losses from Equipment*	<u>150</u>
Total	1,586

*Losses due to unexpected failures in various D₂O-containing components or piping systems which were not covered specifically in the previous items.

5.1.10 D₂O INVENTORY

The D₂O inventory of the SSCR can be broken down into two categories: operating inventory and makeup inventory.

The operating inventory is that inventory required to fill the D₂O-containing fluid systems plus the inventory in various storage tanks. The inventory in the storage tanks is basically needed because, as is described in more detail in Section 5.1.12.8, not all of the coolant which is diluted during a cycle is upgraded during the same cycle and some extra D₂O is needed to provide the beginning-of-cycle D₂O concentration for the next cycle.

The makeup inventory is needed to replenish losses of D₂O from the plant. These losses consist primarily of irrecoverable leakage and effluents to the environment generated during the normal operation of the plant. Makeup inventory has to be provided over the forty-year lifetime of the plant.

5.1.10.1 Operating Inventory

5.1.10.1.1 Inventory in Reactor Coolant System, Storage Tanks and Auxiliary System Piping

One of the factors influencing the D₂O inventory required for the SSCR is the D₂O concentration in the Reactor Coolant System (RCS) at beginning-of-cycle (BOC). The BOC D₂O concentration varies during the initial core

cycles to reflect the gradual approach to an equilibrium core cycle. If 60 wt% is the BOC concentration of an equilibrium cycle, then the BOC concentration of the first cycle is 63 wt%, drops to 55 wt% for the second cycle and reaches 60 wt% at the eighth cycle. This initial reduction of the BOC D_2O concentration has the effect that the additional D_2O inventory in the storage tanks, which is provided to replace the as yet un-upgraded coolant at the beginning of each cycle and which is in excess of the RCS inventory, can be provided gradually from the second to the equilibrium cycle instead of being required at the beginning of the second cycle. This effect is illustrated quantitatively in Table 5.1.10-1 which lists the required D_2O inventory in the RCS and Storage Tanks over the first eleven cycles. The combined RCS and Tank D_2O inventory for the first cycle is 65,520 gallons (of 100% D_2O), drops to 63,706 gallons for the second cycle and reaches a steady state value of 69,468 gallons for the tenth cycle. For the determination of the overall D_2O inventory of the plant the equilibrium BOC value of 69,468 gallons of pure D_2O was used of which 62,400 gallons is in the (cold) RCS and 7,068 gallons are in storage tanks.

It is difficult to determine without detailed equipment arrangements the D_2O inventory which is held up in the piping networks of the Auxiliary Systems. For the purposes of the inventory evaluation it was assumed that 5% of the hot inventory of the RCS piping (41,400 gallons) or 2,070 gallons of 100% D_2O are held up in the Auxiliary Systems.

TABLE 5.1.10-1

D₂O INVENTORY REQUIREMENTS FOR REACTORCOOLANT SYSTEMS AND STORAGE TANKS(Gallons of Pure D₂O)

<u>CYCLE</u>	<u>BOC D₂O CONC.</u>	<u>D₂O REQUIRED IN RCS (GALLONS)</u>	<u>D₂O IN TANKAGE (GALLONS)</u>	<u>TOTAL INVENTORY (GALLONS)</u>	<u>D₂O TO BE ADDED (GALLONS)</u>
1	.63	65,520	0	65,520	0
2	.55	57,200	6506	63,706	-1814 (excess in HWST*)
3	.56	58,063	6737	64,800	-720 (excess in HWST)
4	.57	58,864	6497	65,361	-159 (excess in HWST)
5	.58	59,800	6823	66,623	1103
6	.58	60,632	6864	67,496	873
7	.59	61,526	6968	68,494	998
8	.60	62,400	7064	69,464	970
9	.60	62,400	7060	69,460	-4 (excess in HWST)
10	.60	62,400	7068	69,468	4
11	.60	62,400	7068	69,468	0
40	.60	62,400	7068	69,468	0

- Notes:
1. RCS Volume 104,000 gallons
 2. Concentrations are wt. fractions
 3. Values prior to heatup for the listed cycle

*Heavy Water Storage Tank

5.1.10.1.2 Inventory In D_2O Upgrader Facility

The D_2O inventory in the Upgrader consists of holdup on the sieve trays and downcomers in the towers, normal working inventories in the lower "boot" sections of the towers, inventories in various storage tanks, and holdup in miscellaneous equipment and piping.

The breakdown of D_2O inventory in the Upgrader is shown in Table 5.1.10-2. Referring to Table 5.1.10-2, it is seen that the majority of the D_2O inventory is in the towers. It is noted that there is a wide variation possible in D_2O inventory in the system, varying from 2400 to 4900 gallons (as 100% D_2O).

In developing Table 5.1.10-2, it was assumed that the distillation towers are loaded to 80-100 percent of normal flows, that steady state operation has been reached, that normal working levels are maintained in the tower "boots", and that the Drain Tank is empty. The D_2O inventory in the Upgrader is then essentially a function of the feed concentration to the distillation towers (which affects the concentration profile in the towers and hence the D_2O inventory in the towers) and the inventories in the Feed and Bottoms Product Storage Tanks.

The D_2O inventory in the distillation towers is composed of holdup on the sieve trays and in the downcomers, and the working inventory in the "boot" sections. Varying the feed concentration changes the steady-state concentration profile throughout the tower system and impacts significantly on the D_2O inventory as can be seen in Table 5.1.10-3.

The D_2O inventory in storage tanks is essentially determined by the inventories in the Upgrader Feed Tank and Bottoms Product Storage Tank (assuming that the Drain Tank is empty). The range in D_2O inventory indicated in Table 5.1.10-2 shows the variation between a minimum feed concentration/minimum level situation in the storage tanks, and a maximum feed concentration/maximum level situation.

For the purpose of determining the overall SSCR inventory, the D_2O upgrader inventory was taken as 2600 gallons of 100% D_2O based upon an estimated time-averaged D_2O concentration of about 8 wt%.

TABLE 5.1.10-2
UPGRADER D₂O INVENTORY

	<u>Gallons (100% D₂O)</u>		
	2 wt% D ₂ O Feed, FB-101** and FB-103** <u>at minimum levels</u>	10 wt% D ₂ O Feed, FB-101** and FB-103** <u>at midpoint levels</u>	60 wt% D ₂ O Feed, FB-101** and FB-103** <u>at maximum levels</u>
Towers			
DA-101*	50	80	150
DA-102*	200	350	950
DA-103*	<u>1700</u>	<u>1750</u>	<u>2200</u>
Towers, Total	1950	2180	3300
Storage Tanks	100	250	900
Exchangers, Drums, Pumps	250	300	500
Piping	<u>100</u>	<u>120</u>	<u>200</u>
Total Upgrader Inventory	2400	2850	4900

*For designation of towers refer to Section 5.1.8-D₂O UPGRADER DESIGN

**FB-101: Feed Storage Tank
FB-103: Bottoms Storage Tank

TABLE 5.1.10-3

D₂O INVENTORY IN DISTILLATION TOWERS*

Basis: 3000 gallons liquid holdup/tower
on trays and downcomers plus
300 gallons liquid holdup in "boot"

	Gallons (100% D ₂ O)		
	2 wt% D ₂ O Feed	10 wt% D ₂ O Feed	60 wt% D ₂ O Feed
DA-101**			
Trays, Downcomers	44	65	120
Boot	6	15	30
Total	50	80	150
DA-102**			
Trays, Downcomers	160	290	800
Boot	40	60	150
Total	200	350	950
DA-103**			
Trays, Downcomers	1450	1500	1950
Boot	250	250	250
Total	1700	1750	2200
TOTAL TOWER INVENTORY	1950	2180	3300

* The total H₂O/D₂O vapor inventory (neglected in the table) is approximately 12 gallons per tower. The D₂O content of this vapor inventory is approximately 0.1-0.5 gallons for DA-101, 0.5-4 gallons for DA-102, and 6-9 gallons for DA-103, depending upon the Upgrader feed concentration.

** For designation of towers refer to Section 5.1.8- D₂O UPGRADER DESIGN

5.1.10.2 Makeup Inventory

As mentioned above, irrecoverable losses of D_2O from the plant are replenished by the makeup inventory. The average annual makeup inventory is estimated to amount to 1586 lb/year of 100% D_2O . A more detailed breakdown of the D_2O losses is provided in Section 5.1.9- D_2O LEAKAGE AND LOSSES. The annual makeup inventory has to be made up over the plant life of forty years and totals to about 63,500 lbs of 100% D_2O .

5.1.10.3 Total D_2O Inventory

The total D_2O inventory for the SSCR is the sum of the operating inventory (described in the previous Section 5.1.10.1) and makeup inventory (see previous Section 5.1.10.2). For convenience, a breakdown of these inventories is provided in Table 5.1.10-4. The operating inventory amounts to about 673,000 lb of 100% D_2O . This is about 19% more than the cold D_2O inventory of the reactor coolant system at the beginning of an equilibrium cycle. The makeup inventory over a forty year plant life is about 63,500 lb of 100% D_2O . The sum of the operating and makeup inventory amounts to about 736,500 lb of 100% D_2O . This is about 30% more than the cold reactor coolant system inventory.

TABLE 5.1.10-4
D₂O INVENTORY FOR SSCR

	<u>Lbs (100% D₂O)</u>
1. Operating Inventory	
Reactor Coolant System (at cold conditions)	566,300
Storage Tanks	64,100
D ₂ O Upgrader	23,600
Auxiliary Piping System	<u>18,800</u>
Total	672,800
	<u>lbs/year of 100% D₂O</u>
2. Makeup Inventory	
Unrecoverable Leakage from RCS	205
From Auxiliary Systems	300
From Upgrader	<u>22</u>
	527
Loss During Refueling	
Refueling Cavity Evaporation	45
Loss to Refueling Water Storage Tank	100
Loss to Spent Fuel Pool	<u>316</u>
	461
Coolant Sampling	65
Change of Ion Exchanger Resins	168
Degasifier	135
D ₂ O Upgrader Vent Loss	80
Reserve For Unforeseen Losses from Equipment	<u>150</u>
Total	1586

Over 40 year plant life the annual makeup inventory of 1586 lb/year amounts to 63,440 lb of 100% D₂O.

5.1.11 REFUELING

5.1.11.1 General

When refueling a conventional PWR, the reactor coolant system is drained into a holdup tank until the water level in the reactor vessel drops to one foot below the vessel flange. Then the vessel head is removed, the refueling cavity is filled with water from the refueling water storage tank and the reactor is ready for refueling.

This procedure is not feasible for the SSCR. At the end of the plant life, the tritium activity in the residual water volume remaining in the reactor vessel of a SSCR is at least an order of magnitude higher than for a PWR. Consequently, the tritium activity in the containment building atmosphere due to evaporation from the refueling cavity surface could exceed the Maximum Permissible Concentration and the necessary access to the containment building for refueling and other maintenance work would be feasible only with protective clothing. Furthermore, the residual water volume in the reactor vessel still contains D_2O , about 2 wt% at the end-of-cycle and up to 60 wt% when opening of the reactor vessel would be required earlier in the cycle. Due to its D_2O content, the value of this residual volume ranges from about three quarters of a million dollars at the end-of-cycle to about 20 million at the beginning. While filling the reactor cavity, the residual volume would be diluted with the light water from the refueling water storage tank and, because of its capital value, would have to be salvaged by upgrading in the D_2O upgrader.

Instead of developing a completely new refueling procedure, the PWR procedure was modified for the SSCR. The basic idea is to minimize the residual water volume in the vessel before filling the refueling cavity. This is accomplished by draining to the centerline of the main coolant pipes and by exchanging the D_2O containing residual water volume via flushing with light water before opening the reactor vessel. Thereafter, refueling would proceed essentially as for the PWR. This procedure is schematically shown in Figure 5.1.11-1. In the following sections some of the details of the SSCR refueling procedures are discussed. The operation of various fluid systems during this procedure are described in Chapter 5.1.12 - COOLANT MANAGEMENT OPERATIONS.

5.1.11.2 Boration and Cooldown of the Reactor Coolant System

Prior to cooldown, the reactor coolant system is borated to attain the necessary shutdown margin by feed and bleed using borated light water (0.5 wt% D_2O maximum). During cooldown, the contraction volume is made up using the shutdown concentration borated light water. By these two operations, the end-of-cycle heavy water concentration of 2 wt% is reduced to approximately 1.5 wt%.

5.1.11.3 Drainage of the Reactor Coolant System

Following cooldown, the reactor coolant system is drained at a rate of about 130 gpm via the shutdown cooling system and letdown heat exchanger to the RCS Loop Drain Tank until the water level reaches the hot leg centerline.

Figure 5.1.11-1
SSCR-STUDY REFUELING-COOLANT EXCHANGE

This draining is similar to the operation used in a conventional PWR for draining to a level one foot below the reactor vessel flange. Coolant in the pump suction pipes is drained by opening the drains to the reactor coolant drain tank.

5.1.11.4 Flushing of Reactor Vessel Residual Volume And Filling of Refueling Cavity

After the draining operations, the residual coolant volume (about 28,000 gal) in the reactor vessel with a D_2O concentration of about 1.5 wt% is flushed out of the reactor vessel with borated (about 4000 ppm Boron) light water from the refueling water storage tank (RWST). This flush water from the RWST is pumped into the reactor vessel by a low pressure safety injection pump at a rate of about 5000 gpm. The water leaving the reactor vessel flows by gravity to the reactor vessel flush tank (RVFT).

The hydraulic effectiveness of the flushing operation is, of course, the key item of the entire SSCR refueling operation. The density difference between the residual and flush water is expected to be minimal because the lighter flush water is colder than the heavier but warmer residual water. It appears that the hydraulics of the flushing operation should be bounded by two extreme cases: a piston/plug flow situation with minimal mixing which would be most effective and a feed/bleed situation which would be least effective. Experimental study and confirmation of the flushing procedure is essential to qualify the hydraulics of the process sufficiently in the event a more detailed design of the SSCR is to be prepared. For the

sizing of the tanks which supply or store the flush water it was conservatively assumed that the coolant exchange would require a feed/bleed-type procedure. For refueling at the end-of-cycle, three flush volumes (about 90,000 gal) were assumed. This results in a final average D_2O concentration of about 0.2 wt% in the residual volume and about 0.5 wt% in the water in the reactor vessel flush tank. If opening of the reactor vessel anytime during the cycle became necessary, it was assumed that six flush volumes (about 180,000 gal) would be required at the beginning-of-cycle when the D_2O concentration in the coolant is highest. The tanks and the fluid systems are sized to accommodate the three flush volumes on a regular basis. Six flush volumes can also be accommodated; however, although some additional transfer operations between storage tanks are necessary.

After the flushing operation, the reactor vessel head is taken off. Subsequently the reactor vessel, the coolant system and the refueling cavity are filled (about 600,000 gal required) from the RWST using the same procedure as in the conventional PWR. When making the assumption that the residual reactor vessel volume is exchanged via a feed/bleed process, some D_2O remains in the reactor vessel and becomes mixed with the water from the RWST when filling the refueling cavity. The resulting D_2O concentration in the refueling cavity is estimated to be about 0.003 wt% during the first refueling and about 0.06 wt% during the last refueling after 40 years plant life. The effect on tritium activity in the containment building atmosphere from evaporation of water containing D_2O (and tritium) from the refueling cavity surface is described in Chapter 5.4.4 - TRITIUM EXPOSURE DURING NORMAL OPERATION.

After refueling, the refueling cavity is drained to the RWST as in a conventional PWR. Because there is a small concentration of D_2O in the refueling cavity volume, the water in the RWST will eventually acquire about the same D_2O concentration.

After the refueling cavity is drained and the water level in the reactor vessel is lowered to the main coolant pipe centerline, the flush volume of 90,000 gal is missing from the RWST and must be replenished. Merely transferring the flush volume from the RVFT to the RWST would raise the D_2O concentration in the RWST because the flush water contains about 0.5 wt% D_2O . In subsequent refuelings the D_2O concentration in the refueling cavity would become too high and the tritium concentration in the containment building would exceed the Maximum Permissible Concentration. To avoid this, the flush water is deuterated and processed in the D_2O upgrader in a special operation where the feed stream is separated into a D_2O -lean overhead stream of 0.1 wt% and D_2O -rich stream of 17 wt%. The 0.1 wt% D_2O -lean overhead is then used to replenish the RWST.

In the following sections several typical plant operations are described, which require adjustment of the D_2O and/or boron concentration in the coolant. These operations are:

- 1) heavy water dilution during a typical fuel cycle,
- 2) cold shutdown anytime during a fuel cycle,
- 3) heatup and return to power following cold shutdown during fuel cycle,
- 4) Cold shutdown for refueling at end-of-cycle,
- 5) Cold shutdown anytime during a fuel cycle with removal of reactor vessel head,
- 6) Operation of the boric acid - D_2O separation unit, and
- 7) D_2O upgrader operation.

In addition, a final section is provided which describes the integrated operation of the plant system used for coolant management during a fuel cycle.

5.1.12.1 Heavy Water Dilution During a Typical Fuel Cycle

The heavy water concentration in the Reactor Coolant System (RCS) is reduced approximately linearly by a feed and bleed process from approximately 60 wt percent at the beginning-of-cycle (BOC) to 2 wt percent at the end-of-cycle (EOC) during a typical fuel cycle.

5.1.12.1.1 Dilution Feed

Dilution feed for reducing the RCS heavy water concentration is obtained from the Primary Grade Water System (PGWS) as shown on Figure 5.1.12-1. At BOC, the coolant from the previous cycle which has not been upgraded is held in the PGWS Letdown Tanks. The coolant in these tanks, with a concentration ranging from approximately 6 to 1.3 wt percent D_2O , is used initially for dilution feed. A Letdown Tank Transfer Pump feeds the required volume for dilution into the RCS via the Volume Control Tank (VCT) and Charging Pumps in the CVCS. This coolant from the previous cycle is sufficient to dilute the RCS D_2O concentration to approximately 6 wt percent D_2O . When this feed volume is depleted, water from the Light Water Storage Tank (0.5 wt percent D_2O) is used for dilution feed to the RCS via a Light Water Transfer Pump until EOC.

5.1.12.1.2 Dilution Bleed

Bleed volume from the dilution operation described above is letdown via the CVCS to the BRS and is stored in a Holdup Tank (HT) as shown in Figure 5.1.12-2. When approximately 35,000 gal of letdown is collected in a HT, the volume is deborated using one of the two methods described below:

Boron Concentration Greater Than 10 ppm. Letdown is pumped by the HT Transfer Pump to the Deborating Surge Tank. The Deborating Surge Tank contents are then fed to a Boron Evaporator(s) via a Boron Evaporator Feed Pump(s) where distillate is separated from the boric acid and collected in a Boron Test Tank where a sample is taken. If a further reduction in boron

concentration is required, the Boron Test Tank contents are pumped to a PGWS Letdown Tank via the Boron Demineralizers. Note that series, parallel, or single demineralizer operation is possible depending on boron reduction/flow requirements needed. If a further reduction in boron concentration is not required, the Boron Demineralizers are bypassed and the distillate is pumped directly to a PGWS Letdown Tank.

During deboration, the evaporator bottoms are concentrated to 12 wt percent boric acid. These bottoms are transferred to the Boric Acid - D₂O Separation Unit (BDSU) for processing as described in Section 5.1.12.6.

Boron Concentration Less Than 10 ppm. If the boron concentration in the HT is less than 10 ppm, the Boron Evaporators may be bypassed, and deboration may be accomplished by the Boron Demineralizers. In this mode, the HT Transfer Pumps transfer the letdown from the HT to a PGWS Letdown Tank via the Boron Demineralizers. Again, series, parallel, or single demineralizer operation may be selected depending upon the boron reduction/flow requirements.

5.1.12.2

Cold Shutdown Anytime During Fuel Cycle

If the reactor must be brought to cold shutdown conditions at any intermediate stage of the fuel cycle, the RCS is borated prior to cooldown to maintain the reactor at 5% Δp subcritical during cold shutdown (930 ppm boron at BOC decreasing linearly to 827 ppm boron at EOC).

5.1.12.2.1

Boration

A 35,000 gal Boric Acid Tank and a 35,000 gal Mixing Tank are provided in the CVCS. The Boric Acid Tank stores 12 wt percent boric acid with approximately 0.5 wt percent D_2O ; the Mixing Tank is used to establish an injection volume of borated water with a D_2O concentration equal to that of the RCS at the time of shutdown.

Establish Borated Injection Volume Prior To Boration. The injection volume of borated water is established as shown on Figure 5.1.12-3. Previously letdown reactor coolant stored in the PGW Letdown Tanks (D_2O concentration greater than that existing in the RCS) is mixed in the Mixing Tank with 12 wt percent boric acid from the Boric Acid Tank. The volume of boric acid selected is that quantity which when mixed with the RCS volume at cold shutdown (104,000 gal) results in an RCS boron concentration equal to that required to maintain 5% Δp subcritical cold shutdown (4,610 gal of 12 wt percent boric acid at BOC decreasing approximately linearly to 4,099 gal at EOC). The D_2O concentration in the above mixture is then adjusted to be equal to that of the RCS by adding water from a Light Water Storage Tank (0.5 wt percent D_2O) or from the Heavy Water Storage Tank (85 wt percent D_2O) as necessary.

Boration Prior to Cooldown. Boration of the RCS is accomplished as shown in Figure 5.1.12-4. The injection volume from the Mixing Tank is fed via a Boric Acid Transfer Pump to the VCT. From the VCT it is injected into the RCS via the Charging Pumps. The letdown during boration is temporarily stored in a Holdup Tank. Boration continues until the RCS boron concentration is equal to that required to maintain 5% Δp subcritical shutdown.

5.1.12.2.2 Cooldown

As the RCS is cooled down, a volume of approximately 35,000 gal is required to make up for contraction of the RCS. This makeup volume must be the same D_2O concentration as the RCS and at a boron concentration at least equal to the RCS following boration. The Mixing Tank is used to establish this contraction volume as described below.

Establish Borated Contraction Volume. The 35,000 gal of borated heavy water is established in the Mixing Tank as shown in Figure 5.1.12-5. The borated reactor coolant letdown during boration (temporarily stored in a Holdup Tank) is transferred to the Mixing Tank via a Holdup Tank transfer pump bypassing the boron evaporators. This volume when mixed with the borated water remaining in the Mixing Tank produces 35,000 gal with a D_2O concentration equal to the RCS and with a boron concentration equal to that of the RCS following boration.

Injection of Contraction Volume During Cooldown. As the RCS is cooled down, the contraction volume is pumped to the VCT by a Boric Acid Transfer Pump where it is injected into the RCS via the Charging Pumps as shown in Figure 5.1.12-6.

COMPOSITE OF THE
BORON RECOVERY, PRIMARY GRADE WATER AND
CHEMICAL AND VOLUME CONTROL SYSTEMS
SPECTRAL SHIFT CONTROLLED REACTOR

Figure 5.1.12-5
Cooldown

COMPOSITE OF THE
BORON RECOVERY, PRIMARY GRADE WATER AND
CHEMICAL AND VOLUME CONTROL SYSTEMS
SPECTRAL SHIFT CONTROLLED REACTOR

Figure 5.1.12-6
Cooldown

5.1.12.3 Heatup and Return to Power Following Cold Shutdown
 During Fuel Cycle

5.1.12.3.1 Heatup

The flow path during heatup is shown in Figure 5.1.12-7. During RCS heatup, the resulting expansion volume is letdown to a Holdup Tank. This entire volume of letdown is processed by the Boron Evaporators. The overhead stream is stored in one of the Letdown Tanks while the bottoms stream is transferred to the BDSU for processing as described in Section 5.1.12.6. Thereafter, approximately 12,000 gal of the overhead are used as makeup during the deboration phase (described below); the remaining overhead (depending on its D_2O concentration) is either processed by the D_2O Upgrader during power operation or used as dilution feed for the next fuel cycle.

5.1.12.3.2 Deboration

RCS deboration is accomplished as shown on Figure 5.1.12-8. Reactor coolant is letdown to the Deborating Surge Tank at a flow rate equal to the combined rate of the three charging pumps (132 gpm). The Boron Evaporator Feed Pumps take suction from the Deborating Surge Tank to feed the three Boron Evaporators (135 gpm total combined capacity) which separate the boric acid from the reactor coolant. The evaporator distillate is collected, transferred to the Boron Test Tanks, and pumped directly to the RCS via the VCT and Charging Pumps. This process continues until the boron concentration in

the RCS is approximately 300 ppm, at which time criticality may be achieved. Following criticality, deboration continues in a similar manner until the boron concentration in the RCS is reduced to 10 ppm or less.

When deboration is complete, letdown flow to the Deborating Surge Tank is terminated. The contents of the Deborating Surge Tank are then processed by the Boron Evaporators and the distillate is collected in the Boron Test Tanks. This volume is then transferred via a Boron Demineralizer to a PGW Letdown Tank for storage.

During deboration, the evaporator bottoms are concentrated to 12 wt percent boric acid. These bottoms are transferred to the BDSU for processing as described in Section 5.1.12.6.

During deboration, makeup water is required to replace: 1) the volume of water contained in the Boron Evaporator bottoms; and 2) the volume of water contained in the deborating subsystem. This makeup water is fed from the PGWS Letdown Tank (containing the deborated expansion volume processed during heatup) to the VCT by the Letdown Tank Transfer Pump where it is injected into the RCS via the Charging Pumps.

Figure 5.1.12-7
Heatup

COMPOSITE OF THE
BORON RECOVERY, PRIMARY GRADE WATER AND
CHEMICAL AND VOLUME CONTROL SYSTEMS
SPECTRAL SHIFT CONTROLLED REACTOR

5.1.12.4 Cold Shutdown for Refueling at End-Of-Cycle (EOC)

Refueling is done with essentially light water to minimize the tritium radiological hazard and the loss of heavy water due to evaporation.

5.1.12.4.1 Boration

Prior to cooldown, the RCS is borated to attain the required shutdown margin (827 ppm) as described in Section 5.1.12.2.1 with the following modification: because the intent during refueling is to minimize as much as possible the D_2O concentration in the RCS, the establishment of an injection volume with a D_2O concentration equal to that of the RCS is not desirable. Therefore, the boration volume is established in the Mixing Tank using water from the Light Water Storage Tank and 12 wt percent boric acid from the Boric Acid Tank as shown in Figure 5.1.12-9 to obtain approximately 10,400 gal at a boron concentration of 6,200 ppm which is the maximum attainable without precipitation of boric acid in the CVCS. (Note: Using the maximum boron concentration minimizes feed and bleed, and hence volumes to be processed.) Feeding and bleeding this volume results in an RCS concentration at the required 827 ppm boron.

5.1.12.4.2 Cooldown

Cooldown of the RCS proceeds as described in Section 5.1.12.2.2 with the following modification: Again, because the intent during refueling is to minimize as much as possible the D_2O concentration in the RCS, it is not desirable to establish a contraction volume with a D_2O concentration equal

to that of the RCS. Therefore, the contraction volume is established by mixing water from a Light Water Storage Tank with 12 wt percent boric acid in the Mixing Tank to obtain 35,000 gal at the required boron concentration of 827 ppm as shown in Figure 5.1.12-10 instead of reusing the letdown from boration as in a cooldown during the fuel cycle. (Note: The letdown bleed volumes produced during this refueling shutdown are eventually processed in the boron separator and used in the next cycle as feed for the D_2O dilution process). The established contraction volume is pumped to the VCT via a Boric Acid Transfer Pump where it is injected into the RCS by the charging pumps.

5.1.12.4.3 RCS Drain Prior to Reactor Flush

Following the cooldown, in order to minimize the D_2O concentration further, the RCS is drained as described below and flushed as described in Section 5.1.12.4.4.

Drain RCS to Hot Leg Centerline. While maintaining a nitrogen blanket on the RCS, the RCS is drained to the reactor vessel hot leg centerline with the drain volume being directed to the RCS Loop Drain Tank via a Shutdown Cooling Heat Exchanger and letdown line (see Figures 5.1.12-11 and 5.1.12-12). This is the same procedure as is used for draining the RCS to one foot below the flange in a conventional PWR (see SWESSAR/CESSAR) except for the nitrogen blanket which is required as explained below.

Drain Reactor Coolant Pump Suction Cold Legs. After draining the RCS as described above, the reactor coolant in the reactor coolant pump suction cold legs is removed from the RCS by opening the drains to the Reactor Coolant Drain Tank and then pumping it to the RCS Loop Drain Tank. The nitrogen blanket which is maintained during these draining operations is required to prevent oxygen from entering the hydrogenated Reactor Plant Gaseous Vents and Drain Systems (i.e., the Reactor Coolant Drain Tank).

5.1.12.4.4 Reactor Vessel Flush

The rest volume in the reactor vessel is flushed with RWST water as shown in Figure 5.1.12-13. To initiate the flush, the Low Pressure Safety Injection (LPSI) Pump suction is aligned to the RWST, and the isolation valve to the Reactor Vessel Flush Tank (RVFT) is opened. During the flush, the discharge of the LPSI Pump is throttled to maintain a flush rate of approximately 5,000 gpm. This flow rate is the minimum required to remove decay heat and assures that the coolant leaving the vessel will gravity-drain to the RVFT. When approximately 90,000 gal of flush water has entered the RVFT, its isolation valve is shut; thereby, causing the RCS to refill with water from the RWST. When the RCS is filled, the LPSI Pump suction is realigned for normal shutdown cooling. At this time, the N_2 blanket established during the RCS drain operation may be secured.

During the above flush and fill operation, water from the RWST is used. The boron concentration in the RWST is maintained at between 4,000 and 4,400 ppm. Therefore, during this flush operation the boron concentration in the RCS is increased to approximately 4,000 ppm, providing adequate shutdown margin for refueling (greater than the 2,142 ppm required).

The volume of coolant contained in the RVFT is transferred from the reactor vessel without purification; therefore, the system is designed so that the contents of the RVFT can be recirculated by the RVFT Transfer Pump through the Cesium Removal Ion Exchangers for purification until acceptable radiological levels for reuse as makeup water are obtained.

5.1.12.4.5 Reactor Cavity Fill

In order to reduce the evaporation of reactor cavity water during refueling (to minimize tritium concentration hazard in the containment), normal shutdown cooling is continued until the RCS is approximately 120°F. At this time, the reactor vessel head is removed, and the reactor cavity is flooded with RWST water in the same manner as conventional PWR (see SWESSAR/CESSAR).

5.1.12.4.6 Reactor Cavity Drain After Refueling

After refueling, the contents of the reactor cavity are transferred to the RWST via the Fuel Pool Cooling and Purification System in the same manner as a conventional PWR (see SWESSAR/CESSAR), and the reactor vessel head is replaced. Thereafter, the RCS is drained in the same manner as described in Sections 5.1.12.4.3 except that the drain volume is directed to the RWST instead of the RCS Loop Drain Tank.

5.1.12.4.7 Established Required D_2O Concentration in RCS Prior to Heatup and Refilling of RCS

After refueling, the D_2O concentration in the RCS is approximately 0.06 wt percent or less. The required D_2O concentration in the RCS at BOC is normally 60 wt percent D_2O . This D_2O concentration is achieved by draining (as described above) the RCS to the reactor vessel nozzle center-lines (approximately 28,000 gal remaining) and refilling with the proper proportions of water at 85 wt percent D_2O from the Heavy Water Storage Tank and coolant stored in the Letdown Tanks to achieve the required D_2O concentration in the RCS.

Refilling of the RCS to achieve the required D_2O concentration at BOC is accomplished as shown in Figure 5.1.12-14. The RCS is filled and vented by transferring the necessary amount of: 1) 85 wt percent D_2O from the HWST via the Heavy Water Transfer Pump; and 2) deborated coolant from a Letdown Tank via a Letdown Tank Transfer Pump to achieve the proper BOC D_2O concentration in the RCS when mixed with the rest volume remaining in the reactor vessel. Boration of this feed volume is not required because the dilution attained will not decrease the boron concentration to less than the 930 ppm required to maintain the required cold shutdown margin.

After the RCS has been filled and vented, the N_2 blanket established during the draining operation is secured.

5.1.12.4.8 Heatup and Return to Power

RCS heatup and return to power is completed in the same manner is described in Section 5.1.12.3.

5.1.12.4.9 Process RVFT Volume and Replenish RWST

Following refueling, approximately 90,000 gal of flush water remains in the RVFT, and the RWST volume is depleted by an equal amount (Note: This volume is not required for ECCS operation). During power operation, the volume of water in the RVFT is depleted by the BRS as shown in Figure 5.1.12-15. Since the D_2O concentration of this volume is approximately 0.5 wt percent, the boron evaporator overhead is transferred to a Light Water Storage Tank. Water from the Light Water Storage Tank is then transferred to the Upgrader Facility, where overheads of 0.1 wt percent D_2O are produced (reference Section 5.1.12.7.2). These overheads are pumped directly to the RWST until the 90,000 gal of flush water has been replenished. Since these overheads are boron free, boric acid must be added to the RWST to maintain the boron concentration between 4,000 and 4,400 ppm. Boric acid is added to the RWST from the CVCS as in a conventional PWR; i.e., via the Boric Acid Batching Tank with demineralized water to minimize accumulation of D_2O in the RWST (see SWESSAR/CESSAR). During the above processes, the bottoms of the boron evaporators are transferred to the BDSU, the bottoms of the Upgrader to the HWST.

- NOTES:
1. THE SYSTEM SHOWN IS TYPICAL FOR ONE RCS LOOP, THE OTHER LOOP IS ARRANGED SIMILARLY.
 2. REFER TO THE COMPOSITE OF THE BORON RECOVERY, PRIMARY GRADE WATER CHEMICAL, VOLUME CONTROL SYSTEMS FOR FURTHER DETAILS.

Figure 5.1.12-11
Drain

DRAIN TO HOT LEG & ---

- NOTES:
1. THE SYSTEM SHOWN IS TYPICAL FOR ONE RCS LOOP, THE OTHER LOOP IS ARRANGED SIMILARLY.
 2. REFER TO THE COMPOSITE OF THE BORON RECOVERY, PRIMARY GRADE WATER CHEMICAL, VOLUME CONTROL SYSTEMS FOR FURTHER DETAILS.

Figure 5.1.12-13
Reactor Vessel Flush

FLUSH ADDITION

Figure 5.1.12-15
Reactor Vessel in
Flush Tank Processing

5.1.12.5 Cold Shutdown Anytime During Fuel Cycle with Removal of Reactor Vessel Head

A cold shutdown anytime during the fuel cycle to remove the reactor vessel head is accomplished in the same manner as a refueling shutdown at EOC as described in Section 5.1.12.4, except that: 1) a larger flush volume is required to reduce the D_2O concentration in the rest volume of the reactor value existing in the RCS at the time of shutdown prior to heatup, 3) the boration feed and bleed volume is slightly higher depending on the time in core life when the shutdown occurs, and 4) the deborated flush volumes D_2O concentrations are greater than 0.5 wt percent.

5.1.12.5.1 Reactor Vessel Flush

The RWST is sized to provide a total flush volume of 180,000 gal which is the maximum flush volume required (necessary when the RCS is at 60 wt percent D_2O). The RVFT, however, is sized for a 90,000 gal flush volume maximum; therefore flushes greater than 90,000 gal must be completed in two steps as shown in Figure 5.1.12-16. The first 90,000 gal of flush proceeds as described in Section 5.1.12.4.4 except that the LPSI Pump is realigned for normal shutdown cooling after the 90,000 gal flush is complete before refilling the RCS. The first 20,000 gal from the RVFT are then transferred to the Deboration Surge Tank via the RVFT transfer pump, and are immediately processed by the Boron Evaporators and stored in a Letdown

Tank. The remaining 70,000 gal are transferred by the RVFT Transfer Pump to the Holdup Tanks for storage. When the RVFT is empty, a second reactor vessel flush again proceeds as described in Section 5.1.12.4.

5.1.12.5.2 Processing RVFT

The reactor vessel flush volumes are processed in the same manner as described in Section 5.1.12.4.9 except:

- a. The deborated flush volumes cannot be transferred directly to a Light Water Storage Tank because their D_2O concentrations are greater than 0.5 wt percent. These volumes must be first sent to Letdown Tanks and then processed by the D_2O Upgrader. The resulting overheads are transferred to a Light Water Storage Tank, and the bottoms are transferred to the Heavy Water Storage Tank.
- b. The D_2O Upgrader must process a sufficient quantity of feed from the Light Water Storage Tank to replenish the flush volume in the RWST with 0.1 wt percent D_2O (up to 180,000 gal).

5.1.12.5.3 Re-establish Required D_2O Concentration In The RCS After Replacing Vessel Head

In this case, the contents of the RCS Loop Drain Tank (at a D_2O concentration less than the RCS Concentration at the time of shutdown) are processed by the Boron Evaporators and temporarily stored in the PGWS Letdown Tanks, as shown in Figure 5.1.12-16, for reuse as fill material.

At this point, two options as shown in Figures 5.1.12-18 (Resumption of Normal SSCR Operation), and 5.1.12-19 (Resumption of Modified SSCR Operation) are available:

Resumption of Normal SSCR Operation. The coolant from the RCS Loop Drain Tank, not having a sufficiently high concentration to reestablish the D_2O concentration in RCS equal to that at the time of shutdown cannot be used solely for filling. Therefore, some fill material with a high D_2O concentration must also be used. This high concentration fill material normally is available in the HWST or a Letdown Tank (i.e., the expansion volume resulting from heatup after refueling, or the dilution bleed material generated during the cycle and not yet upgraded). The RCS is filled and vented as shown in Figure 5.1.12.18. The proper proportion of high concentration D_2O (from the HWST or a Letdown Tank) is transferred to the RCS with the proper proportion of deborated RCS Loop Drain Tank Volume (temporarily stored in Letdown Tanks) to fill and vent the system and to achieve a final RCS D_2O concentration equal to that existing at the time of shutdown. RCS heatup and return to power is completed in a normal manner as described in Section 5.1.12.3.

Resumption of Modified SSCR Operation (Partial Boration of RCS). The RCS may be filled and vented as shown in Figure 5.1.12-19. In this case, only the deborated RCS Loop Drain Tank volume (temporarily stored in Letdown Tanks) is used as fill material. This produces a final RCS D_2O concentration less than that which existed at the time of shutdown resulting in a corresponding increase in positive reactivity. To compensate for this additional reactivity, the heatup, return to power, and power operation of the SSCR

are modified. After heatup and return to power, the RCS boron concentration is maintained at a level to provide the reactivity holddown equivalent to the amount the RCS is deficient in D_2O for the existing time in core life (i.e., partial deboration). During power operation, the boron concentration in the RCS is decreased by the Boron Evaporators (D_2O concentration remaining constant) to compensate for fuel burnup until the point in core life when the D_2O concentration existing in the RCS is proper to resume normal SSCR operations.

Figure 5.1.12-18
Establishing D₂O Concentration

5.1.12.6 Operation of the Boric Acid-D₂O Separation Unit (BDSU)

The Boron Evaporators in the BRS are designed to concentrate their bottoms to 12 wt percent boric acid. This boric acid, however, contains varying concentrations of D₂O (depending on the D₂O concentration of the letdown processed). To minimize the amount of D₂O held up in the boric acid, a BDSU has been added to the BRS to extract the D₂O from this boric acid.

The operation of the BDSU is represented by Figure 5.1.12-20. The BDSU Evaporator receives 12 wt percent boric acid from the Boron Evaporator Bottoms via the Boron Evaporator Bottoms Pump. In this evaporator, the Boric acid is slowly boiled to dryness. The distillate is condensed in the BDSU condenser and is collected in the BDSU Distillate Tank where a sample is taken. If the boron concentration of the distillate is less than 10 ppm, the distillate is pumped through a Boron Demineralizer to a PGWS Letdown Tank with a D₂O concentration approximately equal to that in the Distillate Tank. If the boron concentration is greater than 10 ppm, the Distillate Tank is pumped to a Holdup Tank for reprocessing by the Boron Evaporators in the same manner as reactor letdown.

The boric acid remaining in the evaporator is normally reused by mixing with primary grade water from the Light Water Storage Tank until a 12 wt percent solution is attained. The evaporator bottoms are then transferred to the Boric Acid Tank in the CVCS with the BDSU Evaporator Bottoms Pump for reuse during boration operations.

When it is necessary to reduce the boric acid inventory in the RCS, the boric acid remaining in the evaporator may be disposed of by the Radioactive Waste Management Systems. In this case, water from the Radioactive Liquid Waste System is mixed with the BDSU Evaporator bottoms until a 12 wt percent solution is attained. These bottoms are then transferred via the BDSU Evaporator Bottoms Pump to the Radioactive Waste Management Systems for ultimate disposal.

Figure 5.1.12-20

BDSU Processing

5.1.12.7 D₂O Upgrader Operation

During power operation, the Upgrader Facility separates light water from heavy water. During each fuel cycle, the Upgrader Facility operates under two different modes: 1) processing of RCS letdown and 2) processing of flush water to replenish the RWST volume after refueling. These two modes are described below.

5.1.12.7.1 Processing RCS Letdown

All RCS Letdown is processed by the BRS and is stored in the Letdown Tanks. Letdown stored in these tanks with a D₂O concentration of less than approximately 6 wt percent is reused for feed material as discussed in Section 5.1.12.1.1. The remaining letdown is ultimately processed by Upgrader Facility.

Where RCS Letdown is processed by the Upgrader Facility, a Letdown Tank Transfer Pump takes suction on a Letdown Tank and provides feed material to the Upgrader. A NH₄OH Chemical Addition Unit via an eductor adds NH₄OH to the feed stream to provide a pH of approximately 7.5 (minimum) to minimize corrosion in the Upgrader distillation towers. The Upgrader processes the feed and produces a bottom stream of 85 wt percent D₂O and an overhead stream of 0.5 wt percent. The feed, bottoms and overhead flow rates are functions of the D₂O concentration in the feed stream, see Figure 5.1.12-21. The design feed flow rate of the D₂O upgrader is 1725 gal/day of a D₂O

concentration in the feed of 5 wt percent. The overhead stream is transferred to the Light Water Storage Tanks while the bottoms stream is transferred to the Heavy Water Storage Tank for use in the next core cycle.

5.1.12.7.2 Processing Flush Water

During power operation, the water used to flush the reactor vessel rest volume prior to refueling is processed by the BRS and is stored in the Light Water Storage Tank as described in Sections 5.1.12.4.9 and 5.1.12.5.2.

The flush water is processed by the Upgrader as shown in Figure 5.1.12-22. A Light Water Transfer Pump takes suction from a Light Water Storage Tank (0.5 wt percent D_2O) and feeds the Upgrader. The NH_4OH is added to the feed to raise its pH to 7.5 (minimum) to minimize corrosion in the Upgrader. In this case, the Upgrader is operated to produce overheads at 0.1 wt percent D_2O and bottoms of 17 wt percent D_2O . The overheads are directed to the RWST and the bottoms are transferred to a Letdown Tank for reprocessing in the same manner as RCS Letdown (described in previous Section 5.1.12.7.1). This process continues until the volume of water used during the flush process is replenished in the RWST. The feed flow rate of the Upgrader during this mode of operation is about 2,000 gal/day at a D_2O concentration in the feed stream of 0.5 wt percent.

COMPOSITE OF THE
BORON RECOVERY, PRIMARY GRADE WATER AND
CHEMICAL AND VOLUME CONTROL SYSTEMS

SPECTRAL SHIFT CONTROLLED REACTOR
Figure 5.1.12-22
Flush Water Processing

5.1.12.8 Management Of Holding Tanks And D₂O Upgrader Operation

5.1.12.8.1 Introduction

The reduction of the D₂O concentration in the Reactor Coolant System (RCS) by the feed and bleed method results in water volumes which are discharged from the RCS and contain D₂O. Generally these water volumes have to be upgraded so that sufficient quantities of high D₂O concentration water is available for the beginning of the next core cycle. Assuming a 60 wt% D₂O concentration in the RCS at the beginning-of-cycle and a linear reduction to 2 wt% at the end-of-cycle, the average daily D₂O concentration reduction is about 0.19 wt%/day. The magnitude of the bleed volumes to achieve this reduction increases drastically as the D₂O concentration in the RCS decreases, e.g., from about 250 gal/day at beginning-of-cycle to about 8600 gal/day at end-of-cycle.

The bleed volumes are the feed flow for the D₂O upgrader and their order-of-magnitude variation makes the sizing of the D₂O upgrader difficult. Sizing for the maximum feed rate is unfeasible because of the extremely high capital cost of such a facility. Thus, the upgrader size is chosen smaller, closer to a cycle-average capacity. Consequently, the upgrader will not be able to process all the bleed volumes towards the end of a cycle and holding tanks are required to store the unprocessed bleed material until the beginning of the next cycle when the upgrader capacity is greater than the daily bleed volumes. In general, when sizing the upgrader an

optimum has to be found between upgrader capital and operating cost on one side and cost of the D_2O inventory tied up in the holding tanks and cost of the holding tanks on the other side. A simplified analysis at the start of the SSCR Design Study, indicated a flat optimum for upgrader sizes between 1300 gallons/day to 1700 gallons/day feed flow. The upgrader design feed flow rate was later chosen as 1725 gal/day at 5 wt% D_2O concentration from the view point of increased operational flexibility; however the water management patterns described in the following sections are based on a slightly smaller upgrader of 1500 gal/day feed flow at 5 wt% D_2O concentration. A power generation cycle of 300 days was used in the analysis.

The upgrader water management evaluation found that by reusing water with low D_2O concentrations directly, without upgrading, as feed for the RCS when the D_2O concentration in the RCS is high, a reduction in the total amount of water to be upgraded can be achieved. This feature is included in the water management program and water with less than about 6 wt% D_2O (bleed from the RCS during the last portion of the cycle or flush water from the coolant exchange before refueling) is not upgraded but stored and used as RCS feed in the next cycle.

5.1.12.8.2 Holding Tank Management During Equilibrium Core Cycle

The beginning-of-cycle (BOC) D_2O concentration in the RCS varies during the initial core cycles to reflect the gradual approach to an equilibrium cycle. If 60 wt% is the BOC concentration of an equilibrium cycle, then the BOC concentration of the first cycle is 63 wt%, drops to 55 wt% for the second

cycle and gradually increases to its equilibrium value. The following section describes in detail the water management envisioned for the SSCR during an equilibrium core cycle. For simplicity, it was assumed that no cold shutdowns occur during the cycle, in Section 5.1.12.8.3 the impact of cold shutdown is evaluated.

Once equilibrium conditions are obtained, the BOC D_2O concentration of the RCS remains constant at 60 wt%. After a few cycles at this BOC concentration, an equilibrium is achieved where: 1) the total D_2O inventory remains constant throughout the rest of plant life (neglecting irrecoverable D_2O losses) and 2) the water management program attains a repeatable pattern. A detailed breakdown of the water inventory at the beginning of this cycle and a following cycle is shown in Table 5.1.12-1. The only difference between these cycles is the location of the volumes of water in the Letdown Tanks (LDT's). The total water inventories and their D_2O concentrations remain the same.

Figure 5.1.12-23 and 5.1.12-24, sheets 1 through 4 provide a composite bar chart of the water management activities throughout the equilibrium fuel cycle, including refueling. (Processing by the BRS has been omitted because its processing time is negligible in the time frame shown). This composite is composed of 5 water management activities: 1) D_2O dilution of RCS by feed and bleed; 2) processing of RCS Loop Drain Tank (RCSLDT); 3) processing of Reactor Vessel Flush Tank (RVFT); 4) Upgrader operation during fuel cycle; and 5) refueling operations (including cold shutdown and heat up for the next cycle).

Feed and Bleed Operation. Table 5.1.12-2 provides the results of an analysis of the feed and bleed throughout the equilibrium core cycle illustrated in Figure 5.1.12-24 (sheet 1 through 3). Initially, LDT's 3, 4, 5, and 6 are used as feed material for heavy water dilution of the RCS. The letdown during this process is directed to LDT's 7, 8, 3, and 4, respectively, until LDT No. 6 is empty (approximately day 232 of the fuel cycle). At this time LDT's 1, 2, and 7 (RCSLDT volume processed as described in paragraph below), are used as feed, the letdown being directed to LDT No's. 4, 8, and 6, respectively. This process continues until LDT No. 7 is empty (approximately day 278). At this time, the letdown with low D_2O concentration remaining from the previous cycle has been used as feed; therefore, the LWST provides feed material, and the letdown is directed to LDT No's. 6, 5, and 1, respectively. In summary, 271,397 gal of feed material is used; 180,160 gal of letdown from the previous cycle and 91,237 gal from the LWST.

Processing of RCS Loop Drain Tank (RCSLDT). The water stored in the RCSLDT (73,363 gal) after being processed by the BRS is used for dilution feed during the cycle; however, sufficient storage is not available in the LDT's at the beginning cycle. Therefore, this water remains in the RCSLDT until LDT's are available for storage. In this case, this water is processed by the BRS and then transferred to LDT's 1, 2, and 7 after the water originally stored in these tanks is processed by the Upgrader as described below. These transfers are completed on days 170 (35,000 gal), 190 (35,000 gal), and 210 (3,363 gal) respectively. This water is then used for dilution feed as described above.

Upgrader Operation. Table 5.1.12-3 represents the Upgrader operation throughout the equilibrium fuel cycle illustrated in Figure 5.1.12-24 (sheet 1 through 3). Upgrader operation starts at day 88 and continues throughout the remainder of the cycle. The first volume processed is the reactor vessel flush volume. Early in the cycle, the RVFT (approximately 0.5 wt percent D_2O) is deborated by the BRS and is transferred to a Light Water Storage Tank. The intent of the Upgrader operation at this time is to provide 90,000 gal to the RWST at 0.1 wt percent D_2O . To achieve this volume, 92,192 gal (90,000 gal of this originated from the RVFT) are fed from the LWST to the Upgrader. This feed volume provides 90,000 gal of overheads (0.1 wt percent) and 2,182 gal of bottoms (17 wt percent). The overheads are transferred to the RWST, and the bottoms are transferred to LDT No. 3. This operation is complete on day 136. The Upgrader is then fed from LDT No. 3. The resulting bottoms (85 wt percent) and overheads (0.5 wt percent) are transferred to the HWST and LWST, respectively. This process is completed on day 138.

From this point on, the Upgrader processes RCS letdown produced by dilution feed and bleed; the bottoms are transferred to the HWST and the overheads are transferred to the LWST. The first volume processed is the bleed material remaining from the previous cycle with D_2O concentration too high to be used as feed (LDT No. 1, 23,034 gal at 8 wt percent D_2O). This is completed on day 156. The next volume is the heatup expansion volume (LDT No. 2; 35,000 gal at 60 wt percent D_2O). This is completed on day 176. From day 176 to day 300, the Upgrader processes the bleed material with high D_2O concentration generated during the current cycle: LDT No. 7 (35,000 gal at 48.2 wt percent), LDT No. 8 (35,000 gal at 30.7 wt percent),

LDT No. 3 (35,000 gal at 19.6 wt percent), LDT No. 4 (35,000 gal at 12.6 wt percent), and part of LDT No. 8 (11,966 gal at 8.06 wt percent). The total overheads at 0.5 wt percent and bottoms at 85 wt percent processed during the cycle are 138,819 gal and 73,363 gal, respectively. The Upgrader operates approximately 212 days out of the 300 day fuel cycle.

The following operations represented by Figure 5.1.12-24 (Sheet 4) occur during refueling:

Boration. Prior to cooldown, the RCS is borated to 827 ppm boron. Approximately 10,400 gal of letdown at 1.89 wt percent D_2O are generated. This letdown is deborated by the BRS and is stored in LDT No. 3. The RCS D_2O concentration after boration is approximately 1.79 wt percent. For simplicity, the total feed volume was assumed to come from the LWST. This has no effect on the inventory analysis because the boric acid solution stored in the boric acid tank is produced from water stored in the LWST.

Cooldown. Thirty-five thousand gallons of water (0.5 wt percent D_2O) from the LWST's is used for makeup during cooldown. This process decreases the RCS D_2O concentration to approximately 1.36 wt percent D_2O . For simplicity, the total feed volume was assumed to come from the LWST. This has no effect on the inventory analysis because the boric acid solution stored in the boric acid tank is produced from water stored in the LWST.

RCS Drain. 73,363 gal (1.36 wt percent D_2O) is drained from the RCS to the RCSLDT.

Reactor Vessel Flush. 90,000 gal from the RWST are flushed through the Reactor Vessel to the RVFT yielding 90,000 gal at approximately 0.5 wt percent D_2O in the RVFT.

RCS and Reactor Cavity Fill. 605,373 gal are transferred from the RWST to the RCS and Refueling Cavity (76,373 gal to refill the RCS; 529,000 gal to fill the Reactor Cavity).

RCS and Reactor Cavity Drain. After refueling, 605,373 gal are transferred back to the RWST, leaving 27,627 gal in the RCS.

Prepare Fill Volume. To refill the RCS 76,373 gal are required. 73,363 gal are stored in the HWST. The remaining 3,010 gal (1.36 wt percent) are provided by the RCSLDT, and temporarily stored in LDT No. 4.

RCS Refill Prior to Startup. LDT No. 4 and 73,363 gal from the HWST is used to refill the RCS. This mixture yields a 60 wt percent D_2O concentration in the RCS when mixed with the reactor vessel rest volume (27,627 gal with less than 0.06 wt percent D_2O).

RCS Heatup. During RCS heatup 35,000 gal at 60 wt percent D_2O (heatup expansion volume) are letdown, and after processing by the BRS are stored in LDT No. 4.

TABLE 5.1.12-1
D₂O INVENTORY EQUILIBRIUM CYCLE

	BEGINNING OF CYCLE			BEGINNING OF NEXT CYCLE		
	<u>Gallons</u>	<u>Conc.</u>	<u>D₂O</u>	<u>Gallons</u>	<u>Conc.</u>	<u>D₂O</u>
1. Reactor Coolant System	69,000	.60	41,400	69,000	.60	41,400
2. LWST	2,182	.005	11	4,182	.005	11
3. HWST	0	.85	0	0	.85	0
4. RCS Loop Drain Tank	73,363	.0136	999	73,363	.0136	999
5. Reactor Vessel Flush Tank	90,000	.0042	375	90,000	.0042	375
6. Letdown Tanks:						
1.	23,034	.0806	1,857	26,397	.0233	615
2.	35,000	.60	21,000	0	0	0
3.	35,000	.0525	1,838	10,400	.0189	197
4.	35,000	.0336	1,176	35,000	.60	21,000
5.	26,397	.0233	615	35,000	.0336	1,176
6.	10,400	.0189	197	35,000	.0525	1,838
7.	0	0	0	0	0	0
8.	<u>0</u>		<u>0</u>	<u>23,034</u>	.0806	<u>1,857</u>
7. Total	399,376		69,468	399,376		69,468

- Note:
- (1) all gallons are taken at 120°F
 - (2) RCS volume solid is 104,000 gallons
 - (3) Inventory analysis taken with RCS at Normal Operating Temperature, BOC
 - (4) all concentrations are wt fraction

TABLE 5.1.12-2
EQUILIBRIUM CYCLE FEED AND BLEED OPERATION

Bleed Storage (Letdown Tank No.)	Day Letdown Starts	Day Letdown Ends	Tank Providing Feed Material	Feed Volume (gallons)	Feed D ₂ O Conc.	Day Feed Starts	Day Feed Ends	RCS Conc. When Feed Starts	RCS Conc. When Feed Ends	Average Conc. of Bleed Stored
7	0	112.7	LDT #3	35,000	.0525	0	112.7	.60	.3322	.482
8	112.7	184.4	LDT #4	35,000	.0336	112.7	184.4	.3822	.2435	.307
3	184.4	230	LDT #5	26,397	.0233	184.4	220.6	.2435	.1735	.196
			LDT #6	8,603 (35,000)	.0189	220.6	230	.1735	.1554	
4	230	259.2	LDT #6	1,797	.0189	230	231.8	.1554	.1519	.125
			LDT #1	33,203 (35,000)	.0136	231.8	259.2	.1519	.099	
8	259.2	275.7	LDT #1	1,797	.0136	259.2	260.3	.099	.0968	.0806
			LDT #2	33,203 (35,000)	.0136	260.3	276.7	.0968	.065	
6	276.7	288.9	LDT #2	1,797	.0136	276.7	277.4	.065	.0637	.0525
			LDT #7	3,860	.0136	277.4	278.6	.0637	.0613	
			LWST #1	29,840 (35,000)	.005	278.6	288.9	.0613	.04153	
5	288.9	296.4	LWST #1	35,000	.005	288.9	296.4	.04153	.02699	.0366
1	296.4	300	LWST #1	26,397 271,397	.005	296.4	300	.02699	.02	.0233

Note: All concentrations are wt fractions

TABLE 5.1.12-3
EQUILIBRIUM UPGRADER OPERATION

Feed Tank	Feed Tank Conc	Total Feed Vol. (gal.)	Upgrade Rate (gpd)	Day Upgrade Start	Day Upgrade End	Bottoms Produced (gal. @ .85)	Bottoms Storage Location	Overhead Produced (gal. @ .005)	Overhead Storage Location
LWST #2	.005	92,182	1,905	88.3	136.7	(2,182)*	LDT #3	(90,000)**	RWST
LDT #3	.17	2,182	1,140	136.7	138.6	426	HWST	1,756	LWST
LDT #1	.0806	23,034	1,305	135.6	156.6	2,061	HWST	20,973	LWST
LDT #2	.60	35,000	1,792	156.6	176.2	24,645	HWST	10,355	LWST
LDT #7	.482	35,000	1,440	176.2	200.5	19,760	HWST	15,240	LWST
LDT #8	.307	35,000	1,170	200.5	230.4	12,512	HWST	22,488	LWST
LDT #3	.196	35,000	1,132	230.4	261.3	7,913	HWST	27,057	LWST
LDT #4	.126	35,000	1,185	261.3	290.8	4,973	HWST	30,027	LWST
LDT #3	.0806	<u>11,966</u>	1,305	290.3	300	<u>1,073</u>	HWST	<u>10,393</u>	LWST
		304,364				73,363		138,819	LWST's

* bottoms at 17 wt%

** overheads at 0.1 wt%

Note: all concentrations are wt fractions

Figure 5.1.12-23
LEGEND FOR FIGURE 5.1.12-24

A. DURING FUEL CYCLE (SHEETS 1, 2, AND 3)

1. FEED & BLEED

2. UPGRADER

a. TANK USED AS FEED

b. BOTTOMS & OVERHEAD STORAGE

3. SUPPLEMENTARY DRAIN & FILL OPERATIONS (i.e., RVFT & RCSLDT)

B. DURING REFUELING OPERATIONS (SHEET 4)

5-211

NOTE: (1) BOTTOMS TEMPORARILY STORED IN LDT No. 8 UNTIL LDT No. 3 IS EMPTY

WATER MANAGEMENT AND D₂O INVENTORY FOR EQUILIBRIUM FUEL CYCLE (DURING POWER OPERATIONS)

Figure 5.1.12-24

Sheet 1

5-212

5-213

WATER MANAGEMENT AND D₂O INVENTORY
FOR EQUILIBRIUM FUEL CYCLE
(DURING POWER OPERATIONS)

Figure 5.1.12-24

Sheet 3

5-214

MANAGEMENT AND D₂O INVENTORY
FOR EQUILIBRIUM FUEL CYCLE
(DURING REFUELING OPERATIONS)

Figure 5.1.12-24

5.1.12.8.3 Effects of Cold Shutdowns During Equilibrium Core Cycle

The water management systems for the SSCR have been designed to minimize the impact of cold shutdowns during the fuel cycle. This has been accomplished by designing the systems so that letdown created during the cycle is used as makeup for boration and for contraction during cold shutdown. Recycling letdown in this manner minimizes the additional volume resulting from a cold shutdown to be processed by the Upgrader.

It was assumed that three cold shutdowns take place during the cycle. These shutdowns occur at 30 percent, 60 percent, and 90 percent of the fuel cycle. The additional bleed volume, equivalent to 20.7 days of Upgrader operation can be processed during power operation by starting the Upgrader earlier in the fuel cycle, or in the case of shutdowns late in the fuel cycle, by operating the Upgrader using auxiliary steam during cold shutdown or during refueling.

5.1.13 LOAD CHANGE CAPABILITY

5.1.13.1 General

In this chapter the load change capabilities of the SSCR are evaluated and compared with those of the System 80 design. Load changes in the System 80 are effected primarily by changes in the boron concentration of the coolant as well as by some movement of control rods and transient variation of the average coolant temperature. Effecting load changes in the SSCR by variation of the coolant D_2O concentration is impractical because a large excess D_2O inventory and a prohibitively large D_2O upgrader facility would be needed. Variation of the coolant boron concentration as in the conventional PWR is also impractical, because the SSCR requires no boron for base-load operation and huge water volumes would be created during the deboration phase of a load change which subsequently would need to be processed by the boron evaporators. Thus, a study was performed to evaluate the load change capability of the SSCR relative to System 80, using movement of the regulating control rods and transient variation of the average coolant temperature as the means of effecting load changes in the SSCR.

In order to establish the basis for comparison, the set of System 80 Performance Related Design Basis Events (PRDBEs) is described briefly in the following section. The performance capability of the SSCR design is then compared with the System 80 capability for each of the PRDBEs.

5.1.13.2 Load Change Capability Design Basis Events

A PRDBE is defined as an occurrence that is to be accommodated in the design within specified acceptance criteria. A PRDBE is initiated from steady state operating conditions and continues until the Nuclear Steam Supply System (NSSS) reaches a stable operating condition at which the plant can be operated indefinitely. All four reactor coolant pumps are in operation during these events.

A set of PRDBEs exists for the System 80 design which establishes the minimum performance capabilities that the design is to have. To assure that these capabilities are provided in the System 80 design, the results of the PRDBE analysis are translated into component and system design requirements. The System 80 PRDBEs are subdivided into three categories, namely steady state PRDBEs, load change PRDBEs and PRDBEs related to high NSSS availability. In the following study the SSCR capabilities are evaluated only with respect to the System 80 load change PRDBEs.

The System 80 load change PRDBEs are the following:

a) Rapid Transients

- 1) +10% steps,
- 2) +5% per minute ramps,
- 3) Full load rejections,
- 4) Turbine trip without reactor trip,
- 5) Load rejection from any load to any lower load without reactor trip;

- b) Daily load cycling; and
- c) Return-to-power events.

5.1.13.3 Rapid Transient Load Change Capabilities

This category of load change capabilities includes +10% steps, +5% per minute ramps, full load rejections, turbine trip without reactor trip and partial load rejection. These are all relatively rapid transients as compared to daily load cycles and return-to-power events. In order to accommodate these transients the SSCR and System 80 both utilize changes in reactor coolant average temperature, T_{avg} , during the initial seconds of the transient and then use CEA motion for the needed reactivity control.

The SSCR has a more negative moderator temperature coefficient (MTC) of reactivity than the System 80 design and its ratio of Doppler coefficient over MTC is more favorable (smaller) than System 80. The SSCR therefore has a better intrinsic capability for accommodating these rapid transients than System 80. Figure 5.1.13-1 depicts this characteristic of the SSCR as compared to the System 80. This figure shows the initial and final operating points of the NSSS for a load decrease transient using only the T_{avg} changes for adjusting reactivity (i.e., no CEA motion or changes in the D_2O or boron concentration are made). The reactor power decreases because of the negative reactivity added by the increases in T_{avg} . For the same load decrease, Figure 5.1.13-1 shows that the T_{avg} increase for the SSCR using UO_2 fuel is smaller than the increase for System 80. The T_{avg} increase for the SSCR using U/ThO_2 fuel is larger than that for System 80 at EOC (End of

Figure 5.1.13-1
 REACTOR POWER RESPONSES UPON 10% STEP
 LOAD DECREASE (REDUCTION IN TURBINE
 ADMISSION VALVE AREA) NO CEA MOTION

Fuel Cycle) conditions, but this increase is still smaller than the worst case System 80 BOC (Beginning of Fuel Cycle) conditions. Thus, the SSCR performance requirements on components and systems to accommodate the T_{avg} increases during these load decrease transients are less demanding than the equivalent System 80 requirements.

The +10% step load change PRDBE is a step increase in steam flow to the turbine up to +10% of rated with the initial and the final steam flows between 15% and 100% of rated. The +5% per minute ramp PRDBE is a ramp increase in steam flow to the turbine at a rate up to +5% of rated per minute with the initial and the final steam flows between 15% and 100% of rated. The full load rejection PRDBE is a reduction in steam flow to the turbine from rated steam flow to the steam flow needed by the turbine for it to supply only the plant electrical load requirements (approximately 5% of rated) at a rate up to a step change. The turbine trip without reactor trip PRDBE is a step change in steam flow to the turbine from any flow to zero. The load rejection from any load to any lower load (usually called a partial load rejection) without reactor trip PRDBE is a reduction in steam flow to the turbine from any flow to any lower flow at a rate up to a step reduction.

The Daily Load Cycle (DLC) PRDBE is a decrease in steam flow from rated to 50% of rated in two hours, remaining at 50% for six hours and then returning to the rated output in another two hours.* This cycle is repeated on a

*This load cycle is usually designated 100-50-100".

15% power and at a rate up to 5% of rated per minute from 15% on to the rated output).

The acceptance criteria for System 80 PRDBEs specify that they be accommodated:

- a) without resulting in more than normal service effects on any NSSS component or system, or
- b) without violating any of the Limiting Conditions for Operation, (which is a set of conditions within which the plant has to be operated), or
- c) without actuating the Engineered Safety Features System, or
- d) without actuating the Reactor Protection System, or
- e) without lifting either the primary or secondary safety valves.

When designing the NSSS to have the performance capabilities defined by the PRDBEs the basic assumption is made that all automatic NSSS process control systems, which include Reactor Regulating System, Pressurizer Pressure Control System, Pressurizer Level Control System, Steam Bypass Control System and Reactor Power Cutback, are operating normally in the automatic mode. This assumption has also been used in the evaluation of the SSCR load change capabilities. In addition, the SSCR load change capability evaluation assumes that the regulating Control Element Assemblies (CEAs) are available for extensive use during load changes as the primary means for adjusting reactivity and that neither D_2O nor boron concentration change is used in accommodating the load changes.

The next three sections discuss the load change capabilities of the SSCR and the comparison of these capabilities with those of the System 80 in the three types of load changes, namely, rapid transients, daily load cycles, and return-to-power events.

On the step and ramp load increases, the NSSS initially provides the additional steam flow output using its stored energy which results in lowering coolant temperatures and pressures. When the difference between the measured average coolant temperature and a reference temperature reaches a preselected value in the reactor regulating system, the regulating CEAs are withdrawn as necessary to maintain the T_{avg} on its program. Because of the larger power defect (reactivity needed to raise the power) as compared to that of the System 80, the SSCR design needs more available regulating CEA reactivity for achieving the same power increase and maintaining the T_{avg} on its program. With proper positioning of the regulating CEAs before the power increases, the SSCR and System 80 have comparable capabilities for accommodating load increases of +10% steps and +5% of rated per minute ramps between 15% and 100% of rated power. In summary, the SSCR design has performance capabilities comparable to System 80 for accommodating rapid load changes.

5.1.13.4 Daily Load Cycling

5.1.13.4.1 System 80

In the case of System 80, the 100-50-100 Daily Load Cycle (DLC) can be performed using several control strategies which combine the use of CEA motion, boron concentration changes and T_{avg} changes to provide the required reactivity adjustment. Figure 5.1.13-2 shows the System 80 reactor power,

Figure 5.1.13-2
SYSTEM 80 DAILY LOAD CYCLE
(100 - 50 - 100) TRANSIENTS

xenon level, power defect, CEA reactivity, boron concentration and Tav_g responses as a function of time for this 100-50-100 DLC at BOC and EOC conditions.

CEAs are inserted to decrease the power level. Once the reactor power reaches 50% of rated the CEAs stay inserted while the boron dilution starts to compensate for the xenon buildup. At BOC conditions the boron dilution is effective and Tav_g is maintained on its program. However, near the EOC conditions the dilution effect is ineffective even at full letdown flow rate. In order to make up for the xenon buildup, Tav_g is allowed to decrease to provide additional positive reactivity. When the xenon level peaks, the dilution effect catches up and the Tav_g returns to its program. At the time of the power increase, the CEAs are withdrawn and the boron dilution is used with full letdown flow. Again, these two reactivity sources plus the xenon decay and burnout can not provide a sufficient positive reactivity rate for overcoming the power defect. Thus Tav_g drops again to temporarily provide needed positive reactivity. This process is called Tav_g Drop Loading. Later in the transient the xenon burnout process generates excessive positive reactivity and boration becomes necessary. The reactor coolant water replaced during the boration-deboration process has to pass through the boron processing equipment. Near the EOC conditions, at approximately 90% depletion of fuel cycle, the amount of reactor coolant water which requires processing for the 100-50-100 DLC operation exceeds the processing equipment capacity and the 100-50-100 DLC capability becomes limited.

5.1.13.4.2 SSCR

For the SSCR, the 100-50-100 DLC can be performed throughout the entire fuel cycle with moderate use of Tav_g drop loading and a short delay in reaching full power. Figure 5.1.13-3 shows the reactor power, xenon level, power defect, CEA reactivity and Tav_g responses as a function of time for the SSCR during this 100-50-100 DLC operation. CEAs are inserted to reduce the power, withdrawn as necessary to compensate for the xenon buildup when the reactor power is at the low power level and withdrawn to initiate the power increase.

At the BOC conditions, the power defect is large. The CEA insertion is large and sufficient for compensating the xenon buildup later. However, CEAs become fully withdrawn before the reactor power reaches 100%. For the SSCR using UO₂ fuel this occurs when the reactor power is approximately 80% of rated. The Tav_g drop loading is then used to continue the power increase process. Because of the highly negative MTC, the magnitude of Tav_g drop is small as compared to the System 80 case. The reactor power reaches 100% with the Turbine Admission Valves (TAVs) wide open, assuming 5% oversized valves. For the SSCR using U/ThO₂ fuel the CEAs become fully withdrawn when the reactor power is about 86% of rated. The Tav_g drop loading ends at 99% power with TAV wide open. For both fuel types the Tav_g returns to its program about one hour after the reactor power reaches 100%.

At the EOC conditions, the CEAs are almost fully withdrawn in order to provide positive reactivity for compensating the xenon buildup after the

Figure 5.1.13-3
SSCR DAILY LOAD CYCLE
(100 - 50 - 100) TRANSIENTS

reactor power reaches 50% of rated for the SSCR using UO_2 and U/ThO_2 fuels. At the time of increasing the reactor power the CEAs are withdrawn to all-rods out conditions and then the Tav_g drop loading is used to overcome the power defect. The Tav_g drop loading ends at approximately 94% and 92% of rated power respectively for the SSCR using UO_2 and U/ThO_2 fuels when all TAVs become wide open. Eventually the reactor power reaches 100% after a short delay when the xenon destruction continues and provides needed positive reactivity. The magnitude of Tav_g drop is moderate with the lowest Tav_g close to the Tav_g value at no load. For all cases, the xenon decay continues after the reactor power reaches 100% and the Tav_g returns to the program. Therefore, some CEA insertion is needed in order to maintain the reactor at rated power.

5.1.13.5 Return-to-Power Capabilities

A limiting case for the evaluation of the return-to-power (RTP) PRDBE is to start from 100% steam flow, reduce steam flow rapidly to the zero flow and return to rated output at the design ramp rate, with the assumption that all CEAs are fully withdrawn before power reduction.

The System 80 NSSS design has full RTP capabilities up to 65% of an equilibrium fuel cycle, e.g., return from zero to rated power at the maximum design ramp rates without any delay. Thereafter, the capability is slightly limited. The limitation comes from the xenon buildup after the power reduction. The power level achievable at the design rate can be determined for an NSSS design for two separate cases. In the first case Tav_g is being maintained on its program during the return to power. The second case allows Tav_g to temporarily decrease below its programmed value to override xenon.

The most distinctive difference in the RTP capabilities between the System 80 and the SSCR exists early in a fuel cycle. At BOC the System 80 has full RTP capabilities, while the SSCR has only partial RTP capabilities. In System 80 as well as in the SSCR the CEAs which were inserted to reduce the reactor power from full power to zero power are partially withdrawn to compensate for the xenon buildup. As a result, when returning to power the CEA's will become fully withdrawn before rated power is reached. If this situation occurs in System 80, additional reactivity can be gained by reducing the boron concentration in the coolant and rated power can be reached without delay. In the SSCR, additional reactivity is not immediately available, because the D_2O concentration is to be maintained and there is no boron in the coolant. Thus, the SSCR can not return to power as quickly as System 80. As the reactor power increases, the removal rate of xenon also increases which adds positive reactivity. The reactor power will eventually reach the rated level when the xenon returns to its original level. The RTP process can be accelerated through gaining extra positive reactivity by allowing the Tav_g to drop off its program. The Tav_g drop loading is especially beneficial in the SSCR case because of its highly negative MTC. As the fuel depletion progresses the difference between the System 80 and the SSCR reduces and their RTP capabilities become comparable.

5.1.13.6 Summary of Load Change Capability Comparisons Between SSCR and System 80

For rapid transient load change PRDBEs including (1) +10% steps, (2) +5% per minute ramps, (3) full load rejections, (4) turbine trip without reactor trip, (5) load rejections from any load to any lower load without reactor trip, the SSCR has load change capabilities comparable to those of System 80.

For the 100-50-100 daily load cycle PRDBE with Tavg drop loading, both the SSCR and System 80 also have comparable capabilities.

In the return-to-power PRDBE, the SSCR has considerably more limited capability than System 80 at the beginning of a cycle and may lose the capability of overriding the xenon peak after a power reduction (poison itself out) in some situations unless the design is modified to overcome this problem. This means that downtimes for the SSCR could be longer than for System 80 which would result in poorer plant availability.

5.1.13.7 Requirements On Components and Systems

Based on the preceeding evaluation of the load change capabilities of the SSCR, performance requirements for SSCR components and systems can be identified. Because of the scoping nature of the present study, the requirements are qualitative rather than quantitiative.

In order to provide load change capabilities equivalent to System 80, the SSCR components and systems need to meet the following requirements:

1. In order to avoid excessive downtime, the SSCR should be designed to prevent xenon poison-out situations. Several viable means of doing so have been identified and are listed below:
 - a. The SSCR could be designed to operate at power within an enlarged range of reactor coolant temperatures so that positive reactivity

from the moderator feedback effect would be available to overcome the xenon buildup effect. A preliminary estimate is that the ability to drop reactor coolant temperature to 450°F would provide the reactivity necessary to return the reactor to power even at the peak xenon conditions after a rapid shutdown from rated power. However, the feasibility of a ΔT_{avg} drop of this extent with respect to component integrity needs to be ascertained.

- b. The SSCR could be designed such that the number of reactor trips experienced during operation would be reduced considerably relative to current PWR designs. More sophisticated control and monitoring systems would be required to do this by steering the plant away from the trip envelope during various design basis events.
- c. If feasible, the SSCR should be designed and/or operated such that it can return to power quickly following a reactor trip in order to avoid xenon poison-out conditions.

- 2. The evaluation of the load change capability of the SSCR assumed extensive use of the regulating control rods. The reactor design, including fuel, CEAs, monitoring and protective system designs, etc., must permit this extensive use of regulating CEAs near full power. In particular, it is necessary for the SSCR to allow a substantial increase over System 80 in allowable CEA insertion over the entire power range. This requires that the impact on core thermal margin of significant short or long term CEA insertion be evaluated. Furthermore, it requires that the fuel design be capable of withstanding the effects of frequent control rod movement (e.g., pellet-cladding interaction).

5.1.14 HEAT BALANCE

5.1.14.1 Heat Transferred In Steam Generators

A review of the heat balance of the Reactor Coolant System (RCS) of System 80 indicated that by using more up-to-date plant parameters the amount of heat which could be transferred in the steam generators to the secondary system could be increased from 3817 MW(th) to 3821.6 MW(th). Using 3821.6 MW(th) as the (revised) thermal output of the System 80 RCS as a basis, the changes in thermal output for the RCS of the SSCR were determined. The differences between System 80 and the SSCR are essentially due to higher pumping power requirements and lower primary system leakage of the SSCR.

Due to the higher density of the D₂O-containing coolant, the power consumption of the Reactor Coolant Pumps (RCP's) increases. Using 31 wt% as the average D₂O concentration in the coolant, the power consumption of the RCP's increases by 0.84 MW(e), resulting in a total power requirement for the Primary System of the SSCR of about 29.6 MW(e).

The higher power requirements of the RCP's of the SSCR also result in an increase in heat input into the primary coolant by the RCP's of about 0.8 MW(th).

Thermal energy losses due to leakage from the RCS will be smaller for the SSCR than for System 80. The RCS of the SSCR will be more leak tight to minimize D_2O losses because of the associated radiological hazard from tritium as well as because of the cost of D_2O . A leakage rate in the order of 10 gallons/day appears feasible and 10 gallons/day was assumed for the heat balance evaluation. The thermal energy loss with this leakage is very small and was neglected in the heat balance for the SSCR.

Based on the foregoing discussion, the thermal output of the RCS of the SSCR can be derived from the System 80 values via the following tabulation:

Revised System 80 thermal output:	3821.6 MW(th)
Differential in heat input from RCP's:	+0.8 MW(th)
Differential from negligible RCS leakage:	<u>+0.7 MW(th)</u>
Heat Transferred in steam generators of SSCR:	3823.1 MW(th)

5.1.14.2 Plant Heat Balance

The plant heat balance at rated power from SWESSAR/CESSAR was used as reference cycle for the SSCR. The following changes became necessary for the SWESSAR/CESSAR reference cycle:

- a. extraction of 100,000 lb/h steam from the high pressure turbine exhaust to provide process steam for the D_2O upgrader,

- b. extraction of 3,000 lb/h steam from the first stage reheater to provide steam heating of the D₂O upgrader building,
- c. increased thermal output of the RCS from 3817 MW(th) to 3823 MW(th), and
- d. increased requirements for the auxiliary power of the entire plant equipment from 73 MW(e) to 77.8 MW(e), primarily for pumping higher density water.

With these modifications, the thermal and electrical output of SWESSAR/CESSAR and the SSCR can be compared in the following manner:

	<u>SWESSAR/CESSAR</u>	<u>SSCR</u>
RCS Thermal Output, MW(th)	3817	3823
Gross Electrical Generation, MW(e)	1322	1318
Plant Auxiliary Power Requirements, MW(e)	73	77.8
Net Electrical Generation, MW(e)	1249	1240.2

The tabulation shows that the RCS thermal output of the SSCR is 6 MW(th) higher than that of SWESSAR/CESSAR. This should translate into an increase in gross electrical output of about 2 MW(e). However, the gross electrical output of the SSCR is 4 MW(e) lower. This is the consequence of the extraction of steam for the D₂O upgrader facility which is equivalent of a loss of 6 MW(e). The overall net electrical output of the SSCR is about 9 MW(e) less than that of SWESSAR/CESSAR.

It should be noted that the reference thermal cycle shown for SWESSAR/CESSAR is conservative. It is a generic cycle developed for license applications, enveloping many site conditions, and conservatively designed from the standpoint of equipment reliability. In section 10.1 of SWESSAR it is indicated that the turbine cycle design has flexibility to allow optimization for specific site environmental conditions and economic parameters.

Modifications for cycle efficiency improvement can be made in the following areas:

- a. Reduction of condenser back pressure,
- b. Increase in the number of feedwater heating stages, and
- c. Increase in the number of reheat stages.

Examples of typical cycle improvements are shown in Table 5.1.14-1. Assuming an improved cycle with 2.5 in Hg condenser back pressure, seven feedwater heating stages, two reheat stages, and 73 MW(e) auxiliary power requirements, the net electrical output of an improved SWESSAR/CESSAR PWR would be 1266.5 MW(e), while the net electrical output of the comparable improved SSCR would be 1257.7 MW(e) as shown in the following tabulation:

	IMPROVED SWESSAR/CESSAR	IMPROVED SSCR
NSS Thermal Input, Mw(t)	3821.6	3823
Gross Electrical Generation, MW(e)	1341.1	1335.5
Auxiliary Power, MW(e)	73	77.8
Net Electrical Generation, MW(e)	1268.1	1257.7

TABLE 5.1.14-1

EXAMPLES OF POSSIBLE TURBINE CYCLE IMPROVEMENTS
FOR SWESSAR/CESSAR

	<u>SWESSAR/CESSAR</u>	<u>SWESSAR/CESSAR IMPROVED CYCLE</u>	<u>SWESSAR/CESSAR IMPROVED CYCLE</u>
a. Condenser back- pressure reduction (improvement in gross generation)	3.0 in HgA BASE	2.5 in HgA +10 MW(e)	2.5 in HgA +10 MW(e)
b. Stages of feedwater heating (improvement in gross generation)	6 BASE	7 +3.0 MW(e)	8 +6.0 MW(e)
c. Stages of reheat (improvement in gross generation)	1 BASE	2 +4.5 MW(e)	2 +4.5 MW(e)
Total potential improvement in gross generation	BASE	+17.5 MW(e)	+20.5 MW(e)

5.2.1

GENERAL

The performance of the SSCR has been evaluated on two bases: the fuel resource requirements and the power generation costs. In view of the finite uranium ore supply and the delayed development of the breeder reactor, the uranium resources required by various fuel cycles are of increasing importance. In addition, as the price of uranium ore increases, fuel cycles which utilize larger quantities of ore will become less desirable from an economic standpoint since fuel costs will contribute a larger fraction to the total power generation costs. Power costs are an important factor in fuel cycle evaluations, since they determine, to a large extent, which power plant type a utility chooses. The power generation costs for a proposed alternate concept or fuel cycle would have to be less than present nuclear power costs in order to provide sufficient economic incentive for a utility to deploy a new concept or fuel cycle.

5.2.2

DESCRIPTION OF FUEL CYCLES

The SSCR fuel cycles analyzed for this study are the UO_2 once-through fuel cycle, the denatured U-235/ ThO_2 cycle, the denatured U-233/ ThO_2 cycle, and the plutonium/ ThO_2 cycle. The uranium once-through fuel cycle is presently employed in nearly all the light water reactors operated in the U.S. and the rest of the world. The uranium loaded is burned, discharged, and ultimately stored at a waste disposal site as shown in Figure 5.2.2-1(a).

Figure 5.2.2-1 (a)

UO₂ ONCE THROUGH CYCLE (FIVE BATCH, EXTENDED BURNUP)

Figure 5.2.2-1 (b)

DENATURED U-235/THORIUM CYCLE
(PLUTONIUM STORED)

Figure 5.2.2-1(c)
DENATURED U-233/THORUM CYCLE (PLUTONIUM STORED)

Figure 5.2.2-1(d)
PLUTONIUM/THORUM CYCLE (U-233 SOLD)

The optimization study on once-through uranium fuel cycles indicated that improved fuel performance could be gained through a combination of increased discharge burnup and the use of a greater number of in-core batches. Hence, in addition to the traditional three-batch, once-through fuel cycle, an SSCR employing an extended burnup, five-batch fuel cycle was evaluated.

The thorium fuel cycle has long been regarded as a more resource efficient cycle than the uranium cycle when bred fissile material is recovered from the spent fuel and subsequently recycled. Since thorium contains no natural fissile isotope, it has been envisioned that a material with a high fissile content (e.g., highly enriched uranium or reactor grade plutonium) would be mixed with the thorium to provide the necessary enrichment. However, recent concern for the possible diversion of this concentrated fissile material has prompted the investigation of alternate fueling schemes, mainly in the form of denatured uranium/thorium fuel cycles. Two such thorium fuel cycles have been assessed in the present study of the spectral shift control concept. In addition, a more traditional thorium fuel cycle, the plutonium/thorium cycle, has also been evaluated as a transmutter reactor which consumes plutonium generated elsewhere and produces U-233 for subsequent use in the denatured cycles.

The denatured U-235/ThO₂ cycle is viewed as the near-term denatured option since the fissile component (U-235) can be obtained from mined uranium. In this fuel cycle, natural uranium enriched to 20 wt% is blended with thorium; the overall fissile enrichment of fuel is obtained by adjusting the uranium-to-thorium ratio. Enriched uranium provides both the initial fuel to start the cycle and the makeup fuel to sustain the cycle.

Since the conversion ratio is less than unity in the SSCR, the fissile enrichment of the uranium recovered in the denatured U-235/thorium fuel cycle is less than that loaded. In the UO₂ fuel cycle, it has been envisioned (assuming recycle would be allowed) that recovered uranium would be returned to the enriching plant for reenrichment. This would not be done with U-233 bearing fuels because of their high radioactivity which results from U-232 contamination. In addition, the value of U-233 as a reactor fuel is considerably greater than that of U-235 and therefore it would be desirable to minimize the loss of U-233 in the enrichment plant tail stream. Thus, the need for a reenrichment strategy for denatured fuel arises.

Fully enriched uranium was selected as the topping material to reenrich the recycled fuel up to the maximum allowable enrichment limit (i.e., based on the criteria of less than 20 wt% U-235 or 12 wt% U-233 or a weighted average of the two enrichment limits when both species are present). This strategy minimizes the introduction of U-238 into the fuel cycle and thus keeps the production of plutonium as small as possible. In addition to the recycled fuel reenriched in the above manner, some makeup fuel at 20 wt% U-235 or less is needed to complete the material balance. By maintaining

segregation between the recycled fuel stream and the makeup fuel stream, some economic benefits can be achieved. The recycled stream will require a remote fabrication process because of U-232 contamination, while the makeup stream can be fabricated using contact methods. Since the makeup fuel stream represents less than 10% of the total U-235 makeup to the system, it may be desirable to eliminate this separate stream at a slight economic penalty.

Figure 5.2.2-1(b) is a schematic diagram of the facilities required to support the denatured U-235/ThO₂ fuel cycle. In this diagram the plutonium bred in the cycle goes to a storage facility for possible future use. An alternative to this strategy would be to use the plutonium as soon as possible, for example in the plutonium/ThO₂ burner reactor described below.

The denatured U-233/ThO₂ cycle represents the long-term option for use of thorium in a proliferation-limited nuclear economy. Because of the excellent nuclear properties of U-233, it is well suited for use in the SSCR. Its relatively constant capture-to-fission ratio (α) in the SSCR spectrum results in a good utilization of fissile material. However, since U-233 does not occur in nature, it must be produced by conversion of thorium in a neutron flux. This can be accomplished most easily by the irradiation of plutonium/thorium fuel or by the use of a thorium blanket in a power reactor.

The denatured U-233/ThO₂ cycle employs 12 wt% fissile uranium blended with thorium oxide to attain the required overall fissile enrichment of the fuel. The U-233, which is obtained in an almost pure form (greater than

92 wt%), is denatured by adding depleted uranium either at the reprocessing plant when recovered from the spent fuel or at the fabrication plant prior to manufacture of the fuel rods.

As with the denatured U-235/ThO₂ cycle, plutonium is bred along with additional U-233 since a significant quantity of U-238 is present in the cycle. Again this plutonium can either be stored or used to displace some of the fissile makeup requirements of the cycle. Figure 5.2.2-1(c) shows the facilities required in this fuel cycle. A secure nuclear energy center would probably be desirable for use with this fuel cycle to provide a proliferation resistant place in which to denature the source U-233 and to burn the bred plutonium.

The final fuel cycle evaluated in this study was the plutonium/thorium cycle; this cycle is shown schematically in Figure 5.2.2-1(d). As mentioned above, this cycle is the most direct method of obtaining U-233. Since plutonium is bred in all of the previous fuel cycles, a source of fissile material is readily available for this cycle. Because the plutonium is chemically separable from the thorium, particularly in the fresh fuel, a secure facility is required to deploy this cycle. Another alternative would be to spike the plutonium with a radioactive source, but this may result in substantial economic penalties due to the fabrication and handling of highly radioactive fuel.

In Section 5.2.3 the resource utilization of the four fuel cycles described above will be assessed in terms of their thirty-year resource requirements. In Section 5.2.4 the fuel cycle and power generation costs resulting from the deployment of these fuel cycles in the SSCR will be presented. Finally, in Section 5.2.5, the performance of the PWR and SSCR operating on the same fuel cycles will be compared. The detailed 30-year mass flow data sheets and fuel inventory data sheets are contained in Section 7.9. From these detailed data sheets, the resource requirements and economics of the SSCR have been derived.

5.2.3 RESOURCE UTILIZATION

The thirty-year fissile requirements of the five fuel cycles analyzed are shown in Table 5.2.3-1. The benefits of spectral shift control can best be seen in fuel cycles which take full advantage of the improved conversion ratio it affords, that is, in fuel cycles which employ recycle of bred fissile material. The denatured U-233/ThO₂ fuel cycle exhibits the lowest fissile requirements. The net total fissile requirement of this cycle is only about 22% of that of the standard UO₂ once-through cycle, while that of the improved UO₂ once-through cycle is 84% of that of the standard cycle.

TABLE 5.2.3-1

THIRTY-YEAR FISSILE REQUIREMENTS FOR THE SSCR

<u>Fissile Isotopes</u>	<u>UO₂ Once-Through Cycle, Standard Fuel Management</u>	<u>UO₂ Once-Through Cycle Improved Fuel Management</u>	<u>(kg/GWe) Denatured U-235/ThO₂ Cycle, Plutonium Sale or Storage</u>	<u>Denatured U-233/ThO₂ Cycle, Plutonium Sale or Storage</u>	<u>Plutonium/ThO₂ Cycle, U-233 Sale</u>
U-233				9,940	(8,642)
U-235	22,858	19,575	14,425	190	
Pu-239			(1,458)	(1,670)	21,551
Pu-241			(410)	(475)	5,074
<u>Cumulative Total</u>	<u>22,858</u>	<u>19,575</u>	<u>12,557</u>	<u>7,985</u>	<u>17,983</u>
<u>Credits^(a)</u>					
U-233			1,917	(2,536)	323
U-235	(925)	(1,053)	1,476	(387)	
Pu-239			82	(270)	2,760
Pu-241			11	(63)	1,875
<u>Net Total</u>	<u>21,933</u>	<u>18,522</u>	<u>9,071</u>	<u>4,729</u>	<u>13,025</u>

NOTES: (a) Credits include ex-core inventory and final core inventory; for fuels which are recycled, losses are included.

The resource requirements developed for this study are based on an annual refueling schedule and a 75% capacity factor. Material losses due to conversion and fabrication were assumed to be 1.5% as were the losses for reprocessing and reconversion. The diffusion plant tails assay was selected to be 0.20 wt%. For the standard three-batch fuel management, a particular fuel batch remains in core for three years, except for the initial two batches. In those cycles where recycled fuels are employed, fuel resides outside the core for two years for cooling, reprocessing and refabrication. The fuel cycle data is presented in terms of generations zero through five. Each generation encompasses five years except the zeroth which spans three years of operation. For the cases employing recycle, each succeeding generation represents a given pass of recycle fuel through the reactor. Since the behavior and composition of fuel batches belonging to a specified generation are similar, the results are presented in this manner.

As an aid in evaluating the fuel cycles analyzed in this study, several tables are presented including uranium and separative work, average batch inventories, apparent fissile consumption and integral conversion ratios. The performance of the SSCR will be discussed in the subsequent sections in terms of the data summarized in these tables.

5.2.3.1 UO₂ Once-Through Fuel Cycle

The UO₂ once-through fuel cycle is the most costly in terms of resource requirements, since the discharged fissile material is not recovered. With this fuel cycle the most efficient utilization of uranium occurs when the

fissile material loaded and the percentage of initial fissile which is eventually discharged are minimized. Two fuel management schemes were employed in the once-through fuel cycle mode: a three-batch scheme and an improved five-batch scheme. Since annual refueling is maintained in both fuel management schemes, the five-batch scheme has an average discharge exposure of 50,650 MWd/t as compared to an average exposure of 30,400 MWd/t for the three-batch scheme.

The U_3O_8 requirements of the UO_2 once-through fuel cycles are listed in Table 5.2.3-2(a) while the separative work requirements are given in Table 5.2.3-2(b). Although the equilibrium fissile enrichment is 1.10 wt% higher for the five-batch core, the annual equilibrium uranium requirements are 14% lower due to the smaller batch size. The separative work requirements for the two fuel cycles are almost identical because of the balance between the higher fissile enrichment and the lower total loading. The average batch inventories for the equilibrium cycles are shown in Table 5.2.3-3. Approximately 16% of the initial fissile material is discharged in the five-batch core, whereas 26% of the U-235 loaded is discharged in the three-batch core. This indicates a more complete consumption of fissile material in the five-batch core. On the other hand, the apparent fissile consumption (AFC) of the five-batch core is 11% higher than that of the three-batch core when the conventional definition of AFC (cf. footnote (a) to Table 5.2.3-4) is used. This apparent paradox between better uranium utilization and higher fissile consumption is a result of including only fissile material consumed during power operation in the calculation of AFC. The point of fact is that the five-batch core burns more fissile material than the

TABLE 5.2.3-2(a)

U₃O₈ REQUIREMENTS (SHORT TONS/GWe)^a
1260 MWe SSCR

<u>Fuel Cycle</u>	<u>Initial Core</u>	<u>Equilibrium Cycle</u>	<u>30-Year Cumulative</u>
UO ₂ Once-Through Three-Batch	386	175	5,461
UO ₂ Once-Through Five-Batch	390	151	4,769

TABLE 5.2.3-2(b)

SEPARATIVE WORK REQUIREMENTS (METRIC TONS/GWe)^a
1260 MWe SSCR

<u>Fuel Cycle</u>	<u>Initial Core</u>	<u>Equilibrium Cycle</u>	<u>30-Year Cumulative</u>
UO ₂ Once-Through Three-Batch	191	100	3,091
UO ₂ Once-Through Five-Batch	193	100	3,093

^a75% capacity factor (annual refueling); 0.2 wt% tails assay.

TABLE 5.2.3-3

UO₂ Once-Through SSCRTHREE-BATCH AND FIVE-BATCH FUEL CYCLES^aAVERAGE BATCH INVENTORIES

1260 MWe SSCR

<u>Initial Inventories (kg)</u>	<u>Three-Batch Core</u>	<u>Five-Batch Core</u>
U-235	925.1	781.1
U-236	0	0
U-238	33,336.3	19,775.4
Pu-239	0	0
Pu-240	0	0
Pu-241	0	0
Pu-242	0	0
<u>Discharge Inventories (kg)</u>		
U-235	242.2	123.6
U-236	118.1	110.2
U-238	32,459.3	18,958.0
Np-239	3.2	1.9
Pu-239	195.0	132.2
Pu-240	81.0	63.7
Pu-241	48.9	41.0
Pu-242	16.5	18.9

^aEquilibrium Cycle

TABLE 5.2.3-4

APPARENT FISSILE CONSUMPTION (GM FISSILE/MWd)^a1260 MW(e) SSCR

	Generation						Fuel Types
	0 ^b	1	2	3	4	5	
UO ₂ Once-Through Three-Batch	.419	.419	.419	.419	.419	.419	UO ₂ (U-235)
UO ₂ Once-Through Five-Batch	.463	.463	.463	.463	.463	.463	UO ₂ (U-235)
Denatured U-235/ThO ₂	.381	.381	.381	.381	.381	.381	ThO ₂ (D U-235)
	-	.354	.345	.343	.349	.345	ThO ₂ (D U-235 & U-233)
	.381	.357	.348	.345	.351	.347	Average
Denatured U-233/ThO ₂	.202	.191	.191	.195	.194	.192	ThO ₂ (D U-233)
Plutonium/ThO ₂	.485	.462	.466	.486	.491	.499	ThO ₂ (Pu)

^a Apparent Fissile consumption = $\frac{\text{net fissile material consumed during fuel exposure}}{\text{total thermal energy generated during fuel exposure}}$

^b Initial core not included

three-batch core for a given energy output. This results from the lower integral conversion ratio of the five-batch core (about 5% lower than that of the three-batch core as shown in Table 5.2.3-5). The increased loss of neutrons to fission product capture and the poorer eta (η) value of Pu-239 compared to U-235 accounts for this lower conversion ratio in the five-batch core. Nevertheless, in the five-batch core, more of the U-235 loaded is consumed and more of the plutonium bred in the cycle is consumed before the fuel is discharged. These latter facts far outweigh the somewhat poorer neutronic performance of the five-batch core.

5.2.3.2 Denatured U-235/ThO₂ Fuel Cycle

Uranium and separative work requirements of the denatured U-235/ThO₂ fuel cycle are shown in Table 5.2.3-6. Uranium requirements are substantially lower (about 32%) than those of the UO₂ once-through fuel cycles because use of the bred fissile material (i.e., U-233) reduces the amount of U-235 makeup which must be supplied each year. The separative work requirements, however, are about 15% higher than those of the UO₂ once-through cycle due to the use of much higher enrichments in the thorium cycle. In particular, use of 93 wt% uranium to re-enrich the recycle fuel requires about 85% more separative work per kilogram of U-235 product than does 3 wt% uranium. The high separative work per unit of product outweighs the reduced quantity of product, thereby giving a higher overall separative work requirement for the denatured U-235/ThO₂ cycle.

TABLE 5.2.3-5

EQUILIBRIUM CYCLE CORE AVERAGE INTEGRAL CONVERSION RATIOS^a

1260 MW(e) SSCR

<u>Fuel Cycle</u>	<u>Integral Conversion Ratio</u>
UO ₂ Once-Through Three-Batch	0.675
UO ₂ Once-Through Five-Batch	0.644
Denatured U-235/ThO ₂	0.718
Denatured U-233/ThO ₂	0.836
Plutonium/ThO ₂	0.644

^aIntegral Conversion Ratio =
$$\frac{\text{Total fissile atoms bred through capture in fertile materials during exposure}}{\text{Total fissile atoms consumed during fuel exposure}}$$

TABLE 5.2.3-6(a)

U₃O₈ REQUIREMENTS (SHORT TONS/GWe)^aDENATURED U-235/THORIUM CYCLE - PLUTONIUM STORAGE OR SALE

AVERAGE ANNUAL REQUIREMENT

	<u>Initial Core</u>	<u>0^b</u>	<u>Generation</u>					<u>30-Year</u>
			<u>1</u>	<u>2</u>	<u>3</u>	<u>4</u>	<u>5</u>	
20 wt% Makeup	650	259	53	21	17	17	15	1806
93 wt% Re-enrichment	<u>0</u>	<u>0</u>	<u>33</u>	<u>73</u>	<u>79</u>	<u>79</u>	<u>78</u>	<u>1897</u>
Total Requirement	650	259	86	94	96	96	93	3703

TABLE 5.2.3-6(b)

SEPARATIVE WORK REQUIREMENTS (Metric Tons/GWe)^aDENATURED U-235/THORIUM CYCLE - PLUTONIUM STORAGE OR SALE

AVERAGE ANNUAL REQUIREMENT

	<u>Initial Core</u>	<u>0^b</u>	<u>Generation</u>					<u>30-Year</u>
			<u>1</u>	<u>2</u>	<u>3</u>	<u>4</u>	<u>5</u>	
20 wt% Makeup	587	234	48	19	15	15	13	1632
93 wt% Re-enrichment	<u>0</u>	<u>0</u>	<u>35</u>	<u>73</u>	<u>81</u>	<u>79</u>	<u>78</u>	<u>1883</u>
Total Requirement	587	234	83	92	96	94	91	3515

NOTES: (a) 75% capacity factor (annual refueling) 0.2 wt% tails assay.

(b) Zeroth generation represents cycles 2 and 3 reloads.

The better uranium utilization of the denatured U-235/ThO₂ cycle compared to the UO₂ once-through cycle is also reflected in its lower apparent fissile consumption (cf. Table 5.2.3-4) and higher conversion ratio (cf. Table 5.2.3-5). The apparent fissile consumption of the denatured U-235/ThO₂ core is only 17% less than that of the once-through, three-batch core. It should be remembered, however, that the apparent fissile consumption as defined only takes into account the fissile consumed during operation. For the once-through fuel cycle, the fissile material discharged is not recovered and hence is equivalently consumed. If this material were added to that consumed during operation, the apparent fissile consumption for the standard once-through fuel cycle would be 0.889 gm/MWd. The denatured U-235/ThO₂ core would then have an apparent fissile consumption which was 61% less than the once-through cycle. Taking into account refabrication and reconversion losses, and inventory changes from generation to generation in the thorium core, the 61% lower apparent fissile consumption of the thorium core translates into approximately a 54% ore savings compared to the once-through fuel cycle when credit is taken for all fissile material.

The resource requirements listed in Table 5.2.3-6 do not include credits for the fissile material in the ex-core part of the fuel cycle, for the fissile material residing in the final core (i.e., at the end of 30 years operation), or for the fissile plutonium which is not recycled. Table 5.2.3-7 shows the effect of including these credits on the overall resource requirements. By including these credits, an additional saving of about 30% can be obtained for both uranium and separative work. It should be noted

TABLE 5.2.3-7(a)

DENATURED U-235/ThO₂ U₃O₈ REQUIREMENTSEFFECT OF INCLUDING END OF LIFE AND PLUTONIUM CREDITS

	<u>Short Tons/GWe^a</u>
30-Year Cumulative Requirement	3703
Fissile Plutonium Credit ^b	520
Ex-Core Credit ^c	320
Final Core Credit ^d	<u>340</u>
30-Year Net Requirement	2523

TABLE 5.2.3-7(b)

DENATURED U-235/ThO₂ SEPARATIVE WORK REQUIREMENTSEFFECT OF INCLUDING END OF LIFE AND PLUTONIUM CREDITS

	<u>Metric Tons/GWe^a</u>
30-Year Cumulative Requirement	3515
Fissile Plutonium Credit ^b	500
Ex-Core Credit ^c	280
Final Core Credit ^d	<u>290</u>
30-Year Net Requirement	2445

Notes: (a) 75% capacity factor (annual refueling); 0.2 wt% tails assay
 (b) Fissile plutonium produced during 30 year lifetime
 (c) Two years of discharged fuel excluding plutonium
 (d) Fuel remaining in final core at end of thirtieth year

that these net resource requirements are representative of the potential long term benefits of recycling all the bred fissile material. It takes a number of years after the 30-year lifetime of a particular reactor before the credits listed in Table 5.2.3-7 can be realized because of the lag time involved with fully utilizing the available fissile material.

The fuel inventories for the makeup and recycle fuel are shown in Tables 5.2.3-8(a) and 5.2.3-8(b), respectively. As can be seen from the data presented in this table, the enrichment of the recycle fuel gradually increases with each pass of recycle fuel through the core (i.e., with each generation). This increase in fissile loading is required to compensate for the buildup of the minor uranium isotopes, primarily U-234 and U-236. As the fissile loading increases with each succeeding generation, so does the U-238 loading. The higher U-238 loading coupled with the harder neutron spectrum of the latter generations results in the production of about 7% more plutonium in generation five than in generation one.

As was mentioned previously, the recycle fuel fraction in the denatured U-235/ ThO_2 cycle is very large because of the addition of highly-enriched uranium as a topping material to the recycle fuel. As seen in Table 5.2.3-9, the recycle fuel eventually constitutes about 94% of the total fuel in the core. For this reason, it may be more expedient to make the entire reload fuel batch of the same composition. This can be accomplished by adding more highly enriched uranium as a topping re-enriching material to the recycle fuel. The enrichment of this topping material would have to be somewhat less than the 93 wt% used in the present scheme in order that the proliferation limit on the overall fissile content of the fuel can still be met.

TABLE 5.2.3-8(a)

INVENTORIES FOR DENATURED U-235/ThO₂ MAKEUP FUEL

	<u>Initial Inventory</u> <u>(Kg/Initial TeM)</u>	<u>Discharge Inventory</u> <u>(kg/Initial TeM)</u>
U-232	0	0.029
U-233	0	10.559
U-234	0	1.296
U-235	41.089	13.477
U-236	0	4.979
U-238	164.365	156.713
Pa-233	0	0.954
Pu-239 ^a	0	1.869
Pu-240	0	0.570
Pu-241	0	0.557
Pu-242	0	0.169
Th-232	794.545	773.541

TABLE 5.2.3-8(b)
AVERAGE INVENTORIES FOR DENATURED U-235/ThO₂ RECYCLE FUEL

	GENERATION				
	<u>1</u>	<u>2</u>	<u>3</u>	<u>4</u>	<u>5</u>
Fissile Enrichment (wt%)	3.78	3.79	3.87	3.93	3.99
Initial Inventories (kg/Initial TeM)					
U-232	0.030	0.045	0.052	0.057	0.060
U-233	12.689	15.499	15.779	16.259	16.745
U-234	1.560	3.332	4.694	5.751	6.967
U-235	25.091	22.398	22.906	23.019	23.134
U-236	5.488	8.033	9.826	11.546	13.653
U-238	173.370	177.239	177.886	178.751	182.809
TH-232	781.792	773.496	768.908	764.610	756.626
Discharge Inventories (kg/Initial TeM)					
U-232	0.044	0.052	0.056	0.057	0.059
U-233	13.845	14.591	14.806	14.878	15.118
U-234	3.153	4.559	5.576	6.377	7.339
U-235	8.656	8.259	8.887	9.178	9.608
U-236	7.789	9.725	11.381	12.955	14.867
U-238	165.440	169.238	169.925	170.786	174.792
Pu-239 ^a	1.942	2.008	2.048	2.084	2.144
Pu-240	0.608	0.619	0.623	0.621	0.633
Pu-241	0.571	0.576	0.582	0.579	0.588
Pu-242	0.172	0.166	0.169	0.156	0.156
Th-232	761.049	753.274	748.918	745.076	737.343
Pa-233	0.955	0.931	0.916	0.898	0.886

a. Includes Np-239

TABLE 5.2.3-9

FUEL TYPE FRACTIONS FOR THE DENATURED U-235/ThO₂ CYCLE

1260 MW(e) SSCR

Fuel Type	GENERATION					
	0	1	2	3	4	5
Denatured U-235/ThO ₂ Makeup Fuel	1.000	0.106	0.082	0.064	0.064	0.057
Denatured U-235/ThO ₂ Recycle Fuel	0	0.894	0.918	0.936	0.936	0.943

The denatured U-233/ThO₂ cycle is fueled entirely with U-233 supplied from some outside source (e.g., a plutonium/ThO₂ burner reactor or a core blanket composed of thorium). The worth of U-233 as a thermal reactor fuel can best be seen through the low apparent fissile consumption and high conversion ratio of this fuel cycle compared to cycles employing other isotopes as their primary fissile species. As shown in Table 5.2.3-4, the apparent fissile consumption of this fuel cycle is 45% smaller than that of its nearest competitor, the denatured U-235/ThO₂ cycle, and 54% smaller than that of the UO₂ once-through cycle. Similarly, the integral conversion ratio of this fuel cycle (cf. Table 5.2.3-5) is 16% higher than that of its nearest competitor and 24% higher than that of the UO₂ once-through cycle. The excellent performance of this fuel is a result of the small capture-to-fission ratio of U-233 (less than half that of U-235) and its relatively constant value over the range of neutron energy spectra encountered in the SSCR.

The fissile requirements of the denatured U-233/ThO₂ fuel cycle are shown in Table 5.2.3-10. The quantity of U-233 loaded and discharged each year is essentially constant once recycle begins in year four. This results in an almost constant annual makeup requirement of about 250 kg U-233/GWe. The 30-year U-233 makeup requirement shown in Table 5.2.3-10 does not include a credit for the two ex-core batches or for the fuel contained in the final core. With these credits included, the 30-year U-233 requirement is only about 7,400 kg/GWe compared to the 9,940 kg/GWe which is shown in Table 5.2.3-10.

TABLE 5.2.3-10
DENATURED U-233/ThO₂ FUEL CYCLE
AVERAGE ANNUAL FISSILE REQUIREMENTS
(kg/GWe)

	Initial	GENERATION						30-Year
	<u>Core</u>	<u>0</u>	<u>1</u>	<u>2</u>	<u>3</u>	<u>4</u>	<u>5</u>	<u>Requirement</u>
U-233 Loaded	1704	715	706	708	711	716	716	22,344
U-233								
Discharged ^a	426	452	466	468	469	477	478	14,023
U-233 Makeup								
Requirement ^{a,b}	1704	715	261	249	250	254	246	9,940
Plutonium								
Fissile								
Discharged ^c	35	71	71	73	74	75	76	2,145
Plutonium								
Total								
Discharged ^c	43	95	94	97	99	100	101	2,848

NOTES: (a) Includes Pa-233

(b) Includes fuel cycle process losses; ex-core fuel and final core inventory credits not included.

(c) Losses due to reprocessing and Pu-241 decay not included.

The isotopic composition of the denatured U-233/ThO₂ fuel is shown in Table 5.2.3-11 as a function of recycle generation. Because of the small capture-to-fission ratio of U-233, the higher uranium isotopes are slow to buildup, resulting in a fissile loading requirement which is almost constant with recycle generation. The increase in fissile enrichment is about 2% over the six generations (generation zero to generation five) shown in Table 5.2.3-11. The production of plutonium is about 15% greater in the denatured U-233/ThO₂ cycle than in the denatured U-235/ThO₂ cycle because of the higher U-238 loading required to maintain the U-233 content of the uranium component below 12 wt% as compared to the 20 wt% limit for U-235.

TABLE 5.2.3-11

INVENTORIES FOR SSCR DENATURED U-233/ThO₂

	GENERATION					
	0	1	2	3	4	5
Fissile						
Enrichment (wt%)	2.84	2.87	2.87	2.88	2.90	2.90
Initial Inventories (kg/Initial TeM)						
U-232	0.066	0.066	0.066	0.066	0.066	0.066
U-233	28.459	28.651	28.716	28.849	29.038	29.037
U-234	2.301	5.506	7.851	9.619	10.936	12.009
U-235	0.771	1.253	1.879	2.387	2.782	3.089
U-236	0.014	0.215	0.496	0.818	1.157	1.489
U-238	211.942	208.461	209.690	210.618	211.312	211.745
Th-232	756.484	755.880	751.302	747.644	744.707	742.565
Discharge Inventories (kg/Initial TeM)						
U-232	0.045	0.063	0.063	0.062	0.062	0.062
U-233	17.487	17.929	18.035	18.125	18.424	18.503
U-234	4.821	7.131	8.889	10.241	11.277	12.116
U-235	1.148	1.753	2.267	2.666	2.970	3.211
U-236	0.221	0.497	0.828	1.168	1.503	1.821
U-238	202.301	199.029	200.240	201.186	201.926	202.371
Pu-239a	2.197	2.229	2.291	2.337	2.374	2.398
Pu-240	0.745	0.712	0.739	0.738	0.741	0.742
Pu-241	0.664	0.637	0.668	0.668	0.672	0.673
Pu-242	0.219	0.201	0.201	0.193	0.090	0.188
Th-232	734.524	734.241	730.170	726.974	724.212	722.234
Pa-233	0.984	0.966	0.942	0.923	0.914	0.905

a. Includes Np-239

5.2.3.4

Plutonium/ThO₂ Fuel Cycle

The nuclear properties of the fissile plutonium isotopes are such that the plutonium/ThO₂ fuel cycle exhibits the highest apparent fissile consumption (cf. Table 5.2.3-4) and the lowest conversion ratio (cf. Table 5.2.3-5) of any of the cycles examined. In particular, the capture-to-fission ratios of Pu-239 and Pu-241 in the thermal energy range are about four times that of U-233 and twice that of U-235. This results in a rather large consumption of fissile plutonium per unit of energy extracted.

The fissile requirements of the plutonium/ThO₂ cycle are summarized in Table 5.2.3-12. The 30-year fissile plutonium makeup requirement is over two and one-half times the size of the U-233 makeup requirement in the denatured U-233/ThO₂ cycle. This much larger fissile makeup requirement is due to two factors: (1) the higher conversion ratio in the U-233 fueled cycle and (2) the fact that the core fissile inventory is increasing each generation (about 50% over the six generations).

The isotopic composition of the plutonium/ThO₂ fuel is shown in Table 5.2.3-13 as a function of recycle generation. The increase in core fissile inventory can readily be seen by the increase in fissile enrichment from one recycle generation to the next. The cause of this inventory increase is the buildup of the non-fissile plutonium isotopes, Pu-240 and Pu-242. The isotope Pu-242 is particularly detrimental since it acts as a poison to the system. The isotope Pu-240, although causing the inventory to increase, helps the fuel resource requirements through conversion to Pu-241.

TABLE 5.2.3-12

PLUTONIUM/ThO₂ FUEL CYCLE
AVERAGE ANNUAL FISSILE REQUIREMENTS
(kg/GWe)

	Initial	GENERATION						30-Year
	<u>Core</u>	<u>0</u>	<u>1</u>	<u>2</u>	<u>3</u>	<u>4</u>	<u>5</u>	<u>Requirement</u>
Plutonium Loaded								
Fissile	2410	991	1203	1306	1410	1484	1550	42,254
Total	3410	1402	1953	2269	2556	2792	3000	75,063
Plutonium Discharged								
Fissile	297	394	517	623	716	790	853	19,252
Total	525	810	1153	1444	1708	1930	2126	45,320
Plutonium Makeup ^{a, b}								
Fissile	2410	991	834	823	828	813	798	26,625
Total	3410	1402	1177	1165	1173	1153	1134	37,492
U-233 Discharged ^c	172	307	301	296	290	287	283	8,642

-
- NOTES: (a) Includes fuel cycle process losses and 2 year decay of Pu-241.
(b) Ex-core inventory credit and final core credit not included.
(c) Includes Pa-233; fuel cycle process losses not included.

TABLE 5.2.3-13
INVENTORIES FOR SSCR PLUTONIUM/ThO₂

	GENERATION					
	<u>0</u>	<u>1</u>	<u>2</u>	<u>3</u>	<u>4</u>	<u>5</u>
Fissile						
Enrichment						
(wt%)	4.48	4.88	5.30	5.72	6.02	6.28
Initial Inventories (kg/Initial TeM)						
Pu-239	36.533	35.605	37.089	39.656	41.400	43.096
Pu-240	15,328	23,866	29.403	34.326	38.897	42.968
Pu-241	8.266	13.195	15.910	17.552	18.817	19.802
Pu-242	3.265	6.605	9.669	12.179	14.199	15.864
Th-232	936.607	920.729	907.928	896.286	886.686	878.269
Discharge Inventories (kg/Initial TeM)						
U-232	0.021	0.019	0.016	0.015	0.014	0.013
U-233	11.557	11.383	11.272	11.027	10.898	10.769
U-234	1.126	1.036	0.972	0.903	0.866	0.832
U-235	0.177	0.162	0.151	0.138	0.131	0.124
U-236	0.009	0.008	0.006	0.005	0.005	0.005
Pu-239	8.138	9.991	12.487	14.735	16.559	18.206
Pu-240	12.478	18.245	23.131	27.958	32.213	36.132
Pu-241	7.843	10.993	12.781	14.325	15.501	16.428
Pu-242	4.203	7.345	9.851	11.937	13.676	15.112
Th-232	916.985	902.065	889.863	878.959	869.762	861.719
Pa-233	0.883	0.825	0.786	0.748	0.728	0.710

The increased plutonium inventory in later generations has a secondary effect on the fuel performance of the cycle. Because of the higher plutonium loading, about 8% less U-233 is produced in generation five than is produced in generation zero. Approximately 6% of this reduction can be attributed to the thorium displaced by plutonium as the inventory increases. The remaining 2% can be attributed to the harder neutron spectrum with the more highly loaded plutonium fuel. The reduced thermal flux with the higher plutonium loading results in less thermal captures in Th-232, causing conversion to U-233 to decrease.

Since the economics of power generation determines, to a large extent, a utility's choice of reactor systems, the power costs of the spectral shift control reactor are a measure of the SSCR's competitiveness in the nuclear power industry. The power costs of a nuclear generating station are influenced by many factors, including plant capital costs, the cost of money (interest rates) and fuel costs. Each of these quantities is inconstant and to properly evaluate the power costs, their variability should be included in the analysis. However, it is impossible to model the intangibles such as government policy and technological change which so often are the major factors that affect these variables. Hence, thirty-year levelized costs were calculated for two sets of fixed cost assumptions. Separate calculations were performed to measure the sensitivity of the fuel cycle costs (and hence the power costs) to price variations in ore, fabrication and reprocessing.

To obtain power costs for the SSCR, it is necessary to determine the capital costs, operation and maintenance costs, and the fuel cycle costs for the various fuel cycles investigated in this study. The sections which follow describe the individual components and total power costs for each of the fuel cycles analyzed.

5.2.4.1

Capital Costs

The capital cost estimates for the SSCR are divided into two categories: the capital costs of a typical large current design PWR and the additional

costs required to realize spectral shift control. The PWR cost estimates are based on cost development by United Engineers and Constructors (UE&C) as part of a government sponsored study to facilitate policy and economic decisions regarding electric generating stations in the public and private sectors (1). The PWR costs calculated by UE&C were determined for a 1139 MW(e) plant in terms of July 1976 dollars. These capital cost estimates were escalated using an 8% per year rate to obtain costs in terms of January 1978 dollars and were subsequently scaled (using a 0.7 factor), for a 1270 MW(e) plant to estimate the cost of the reference PWR. Approximately \$26 million was added to the PWR costs to account for the inclusion of spectral shift control in a PWR design. The total SSCR base capital costs are summarized in Table 5.2.4-1. The capital cost increase of an SSCR relative to a conventional PWR is listed in more detail in Table 5.2.4-2. A more detailed breakdown of these add-on costs is contained in Appendix 7.8 - PLANT COST ESTIMATES.

A major contributor to the increased costs in the SSCR is the additional reactor plant equipment. Of the \$16 million direct costs shown in Table 5.2.4-2, approximately \$6.7 million is required for heavy water upgrader system and \$2.3 million for the modifications of systems which facilitate coolant management (e.g., the Primary Grade Water System, the Chemical & Volume Control System, and the Boron Recovery System).

The net electrical output of the SSCR is estimated to be 1260 MW(e). The reduction of about 10 MW(e), relative to the reference PWR, is a consequence of the additional pumping power required to circulate the higher density

TABLE 5.2.4-1
CAPITAL COST ESTIMATE SUMMARY
FOR 1260 MW(e) SSCR^a
(JANUARY 1978 DOLLARS)

<u>Acct. No.</u>	<u>Account Description</u>	<u>PWR Costs</u>	<u>SSCR Addon Costs</u>	<u>Total Costs</u>
20	Land and Land Rights	2,414,000	-	2,414,000
21	Structures and Improvements	122,364,600	5,469,250	127,833,850
22	Reactor Plant Equipment	161,116,300	9,277,900	170,394,200
23	Turbine Plant Equipment	134,330,600	892,400	135,223,000
24	Electric Plant Equipment	47,591,500	246,000	47,837,500
25	Miscellaneous Plant Equipment	14,247,200	474,200	14,721,400
26	Main Cond. Heat Reject System	26,057,800	-	26,057,800
91	Construction Services	84,532,600	5,085,000	89,617,600
92	Main Office Engineering and Service	59,410,500	1,564,000	60,974,500
93	Field Office Engineering and Service	34,546,700	-	34,546,700
	SSCR Contingencies ^b	-	2,661,750	2,661,750
	Owner's Costs	42,376,200	-	42,376,200
	Total Costs	728,988,100	25,670,500	754,658,600

^aExcludes interest during construction.

^bContingencies for modifications necessary to realize spectral shift control.

TABLE 5.2.4-2

1260/ W(e) SSCR CAPITAL COST INCREASE RELATIVE TO PWR (JANUARY 1978 DOLLARS)

Account Number	Account Description	FACTORY		SITE			
		Cost, \$	Labor Hours	Labor Cost, \$	Material Cost, \$	Total Site Cost, \$	Total Cost, \$
21	Structures and Improvements	-	209,603	2,492,600	2,976,650	5,469,250	5,469,250
22	Reactor Plant Equipment excluding D ₂ O Upgrader	1,830,320	48,213	571,496	220,114	791,610	2,550,900
228	D ₂ O Upgrader	2,797,000				3,930,000	6,727,000
23	Turbine Plant Equipment	786,400	9,122	106,000		106,000	892,400
24	Electric Plant Equipment	150,000	7,988	96,000		96,000	246,000
25	Miscellaneous Plant Equipment	361,200	9,709	113,000		113,000	474,200
2	Total Direct Cost	5,924,920	284,635	3,379,096	3,196,764	10,505,860	16,359,750
91	Construction Services						5,085,000
92	Home Office Engineering Service						1,564,000
9	Total Indirect Cost						6,649,000
	Contingency						
	A/E						1,403,750
	D ₂ O Upgrader						1,258,000
	Total Cost (Direct, Indirect and Contingency)	5,924,920	284,635	3,379,096	3,196,764	10,505,860	25,670,500

coolant and the low pressure steam and electrical utilities needed to operate the D₂O upgrader. The total base capital cost of the SSCR is approximately \$599/kW(e), excluding interest and escalation during construction and PWR contingency estimates.

To establish power costs, interest during construction (IDC) must be added to the base capital cost of a power generating station. The IDC depends on several quantities including construction time, payment schedule and the cost of money. Assuming a construction period of 10 years, a deflated effective interest rate of 4.5% and using a payment schedule from Ref. (2), the interest during construction amounts to approximately 19% of the base capital cost. When interest during construction is included, the capital cost in January 1978 dollars is \$715/kW(e). The capital cost estimate given above is based on constant dollars excluding escalation. The use of constant dollars results in a cost estimate which appears to be considerably less than that often quoted in the literature. If escalation is included, the plant costs may be a factor of two higher than those quoted here. The capital cost of a plant delivered at some future date, therefore, will exceed \$715/kW(e) when viewed in terms of the then current dollars because of escalation. For example, the cost of a plant for 1990 operation would be \$1540/kW(e) in 1990 dollars, assuming an 8% escalation rate. However, a 1990 dollar would only be worth about 40% of the value of a 1978 dollar if there was an 8% inflation rate.

The ground rules used to prepare the capital costs for both the reference PWR and the SSCR are summarized below.

1. The costs were developed in January 1978 dollars for plants deliverable on January 1, 1978.
2. The estimate was developed for a single plant located with sufficient land area to accommodate a second unit.
3. The costs were based on a "Middletown U.S." site with its associated labor and material costs.
4. The design of the main heat rejection system is based on the use of mechanical draft evaporative cooling towers.
5. The plant has onsite storage capacity for four-thirds of a nuclear reactor core.
6. The plant design lifetime is 30 years.
7. The U.S. licensing and design criteria are utilized to the greatest possible extent.
8. The following items were not included.
 - a. Main transformer, switchyard and transmission facility costs.
 - b. Escalation during construction.
 - c. Initial fuel loading and heavy water inventory.

- d. Off-site waste disposal.
- e. Nuclear liability insurance.
- f. Contingency allowance (PWR).

5.2.4.2 Operation and Maintenance Costs

The operation and maintenance (O&M) costs were not independently calculated. It is logical, however, to use the PWR operation and maintenance costs as a basis. Additional costs must be added to account for D₂O losses and maintenance on systems with higher radiological risks due to the presence of tritium.

The D₂O losses during operation are not expected to be significant from an economic standpoint. However, plant maintenance will be more difficult in view of the higher release of tritium. Ref. (3) gives a value of 1.76 mills/kWh in January 1978 dollars for the O&M costs of a 1270 MW(e) PWR operating with a 75% capacity factor. This O&M cost should be viewed as that of a mature, standardized industry. In 1976, O&M costs for nuclear power plants were about 2.3 mills/kWh averaged over the U.S. utility industry. However, this average included a number of plants which had major maintenance expenses. If these high maintenance plants are excluded, the industry averaged O&M costs drop to about 1.2 mills/kWh. Presumably, by standardizing plant designs and by learning through experience with earlier designs, the O&M costs can be reduced below those achieved in 1976. The operation and maintenance costs for the SSCR are estimated to be about 1.85 mills/kWh. This represents an increase of about 0.09 mills/kWh relative to the O&M costs for a conventional PWR.

The same near term and long term economic assumptions used in the SSCR optimization study were used here to compute the individual fuel cycle costs. These cost sets have been established by the U.S Department of Energy for its Nonproliferation Alternative Systems Assessment Program (NASAP) and are summarized in Table 5.2.4-3. The near term cost assumptions represent the economic environment of the present nuclear industry but with developed fuel cycle processes, e.g., reprocessing and refabrication. The U_3O_8 cost is relatively low, \$40/lb, while reprocessing and refabrication costs are high. The long term costs reflect a nuclear industry which is well established. Reprocessing and fabrication facilities have high throughputs so that the economies of scale result in more economical unit costs. Under the long term cost assumptions, the price of uranium is \$100/lb, reflecting the increased scarcity of U_3O_8 and the reliance on lower grade ores to supply the growing demand.

The fuel cycle costs presented here are based on thirty-year levelized cost calculations. All fuel cycles were analyzed using the same cost assumptions and fuel cycle cost computer program. The characteristics of each batch, including material inventories for feed and discharge, and batch burnup values were used in the fuel cycle cost calculations. For fuel cycles utilizing recycled fuel, i.e., denatured $U-235/ThO_2$ cycle, the denatured $U-233/ThO_2$ and the plutonium/ ThO_2 cycle, a value was assigned to the bred fissile material. One method of defining a value for bred fuel is through indifference calculations. These calculations determine

TABLE 5.2.4-3

ECONOMIC AND RESOURCE PARAMETERS

	Near Term ^a	Long Term ^b
Net Power Rating, MW(e)		
SSCR	1260	
PWR	1270	
Nominal Capacity Factor, %	75	
U ₃ O ₈ Costs, \$/lb	40	100
Thorium, \$/lb	15	
Conversion, \$/kg	4	
Separative Work, \$/kg SWU	100	
Fabrication Costs, \$/kg HM		
LE U5-U8	106	
DE U5-U8	139	123
DE U3-U8	664	574
Pu-Th	438	379
ThO ₂ Reprocessing Costs, \$/kg HM	241	154
Shipping and Waste Storage, \$/kg HM	85	
Throwaway Cost, \$/kg HM	150	
D ₂ O Price, \$/lb.	80	
Enrichment Tails Assay, wt%	0.2	
Reactor Plant Life, years	30	
Discount Rate, %	4.525	
Fixed Charge Rate, %	10	
Carrying Charge Rate, %	8	

^aAssumes a developed industry^bAssumes high capacity industry

the value of the bred fuel at which the reactor operator is indifferent as to whether he sells the fuel or recycles it in his reactor. Furthermore, the indifference value is the price below which an operator would be willing to buy the bred fuel on the open market.

The indifference calculations consist of determining the value of the bred fuel which yields equal fuel cycle costs for a reactor recycling or selling this material. The value of the bred material determined by this method is a function of how (where) it is used and of the particular economic parameters assumed in the calculations. As it becomes more expensive to use the bred material (e.g., as the refabrication unit costs increase), the value of this material decreases. Similarly, as the bred material displaces more costly fuel (e.g., as the price of U_3O_8 increases), the value of the bred material increases.

Indifference values were calculated for fissile plutonium by determining at what plutonium price the fuel cycle costs for the UO_2 once-through SSCR, with a credit for recovered uranium and plutonium, equaled the fuel cycle costs for the self-generated plutonium recycle SSCR. The plutonium indifference values so determined were \$17.42/gm Pu fissile and \$46.24/gm Pu fissile for the near-term and long-term cost assumptions, respectively.

Indifference values were also calculated for the U-233 recycled in the denatured U-235/ThO₂ fuel cycle. This calculation consisted of determining at what U-233 price the fuel cycle costs of the denatured U-235/ThO₂ once-through cycle (with credits for U-233, residual U-235, and fissile plutonium) were equal to the fuel cycle costs of the denatured U-235/ThO₂ cycle with recycle of recovered uranium. Since plutonium is not recycled in this fuel cycle, credits for discharged plutonium, equal to the plutonium indifference values given above, were assigned. (Figure 5.2.4-1 shows the fuel cycle cost curves used to determine the U-233 indifference values). The values of U-233 obtained by this method were \$4.50/gm and \$42.00/gm for the near-term and long-term cost assumptions, respectively.

5.2.4.3.1 Thirty-Year Levelized Fuel Cycle Costs

The thirty-year component levelized costs are summarized in Table 5.2.4-4 for the UO₂ once-through fuel cycles and the denatured U-235/ThO₂ fuel cycle. The improved UO₂ once-through fuel cycle (five-batch, extended-burnup fuel cycle) is the most economical for both sets of cost assumptions because of its low uranium ore requirements and its smaller throughput of fuel. This is shown in the UF₆ charges which are 15.6% lower than the standard UO₂ once-through cycle (three-batch, standard burnup fuel cycle) and in the lower fabrication and spent fuel disposal costs. The levelized fuel cycle costs for the denatured U-235/ThO₂ cycle are 3.0 mills/kWh higher than those of the improved UO₂ once-through cycle in the near-term and 2.0 mills/kWh in the long term. The cost penalty for this cycle is primarily due to the very high fabrication costs of the U-232 bearing recycle fuel. The fabrication costs for the denatured U-235/ThO₂ cycle are

Figure 5.2.4-1

DENATURED U^{235}/ThO_2 SSCR
U-233 INDIFFERENCE VALUES

TABLE 5.2.4-4

SSCR THIRTY YEAR LEVELIZED
FUEL CYCLE COSTS (mills/kWh)^a

<u>Fuel Cycle</u>	<u>UF₆</u>	<u>SWU</u>	<u>Thorium</u>	<u>Fab</u>	<u>Plutonium Credit</u>	<u>Back End^b Direct</u>	<u>Indirect</u>	<u>Total Fuel Cycle Cost</u>
UO ₂ Once-Through	2.24	1.52	-	0.47	-	0.63	0.70	5.56
Three-Batch	(5.48)	(1.52)	-	(0.47)	-	(0.63)	(1.37)	(9.47)
UO ₂ Once-Through	1.89	1.45	-	0.30	-	0.39	0.86	4.89
Five-Batch	(4.63)	(1.45)		(0.30)		(0.39)	(1.62)	(8.39)
Denatured U-235/ThO ₂								
Sale or Storage of Plutonium	1.44	1.78	0.11	1.96	-0.17	1.23	1.50	7.85
	(3.64)	(1.78)	(0.11)	(1.67)	(-0.45)	(0.90)	(2.66)	(10.31)

NOTES:

- a Values without parentheses are for near term cost assumptions, while those with parentheses are for long term cost assumptions.
- b Disposal costs for UO₂ once-through fuel cycles; reprocessing costs for denatured U-235/ThO₂ fuel cycle.
- c Near term Pu value = \$17.42/gm; U-233 value = \$4.50/gm.
Long term Pu value = \$46.24/gm; U-233 value = \$42.00/gm.

about six times those of the improved UO_2 once-through cycle. The backend (reprocessing) costs and the indirect costs (carrying charges) also contribute significantly to the higher costs of the denatured U-235/ ThO_2 fuel cycle.

Fuel cycle costs for the denatured U-233/ ThO_2 cycle are shown in Table 5.2.4-5 and Figure 5.2.4-2. In Table 5.2.4-5, the component costs are given for an assumed U-233 value of \$30/gm. At this fissile material charge, the fabrication costs represent over 40% of the total fuel cycle cost; the value of U-233 would have to be about \$60/gm before the U-233 charge would equal the fabrication charge. Since the value of U-233 is somewhat arbitrary without a systems analysis of the nuclear economy in which it is used, the fuel cycle costs of the denatured U-233/ ThO_2 SSCR are shown as a function of U-233 cost in Figure 5.2.4-2. In this figure, two curves are shown: one for the near term cost assumptions and the other for the long term cost assumptions. To obtain fuel cycle costs comparable to those of the improved UO_2 once-through cycle, the cost of U-233 would have to be less than or equal to about \$10/gm and \$69/gm for the near-term and long-term cost assumptions, respectively.

TABLE 5.2.4-5

DENATURED U-233/ThO₂ SSCR30-YEAR LEVELIZED FUEL CYCLE COSTS

(mills/kWh)

	<u>U-233</u> <u>CHARGE</u>	<u>FABRICATION</u>	<u>THORIUM</u>	<u>BACK-END</u> <u>DIRECT</u>	<u>PLUTONIUM</u> <u>FISSILE</u> <u>CREDIT</u>	<u>INDIRECT</u> <u>CHARGE</u>	<u>TOTAL</u> <u>FUEL CYCLE COST</u>
Near Term Cost Assumptions							
Pu = \$17.42/gm fissile	1.19	2.63	0.11	1.24	-0.19	1.39	6.37
U-233 = \$30/gm							
Long Term Cost Assumptions							
Pu = \$46.24/gm fissile	1.19	2.27	0.11	0.91	-0.51	1.45	5.42
U-233 = \$30/gm							

Figure 5.2.4-2
DENATURED U-233/ThO₂ SSCR FUEL CYCLE COSTS

Fuel cycle costs for the plutonium/ ThO_2 cycle are shown in Table 5.2.4-6 and Figure 5.2.4-3. As for the denatured U-233/ ThO_2 cycle, the component costs are given in Table 5.2.4-6 for an assumed value of the primary fissile fuel (i.e., plutonium for this cycle). For a plutonium cost of \$30/gm fissile, the fissile charge represents the major component of the fuel cycle costs, 40% in the near-term and over 50% in the long-term. This large fractional contribution to the total fuel costs is a result of the high fissile consumption of the plutonium/ ThO_2 transmutter reactor. The next largest component of the fuel cycle costs is that due to carrying charges (i.e., indirect charges). The magnitude of this component is a consequence of the large fissile inventory required in this highly absorbing fuel. Again since the value of plutonium is a function of the overall nuclear system, the fuel cycle costs of the plutonium/ ThO_2 SSCR have been parameterized as a function of plutonium fissile cost. Figure 5.2.4-3 shows the fuel costs for both the near-term and long-term cost assumptions. From this figure it can be seen that in order to obtain fuel cycle costs comparable to those of the improved UO_2 once-through cycle, plutonium must be available for less than about \$11/gm fissile in the near-term and \$44/gm fissile in the long-term.

By using the fuel cycle costs for the denatured U-233/ ThO_2 cycle and the plutonium/ ThO_2 cycle an indifference type calculation can be defined which gives equal fuel cycle costs for these two cycles. Four sets of U-233 and fissile plutonium values are selected and fuel cycle costs determined for each set of values. The intersections of the fuel cycle cost curves defines the values of the bred material where cost equivalence is obtained. Figure 5.2.4-4 shows the functional relationship between the values of U-233 and

TABLE 5.2.4-6

PLUTONIUM/ThO₂ SSCR30-YEAR LEVELIZED FUEL CYCLE COSTS

(mills/kWh)

	PLUTONIUM FISSILE <u>CHARGE</u>	<u>FABRICATION</u>	<u>THORIUM</u>	BACK-END <u>DIRECT</u>	U-233 <u>CREDIT</u>	INDIRECT <u>CHARGE</u>	TOTAL <u>FUEL CYCLE COST</u>
Near Term Cost Assumptions							
U-233 = \$4.50/gm	3.27	1.73	0.13	1.17	-0.23	1.99	8.06
Pu = \$30/gm fissile							
Long Term Cost Assumptions							
U-233 = \$42.00/gm	3.27	1.50	0.13	0.86	-1.93	2.29	6.12
Pu = \$30/gm fissile							

Figure 5.2.4-3
PLUTONIUM/ ThO_2 SSCR FUEL CYCLE COSTS

Figure 5.2.4-4

FUNCTIONAL RELATIONSHIP BETWEEN FISSILE PLUTONIUM
VALUE AND U-233 VALUE FROM DENATURED U-233/ThO₂
AND PLUTONIUM/ThO₂ FUEL CYCLES

fissile plutonium determined in this way. The value of U-233 is about 33% higher than the value of fissile plutonium. Essentially the same result is obtained using the near-term cost assumptions or the long-term cost assumptions.

5.2.4.3.2 Fuel Cycle Cost Sensitivity

The effects of variations in cost parameters on thirty-year levelized fuel cycle costs were investigated. The price of recycle fuel fabrication, reprocessing, U_3O_8 , and fissile material credit were individually varied and the cost calculations were rerun for each perturbation of the long term cost set. Table 5.2.4-7 summarizes the results of these calculations. The following observations concerning the data in this table are worth noting:

- a. The denatured U-235/ ThO_2 cycle is less sensitive to the price of U_3O_8 than the improved UO_2 once-through cycle because of the lower ore demands of the former cycle. However, the reduction is only about half that expected from the lower uranium demands. This is a result of the high initial loads required in the denatured U-235/ ThO_2 cycle compared to the improved UO_2 once-through cycle. Because of the present worthing of early expenditures, the full benefit of the lower uranium requirements does not translate into an equivalent economic benefit.
- b. The denatured U-233/ ThO_2 cycle has the greatest sensitivity to a change in the recycle fuel fabrication cost. This results from the fact that all fuel loaded in this cycle is fabricated as recycle fuel.

TABLE 5.2.4-7
FUEL CYCLE COST SENSITIVITY-
CHANGE IN FUEL CYCLE COSTS^a
(mills/kWh)

<u>Fuel Cycle</u>	<u>U₃O₈</u>	<u>Recycle Fuel Fabrication</u>	<u>Reprocessing Costs</u>	<u>Fissile^b Credit</u>
				<u>Increased</u>
	<u>Increased 60%</u>	<u>Decreased 25%</u>	<u>Decreased 25%</u>	<u>\$20/gm</u>
UO ₂ Once-Through Improved	+3.51	-	-	-
Denatured U-235/ThO ₂	+3.14	-0.45	-0.19	-0.16
Denatured U-233/ThO ₂	-	-0.66	-0.19	-0.18
Plutonium/ThO ₂	-	-0.44	-0.19	-0.75

- NOTES: a. Base costs for this comparison are those using long-term cost assumptions including Pu value of \$46.24/gm fissile and U-233 value of \$42.00/gm.
- b. Fissile plutonium for the denatured U-235/ThO₂ cycle and the denatured U-233/ThO₂ cycle; U-233 for the plutonium/ThO₂ cycle.

The cost saving from the initial batches, in particular, makes this fuel cycle more sensitive than the denatured U-235/ThO₂ cycle. Since the refabrication costs for the denatured U-233/ThO₂ cycle are about 50% higher than those of the plutonium/ThO₂ cycle, the sensitivity of the former cycle to changes in refabrication costs are about 50% greater.

- c. The sensitivity to changes in the cost of reprocessing is the same for each cycle. This results from the fact that approximately the same amount of fuel is reprocessed each year. Also, the reprocessing unit costs were assumed equal for the three thorium cycles.
- d. The plutonium/ThO₂ cycle has the greatest sensitivity to the value of the bred fissile material which is not recycled. In this fuel cycle, over three times the quantity of nonrecycled fissile material is bred. This leads to a fissile credit for this cycle which is much larger than that of the other fuel cycles.

5.2.4.4 Heavy Water Costs

The heavy water inventory costs were based on two assumptions: a 60 wt% heavy water concentration at beginning of cycle and a cost of \$80/lb D₂O. For some of the fuel cycles analyzed, the initial heavy water concentration was below this assumed value. The actual D₂O concentration required to control reactivity depends on the specific fuel cycle, the fuel management scheme, and on the exact procedure utilized for coolant dilution. For example, it appears practical to use soluble boron rather than D₂O to control the reactivity associated with the buildup of equilibrium xenon following the

recovery from cold shutdown. This procedure, if adopted, would reduce somewhat the D_2O inventory requirements. Using this assumed value was felt to be a conservative practice and allowed one single value for the cost of heavy water. The \$80/lb D_2O cost is below present costs for reactor grade heavy water. However, since the SSCR requires less concentrated D_2O , 60 wt% versus 99.97 wt%, it should be possible to realize a saving in the heavy water production cost. This is particularly true if a facility designed specifically to produce less than reactor grade D_2O were utilized. Based on the above two assumptions and the reactor plant inventory summary in Section 5.1.10 - D_2O INVENTORY, the heavy water inventory cost amounts to approximately \$54 million. This represents a cost of 0.65 mills/kWh when a 10% fixed charge rate is used.

5.2.4.5 Total Power Costs

Total power costs for the SSCR are obtained by adding the capital costs (including interest during construction), the O&M costs, the fuel cycle costs and the heavy water costs. These costs are summarized in Table 5.2.4-8 for the fuel cycles analyzed in this study. Since the only component cost which varies for the different fuel cycles is the fuel cycle cost, the same trends as observed for the fuel cycle costs occur for the power costs. The power costs shown for the denatured U-233/ThO₂ cycle and the plutonium/ThO₂ cycle are based on fuel cycle costs for which the primary fissile species was arbitrarily assigned a value of \$30/gm. To obtain a more realistic intercomparison of power costs, a more representative value for the fissile material must be used. This value should be based on a systems analysis of the overall nuclear economy in which supplier and user reactors (of the various bred fissile materials) are included.

TABLE 5.2.4-8

TOTAL BUSBAR POWER COSTS FOR A 1260 MW(e) SSCR^a

(mills/kWh)

	CAPITAL	O&M	FUEL CYCLE	D ₂ O	POWER
<u>FUEL CYCLE</u>	<u>COST</u>	<u>COST</u>	<u>COSTS</u>	<u>COST</u>	<u>COSTS</u>
UO ₂ Once-Through	10.88	1.85	5.56	0.65	18.94
Three-Batch			(9.47)		(22.85)
UO ₂ Once-Through	10.88	1.85	4.89	0.65	18.27
Five-Batch			(8.39)		(21.77)
Denatured U-235/ThO ₂ ^{b,d}	10.88	1.85	7.85	0.65	21.23
Sale or Storage of Plutonium			(10.31)		(23.69)
Denatured U-233/ThO ₂ ^{b,c}	10.88	1.85	6.37	0.65	19.75
Sale or Storage of Plutonium			(5.42)		(18.80)
Plutonium/ThO ₂ ^{d,e}	10.88	1.85	8.06	0.65	21.44
Sale of U-233			(6.12)		(19.50)

- NOTES:
- The values without parentheses are for the near-term cost assumptions while the values with parentheses are for the long-term cost assumptions.
 - Plutonium value of \$17.42/gm fissile in near-term and \$46.24/gm fissile in long-term.
 - U-233 value of \$30/gm used.
 - U-233 value of \$4.50/gm in near-term and \$42.00/gm in long-term.
 - Plutonium value of \$30/gm used.

The feasibility of SSCR deployment can best be measured through a comparison with light water reactor performance. The SSCR can be developed using presently known engineering principles and with relatively few modifications to present PWR design as shown in Section 5.1 - DESCRIPTION OF REFERENCE DESIGN. This section examines the competitiveness of SSCR relative to the PWR in terms of resource requirements and power costs.

5.2.5.1

Comparison of Resource Requirements

The thirty-year U_3O_8 requirements of the SSCR and the PWR are given in Table 5.2.5-1 for the UO_2 once-through and the denatured U-235/ ThO_2 fuel cycles. For both the standard once-through cycle and the five-batch core, utilization of spectral shift control reduces uranium requirements about 11%, averaged over thirty years. The uranium requirements of a SSCR operating with denatured U-235/ ThO_2 are similarly about 11% lower than the PWR requirements. The percentage gain is limited in this fuel cycle by the fact that a large fraction of the bred fissile material (i.e., fissile plutonium) is not recycled. If full use of all bred material were employed the resource savings relative to the PWR would be larger. The separative work requirements for the three U-235 fueled cycles are summarized in Table 5.2.5-2; the same trend in separative work requirements as observed in U_3O_8 requirements is evident.

TABLE 5.2.5-1
SSCR AND PWR COMPARISON
THIRTY-YEAR U_3O_8 REQUIREMENT^a
 (SHORT TONS U_3O_8 /GW(e))

	<u>INITIAL CORE</u>	<u>EQUILIBRIUM CYCLE</u>	<u>THIRTY-YEAR REQ.</u>
UO_2 Once-Through, Three-Batch			
PWR	394	194	6040
SSCR	386	175	5461
UO_2 Once-Through, Five Batch			
PWR	471	171	5395
SSCR	390	151	4769
Denatured U-235/ ThO_2 ^b			
PWR	664	113	4180
SSCR	650	93	3703

NOTES:

^a0.20 wt% diffusion plant tails assay; 75% capacity factor (annual refueling).

^b Sale or storage of bred plutonium.

TABLE 5.2.5-2
SSCR AND PWR COMPARISON
THIRTY-YEAR SEPARATIVE WORK
REQUIREMENTS^a (METRIC TONS/GW(e))

	<u>INITIAL CORE</u>	<u>EQUILIBRIUM CYCLE</u>	<u>THIRTY-YEAR REQ.</u>
UO ₂ Once-Through, Three-Batch			
PWR	200	117	3600
SSCR	191	100	3091
UO ₂ Once-Through, Five Batch			
PWR	264	118	3670
SSCR	193	100	3093
Denatured U-235/ThO ₂ ^b			
PWR	603	107	4000
SSCR	587	91	3515

NOTES:

^a0.20 wt% diffusion plant tails assay; 75% capacity factor (annual refueling).

^b Sale or storage of bred plutonium.

A comparison of SSCR and PWR resource requirements for the denatured U-233/ThO₂ and plutonium/ThO₂ fuel cycles is given in Tables 5.2.5-3 and 5.2.5-4, respectively. For the denatured U-233/ThO₂ cycle, 17% less U-233 is required for the SSCR over the 30-year lifetime while producing an additional 16% fissile plutonium. The total net fissile requirements of the SSCR are about 27% lower than those of the PWR when both U-233 and fissile plutonium are considered.

In contrast to the significantly improved performance of the SSCR operating on the denatured U-233/ThO₂ cycle, the plutonium/ThO₂ fueled SSCR shows only marginal improvement over the PWR. The net 30-year fissile plutonium requirement of the SSCR is only 1% lower than that of the PWR, while the U-233 production is 11% higher. The total net fissile requirements of the SSCR are thus only about 8% lower than those of the PWR.

TABLE 5.2.5-3

DENATURED U-233/ThO₂ FUEL CYCLESSCR AND PWR COMPARISON OF FISSILE REQUIREMENTS

(kg/GWe)

	PWR	SSCR
U-233 REQUIRED		
Initial Core Charge	1884	1704
Equilibrium Cycle Makeup	303	245
30-Year Cumulative	11500	9940
30-Year Net ^a	8970	7405
PLUTONIUM DISCHARGED		
Equilibrium Cycle	65	76
30-Year Cumulative	1845	2145
30-Year Cumulative With Final Core Credit ^b	1940	2252

Notes:

- a. Includes credit for U-233 in ex-core fuel batches and in fuel in final core.
- b. Includes credit for plutonium in partially burnt fuel in final core.

TABLE 5.2.5-4

PLUTONIUM/ThO₂ FUEL CYCLESSCR AND PWR COMPARISON OFFISSILE REQUIREMENTS

(kg/GWe)

	PWR	SSCR
FISSILE PLUTONIUM REQUIREMENTS		
Initial Core Charge	2465	2410
Equilibrium Cycle Makeup	781	798
30-Year Cumulative	27054	26625
30-Year Net ^a	22177	21990
U-233 DISCHARGED		
Equilibrium Cycle	256	283
30-Year Cumulative	7778	8642
30-Year Cumulative With Final Core Credit ^b	8066	8965

NOTES:

- a. Includes credit for fissile plutonium in ex-core fuel batches and in fuel in final core.
- b. Includes credit for U-233 in partially burnt fuel in final core.

The thirty-year levelized power costs are summarized in Table 5.2.5-5. Using the near term cost assumptions, the most economical SSCR is the once-through five batch core. However, its power costs are still 0.47 mills/kWh above that of the five-batch PWR. The savings in fuel cycle costs due to the lower uranium and SWU demands of the SSCR are offset by the heavy water costs and, hence, the fuel cycle costs (when the costs of heavy water are included) are almost identical for the once-through fuel cycles. For the long term cost assumptions the SSCR is competitive with the PWR. The higher cost of U_3O_8 (\$100/lb) in this cost set results in a fuel cycle cost savings which is sufficient to offset the D_2O costs and the higher capital and O&M costs.

No cost comparison is made for the denatured $U-233/ThO_2$ and the plutonium/ ThO_2 fuel cycles because of the strong sensitivity to the price of the primary fissile material. Based on the fuel resource utilization of the two cycles, it is felt that the denatured $U-233/ThO_2$ fueled SSCR will be competitive with the similarly fueled PWR while the plutonium/ ThO_2 fueled SSCR will not.

TABLE 5.2.5-5
COMPARISON OF TOTAL POWER COSTS
FOR PWR AND SSCR^a
(mills/kWh)

	PWR				SSCR			
	Capital Cost	O&M	Fuel Cycle	Total	Capital Cost	O&M	Fuel Cycle ^b	Total
Once-Through	10.50	1.76	6.20	18.46	10.88	1.85	6.20	18.94
Three-Batch			(10.59)	(22.85)			(10.12)	(22.85)
Once-Through	10.50	1.76	5.54	17.80	10.88	1.85	5.54	18.27
Five-Batch			(9.50)	(21.76)			(9.04)	(21.77)
Denatured	10.50	1.76	8.25	20.51	10.88	1.85	8.50	21.23
U-235/ThO ₂ SSCR ^c			(10.95)	(23.21)			(10.96)	(23.69)

^aValues without parentheses are for the short term cost assumptions, while those in parentheses are for long term cost assumptions.

^bD₂O cost included in fuel cycle costs.

^cIndifference values used for fissile plutonium and U-233.

REFERENCES FOR SECTION 5.2

"Resource Utilization and Economics of SSCR Reference Design"

1. "Capital Cost: Pressurized Water Reactor Plant", NUREG-0241, June 1977.
2. D. R. Haffner, et. al., "Thorium Assessment Program Studies", TC-1064, Hanford Engineering Development Laboratory, February 9, 1978.
3. Draft Version - NASAP Provisional Data Base for Second Rolling Report, DOE-FCE, July 21, 1978.

5.3

FUEL MANAGEMENT IN THE SSCR

5.3.1

INTRODUCTION

Reactivity control in the conventional PWR is achieved by the use of soluble boron in the reactor coolant to capture the excess neutrons generated throughout the operating cycle. The soluble boron concentration is adjusted in proportion to the excess reactivity of the core which is generally highest at beginning of cycle. As excess reactivity is reduced by fuel depletion and fission product buildup, the soluble boron concentration is likewise reduced through dilution with pure light water.

In the SSCR, however, reactivity control is achieved by the introduction of heavy water into the reactor coolant. Due to the poorer moderating properties of heavy water, the neutron spectrum is shifted to higher energies. At higher D_2O concentrations, neutrons are less likely to thermalize and more likely to be captured epithermally in the resonances of the fertile material (predominantly U-238 in uranium-based fuels and Th-232 in thorium-based fuels). The heavy water concentration is gradually decreased during the operating cycle by dilution with light water.

The heavy water concentration required to control the excess reactivity of the core is dependent on the fuel type loaded, the cycle being considered and the fuel management strategy employed. The heavy water concentration at the beginning-of-cycle (BOC) of the initial core is typically 50 to 60 wt% with equilibrium xenon built-in. This concentration decreases

almost linearly with burnup to about 2 wt% at end-of-cycle (EOC). The 2 wt% EOC heavy water concentration was selected as a practical lower limit which can be economically attained. Very little additional reactivity is tied up in this last 2 wt% heavy water. In addition, sizing requirements of the D₂O upgrader become prohibitive as the heavy water concentration becomes lower. After the initial cycle, the buildup of fission products and other parasitic isotopes reduces the BOC heavy water requirement. This ranges from about 38 wt% for UO₂ once-through cycle to 59 wt% for the denatured U-233/ThO₂.

Since the SSCR is basically a modified PWR, the fuel management aspects of the two reactors are quite similar. Partial batch refueling, with fresh fuel loaded on the periphery of the core, is the dominant theme in the fuel management of either reactor. The reactor core for both concepts is housed in a pressure vessel which is sealed during power production. Periodically, the reactor is shutdown and the reactor vessel is opened for replacement of spent fuel. This provides a strong economic incentive to refuel as infrequently and as quickly as possible. Due to annual load variations, utilities traditionally refuel their reactors once a year at low-demand periods. At these times, typically one-third of the fuel, that with the highest exposure, is replaced. This batch size was selected as a compromise between ease of fuel management and economics.

In addition to the conventional three batch fuel management scheme, an improved scheme employing five in-core fuel batches and extended burnup will be discussed in this section. This fuel management scheme was shown

in Section 5.2 to give a significant improvement in uranium utilization compared with the traditional three batch scheme for the UO_2 once-through fuel cycle. The results of calculations for the SSCR employing both three and five batch fuel management will be compared with similar calculations for the conventional PWR.

All spatial calculations used in the analysis of the SSCR were performed with a coarse mesh (16 nodes per assembly), four neutron energy group PDQ model. Since an actual assembly contains a 16 by 16 fuel rod array with five large (2 by 2 array locations) water holes, the one pin power peaks could not be calculated directly with this model, but were interpolated from a spline fit of the coarse mesh data. Consequently, the coarse mesh model could not explicitly determine the power distribution near water holes in the way a fine mesh model can. This shortcoming can result in an underprediction of pin peaks near the water holes. However, experience has shown that in PWR calculations the coarse mesh pin peaking values are typically within 3 percent of fine mesh calculations where each fuel pin and water hole are represented explicitly.

5.3.2 UO_2 ONCE-THROUGH FUEL CYCLES

5.3.2.1 UO_2 Once-Through SSCR (Three Batch)

Three batch fuel management for the once-through SSCR closely resembles that for the PWR. Annually, one third of the fuel, that with the highest exposure, is removed from the core. This fuel generally has spent its last two operating cycles resident near the center of the core where once and

twice burned fuel are loaded in approximately a checkerboard pattern. The fresh fuel is placed on the periphery of the core to help achieve a more uniform power distribution. Since the entire fresh batch cannot be accommodated on the core periphery, it is often split into two sub-batches of differing enrichments and reactivities, the lower enrichment fuel being loaded into interior core locations.

In the remainder of this section, data is presented for the initial three cycles of operation for the SSCR. By the end of cycle three, the core is in a near-equilibrium state and the isotopic depletion given in Section 5.3.3.1 adequately describes the fuel's behavior.

Due to the low average enrichment of the initial core (which results from the relatively low fission product inventory throughout the cycle) and the subsequent rapid replacement of high reactivity U-235 by low reactivity plutonium, the reactivity of this core falls off quickly. If the equilibrium cycle length of 10,130 MWd/t were employed, high reactivity splits between fresh and once burned fuel would result at the beginning of Cycle 2, leading to unacceptably high core power peaking. In order to avoid this consequence, sufficient fissile material is loaded to extend the first cycle from 10,130 MWd/t to 12,250 MWd/t. The increase in excess reactivity of Cycle 1 over Cycle 2 is apparent from the higher D_2O rundown curve in Figure 5.3.2-1. The second cycle loading is reduced in order to obtain a shortened cycle lifetime of 8000 MWd/t. This decrease in Cycle 2 loading reduces the reactivity split between fresh and burnt fuel and thus reduces the core power peaking factors. The combination of the two cycle lengths leaves the accumulated burnup throughout two cycles unchanged at about 20,250 MWd/t.

Cycles 3 and beyond are loaded to achieve an equilibrium cycle exposure of 10,130 MWd/t. The procedure of lengthening first cycle burnup and shortening second cycle burnup to obtain acceptable second cycle power distributions is used for both the three and five batch UO_2 once-through SSCR fuel cycles. Similar cycle energy distribution are also used in the analogous PWR fuel cycles.

Fuel loading patterns and beginning of cycle (with equilibrium xenon) power distributions for Cycles 1 through 3 of the three-batch UO_2 once-through SSCR are given in Figures 5.3.2-2 through 5.3.2-4. Beginning of cycle power distributions are shown in these figures since the highest peaking factors usually occur at this time and subsequently burn out with fuel exposure. These power distributions are comparable to those of the conventional PWR. Table 5.3.2-1 shows resident fuel inventories at beginning and end of cycles 1 through 3 for the three-batch UO_2 once-through SSCR. This table also contains data for a comparable PWR which illustrates the lower makeup enrichments and the higher plutonium production rates of the SSCR as compared to the PWR. The end of cycle fissile loadings are somewhat higher for the SSCR because a larger fraction of this loading consists of plutonium, particularly Pu-239. The value of eta (η) for Pu-239 is less than that for the other fissile isotopes, requiring more of this material for a given reactivity. Table 5.3.2-2 contains detailed material inventories for the fuel loaded in the initial core and discharged at the end of Cycles 1 through 3.

Figure 5.3.2-1
D₂O CONCENTRATION vs CYCLE EXPOSURE UO₂ ONCE THROUGH SSCR
(THREE BATCH FUEL MANAGEMENT)

Box Peaking Factor: 1.30
Pin Peaking Factor: 1.44

- C* - Fresh UO₂ Fuel
(Split High Enrichment -
1.94 w/o)

5-307

FIGURE 5.3.2-3 UNRODDED PLANAR POWER DISTRIBUTION SECOND CYCLE

MAXIMA

Box Peaking Factor: 1.23

Pin Peaking Factor: 1.37

A - UO ₂ Fuel Irradiated One Cycle
(Low Enrichment - 1.44 w/o)	D	3
B - UO ₂ Fuel Irradiated One Cycle
(Middle Enrichment - 1.94 w/o)	D	4
B* - UO ₂ Fuel Irradiated One Cycle
(Split High Enrichment-1.75 w/o)	D	5
C - UO ₂ Fuel Irradiated One Cycle
(Split High Enrichment-2.90 w/o)		
C* - UO ₂ Fuel Irradiated One
Cycle (Split High
Enrichment - 1.94 w/o)		
D - Fresh UO ₂ Fuel
(High Enrichment)		
D* - Fresh UO ₂ Fuel
(Low Enrichment)		

[illegible]

TABLE 5.3.2-1

NEW CORE FUEL MANAGEMENT
CHARACTERISTICS UO₂ ONCE-THROUGH
(THREE BATCH) CYCLE 3800 MW(th)

		<u>First Cycle</u>		<u>Second Cycle</u>		<u>Third Cycle</u>	
		<u>SSCR</u>	<u>PWR</u>	<u>SSCR</u>	<u>PWR</u>	<u>SSCR</u>	<u>PWR</u>
Cycle Lengths (Full Power Days)		331	328	230	243	274	284
Average Makeup U-235 Enrichment (wt%)		2.03	2.18	2.70	3.17	2.70	2.95
Core Inventory (kg)							
U-235	BOC	2085	2198	1854	1990	2001	2145
	EOC	1134	1168	1282	1347	1306	1390
Pu-239	BOC	0	0	303	235	297	253
	EOC	454	359	473	383	498	426
Pu-240	BOC	0	0	76	59	75	67
	EOC	124	100	146	126	157	140
Pu-241	BOC	0	0	38	26	40	32
	EOC	63	45	81	62	89	74
Pu-242	BOC	0	0	6.3	4.2	7.9	7.0
	EOC	11.3	7.9	20.3	16.6	23.8	21.5

TABLE 5.3.2-2

MATERIAL INVENTORIES FOR UO_2 ONCE-THROUGH SSCR INITIAL
CORE (THREE BATCH FUEL MANAGEMENT)

Batch Designation	A	B	B*	C	C*
Number of Assemblies	81	68	12	72	8
Initial Enrichment (wt% U-235)	1.44	1.94	1.75	2.90	1.94
Loading (kg)	34543	28999	5117	30705	3411
Discharge Exposure (MWd/t)	12366	22022	20929	30762	29075
Initial Inventories (kg/Initial TeM)					
U-235	14.258	19.211	17.329	28.717	19.211
U-238	985.742	980.789	982.671	971.283	980.789
Discharge Inventories (kg/Initial TeM)					
U-235	6.240	5.722	5.093	8.873	3.801
U-236	1.409	2.323	2.512	3.647	2.549
U-238	972.103	959.272	961.423	945.883	952.945
Pu-239	4.544	5.305	5.227	5.392	5.000
Pu-240	1.418	2.093	2.052	2.388	2.588
Pu-241	0.723	1.198	1.191	1.466	1.489
Pu-242	0.147	0.364	0.366	0.476	0.615

A straight-forward way to increase fuel utilization in a once-through cycle is to increase the discharge burnup of the fuel. As burnup is increased, the uranium requirements decreased because of the better in situ utilization of uranium and plutonium. With increased burnup, the fuel operates for a longer period before it is discharged from the reactor and hence the quantity of fissile material unproductively residing in spent fuel is reduced. By increasing the number of fuel batches resident in the core, a lower core average burnup at end of cycle can be obtained for a given discharge exposure. Batch loaded cores operate until the buildup of parasitic fission products (which is proportional to burnup) reduces core reactivity to the point where refueling is necessary. Reducing the core average exposure, therefore, reduces the fission product burden and allows the fuel to attain higher burnups before refueling is required. For these two reasons, an improved fuel management scheme was studied which employed five batches with an in-core residency time of five years or 50,650 MWd/t. This fuel management scheme allows annual refueling to be maintained with 20 percent of the core replaced on each reload compared to the conventional 33 percent. As discussed in Section 5.2.1, the five-batch fuel management scheme showed significantly better uranium utilization than the conventional three batch scheme for both the SSCR and the PWR.

Since the cycle lifetime of a reactor with no previous exposure is primarily determined by the fissile loading of the core and only secondarily by the arrangement of the fissile material, the required fuel enrichment of the initial core for the five batch fuel management scheme is essentially the

same as for the three batch fuel management scheme. The D_2O concentrations required to control the excess reactivity throughout the five batch first cycle are, therefore, very similar to those required for the standard three batch fuel management scheme.

In later cycles, however, the leakage is somewhat higher for the five-batch core because of the higher feed enrichment of the reload fuel loaded on the core periphery. This trend of higher core leakage also persists in later cycles. The reactivity of the fresh fuel located on the core periphery is increased, resulting in the production of more power in the peripheral assemblies with the higher feed enrichment (3.8 wt% versus 2.7 wt% in the three batch core). The higher neutron leakage which results from this increased power production near the core boundary reduces the required D_2O concentration a few percent throughout the cycle. In addition, the higher plutonium inventory of the core which results from the increased core average exposure contributes to the lower D_2O requirements of the five batch core compared to the three batch core. The D_2O rundown curves for Cycles 1 through 5 are shown in Figure 5.3.2-5.

As with the three batch core, in order to obtain acceptable power distributions in the second cycle, the first cycle was lengthened while the second cycle was shortened a corresponding amount. Higher feed enrichments than the three batch core were required in order to attain the higher discharge burnups (higher energy extraction) and to accommodate the large accumulation of fission products. Annual equilibrium requirements for this core were determined to be 48 assemblies of 3.80 wt% enrichment; this compares to an annual requirement of 80 assemblies enriched to 2.7 wt% for the three batch core. The fuel management characteristics for the initial five

Figure 5.3.2-5
D₂O CONCENTRATION vs CYCLE EXPOSURE
UO₂ ONCE-THROUGH SSCR (FIVE BATCH FUEL MANAGEMENT)

cycles of the five batch UO_2 once through SSCR are summarized in Table 5.3.2-3. Also shown for comparison are the first five cycles of the five batch PWR. As with the three batch SSCR, the lower fissile loading requirements and the higher rates of plutonium production are readily apparent from the data presented in this table.

Fuel loading patterns and beginning of cycle (with equilibrium xenon) power distributions for Cycles 1 through 5 of the five batch SSCR are shown in Figures 5.3.2-6 through 5.3.2-10. Because of the relatively large number of fuel enrichments, the five batch fuel management scheme did not require that the enrichment of the fuel batches be split in order to obtain acceptable power distributions. This occurs for two reasons: first, the increased number of fuel batches adds flexibility in the choice of the loading/shuffling pattern and, secondly, the reduced fuel batch size allows all fresh fuel to be loaded on the periphery of the core.

If one compares the power distributions of the three and five batch SSCRs, two factors which have opposite effects on core power peaking should be kept in mind. The added flexibility obtained by loading extra fuel batches helps to reduce core power peaking factors. This can be seen in the first core where the five batch SSCR has a 5 percent lower core power peaking factor than the three batch SSCR. After the first cycle, the large reactivity splits between the reload fuel batches (U-235 enrichments of 3.80 wt%) and the partially burned initial core fuel batches cause relatively high peaking factors. This condition persists as long as an appreciable amount of initial core (low enrichment) fuel remains in the core. Thus, the power peaking factors for cycles 2 and 3 are higher for the five batch core than for comparable

TABLE 5.3.2-3

FUEL MANAGEMENT CHARACTERISTICSUO₂ ONCE-THROUGH (FIVE BATCH) CYCLE3800 MW(th)

		<u>First Cycle</u>		<u>Second Cycle</u>		<u>Third Cycle</u>		<u>Fourth Cycle</u>		<u>Fifth Cycle</u>	
		<u>SSCR</u>	<u>PWR</u>	<u>SSCR</u>	<u>PWR</u>	<u>SSCR</u>	<u>PWR</u>	<u>SSCR</u>	<u>PWR</u>	<u>SSCR</u>	<u>PWR</u>
Cycle Length (full power days)		331	392	216	181	274	217	274	245	274	274
Average Makeup Fissile Enrichment (wt%)		2.05	2.53	3.80	3.95	3.80	3.95	3.80	4.30	3.80	4.30
Core Inventory (kg)											
U-235	BOC	2104	2508	1794	1950	2000	2161	2150	2352	2259	2495
	EOC	1159	1288	1296	1485	1360	1583	1490	1694	1586	1821
Pu-239	BOC	0	0	358	320	372	316	397	341	419	368
	EOC	440	397	469	403	505	428	528	458	548	485
Pu-240	BOC	0	0	94.5	89.6	112	100	121	107	128	113
	EOC	123	117	159	138	184	150	188	162	193	174
Pu-241	BOC	0	0	46.4	43.4	60.9	51.0	68.0	56.5	74.4	63.7
	EOC	60.9	56.7	86.3	71.0	104	80.1	111.0	90.7	117.0	97.9
Pu-242	BOC	0	0	8.3	8.3	15.1	13.3	18.8	16.8	20.0	17.3
	EOC	11.2	11.2	23.6	19.6	34.0	26.7	37.6	31.7	38.3	35.3

MAXIMA

Box Peaking Factor: 1.21
Pin Peaking Factor: 1.39

A - Fresh UO_2 Fuel
(Enrichment = 1.45 w/o)

B - Fresh UO_2 Fuel
(Enrichment = 1.65 w/o)

C - Fresh UO_2 Fuel
(Enrichment = 1.75 w/o)

D - Fresh UO_2 Fuel
(Enrichment = 2.00 w/o)

E - Fresh UO_2 Fuel
(Enrichment = 3.40 w/o)

Location						E 1	E 2	E 3	E 4	
Peaking						0.60	0.78	0.84	0.85	
Peaking Factor						1.05	1.19	1.23	1.23	
					E 5	E 6	D 7	D 8	B 9	C 10
					0.58	0.88	0.88	1.01	0.94	0.77
					1.05	1.37	1.11	1.18	1.04	1.09
				E 11	D 12	B 13	D 14	B 15	D 16	A 17
				0.64	0.82	0.93	1.16	1.09	1.21	1.00
				1.17	1.09	1.06	1.28	1.13	1.29	1.03
			E 18	D 19	C 20	D 21	B 22	C 23	A 24	C 25
					0.98	1.21	1.14	1.17	1.04	1.15
					1.10	1.32	1.17	1.20	1.06	1.17
		E 26	B 27	D 28	B 29	C 30	A 31	C 32	A 33	
					1.16	1.19	1.05	1.15	1.03	
					1.17	1.21	1.07	1.19	1.04	
E 34	D 35	D 36	B 37	C 38	A 39	C 40	A 41	C 42		
					1.06	1.15	1.02	1.13		
					1.08	1.19	1.03	1.16		
E 43	D 44	B 45	C 46	A 47	C 48	A 49	B 50	A 51		
						1.02	1.10	0.99		
						1.03	1.13	1.01		
E 52	B 53	D 54	A 55	C 56	A 57	B 58	A 59	B 60		
							0.99	1.08		
							1.00	1.10		
E 61	C 62	A 63	C 64	A 65	C 66	A 67	B 68	A 69		
								0.98		
								0.99		

FIGURE 5.3.2-6 UNRODDED PLANAR POWER DISTRIBUTION FIRST CYCLE

UO_2 ONCE THROUGH FIVE BATCH SSCR
(BEGINNING OF CYCLE EQUILIBRIUM XENON)

FIGURE 5.3.2-7

MAXIMA

Bôx Peaking Factor: 1.31

Pin Peaking Factor: 1.46

A - UO_2 Fuel Irradiated One Cycle
(Enrichment = 1.45 w/o)

B - UO_2 Fuel Irradiated One Cycle
(Enrichment = 1.65 w/o)

C - UO₂ Fuel Irradiated One Cycle
(Enrichment = 1.75 w/o)

D - UO₂ Fuel Irradiated One Cycle
(Enrichment = 3.40 w/o)

E - UO₂ Fuel Irradiated One
Cycle
(Enrichment = 3.40 w/o)

F - Fresh UO_2 Fuel

[illegible]

Box Peaking Factor: 1.41
Pin Peaking Factor: 1.45

G - Fresh UO_2 Fuel

Loading Factor					G 1	G 2	G 3	G 4	
Loading Factor					0.89 1.32	1.15 1.40	1.16 1.42	1.15 1.37	
				G 5	G 6	G 7	G 8	C 9	D 10
				0.69 1.09	1.09 1.40	1.08 1.19	1.15 1.23	0.86 0.88	0.88 0.92
Cycles			G 11	D 12	C 13	F 14	C 15	F 16	C 17
			0.65 0.95	0.65 0.77	0.81 0.88	1.41 1.45	0.87 0.91	1.31 1.38	0.86 0.88
Cycles	G 18	D 19	E 20	F 21	D 22	E 23	C 24	F 25	
			0.92 1.03	1.26 1.41	0.92 0.97	1.09 1.19	0.83 0.86	1.32 1.41	
Cycles	G 26	C 27	F 28	C 29	E 30	F 31	D 32	C 33	
				0.82 0.85	1.06 1.17	1.33 1.38	0.85 0.88	0.79 0.81	
G 34	E 35	F 36	D 37	E 38	C 39	D 40	F 41	D 42	
					0.78 0.81	0.84 0.89	1.25 1.33	0.88 0.91	
G 43	E 44	C 45	E 46	F 47	D 48	E 49	D 50	F 51	
						0.95 1.04	0.83 0.88	1.28 1.36	
G 52	C 53	F 54	C 55	D 56	F 57	D 58	E 59	E 60	
							0.99 1.05	1.04 1.10	
G 61	D 62	C 63	F 64	D 65	D 66	F 67	E 68	B 69	
								0.80 0.82	

FIGURE 5.3.2-8 UNRODDED PLANAR POWER DISTRIBUTION THIRD CYCLE
UO₂ ONCE THROUGH FIVE BATCH SSCR
(BEGINNING OF CYCLE EQUILIBRIUM XENON)

FIGURE 5.3.2-9 UNRODDED PLANAR POWER DISTRIBUTION FOURTH CYCLE

MAXIMA

Box Peaking Factor: 1.34
Pin Peaking Factor: 1.38

C - UO₂ Fuel Irradiated Three Cycles
(Enrichment = 1.75 w/o)

D - UO_2 Fuel Irradiated Three Cycles
(Enrichment = 2.00 w/o)

E - UO_2 Fuel Irradiated Three Cycles
(Enrichment = 3.40 w/o)

F - UO₂ Fuel Irradiated Two Cycles

G - UO₂ Fuel Irradiated One Cycle

H - Fresh UO_2 Fuel[illegible]

MAXIMA

Box Peaking Factor: 1.26
Pin Peaking Factor: 1.36

D - UO_2 Fuel Irradiated Four Cycles
(Enrichment = 2.00 w/o)

E - UO_2 Fuel Irradiated Four Cycles
(Enrichment = 3.40 w/o)

F - UO_2 Fuel Irradiated Three Cycles

G - UO_2 Fuel Irradiated Two Cycles

H - UO_2 Fuel Irradiated One
Cycle

I - Fresh UO_2 Fuel

Location									I 1	I 2	I 3	I 4
Peaking Power									0.77	1.03	1.15	1.17
Peaking Power									1.10	1.31	1.36	1.36
	</											

FIGURE 5.3.2-10 UNRODDED PLANAR POWER DISTRIBUTION FIFTH CYCLE

UO_2 ONCE THROUGH FIVE BATCH SSCR

(BEGINNING OF CYCLE EQUILIBRIUM XENON)

cycles of the three batch core. This same trend was observed for a similarly fueled five batch PWR. In general, it is more difficult to obtain acceptable power distributions during the transition cycles for a five batch high burnup core than for a three batch core. Once the low enrichment fuel is discharged from the core, and only equilibrium reload fuel remains, power peaking factors similar to those of the three-batch core can be achieved.

The detailed material inventories for the initial core fuel (i.e., the fuel loaded in the initial core and subsequently discharged during the first five cycles) are shown in Table 5.3.2-4.

TABLE 5.3.2-4
 MATERIAL INVENTORIES FOR UO_2
 ONCE THROUGH SSCR INITIAL CORE
 (FIVE BATCH FUEL MANAGEMENT)

Batch Designation	A	B	C	D	E
Number of Assemblies	49	48	48	48	48
Initial Enrichment (wt% U-235)	1.45	1.65	1.75	2.00	3.40
Loading (kg)	20897	20470	20470	20470	20470
Discharge Exposure (MWd/t)	12411	20205	28587	36685	48453
Initial Inventories (kg/Initial TeM)					
U-235	14.500	16.500	17.500	20.000	34.000
U-238	985.500	983.500	982.500	980.000	966.000
Discharged Inventories (kg/Initial TeM)					
U-235	6.393	4.905	3.343	2.604	4.967
U-236	1.436	1.992	2.335	2.787	4.990
U-238	972.385	963.376	954.265	932.498	926.848
Pu-239	4.467	5.025	5.315	5.469	6.057
Pu-240	1.397	2.037	2.521	2.857	3.096
Pu-241	.696	1.095	1.449	1.718	1.902
Pu-242	.144	.345	.638	.863	.942

The introduction of thorium as the primary fertile component of the fuel gives rise to a number of generic changes in the fuel management characteristics of a reactor. In comparison to uranium based fuel cycles, thorium cycles typically have a larger thermal absorption cross section and a shallower reactivity rundown with fuel depletion. The high thermal absorption cross section is simply due to the presence of Th-232, whose thermal absorption cross section is nearly three times that of U-238. The shallow reactivity rundown with burnup is due to two factors. First, the conversion ratio in thorium fuels is greater than that in uranium fuels. This higher conversion ratio is due to the larger thermal cross section of Th-232 and to the lower capture to fission ratio of U-233, the primary bred fissile species. Secondly, the production of plutonium in the thorium cycles is at most about one-third that produced in uranium cycles. This plutonium production in the thorium cycles is due to the presence of U-238, which is used to denature the fissile uranium loaded with the thorium. Because of the high absorption cross section of plutonium, the total absorption cross section of the fuel increases with burnup in uranium based cycles. The lower rate of plutonium production in thorium based cycles, results in a slower increase in the total absorption cross section of the fuel and thus a slower decrease in reactivity.

This reactivity depletion characteristic results in a fuel-management behavior which is characteristic of all the thorium-based cores evaluated in this report. The initial operating cycle of the thorium cores is shorter than that usually encountered in uranium fuel cycles.

Because of the low initial reactivity of thorium-based fuels, high loadings would be required to achieve the long first cycle typical of UO_2 -fueled cores. Moreover, with these high initial core loadings, the second cycle would be considerably longer than that for the comparable UO_2 core because of the shallow slope of the reactivity depletion curve. Therefore, a more attractive alternative was to employ a shorter first cycle, so that the total energy produced in the initial three cycles was comparable to that of the UO_2 -fueled core.

For all of the thorium cores, equal cycle exposures were selected for the first three cycles, the value selected being equal to the equilibrium 75% capacity factor (annual refueling) exposure. This can be contrasted to a typical UO_2 fuel cycle which has a long first cycle (e.g., 16% longer than the equilibrium) followed by a short second cycle (e.g., 16% shorter than the equilibrium).

Another consequence of the shallow reactivity depletion curve for thorium fuel is the potential for reduced local power variations resulting from inter-assembly reactivity differences. The shallower slope of the reactivity rundown for thorium fuel means that the reactivity of fresh, once-burnt, and twice-burnt fuel is much closer together compared to that of uranium fuel. Power mismatches between assemblies of different batches will tend to be smaller resulting in smaller power peaking factors.

Apart from the above mentioned similarities, the three thorium-based fuel cycles have fuel management characteristics which are quite distinct from one another. These characteristics are dictated, to a large extent, by the role that these cycles would play in a future nuclear economy. For example,

the denatured U-235/thorium cycle is viewed as the near-term option for use of thorium in a proliferation-limited nuclear economy. In this cycle, natural uranium enriched to 20 wt% is blended with thorium oxide to attain the required fissile content. Enriched uranium provides both the initial fuel to start the cycle and the makeup fuel to sustain the cycle. The denatured U-233/thorium cycle, on the other hand, represents the long-term option for use of thorium since significant stockpiles of U-233 do not exist at the present time. A supply of U-233 would have to be developed by use of thorium-fueled converter reactors before this fuel cycle could be deployed. The plutonium/thorium cycle is viewed as a transmutter cycle which takes as its fissile component plutonium generated in a uranium-based fuel cycle or a denatured thorium cycle and converts Th-232 into U-233. In the sections which follow, the fuel management characteristics of these three fuel cycles will be discussed.

5.3.3.1 Denatured U-235/ThO₂ Cycle

The denatured U-235/ThO₂ fuel cycle is characterized by a very large recycle fuel fraction. This is a consequence of using highly enriched uranium as a topping material to re-enrich the recycle uranium. By this method the U-238 content of the denatured fuel is maintained as small as possible; this allows the maximum amount of U-233 and the minimum amount of plutonium to be bred. Fresh 20 wt% denatured U-235/thorium makeup fuel fills the remainder of the batch. The two fuel components are kept segregated from one another because the recycle fuel is highly radioactive while the makeup fuel is not.

Shuffling patterns and beginning of cycle (with equilibrium xenon) power distributions for the denatured U-235/ThO₂ SSCR are shown on Figures 5.3.3-1 through 5.3.3-4 for cycles 1 through 4. When comparing these to those of the three-batch UO₂ once-through SSCR, the similarity of the fuel loading/shuffling patterns is apparent. The first cycle loading patterns are identical to each other while the differences between the shuffling patterns in second and third cycles are minimal. These differences are mostly in the placement of the low enrichment assemblies of the split batches. In comparison to the UO₂ once-through SSCR, beginning of first cycle core power peaking for the denatured U-235/ThO₂ SSCR was reduced by decreasing the reactivity split between the Batch B (resident for two cycles) and Batch C (resident for three cycles) assemblies. While the beginning of first cycle infinite multiplication factor was 7.7% different for the B and C fuel in the UO₂ core, the difference was only 5.5% for the ThO₂ core. This reduction in reactivity split was a result of the shallow slope of the reactivity rundown with burnup for the ThO₂ fuel as discussed previously. In addition, this shallow reactivity rundown reduced the beginning of cycle excess reactivity; hence, the beginning of cycle D₂O concentration was some 14% (cf. Figure 5.3.3-5) lower for this ThO₂ core than for the UO₂ core. The overall effect on these factors was to produce smaller core power peaking factors in the ThO₂ core compared to the UO₂ core for the first cycle.

Comparisons of second and third cycle power distributions for the UO₂ once-through core and the denatured U-235/ThO₂ core are greatly influenced by the distribution of cycle energies. Due to the more rapid reactivity rundown with burnup of the UO₂ fuel, the first cycle is extended and the second

MAXIMA

Box Peaking Factor: 1.30

Pin Peaking Factor: 1.35

- A. - Fresh DU-235/ThO₂ Fuel
(Low Enrichment 2.6 w/o)

- B - Fresh D U-235/ThO₂ Fuel
(Middle Enrichment 3.55 w/o)

- B* - Fresh D U-235/ThO₂ Fuel
(Split Middle Enrichment 3.21 w/o)

- C - Fresh D U-235/ThO₂ Fuel
(High Enrichment 4.50 w/o)

- C* - Fresh D U-235/ThO₂ Fuel
(Split High Enrichment
3.55 w/o)

[illegible]

MAXIMA

Box Peaking Factor: 1.34

Pin Peaking Factor: 1.43

A - D.U-235/ThO₂ Fuel
Irradiated One Cycle
(Low Enrichment 2.6 w/o)

B - D.U-235/ThO₂ Fuel
Irradiated One Cycle
(Middle Enrichment 3.55 w/o)

B* - D.U-235/ThO₂ Fuel
Irradiated One Cycle
(Split Middle Enrichment 3.2 w/o)

C - D.U-235/ThO₂ Fuel
Irradiated One Cycle
(High Enrichment 4.50 w/o)

C* - D.U-235/ThO₂ Fuel
Irradiated One Cycle
(Split High Enrichment
3.55 w/o)

D - Fresh D.U-235/ThO₂
(High Enrichment)

D* - Fresh D.U-235/ThO₂ Fuel
(Low Enrichment)

							D 1	D 2	D 3	D 4
							.46 .76	.59 .86	.66 .90	.67 .90
					D 5	D 6	D 7	B 8	D 9	B 10
					.47 .79	.70 .96	.89 1.07	.85 .97	1.06 1.18	.91 .99
		D 11	D 12	C 13	B 14	C 15	B 16	C 17		
		.52 .84	.83 1.02	1.03 1.22	.99 1.13	1.21 1.34	1.06 1.10	1.23 1.28		
	D 18	D 19	B 20	C* 21	D* 22	B 23	C 24	B 25		
			.89 1.00	1.02 1.16	1.16 1.23	1.14 1.22	1.29 1.33	1.11 1.13		
	D 26	C 27	C* 28	C 29	C 30	C 31	B 32	B 33		
				1.29 1.37	1.30 1.41	1.34 1.43	1.12 1.15	1.10 1.11		
	D 34	D 35	B 36	D* 37	C 38	B* 39	C 40	B 41	C 42	
						1.11 1.14	1.32 1.42	1.10 1.14	1.23 1.31	
	D 43	B 44	C 45	B 46	C 47	C 48	B* 49	C 50	B 51	
							1.06 1.09	1.22 1.33	1.07 1.09	
	D 52	D 53	B 54	C 55	B 56	B 57	C 58	B* 59	C 60	
								1.00 1.03	1.19 1.25	
	D 61	B 62	C 63	B 64	B 65	C 66	B 67	C 68	A 69	
									.91 .94	

FIGURE 5.3.3-2 UNRODDED PLANAR POWER DISTRIBUTION SECOND CYCLE

DENATURED U-235/THORIUM SSCR
(BEGINNING OF CYCLE, EQUILIBRIUM XENON)

MAXIMA

Box Peaking Factor: 1.31

Pin Peaking Factor: 1.36

B - D. U-235/ThO₂ Fuel
Irradiated Two Cycles
(Middle Enrichment 3.55 w/o)

C - D. U-235/ThO₂ Fuel
Irradiated Two Cycles
(High Enrichment 4.50 w/o)

C* - D. U-235/ThO₂ Fuel
Irradiated Two Cycles
(Split High Enrichment 3.55 w/o)

D - D. U-235/ThO₂ Fuel
Irradiated One Cycle
(High Enrichment)

D* - D. U-235/ThO₂ Fuel
Irradiated One Cycle
(Low Enrichment)

E - Fresh D. U-235/ThO₂ Fuel
(High Enrichment)

E* - Fresh D. U-235/ThO₂ Fuel
(Low Enrichment)

							E 1	E 2	E 3	E 4
							.55 .92	.73 1.05	.80 1.08	.78 1.03
			E 5	E 6	E 7	D 8	E 9	C 10		
			.51 .86	.77 1.06	1.02 1.20	1.12 1.25	1.24 1.35	1.06 1.17		
		E 11	E 12	C 13	D 14	C 15	D 16	C 17		
		.58 .94	.91 1.09	.93 1.07	1.17 1.22	1.17 1.28	1.31 1.36	1.17 1.21		
	E 18	E 19	D 20	D 21	C 22	E* 23	C 24	D 25		
			1.08 1.10	1.08 1.09	1.08 1.21	1.13 1.20	1.19 1.28	1.26 1.30		
	E 26	C 27	D 28	C* 29	D 30	C 31	D 32	C 33		
				.93 .94	1.09 1.15	1.05 1.19	1.19 1.25	1.12 1.19		
E 34	E 35	D 36	C 37	D 38	C* 39	D* 40	C 41	D 42		
					.89 .91	.92 .97	1.03 1.08	1.11 1.15		
E 43	D 44	C 45	E* 46	C 47	D* 48	C 49	D 40	C 51		
						.98 1.03	1.03 1.06	1.01 1.03		
E 52	E 53	D 54	C 55	D 56	C 57	D 58	C 59	D 60		
							.97 1.00	1.03 1.05		
E 61	C 62	C 63	D 64	C 65	D 66	C 67	D 68	B 69		
								.90 .91		

FIGURE 5.3.3-3 UNRODED PLANAR POWER DISTRIBUTION THIRD CYCLE
DENATURED U-235/THORIUM SSCR
(BEGINNING OF CYCLE, EQUILIBRIUM XENON)

MAXIMA

Box Peaking Factor: 1.34

Pin Peaking Factor: 1.40

- C - D. U-235/ThO₂ Fuel Irradiated Three Cycles (High Enrichment)
- D - D. U-235/ThO₂ Fuel Irradiated 2 Cycles (High Enrichment)
- D* - D. U-235/ThO₂ Fuel Irradiated Two Cycles (Low Enrichment)
- E - D. U-235/ThO₂ Fuel Irradiated One Cycle (High Enrichment)
- E* - D. U-235/ThO₂ Fuel Irradiated One Cycle (Low Enrichment)
- F - Fresh D. U-235/ThO₂ Recycle Fuel

						F/ 1	F/ 2	F 3	F/ 4
						.53	.71	.79	.77
						.87	1.01	1.10	1.01
				F/ 5	F/ 6	F 7	E 8	F/ 9	D 10
				.46	.71	.98	1.11	1.22	1.06
				.77	.96	1.17	1.28	1.32	1.18
		F 11	F/ 12	D 13	E 14	D 15	E 16	D 17	
		.53	.81	.87	1.15	1.13	1.34	1.17	
		.86	.95	1.01	1.22	1.23	1.40	1.21	
	F/ 18	F/ 19	E 20	E* 21	D 22	F 23	D 24	E 25	
			.98	.91	1.02	1.33	1.17	1.30	
			1.03	.96	1.15	1.38	1.24	1.32	
	F/ 26	D 27	E* 28	D 29	E 30	D* 31	E 32	D 33	
				.98	1.13	1.01	1.25	1.14	
				1.04	1.24	1.09	1.33	1.20	
F/ 34	F 35	E 36	D 37	E 38	D 39	E 40	D 41	E 42	
					1.02	1.11	1.04	1.16	
					1.05	1.17	1.07	1.19	
F/ 43	E 44	D 45	F 46	D* 47	E 48	D 49	E 50	D 51	
						.99	1.10	1.02	
						1.03	1.14	1.03	
F 52	F/ 53	E 54	D 55	E 56	D 57	E 58	D 59	E 60	
							1.00	1.07	
							1.02	1.09	
F/ 61	D 62	D 63	E 64	D 65	E 66	D 67	E 68	C 69	
								1.00	
								1.03	

FIGURE 5.3.3-4 UNRODDED PLANAR POWER DISTRIBUTION FOURTH CYCLE

DENATURED U-235/THORIUM SSCR
(BEGINNING OF CYCLE EQUILIBRIUM XENON)

Figure 5.3.3-5
D₂O CONCENTRATION vs CYCLE EXPOSURE
DENATURED U-235/ThO₂ SSCR

cycle is shortened in order to obtain acceptable second cycle power distributions. This variation of first and second cycle lengths is not done for the ThO_2 core, resulting in a higher second cycle fissile loading (as compared to a case with a shortened second cycle) with attendant high local reactivity variations and core power peaking factors. The core peaking factor at the beginning of cycle 2 is 7% higher for the ThO_2 core than the UO_2 core. By Cycle 3, both cores have equilibrium cycle lengths and the core power peaking factor comparison is much the same as the Cycle 1 comparison - the UO_2 core having a higher core power peaking factor by about 7%.

From Cycle four and later, however, any comparison between the fuel management characteristics of the denatured U-235/ThO₂ SSCR and the three batch UO₂ once-through SSCR breaks down because of the introduction of recycle fuel in the ThO₂ core. The distinction between recycle fuel and makeup fuel is mainly a consequence of the U-233 contained in the recycled uranium. While this isotope is not significant from a neutronics stand-point, its daughters emit high energy gamma rays making it significant from a radiation shielding point of view. Because of the high fabrication cost of the recycle fuel, economics favor the use of segregated recycle; that is, makeup fuel assemblies are fabricated by normal contact methods while recycle assemblies are fabricated by more expensive "hot-cell" techniques. Therefore, starting with the fourth cycle, the feed of the denatured U-235/ThO₂ SSCR is split into two fuel types--makeup fuel and recycle fuel.

The recycle fuel is typically of higher reactivity than the makeup fuel. This dictates that the recycle fuel be loaded in the low flux locations (e.g., on the core periphery) while the makeup is loaded in the high flux locations. Approximately ten percent of the fuel assemblies are makeup fuel assemblies with the remaining assemblies being recycle fuel assemblies. The shuffling pattern and beginning of cycle (with equilibrium xenon) power distribution for Cycle eight of the denatured U-235/ThO₂ SSCR is shown in Figure 5.3.3-6. This loading pattern is representative of an equilibrium core and will generally be repeated in later cycles. The power distribution is relatively flat with a pin peaking factor of only 1.35. This small peaking factor is typical of later cycles and results from the shallow slope of the reactivity rundown curve and from the fact that U-233 (with a fission cross section comparable to that of U-235), rather than Pu-239, replaces the U-235 that is consumed.

MAXIMA

Box Peaking Factor: 1.31
Pin Peaking Factor: 1.35

G - D.U-235/ThO₂ Makeup Fuel
Irradiated Three Cycles

H - D.U-235/ThO₂ Makeup Fuel
Irradiated Two Cycles

H/- D.U-235/ThO₂ Recycle Fuel
Irradiated Two Cycles

I - D.U-235/ThO₂ Makeup Fuel
Irradiated One Cycle

I/- D.U-235/ThO₂ Recycle Fuel
Irradiated One Cycle

J - Fresh D.U-235/ThO₂ Makeup Fuel

J/- Fresh D.U-235/ThO₂
Recycle Fuel

Location					J/ 1 .54 .88	J/ 2 .71 1.01	J/ 3 .77 1.05	J/ 4 .77 1.01		
Leaking Factor				J/ 5 .48 .78	J/ 6 .72 .99	J/ 7 .97 1.15	I/ 8 1.09 1.23	J/ 9 1.20 1.31	H/ 10 1.02 1.13	
Leaking Factor			J/ 11 .54 .86	J/ 12 .83 .98	H/ 13 .87 .97	I/ 14 1.14 1.20	H/ 15 1.13 1.19	I/ 16 1.31 1.35	H/ 17 1.13 1.17	
Location		J/ 18	J/ 19	I/ 20 .99 1.02	I 21 .98 1.03	H/ 22 1.07 1.16	J 23 1.28 1.33	H/ 24 1.19 1.25	I/ 25 1.27 1.30	
1		J/ 26	H/ 27	I 28	H/ 29 .96 .99	I/ 30 1.13 1.21	H/ 31 1.06 1.15	I/ 32 1.23 1.28	H/ 33 1.11 1.16	
1		J/ 34	J/ 35	I/ 36	H/ 37	I/ 38	H/ 39 .99 1.01	I/ 40 1.10 1.14	H/ 41 1.08 1.12	I/ 42 1.15 1.19
p		J/ 43	I/ 44	H/ 45	J 46	H/ 47	I/ 48	H/ 49 1.05 1.07	I/ 50 1.09 1.13	H/ 51 1.01 1.05
		J/ 52	J/ 53	I/ 54	H/ 55	I/ 56	H/ 57	I/ 58	H/ 59 .98 1.02	I/ 60 1.05 1.08
		J/ 61	H/ 62	H/ 63	I/ 64	H/ 65	I/ 66	H/ 67	I/ 68	G 69 .92 .94

The fuel management characteristics of selected cycles for the denatured U-235/ThO₂ core are shown in Table 5.3.3-1. The data presented in this table demonstrates the low plutonium production in denatured thorium fuels, as compared to the once-through uranium cycle (e.g., an EOC core inventory of 187 kg versus 587 kg for the uranium cycle). Also shown is the fact that as U-233 gradually replaces U-235, the total fissile loading at beginning of cycle (BOC) decreases. This decrease in fissile loading is due to the higher worth of U-233 relative to U-235 in a thermal reactor.

The material inventories for fuel loaded in the initial core are shown in Table 5.3.3-2. This data together with that presented in Section 5.2.3.2 gives the 30 year material inventories for the denatured U-235/ThO₂ cycle.

TABLE 5.3.3-1

FUEL MANAGEMENT CHARACTERISTICS
DENATURED U-235/ThO₂ SSCR

	<u>Cycle 1</u>	<u>Cycle 3</u>	<u>Cycle 5</u>	<u>Cycle 7</u>	<u>Cycle 9</u>
Cycle Length (Full Power Days)	274	274	246	246	246
Makeup Fuel Enrichment (wt% U-235)	3.50	4.10	4.10	4.10	4.10
Number of Makeup Assemblies	241	80	16	8	8
Recycle Fuel Enrichment (wt% uranium fissile)	-	-	3.53	3.55	3.60
Number of Recycle Assemblies	-	-	64	72	72
Core Inventory (kg)					
U-233	BOC	0	394	896	1319
	EOC ^(a)	579	781	1105	1385
U-235	BOC	3273	2949	2188	1865
	EOC	2206	2093	1577	1364
Pu-239	BOC	0	98	89	91
	EOC	112	158	151	155
Pu-241	BOC	0	14.5	12.1	11.9
	EOC	11.8	32.5	28.6	28.5

Notes: (a) Includes Pa-233

TABLE 5.3.3-2

MATERIAL INVENTORIES FOR DENATURED U-235/ThO₂ SSCR INITIAL CORE

Batch Designation	A	B	B*	C	C*
Number of Assemblies	81	68	12	72	8
Initial Enrichment (wt%)	2.61	3.56	3.22	4.51	3.56
Loading HM (kg)	31436	26387	4657	27948	3104
Discharge Exposure (MWd/t)	10808	24316	23511	35201	35093
Initial Inventories (kg/Initial TeM)					
Th-232	869.685	822.069	839.115	774.453	822.069
U-234	.208	.285	.257	.361	.285
U-235	26.063	35.585	32.176	45.108	35.585
U-238	104.044	142.061	128.452	180.078	142.061
Discharged Inventories (kg/Initial TeM)					
Th-232	860.197	805.193	821.323	753.749	797.526
U-233 ^(a)	7.297	10.560	10.460	11.510	12.707
U-234	.605	1.172	1.225	1.472	1.749
U-235	15.488	14.549	12.549	15.536	9.647
U-236	1.946	3.856	3.562	5.452	4.599
U-238	101.691	136.538	123.373	171.319	133.902
Pu-239	1.045	1.675	1.477	2.211	1.745
Pu-240	.225	.460	.419	.663	.572
Pu-241	.138	.409	.377	.635	.563
Pu-242	.017	.095	.094	.183	.206

(a) Includes decay of Pa-233

In the denatured U-233/ThO₂ cycle, all initial and makeup fuel consists of U-233 denatured with depleted uranium. As such, every fuel assembly is radioactive, requiring remote fabrication and special handling of fresh fuel. This results from the fact that all practical sources of U-233 contain the contaminant U-232, the daughters of which emit high energy gamma rays. There is no reason, therefore, to segregate recycle and makeup fuel material since all assemblies must be fabricated using remote hot cell techniques. The in-core fuel management is thus less complicated than that of the denatured U-235/ThO₂ cycle since every assembly in the batch is composed of the same material.

The denatured U-233/ThO₂ core has a much higher initial reactivity than that of the denatured U-235/ThO₂ core. The low capture-to-fission ratio of U-233 as compared to that of U-235 is the cause for this high reactivity. This higher initial reactivity is reflected in a greater D₂O concentration at beginning of cycle. The beginning of cycle D₂O concentrations are typically about 15% higher than those for the denatured U-235/ThO₂ core during the early part of the operating cycle. However, since the total fissile loading is smaller in the denatured U-233/ThO₂ core, each atom of U-233 consumed represents a larger loss of reactivity. Therefore, the reactivity drops off much faster in the denatured U-233/ThO₂ core.

The faster reactivity rundown is seen in Figure 5.3.3-7 which shows the D_2O concentration versus cycle exposure for the initial three cycles of the denatured U-233/ThO₂ core.

The fuel loading patterns and beginning of cycle (with equilibrium xenon) power distributions for Cycles one through three are shown in Figures 5.3.3-8 through 5.3.3-10. A strong similarity in the fuel loading patterns of this core and the denatured U-235/ThO₂ core can be seen by comparing these figures to Figures 5.3.3-1 through 5.3.3-3. However, the power peaking factors are greater in the denatured U-233/ThO₂ core than in the denatured U-235/ThO₂ core. This is due to the more rapid fall off of reactivity with burnup in the U-233 fueled core. The resulting mismatch in reactivity of assemblies belonging to different fuel batches causes larger local power variations.

Once recycle begins, similarities between the two denatured thorium cores end. In the denatured U-235/ThO₂ core the fuel loading patterns and fuel compositions change markedly once recycle begins. However, in the denatured U-233/ThO₂ core, later cycles appear very similar to earlier cycles (e.g., cycle three) since the higher uranium isotopes (higher than U-233) are slow to build up because of the low capture cross section of U-233. This makes the cycle-by-cycle fuel management considerably easier in the U-233 fueled core. Once near-equilibrium conditions have been established, the same fuel shuffling pattern applies to all subsequent cycles.

Figure 5.3.3-7
D₂O CONCENTRATION vs CORE EXPOSURE
DENATURED U-233/ThO₂ SSCR

5-341

MAXIMA

Box Peaking Factor: 1.32

Pin Peaking Factor: 1.36

A - Fresh D U-233/ThO₂ Fuel
(Low enrichment 1.78 w/o)

B - Fresh D U-233/ThO₂ Fuel
(Middle enrichment 2.43 w/o)

B*- Fresh D U-233/ThO₂ Fuel
(Split Middle enrichment
2.20 w/o)

C - Fresh D U-233/ThO₂ Fuel
(High Enrichment
2.75 w/o)

C*- Fresh D U-233/ThO₂ Fuel
(Split High enrichment
2.43 w/o)

							C 1	C 2	C 3	C 4
							.54 .85	.66 .91	.67 .94	.67 .91
					C 5	C 6	C 7	C 8	A 9	B 10
					.55 .90	.80 1.12	.99 1.24	1.08 1.28	.79 .88	.96 1.11
				C 11	C 12	B 13	B 14	A 15	B 16	A 17
				.60 .96	.95 1.24	1.05 1.21	1.16 1.28	.97 1.02	1.17 1.27	.95 1.00
			C 18	C 19	A 20	B 21	A 22	C* 23	A 24	B 25
					.89 .98	1.22 1.32	1.04 1.07	1.28 1.34	1.05 1.07	1.27 1.34
			C 26	B 27	B 28	A 29	B 30	A 31	B 32	A 33
						1.05 1.09	1.32 1.36	1.08 1.10	1.31 1.35	1.07 1.09
		C 34	C 35	B 36	A 37	B 38	A 39	B 40	A 41	B 42
							1.08 1.11	1.30 1.36	1.05 1.07	1.27 1.33
		C 43	C 44	A 45	C* 46	A 47	B 48	A 49	B* 50	A 51
								1.03 1.06	1.07 1.13	.97 1.00
		C 52	A 53	B 54	A 55	B 56	A 57	B* 58	A 59	B* 60
									.94 .96	1.00 1.04
		C 61	B 62	A 63	B 64	A 65	B 66	A 67	B* 68	A 69
										.90 .91

FIGURE 5.3.3-8 UNRODDED PLANAR POWER DISTRIBUTION - FIRST CYCLE

DENATURED U-233/THORIUM SSCR

(BEGINNING OF CYCLE EQUILIBRIUM XENON)

Box Peaking Factor: 1.26
Pin Peaking Factor: 1.37

B - D U-233/ThO₂ Fuel Irradiated one cycle
(Middle enrichment 2.43 w/o)

B* - D U-233/ThO₂ Fuel Irradiated
one cycle
(Split middle enrichment
2.20 w/o)

C - D U-233/ThO₂ Fuel
Irradiated one cycle
(high enrichment
2.75 w/o)

C* - D U-233/ThO₂ Fuel
Irradiated one cycle
(split high enrichment 2.43 w/o)

D - Fresh U-233/ThO₂ Fuel
(high enrichment)

D* Fresh D U-233/ThO₂ Fuel
(low enrichment)

[illegible]

Fuel Batch
Assembly Location

0	0
0.00	
0.00	

Box Peaking Factor
Pin Peaking Factor

MAXIMA

Box Peaking Factor: 1.32

Pin Peaking Factor: 1.48

B - DU-233/ThO₂ Fuel Irradiated Two Cycles

(Middle Enrichment 2.45 w/o)

C - DU-233/ThO₂ Fuel Irradiated Two Cycles (High Enrichment 2.75 w/o)

C*- D U-233/ThO₂ Fuel Irradiated Two Cycles (Split High Enrichment 2.45 w/o)

D - D U-233/ThO₂ Fuel Irradiated One

Cycle (High Enrichment)

D* -D U-233/ThO₂ Fuel Irradiated One Cycle (Low Enrichment)

E - Fresh D U-233/ThO₂ Fuel
(High Enrichment)

E*- Fresh D U-233/ThO₂ Fuel
(Low Enrichment)²

					E 1 .77 1.21	E 2 .98 1.32	E 3 1.04 1.38	E 4 1.01 1.29	
				E 5 .77 1.18	E 6 1.05 1.35	E 7 1.13 1.26	D 8 1.29 1.40	E* 9 1.26 1.34	C 10 1.11 1.19
			E 11 .89 1.33	E 12 1.26 1.40	C 13 1.10 1.16	D 14 1.30 1.41	C 15 1.16 1.24	D 16 1.32 1.47	C 17 1.10 1.17
		E 18	E 19	D 20 1.30 1.36	D 21 1.18 1.27	C 22 1.04 1.15	E* 23 1.15 1.22	C 24 1.05 1.17	D 25 1.16 1.24
		E 26	C 26	D 28	C* 29 .88 .95	D 30 1.00 1.12	C 31 .89 1.01	D 32 1.00 1.15	C 33 .88 .96
E 34	E* 35	D 36	C 37	D 38	C* 39 .71 .77	D* 40 .72 .76	C 41 .75 .82	D 42 .82 .90	
E 43	D 44	C 45	E* 46	C 47	D* 48	C 49 .67 .73	D* 50 .63 .67	C 51 .64 .69	
E 52	E* 53	D 54	C 55	D 56	C 57	D* 58	C 59 .59 .63	D 60 .65 .68	
E 61	C 62	C 63	D 64	C 65	D 66	C 67	D 68	B 69 .54 .54	

FIGURE 5.3.3-10

UNRODDED PLANAR POWER DISTRIBUTION THIRD CYCLE
DENATURED U-233/THORIUM SSCR
(BEGINNING OF CYCLE EQUILIBRIUM XENON)

The fuel management characteristics of the initial three cycles of the denatured U-233/ThO₂ core are shown in Table 5.3.3-3, while the material inventories for the initial core fuel are shown in Table 5.3.3-4. These tables demonstrate the low fissile loading and inventory requirements of the U-233 fueled core compared to the U-235 fueled core. For example, in Cycle three, the denatured U-233/ThO₂ core inventory is 29% less at beginning of cycle and 25% less at end of cycle than that of the denatured U-235/ThO₂ core. Also apparent from a comparison of the above tables is the higher production of plutonium in the denatured U-233/ThO₂ core. This results from the fact that the U-233 bearing fuel has a maximum allowable enrichment of the uranium component of 12 wt% fissile, while the U-235 bearing fuel has an enrichment limit of 20 wt%. More U-238 is therefore loaded per unit of fissile uranium in denatured U-233/ThO₂ cycle than in the denatured U-235/ThO₂ cycle.

TABLE 5.3.3-3

NEW CORE FUEL MANAGEMENT CHARACTERISTICS DENATURED U-233/ThO₂ SSCR

		<u>Cycle 1</u>	<u>Cycle 2</u>	<u>Cycle 3</u>
Cycle length (Full Power Days)		274	274	274
Makeup Enrichment (wt% U-233)		2.27	2.93	2.87
Core Inventory (kg)				
U-233	BOC	2147	2232	2263
	EOC ^(a)	1878	1968	1981
U-235	BOC	58	70	77
	EOC	62	79	87
Pu-239	BOC	0.0	95	108
	EOC	131	162	180
Pu-240	BOC	0.0	19	26
	EOC	29	44	49
Pu-241	BOC	0.0	11	18
	EOC	16	33	39
Pu-242	BOC	0.0	1.3	3.5
	EOC	2.1	7.4	10.4

Note: (a) Includes decay of Pa-233

TABLE 5.3.3-4
MATERIAL INVENTORIES FOR DENATURED U-233/ThO₂ SSCR INITIAL CORE

Batch Designation	A	B	B*	C	C*
Number of Assemblies	81	68	12	72	8
Initial Enrichment (wt% U-233)	1.78	2.43	2.20	2.75	2.43
Loading HM(kg)	31436	26387	4657	27948	3104
Discharge Exposure (MWd/t)	10894	24256	22642	32519	35253
Initial Inventories (kg/Initial TeM)					
Th-232	847.725	796.984	812.040	765.083	791.984
U-233	17.800	24.324	21.970	27.459	24.324
U-234	1.439	1.966	1.777	2.220	1.966
U-235	.482	.659	.595	.744	.659
U-236	.009	.012	.011	.014	.012
U-238	132.555	191.056	163.607	204.481	181.056
Discharged Inventories (kg/Initial TeM)					
Th-232	837.313	774.099	794.230	743.486	765.384
U-233 ^(a)	17.286	18.260	17.925	18.147	17.170
U-234	2.597	4.101	3.768	4.767	4.779
U-235	.540	.858	.762	1.041	1.022
U-236	.064	.147	.129	.204	.221
U-238	129.519	174.186	157.977	194.762	171.08
Pu-239	1.221	1.834	1.474	2.195	1.799
Pu-240	.303	.587	.491	.728	.692
Pu-241	.186	.410	.396	.670	.640
Pu-242	.028	.133	.104	.226	.269

^(a) Includes decay of Pa-233

The plutonium/ThO₂ cycle is viewed as a transmutter cycle which uses as its fissile component plutonium generated in a uranium-based fuel cycle or a denatured thorium cycle and converts Th-232 into U-233. The chemically separable uranium component contained in the spent fuel is almost pure fissile material (greater than 92 wt% U-233) and as such, must be safeguarded. Since this fuel cycle accepts a weapons-usable material--plutonium, and converts it into another weapons-usable material--U-233, it can be deployed only within a secure energy center. However, the recovered uranium can be easily denatured by mixing it with depleted uranium prior to its shipment to reactors outside the energy center.

The introduction of plutonium as the fissile component hardens the neutron spectrum even more than in the denatured thorium cycles. This hard spectrum results in a relatively low beginning of life reactivity and consequently a low beginning of cycle D₂O concentration. Furthermore, since bred U-233 replaces the consumed plutonium as the fuel depletes and since U-233, because of its low capture-to-fission ratio, is more effective at maintaining a high reactivity than plutonium, the reactivity of the plutonium/ThO₂ core falls off more gradually than the other thorium cores. These trends can be observed in Figure 5.3.3-11 which shows the D₂O concentration required to control the excess core reactivity of the plutonium/ThO₂ SSCR as a function of cycle exposure. The curve labeled "cycle 2 and 3" in this figure is representative of the D₂O rundown for later cycles. Only a small decrease in concentration is expected in later cycles as a result of the buildup of higher plutonium isotopes from the recycle of recovered plutonium.

Figure 5.3.3-11
D₂O CONCENTRATION vs CYCLE EXPOSURE
PLUTONIUM/ThO₂ SSCR

As with the denatured U-233/ThO₂ cycle, every assembly in a plutonium/ThO₂ batch is composed of the same fuel material since all plutonium-bearing fuel is radioactive and therefore requires some type of remote or semi-remote fabrication technique. Again, near equilibrium conditions are quickly attained once the initial core fuel (low burnup fuel) is discharged. Power mismatches between adjacent assemblies are not a problem as they can be in cores where only some of the assemblies are plutonium-fueled. Therefore, the familiar problem of requiring enrichment zoning in the mixed-oxide assemblies of a self-generated UO₂ recycle core does not exist in this core.

Fuel loading patterns and beginning of cycle (with equilibrium xenon) power distributions for Cycles one through three of the plutonium/ThO₂ SSCR are shown in Figures 5.3.3-12 through 5.3.3-14, respectively. Because of the small reactivity splits of the initial core fuel, an acceptable power distribution is achieved with only three fuel enrichments. This can be contrasted with previous initial cores which used four or five enrichments to obtain acceptable power distributions. Furthermore, when an enrichment split is used for cycle two reload fuel, a relatively low pin peaking factor (1.35) is obtained. When only a single enrichment is used for cycle three reload fuel, an acceptable peaking factor is still obtained, but at the expense of a higher maximum pin power.

The fuel management characteristics of the initial three cycles of the plutonium/ThO₂ SSCR are shown in Table 5.3.3-5. The buildup of the higher plutonium isotopes and the decrease in the concentration of Pu-239 with

MAXIMA

Box Peaking Factor: 1.24

Pin Peaking Factor: 1.43

A - Low Enrichment Pu/ThO₂ Fuel

B - Middle Enrichment Pu/ThO₂ Fuel

C - High Enrichment Pu/ThO₂ Fuel

					C 1	C 2	C 3	C 4	
					.54 .82	.68 .95	.74 1.03	.75 .98	
				C 5	C 6	C 7	C 8	B 9	C 10
				.53 .83	.75 1.07	.93 1.24	1.07 1.39	.99 1.11	1.15 1.43
			C 11	C 12	B 13	B 14	A 15	B 16	A 17
			.59 .90	.87 1.21	.95 1.11	1.07 1.24	1.00 1.07	1.17 1.29	1.04 1.10
		C 18	C 19	B 20	B 21	A 22	B 23	A 24	B 25
				.98 1.13	1.11 1.26	1.04 1.09	1.21 1.31	1.08 1.12	1.24 1.32
		C 26	B 27	B 28	A 29	B 30	A 31	B 32	A 33
					1.05 1.09	1.22 1.31	1.09 1.13	1.24 1.32	1.10 1.13
C 34	C 35	B 36	A 37	B 38	A 39	B 40	A 41	B 42	
					1.09 1.12	1.23 1.32	1.08 1.12	1.22 1.31	
C 43	C 44	A 45	B 46	A 47	B 48	A 49	A 50	A 51	
						1.07 1.11	1.07 1.09	1.06 1.08	
C 52	B 53	B 54	A 55	B 56	A 57	A 58	A 59	B 60	
							1.05 1.08	1.19 1.29	
C 61	C 62	A 63	B 64	A 65	B 66	A 67	B 68	A 69	
								1.05 1.09	

FIGURE 5.3.3-12 - UNRODDED PLANAR POWER DISTRIBUTION FIRST CYCLE
PLUTONIUM/THORIUM SSCR
(BEGINNING OF CYCLE, EQUILIBRIUM XENON)

Assembly Location

0	0
0.00	—
0.00	—

Box Peaking

Pin Peaking Factro

MAXIMA

Box Peaking Factor: 1.23

Pin Peaking Factor: 1.35

B - Middle Enrichment Pu/ThO₂ Fuel Irradiated
One Cycle

C - High Enrichment Pu/ThO₂ Fuel
Irradiated One Cycle

D - Fresh High Enrichment Pu/ThO₂
Fuel

D*- Fresh Low Enrichment
Pu/ThO₂ Fuel

Location						D 1 .56 .85	D 2 .70 1.03	D 3 .78 1.03	D 4 .81 1.04
Peak Peak Factor									
				D 5 .57 .88	D 6 .78 1.07	D 7 .93 1.21	B 8 .85 .96	D 9 1.14 1.31	C 10 1.11 1.28
2 Fuel Irradiated		D 11 .65 .95	D 12 .93 1.20	B 13 .86 .95	C 14 1.09 1.26	B 15 .97 1.04	C 16 1.20 1.33	B 17 1.02 1.07	
Fuel	D 18	D 19	C 20 1.07 1.18	C 21 1.16 1.30	B 22 1.00 1.05	D* 23 1.23 1.32	B 24 1.02 1.07	C 25 1.19 1.30	
/ThO ₂	D 26	B 27	C 28	B 29 1.04 1.08	C 30 1.22 1.33	B 31 1.03 1.08	C 32 1.21 1.35	B 33 1.04 1.08	
	D 34	D 35	C 36	B 37	C 38	B 39 1.04 1.09	C 40 1.22 1.35	B 41 1.05 1.09	C 42 1.19 1.29
	D 43	B 44	B 45	D* 46	B 47	C 48	B 49 1.05 1.09	C 50 1.20 1.32	B 51 1.01 1.06
	D 52	D 53	C 54	B 55	C 56	B 57	C 58	B 59 1.00 1.06	C 60 1.14 1.26
	D 61	D 62	B 63	C 64	B 65	C 66	B 67	C 68	B 69 .93 1.00

FIGURE 5.3.3-13 - UNRODDED PLANAR POWER DISTRIBUTION SECOND CYCLE

(BEGINNING OF CYCLE, EQUILIBRIUM XENON)

5-352

MAXIMA

Box Peaking Factor: 1.34
Pin Peaking Factor: 1.41

B - Middle Enrichment Pu/ThO₂ Fuel
Irradiated Two Cycles

C - High Enrichment Pu/ThO₂ Fuel
Irradiated Two Cycles

D - High Enrichment Pu/ThO₂ Fuel
Irradiated One Cycle

D*- Low Enrichment Pu/ThO₂
Fuel Irradiated One Cycle

E - Fresh Pu/ThO₂ Fuel

							E 1	E 2	E 3	E 4
							.47 .71	.59 .81	.66 .88	.68 .87
			E 5	E 6	E 7	C 8	E 9	D 10		
			.48 .75	.68 .91	.83 1.00	.87 .98	1.01 1.13	1.04 1.16		
	E 11	E 12	C 13	D 14	C 15	D 16	C 17			
	.56 .84	.82 1.02	.92 1.03	1.10 1.25	1.08 1.14	1.19 1.27	1.12 1.14			
	E 18	E 19	D 20	D 21	C 22	E 23	C 24	D 25		
			1.08 1.22	1.19 1.30	1.15 1.20	1.28 1.32	1.15 1.19	1.22 1.25		
	E 26	C 27	D 28	C 29	D 30	C 31	D 32	C 33		
				1.22 1.27	1.34 1.41	1.19 1.23	1.26 1.35	1.12 1.15		
E 34	E 35	D 36	C 37	D 38	D* 39	D 40	C 41	D 42		
					1.22 1.27	1.27 1.40	1.12 1.17	1.18 1.29		
E 43	C 44	C 45	E 46	C 47	D 48	C 49	D 50	B 51		
						1.09 1.15	1.14 1.28	.85 .91		
E 52	E 53	D 54	C 55	D 56	C 57	D 58	C 59	D* 60		
							.94 1.00	.94 1.07		
E 61	D 62	C 63	D 64	C 65	D 66	B 67	D* 68	B 69		
								.75 .77		

FIGURE 5.3.3-14 - UNRODDED PLANAR POWER DISTRIBUTION THIRD CYCLE

PLUTONIUM/THORIUM SSCR
(BEGINNING OF CYCLE, EQUILIBRIUM XENON)

TABLE 5.3.3-5

NEW CORE FUEL MANAGEMENT CHARACTERISTICS PLUTONIUM/ThO₂ SSCR

		<u>Cycle 1</u>	<u>Cycle 2</u>	<u>Cycle 3</u>
Cycle Length (Full Power Days)		274	274	274
Makeup Enrichment (wt% Fissile Pu)		3.25	4.28	3.77
Core Inventory (kg)				
U-233	BOC	0	333	412
	EOC ^(a)	487	692	741
U-235	BOC	0	1.3	2.7
	EOC	2.3	6.5	8.7
Pu-239	BOC	2476	2260	2209
	EOC	1411	1374	1352
Pu-240	BOC	1039	1251	1323
	EOC	1019	1165	1215
Pu-241	BOC	560	700	740
	EOC	598	691	734
Pu-242	BOC	221	285	311
	EOC	252	307	342

Note: (a) Includes decay of Pa-233

cycle number is apparent. An interesting characteristic shown in Table 5.3.3-5 is the reduced reload fuel enrichment of Cycle three as compared to Cycle two. However, addition of the resident masses of the four fissile isotopes shows that the total beginning of cycle fissile mass is nearly identical for Cycles two and three. The higher loading of U-233 at the beginning of Cycle three apparently compensates for the higher fission product poisoning in Cycle three compared to Cycle two.

The detailed material inventories for the initial core fuel (i.e., the fuel loaded in the initial core and subsequently discharged during the first three cycles) are shown in Table 5.3.3-6. It can be seen from the data in this table that the discharge fuel contains uranium whose fissile content is in excess of 90%. Thus the need for a secure energy center to process this material is apparent.

TABLE 5.3.3-6
MATERIAL INVENTORIES FOR PLUTONIUM/ThO₂ SSCR INITIAL CORE

Batch Designation	A	B	C
Number of Assemblies	77	88	76
Initial Enrichment (wt% fissile Pu)	2.21	3.09	4.48
Loading HM(kg)	29874	34142	29486
Discharge Exposure (MWd/t)	11030	23400	34638
Initial Inventories (kg/Initial TeM)			
Th-232	968.737	956.249	936.631
Pu-239	18.009	25.213	36.539
Pu-240	7.556	10.519	15.331
Pu-241	4.075	5.705	8.233
Pu-242	1.610	2.254	3.266
Discharge Inventories (kg/Initial TeM)			
Th-232	959.948	940.049	916.361
U-233 ^(a)	7.036	10.942	12.685
U-234	.417	.925	1.189
U-235	.033	.124	.192
U-236	.001	.006	.010
Pu-239	7.340	5.531	7.559
Pu-240	7.057	8.484	12.233
Pu-241	4.488	5.698	7.719
Pu-242	2.033	3.099	4.241

(a) Includes decay of Pa-233

Although five different SSCR cores were examined in the present study, detailed calculations for the comparable PWR were available only for the two UO_2 once through cores. The comparison between the fuel management characteristics of the SSCR and the PWR therefore concentrated on these two cores.

5.3.4.1 UO_2 ONCE THROUGH THREE BATCH FUEL MANAGEMENT

Fuel management aspects of the UO_2 once through SSCR are very similar to those of a conventional PWR. Partial batch refueling, with fresh fuel loaded on the periphery of the core and once and twice burned fuel loaded in the interior of the core in a checkerboard pattern, is the predominant fuel management characteristic of either reactor. Additionally, with the possible exception of the low enrichment fuel of the initial core (Batch A), both reactor types typically split the input batches into high and low enrichment sub-batches. This is done in order to gain the flexibility needed to obtain acceptable radial power distributions. One notable difference between the fuel management schemes of the two reactors is that the SSCR does not use burnable poison shim rods. Shim rods are desirable in a PWR to reduce the required soluble boron concentrations, thus maintaining the moderator temperature coefficient sufficiently negative. The SSCR needs no such consideration since the moderator temperature coefficient is much more negative due to the use of D_2O concentration as opposed to the use of soluble boron in the PWR which results in a less negative coefficient. The introduction of poison shim rods into a SSCR would increase parasitic absorption and reduce its conversion ratio. Additionally, poison shim rods are used in the PWR to obtain more acceptable power distributions. The

relatively long diffusion lengths in the SSCR as compared to those of the PWR at high D_2O concentrations (near beginning of cycle) make the SSCR power distributions less sensitive to local reactivity variations. This reduced sensitivity tends to give the SSCR flatter power distributions, further reducing the need for shim rods.

Neutron leakage from a current design PWR is typically about 4 percent with two thirds of this leakage radial and the other third axial. Throughout the operating cycle, the leakage is relatively constant, with only a slight increase as power shifts towards the core periphery near end of cycle. Due to the longer diffusion lengths in the SSCR at high D_2O concentrations, neutron leakage is greater in the SSCR than in the PWR near beginning of cycle. As the core depletes and heavy water concentration is reduced, neutrons thermalize more quickly with attendant shorter diffusion lengths and reduced core leakage. Leakage becomes comparable to that of a PWR near end of cycle when both cores are essentially moderated by pure light water.

Another consequence of longer diffusion lengths in the SSCR, and the subsequent decreased sensitivity to local reactivity variations, is that fresh fuel of a higher reactivity can be loaded in an SSCR than in a PWR and still obtain acceptable power distributions. This makes less traditional fuel management schemes, such as would be used for eighteen month refueling or for the five batch/high burnup strategy, more easily achievable in the SSCR than in the PWR. As the core depletes and the diffusion length decreases, the reactivity of the high enrichment fuel also decreases due to depletion, so that power peaking problems do not develop late in the cycle when the SSCR core looks similar to that of the PWR.

A comparison of five batch fuel management in the SSCR and the PWR shows many similarities with only a few notable differences. These can be traced to the differences between a set of arbitrary assumptions. However, they can be traced to the differences between the core diffusion lengths in the SSCR as compared to the PWR, and to the fact that the PWR employs burnable poison shim rods in the initial core to maintain an acceptable moderator temperature coefficient. For both reactor types, the initial core consists of fuel with five different enrichments, the total split between the enrichments being as large as possible while still achieving acceptable power distributions. Ideally, the highest enrichment would be set equal to the equilibrium batch enrichment; this hastens the approach to equilibrium conditions. Neither reactor could load this enrichment, however, without obtaining excessively high power peaks on the core periphery where the highest enrichment was loaded. Enrichments of only 79 percent and 90 percent of the equilibrium values could be loaded in the PWR and SSCR, respectively. The approach to equilibrium conditions took longer for the PWR (about five cycles) than for the SSCR (about three cycles) because of the greater disparity between the highest enrichment which could be loaded into the initial core and the higher reload enrichment of the equilibrium cycle. This approach to equilibrium was shown on Table 5.3.2-3 for both the PWR and the SSCR. The fact that a batch with an enrichment closer to that of the equilibrium reload batch could be loaded in the SSCR results from the decreased sensitivity of the beginning of cycle core power distribution to local variations in core reactivity. The longer neutron diffusion lengths in the SSCR decrease the power mismatches between adjacent assemblies

and generally promote overall power distributions which are peaked more toward the center of the core.

In addition to the five distinct batches in the PWR initial core, defined by the different enrichments, there are several reactivity splits within the individual batches as a result of poison shim loading. The fuel batches with the three highest enrichments each have two or three shim loadings. Thus, there are fuel assemblies in the PWR initial core with nine different reactivity depletion characteristics. This relatively large number of sub-batches makes fuel management in the five batch PWR more complicated than in the five batch SSCR.

5.4.1

DETERMINATION OF CORE PHYSICS PARAMETERS

The core physics parameters examined in this study include heavy water and soluble boron worths, control rod worths, moderator and fuel temperature coefficients, and delayed neutron fractions and prompt neutron lifetimes. These characteristics form the basis for the safety evaluation of a reactor design and, to a large extent, determine the severity of various postulated accidents and plant transients.

The worth of heavy water, which is used to control the long-term reactivity changes due to fuel depletion and fission product buildup, is important from a safety standpoint because of the potential increase in core reactivity which would result if light water were inadvertently substituted for heavy water. The worth of heavy water determines, to a large extent, the response time available to detect and terminate the injection of light water. Similarly, the worth of soluble boron, which is used to control the reactivity changes due to moderator cooldown and xenon buildup is important when considering light water dilution during shutdown. Also, too low a soluble boron worth could compromise the use of boric acid in the safety injection system as a means of providing auxiliary/emergency reactivity control.

Individual and total control rod worth requirements are based on limiting the effects of anticipated operating transients and accidents. The reactivity available for insertion must be adequate to supply sufficient shutdown margin under all occurrences.

The moderator and fuel temperature coefficients are parameters which affect the inherent safety of the core. In the power operating range, the combined response of the reactivity feedback mechanisms (i.e., the temperature coefficients) to an increase in reactor thermal power must be a decrease in core reactivity. This criterion is usually satisfied by requiring that each of these coefficients be negative.

The delayed neutron fraction (and the prompt neutron lifetime) are controlling parameters in the reactor's initial response to short-term ($\sim$ seconds) power transients, such as occur in rod ejection accidents. Small values of these parameters mean that it may be more difficult to minimize the consequences of certain postulated accidents, since reactor power will increase at a faster rate for a given amount of reactivity addition.

The physics parameters which are important to the safety analysis of the reactor plant have been determined for two SSCR cores: the UO_2 once-through core employing standard PWR fuel management and the denatured U-235/ThO_2 core with recycle of recovered uranium. A comparison of these parameters is made in this section with those of a UO_2 -fueled PWR to determine the probable changes in core operating characteristics resulting from the use of spectral shift control. In the next section, an assessment is made of the impact of these parameters on the safety of the SSCR.

5.4.1.1 UO_2 Once-Through SSCR

The predominant factors which differentiate the core parameters of the UO_2 once-through SSCR from those of a similarly fueled PWR are the increase in

diffusion length and the increase in epithermal flux. Both of these factors are characteristic of the SSCR at beginning of cycle when the D_2O concentration is high, while most core parameters resemble those of the PWR near end of cycle when the coolant is essentially pure light water. A summary of the core physics parameters of the uranium and thorium fueled SSCR and of the uranium fueled PWR is given in Table 5.4.1-1. The values presented in this table are for normal operating conditions; in the discussions which follow, the sensitivity of these parameters to changes in core conditions (e.g., moderator temperature, D_2O concentration, etc.) will be shown.

The delayed neutron fraction is mainly a function of the isotopic content of the fuel and the relative fission rates in these isotopes. Therefore, this parameter shows no discernable difference between reactor types operating on the same fuel cycle. The prompt neutron lifetime, however, is increased with decreasing moderation of the core. Near beginning of cycle, the prompt neutron lifetime is about 19% greater than that of the PWR. As the D_2O concentration is decreased throughout the operating cycle, the prompt neutron lifetime decreases until near end of cycle when it approaches the value of the PWR.

Figure 5.4.1-1 shows a graph of moderator temperature coefficient versus moderator temperature for the SSCR and the PWR at the beginning and the end of an equilibrium cycle. At end of cycle, the moderator properties of the two reactors are nearly identical; the PWR has only a small residual quantity of soluble boron (10 to 20 ppm) remaining in the coolant, while the SSCR has its lowest operating concentration of heavy water (about 2 wt%). The similarity of the moderator composition for the two reactors results in moderator temperature coefficient curves which are virtually indistinguishable.

TABLE 5.4.1-1

COMPARISON OF PWR AND SSCR CORE PHYSICS PARAMETERS

	<u>PWR</u> <u>UO₂ ONCE-THROUGH</u>	<u>SSCR</u> <u>UO₂ ONCE-THROUGH</u>	<u>SSCR</u> <u>DENATURED</u> <u>U-235/ThO₂</u>
Delayed Neutron Fraction			
BOC	0.00625	0.00627	0.00572
EOC	0.00546	0.00541	0.00469
Prompt Neutron Lifetime ($\times 10^{-6}$ sec)			
BOC	21.3	25.4	19.8
EOC	24.8	24.7	21.8
Fuel Temperature Coefficient ⁽¹⁾ ($\times 10^{-5}$ $\Delta\rho/^{\circ}\text{F}$)			
BOC	-1.22	-1.77	-3.25
EOC	-1.31	-1.22	-2.14
Moderator Temperature Coefficient ($\times 10^{-4}$ $\Delta\rho/^{\circ}\text{F}$)			
BOC	-0.7	-5.5	-4.6
EOC	-3.4	-3.8	-2.7
Control Rod Worth (% $\Delta\rho$) ⁽²⁾			
BOC	13.5	16.0	16.9
EOC	13.1	13.0	13.5
Inverse Soluble Boron Worth ⁽³⁾ (ppm/% $\Delta\rho$)			
BOC	113	100	131
EOC	98	107	129
Inverse Heavy Water Worth ⁽⁴⁾ (% D ₂ O/% $\Delta\rho$)			
BOC	-	2.56	2.67
EOC	-	5.80	7.14

- NOTES: (1) At 1500°F Fuel Temperature
 (2) Worst Rod Stuck Out
 (3) Given at 50% D₂O for BOC and 2% D₂O at EOC for SSCR; Burnup Dependence not included, calculations performed with BOC Fuel Composition.
 (4) Given at 47.5% D₂O for BOC and 6% D₂O for EOC.

Figure 5.4.1-1

MODERATOR TEMPERATURE COEFFICIENT vs MODERATOR TEMPERATURE
(PWR AND SSCR UO₂ ONCE-THROUGH FUEL CYCLE; STANDARD FUEL MANAGEMENT SCHEME)

At beginning of cycle, however, the moderator temperature coefficient curves are quite different; the SSCR having a much more negative moderator temperature coefficient than the PWR. At this time in the cycle, the coolant in the PWR contains approximately 1200 ppm boron, which is controlling about 12% reactivity through parasitic absorption. Raising the moderator temperature decreases the moderator density and, at the same time, expels soluble boron from the core. The net result of these two opposing effects is a small negative reactivity insertion. The SSCR, however, contains only a trace amount of soluble boron which remains after the cleanup (deboronation) following cold shutdown. An increase in the moderator temperature will have the single effect of decreasing the moderator density and, as a result of the reduction in the slowing-down power of the moderator, will cause a much larger decrease in core reactivity. Since the majority of the neutron slowing-down occurs as a result of collisions with hydrogen and since there is less hydrogen in the SSCR core when the heavy water concentration is high at beginning of cycle, the expulsion of an atom of hydrogen as a result of moderator temperature changes is more important in the SSCR than in the PWR.

Figure 5.4.1-2 shows a graph of fuel temperature coefficient (Doppler coefficient) versus average fuel temperature for both the SSCR and the PWR at beginning and end of equilibrium cycle. Due to the nearly burnup independent neutron spectrum of the PWR, the curves for the beginning and end of cycle fuel temperature coefficients are very similar. The end of cycle values are slightly higher because of the added Doppler absorption in Pu-240, which has a higher core average concentration at end of cycle than at beginning of cycle. Since the neutron spectra for the PWR and the SSCR at

Figure 5.4.1-2

FUEL TEMPERATURE (DOPPLER) COEFFICIENT vs AVERAGE FUEL TEMPERATURE
(PWR AND SSCR UO₂ ONCE-THROUGH FUEL CYCLE; STANDARD FUEL MANAGEMENT SCHEME)

end of cycle are essentially the same, the fuel temperature coefficients for the two reactors are very close. The small differences between the end of cycle curves is primarily due to differences in the average isotopic composition of the two cores. At beginning of cycle, however, the SSCR has a much harder spectrum than the PWR. As a result of the higher epithermal flux which accompanies this harder spectrum, epithermal absorptions in fertile material are considerably higher and therefore changes in the capture cross sections of the fertile components have more importance. This increased sensitivity to Doppler broadening gives rise to a more negative fuel temperature coefficient at beginning of cycle for the SSCR.

Control rod worth curves as a function of moderator temperature are shown in Figure 5.4.1-3 for the SSCR at beginning and end of cycle. For comparison, similar curves are shown for a conventional PWR in Figure 5.4.1-4. These PWR curves have been adjusted to correct for differences between the control rod patterns for the two reactors. The control rod worths presented in both figures were calculated for all rods inserted except for the most reactive rod which was assumed stuck out of the core. Due to the characteristic "blackness" (near-absolute absorbing) properties of the control rod, control rod worth is highly dependent on the core thermal and epithermal diffusion lengths; that is, the farther an average neutron travels, the more likely it is to encounter a control rod and be absorbed. Due to the longer diffusion lengths at higher moderator temperatures (lower coolant densities) and higher heavy water concentrations, the control rod worth increases with both of these parameters.

Figure 5.4.1-3
CONTROL ROD WORTH* vs MODERATOR TEMPERATURE
(UO₂ ONCE-THROUGH SSCR; STANDARD FUEL MANAGEMENT SCHEME)

* ALL CEAs INSERTED EXCEPT MOST REACTIVE ROD STUCK OUT

Figure 5.4.1-4

TYPICAL CONTROL ROD WORTH* vs MODERATOR TEMPERATURE
(UO₂ ONCE-THROUGH PWR; STANDARD FUEL MANAGEMENT SCHEME)

* ALL CEAs INSERTED EXCEPT MOST REACTIVE ROD STUCK OUT

The inverse D_2O worth for the UO_2 once-through SSCR is shown in Figure 5.4.1-5. Its shape reflects the fact that the reactivity controlled by D_2O is mainly a function of the fractional change in hydrogen concentration which results. At beginning of cycle when the concentration of hydrogen is relatively low, each incremental change in D_2O concentration changes the hydrogen concentration and, thus, the slowing-down power of the moderator significantly. However, at end of cycle when the concentration of hydrogen is high, each incremental change in D_2O concentration has a smaller effect on the slowing-down power. It therefore takes a greater change in D_2O concentration to control a given amount of reactivity at end of cycle than at beginning of cycle. The temperature of the moderator, because of its effect on the concentration of hydrogen in the core, also affects the worth of D_2O . As the moderator temperature decreases, and more hydrogen is present in the core (for a given D_2O concentration), the worth of D_2O also decreases.

The soluble boron worth is, to a large extent, independent of boron concentration, D_2O concentration, and fuel depletion. From previous PWR analyses, it has been found that the concentration of boron has essentially no effect on the boron worth. Furthermore, it has been found that the dependence of boron worth on core exposure is small, on the order of 5-15%. Similarly, in the present study it has been found that the sensitivity of boron worth to D_2O concentration is small, about 7% for the full range of D_2O concentrations encountered in spectral shift control operation.

Figure 5.4.1-5

INVERSE D₂O WORTH vs D₂O CONCENTRATION
(UO₂ ONCE-THROUGH SSCR)
NORMAL OPERATING CONDITIONS

While core characteristics of a uranium-fueled SSCR differ from those of a uranium-fueled PWR primarily because of changes in moderator properties, characteristics of a thorium-fueled SSCR differ from those of a uranium-fueled SSCR due to the nuclear properties of thorium compared to those of uranium. In particular, Th-232 has a relatively large thermal absorption cross section, almost three times that of U-238, and a somewhat smaller resonance absorption cross section, about 15% less than that of U-238.

The kinetics parameters for the denatured U-235/ThO₂ core are shown in Table 5.4.1-1. The delayed neutron fraction (β) of the thorium core is less than that of the UO₂ once-through core. The calculation of β was performed for a near-equilibrium cycle, for which a sizable portion of the fissile material is U-233. The presence of U-233 lowers the delayed neutron fraction because of the small value of β for U-233. The prompt neutron lifetime for the denatured U-235/ThO₂ core is much less than that for the UO₂ once-through core, particularly at beginning of cycle. This is due to the larger core-averaged thermal cross section for the thorium core compared to the uranium core. A larger thermal cross section reduces the thermal diffusion time and consequently the prompt neutron lifetime.

Moderator temperature coefficients as a function of moderator temperature for the denatured U-235/ThO₂ SSCR are shown in Figure 5.4.1-6. The shape and magnitude of the moderator temperature coefficient curves are similar to those of the UO₂ once-through SSCR. However, at higher temperatures, the coefficients are less negative in the thorium core than in the uranium

Figure 5.4.1-6

MODERATOR TEMPERATURE COEFFICIENT vs MODERATOR TEMPERATURE
DENATURED U-235/ThO₂ SSCR

core. This difference is attributable to the lesser importance of the change in epi-thermal captures in Th-232 which occur as a result of the shift in spectrum caused by a change in moderator temperature. Captures in Th-232 are split almost evenly between thermal and epithermal energies, while captures in U-238 and Pu-240 are largely in the epithermal range. In addition, the beta-value (neutrons created per neutron absorbed) of U-233 is significantly higher in the epithermal region than that of U-235 and Pu-239. Thus, changes in spectrum due to changes in moderator temperature have less of an effect on reactivity in the denatured U-235/ThO₂ cycle than in the UO₂ once-through cycle.

Fuel temperature coefficients as a function of average fuel temperature are shown in Figure 5.4.1-7 for the denatured U-235/ThO₂ core. It is interesting to note that the fuel temperature coefficients for this core are more negative than those for the UO₂ once-through SSCR core. This is due to the presence of two resonance absorbers, each with a density less than if they were the single fertile component. When a fuel cycle employs a single fertile absorber, lumping of this absorber into discrete fuel rods causes substantial resonance self shielding with a significant reduction in absorption rate sensitivity to Doppler broadening. Due to the denaturing of U-235, a substantial quantity of U-238 (approximately 15% of the total heavy metal) is added to the fuel. However, because of its relatively low concentration, this U-238 has less self shielding, about half that of the U-238 in the UO₂ once-through cycle; its absorption rate is therefore more sensitive to Doppler broadening. The U-238 in the denatured thorium cycle also displaces thorium, thereby decreasing the concentration of Th-232 and, in turn, the thorium self shielding. This reduction in Th-232 self shielding also increases the sensitivity of the fuel cycle to Doppler broadening.

Figure 5.4.1-7
FUEL TEMPERATURE (DOPPLER) COEFFICIENT vs AVERAGE FUEL TEMPERATURE
DENATURED U-235/ThO₂ SSCR

Control rod worth as a function of moderator temperature is shown in Figure 5.4.1-8. As mentioned previously, control rod worth is primarily a function of the core diffusion length since this parameter determines how likely a neutron is to encounter a control rod (i.e., the further it travels, the more likely it is to encounter a control rod). Because of the larger thermal absorption cross section of thorium and the resulting smaller thermal diffusion length, one would expect that the control rod worth for the thorium core would be less than that of the uranium core. The control rod worths shown in Figure 5.4.1-7, however, are for all rods in except for the most reactive rod stuck out. This type of rod worth is highly dependent on the fuel shuffling pattern chosen in addition to the fuel type employed. By locating fuel with the highest reactivity away from control rod locations, the stuck rod worth is kept low, thereby increasing the net control rod worth. In the case of the denatured U-235/ThO₂ core, a fuel shuffling pattern which maintained a more favorable stuck rod worth was employed.

The inverse D₂O worth for the denatured U-235/ThO₂ SSCR is shown in Figure 5.4.1-9. This curve is very similar to that for the UO₂ once-through SSCR. At low D₂O concentrations (less than 15 wt%), the worth of D₂O is somewhat greater in the thorium core than in the uranium core. Changing the hydrogen concentration (by displacing it with deuterium) in the denatured U-235/ThO₂ core has more of an effect than in the UO₂ once-through core because the thorium core is more under-moderated than the uranium core.

Figure 5.4.1-8

CONTROL ROD WORTH * vs MODERATOR TEMPERATURE
DENATURED U-235/ThO₂ SSCR

* ALL CEAs INSERTED EXCEPT MOST REACTIVE ROD STUCK OUT

Figure 5.4.1-9

INVERSE D₂O WORTH vs D₂O CONCENTRATION
(DENATURED U-235/ThO₂ SSCR)
NORMAL OPERATING CONDITIONS

As mentioned previously the soluble boron worth is nearly independent of boron concentration, D_2O concentration, and fuel depletion. It is, however, an inverse function of the core-averaged thermal absorption cross section. Since the denatured $U-235/ThO_2$ core has a larger thermal cross section than the uranium core, the boron worth is less in the thorium core.

5.4.2 LOSS-OF-COOLANT ACCIDENT

5.4.2.1 Introduction

The evaluation of the Loss-of-Coolant Accident (LOCA) for the SSCR was essentially qualitative in nature and made use of existing analysis results for the System 80TM PWR. For a Safety Analysis Report, LOCA analyses are performed for a spectrum of pipe break sizes ranging from 0.5 ft² cross-section to twice the cross-sectional area of the cold leg pipe. From the information presented in CESSAR⁽¹⁾ it is seen that the double-ended guillotine break of a reactor coolant pump discharge pipe results in the highest peak cladding temperature (2146°F) and cladding oxidation percentage (16.05%). For comparison, the Acceptance Criteria according to the Federal Register⁽²⁾ limit the cladding temperature to 2200°F and the total cladding oxidation to 17% of the total cladding thickness before oxidation.

For the evaluation of the SSCR it was assumed that the double-ended guillotine break of a discharge pipe would also result in the highest cladding temperature and oxidation percentage. Furthermore, core and containment heat transfer parameters of the SSCR were assumed to be unchanged from System 80TM. Also, the emergency cooling system of System 80TM remains unchanged for the SSCR.

5.4.2.2 Loss-Of-Coolant Accident, Standard Burn-Up Fuel

The kinetics parameters (delayed neutron fractions, decay constants, fission product energies and neutron life times) of the SSCR are very similar to those of System 80TM. In addition, the moderator reactivity curve for

System 80TM can be used as a conservative upper bound for the SSCR (see Figure 5.4.2.-1) since the more negative coefficient of the SSCR will result in less severe accident consequences. If the radial and axial peaking factors are the same as those for System 80TM, then the heat generation rate at the hot spot for the SSCR design must be the same as or lower than that for System 80TM. Consequently, for the same core flow rate and hot spot heat transfer, the peak clad temperature for the SSCR design should be similar to that for the System 80TM design. Thus, for both SSCR fuel cycles (UO_2 and U/THO_2), at both the beginning of life and end of life, the peak clad temperature remains below the limiting temperature of 2200°F. The clad oxidation should also be similarly below the limiting value of 17% for the SSCR design.

5.4.2.3 Loss-Of-Coolant Accident, High Burnup Fuel

For the evaluation of the effect of high fuel burnup on LOCA it was assumed that the effect of high burnup on reactor kinetics and core power is negligible. From the point of view of LOCA, the possibility of higher fuel rod internal gas pressures due to high fission gas release has the most significance. For the evaluation of the LOCA behavior it was assumed (based on the analysis presented in Section 5.1.4-FUEL DESIGN), that the internal fuel rod pressure would rise to 3000 psia at the end of exposure. At such internal pressures, there is a possibility for the hottest rod to swell to large diameters and rupture early in blowdown, instead of during reflood as for System 80TM. Although blowdown rupture in itself is not detrimental to cooling of the fuel rod, from the point of view of flow blockage and rod-

Figure 5.4.2-1
REACTIVITY vs MODERATOR DENSITY
SSCR - U/ThO₂ FUEL

to-rod touching (which reduces heat transfer to surrounding cooler rods), it is important to know when rod rupture takes place, and whether or not the ruptured rod touches the surrounding rods at or after rupture. Therefore, a study was conducted to determine the time period (blowdown or reflood) at which the hot rod ruptures and the strains associated with this rupture. For the study, the fuel rod clad temperature and core pressure during the transient were taken to be the same as those generated for the System 80TM design, as shown in Figures 5.4.2-2 and 5.4.2-3. The fuel rod internal pressure was assumed to be constant at 3000 psia throughout the transient until rupture. For the System 80TM design (16 x 16 matrix, 0.382 in. OD fuel rod), the perforation temperature for the clad material (zircaloy) is a function of the clad differential pressure as indicated below:

$$T = 8472 (\Delta P)^{-0.2859} \quad 0 \leq \Delta P \leq 233.5 \text{ psia}$$

$$T = 1842 - 0.2532 (\Delta P) \quad \Delta P > 233.5 \text{ psia}$$

Figure 5.4.2-4 is a graphical representation of this best estimate clad rupture temperature versus differential pressure equation. For clad conditions below this curve, perforation of the clad is precluded. If however, the clad temperature reaches a value high enough to cause the locus of T vs. ΔP for a transient to intersect the curve, clad rupture occurs.

Using Figures 5.4.2-2 and 5.4.2-3, clad temperature and differential pressure values for the SSCR can be estimated. In Figure 5.4.2-4, the curve marked "SSCR" represents this clad temperature/ ΔP curve. It is seen that for the SSCR cladding rupture is expected to occur at about 1320°F, approximately 11 seconds after the pipe break.

Figure 5.4.2-2
CLADDING TEMPERATURE FOR DOUBLE-ENDED
GUILLOTINE DISCHARGE BREAK

Figure 5.4.2-3
SYSTEM 80
CORE PRESSURE FOR DOUBLE-ENDED GUILLOTINE
DISCHARGE BREAK

Figure 5.4.2-4
PERFORATION TEMPERATURE vs CLAD DIFFERENTIAL PRESSURE FOR SYSTEM 80
(BEST ESTIMATE CURVE)

To determine whether or not rod-to-rod touching takes place at rupture, it is necessary to calculate the circumferential strain at the time of rupture. The circumferential strain curve of Figure 5.4.2-5 is a plot of circumferential strain of zircaloy vs. clad differential pressure at rupture. The equation of this curve is:

$$\epsilon = 26.99 + \frac{3024.67}{\Delta P} \text{ (percent)}$$

For the SSCR design, the circumferential strain is

$$\epsilon = 28.4\%$$

The final diameter of the fuel rod can be expressed as:

$$D_f = D_i (1 + \epsilon)$$

where, D_f is the final diameter of the fuel rod, in., and
 D_i is the initial diameter of the fuel rod.

Since $D_i = 0.382$ in., $D_f = 0.49$ in.

An illustration of the relationship of the swollen fuel rod to the unswollen rod is provided in Figure 5.4.2-6. It is seen that the fuel rods do not touch at the time of rupture, although the clearance between them is very small (~ 0.016 in.). It has been found in earlier calculations that rod-to-rod touching will degrade the radiative heat transfer from the hot rod to the

Figure 5.4.2-5
CIRCUMFERENTIAL STRAIN OF ZIRCALOY AS A FUNCTION OF CLAD
DIFFERENTIAL PRESSURE AT RUPTURE FOR SYSTEM 80
(BEST ESTIMATE CURVE)

Figure 5.4.2-6

HIGH BURNUP FUEL
COMPARISON OF FUEL ROD ARRAY DIMENSIONS

cooler surrounding rods. However, for blowdown rupture with no rod touching, the radiative heat transfer is not degraded, and hence no increases in peak cladding temperature and local clad oxidation should be realized.

The general consequences of clad rupture are illustrated by the temperature curves of Figure 5.4.2-2 which shows the temperature variation for the fuel centerline, average fuel and clad during the transient for the System 80TM design.

For System 80TM, the clad ruptures during the reflood period, about 75 seconds after the pipe break. Subsequent to rupture, the clad temperature rise is seen to be slowed down; in fact, the clad temperature is reduced for a short period of time, due to the fact that the heat transfer from the fuel to the clad is reduced right after rupture, while the heat transfer from the clad to the coolant is not affected by clad rupture. Eventually, the gap conductance between the fuel and clad increases and the clad temperature rises, since the clad to coolant heat transfer during the initial portion of the reflood phase is not sufficient to prevent heating-up of the clad. In the SSCR, clad rupture occurs during the blowdown phase, about 11 seconds after the pipe breaks. Since there also is no degradation in heat transfer from the clad to the coolant during the transient for blowdown as a result of rupture, the peak clad temperature for the SSCR design with high discharge exposure should not be any higher than what is calculated for the System 80TM design with the same peak linear heat generation rate as employed for the SSCR design. In comparison to the unruptured rod clad temperature, the clad temperature for the ruptured rod may remain at higher temperature values (closer to the peak value) for a longer time during reflood due to the poorer heat transfer from fuel to clad.

REFERENCES FOR SECTION 5.4.2

"Loss-of-Coolant Accident"

1. Combustion Engineering Standard Safety Analysis Report (CESSAR) Vol. 2, 1975.
2. Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors; Federal Register, Vol. 39, No. 3, January 1974.

5.4.3 ACCIDENT ANALYSIS EXCLUDING LOSS-OF-COOLANT ACCIDENT

5.4.3.1 Introduction

This section presents an evaluation of the response of the Nuclear Steam Supply System (NSSS) of the SSCR to process disturbances and equipment malfunctions. The effects of these incidents were examined to determine their consequences and to compare the relative capabilities of the SSCR and System 80TM to accommodate the consequences of these incidents.

The evaluation of the SSCR was primarily qualitative in nature and relied heavily on comparison with the results of System 80TM safety assessments. The evaluation was patterned after the accident analysis presented in Section 15 of CESSAR-F⁽¹⁾, the Standard Safety Analysis Report for System 80TM. Section 15 of CESSAR-F is structured around an eight by five category/frequency matrix which categorizes the initiating events by type and expected frequency of occurrence. Organization of Section 15 around such a category/frequency matrix permits the consistent application of specific acceptance criteria for each postulated initiating event. Each postulated initiating event has been assigned to one of the eight categories in Table 5.4.3-1. Five frequency groups were used in the category/frequency matrix for CESSAR-F. In order to permit the consistent application of specific acceptance criteria for each postulated initiating event, acceptance guidelines have been established for each frequency group in the category/frequency matrix. Typical frequency groups and acceptance guidelines are presented in Table 5.4.3-2.

TABLE 5.4.3-1

CATEGORY/FREQUENCY MATRIX

Category

- 1 Increase in heat removal by the secondary system
- 2 Decrease in heat removal by the secondary system
- 3 Decrease in reactor coolant system flowrate
- 4 Reactivity and power distribution anomalies
- 5 Increase in reactor coolant inventory
- 6 Decrease in reactor coolant inventory
- 7 Radioactive release from a subsystem or component
- 8 Primary system pressure deviation

Frequency

- 1 Moderate Frequency Events
- 2 Infrequent Events
- 3 Limiting Faults - 1
- 4 Limiting Faults - 2
- 5 Limiting Faults - 3

TABLE 5.4.3-2

CATEGORIZATION VS ACCEPTANCE GUIDELINE SCHEME

<u>PROBABILITY GROUP CATEGORIES</u>	<u>MODERATE FREQUENCY EVENTS</u>	<u>INFREQUENT EVENTS</u>	<u>LIMITING FAULT -1</u>	<u>LIMITING FAULT -2</u>	<u>LIMITING FAULT -3</u>
EVENT PROBABILITY	MAY OCCUR DURING A CALENDAR YEAR	MAY OCCUR DURING A PLANT LIFETIME	NOT LIKELY TO OCCUR DURING A PLANT LIFETIME	LOW PROBABILITY OF OCCURRING DURING A PLANT LIFETIME	EXCEEDINGLY LOW PROBABILITY OF OCCURRING DURING A PLANT LIFETIME
ACCEPTANCE GUIDELINES:					
ON RADIATION RELEASE OR FUEL PERFORMANCE	PEAK PIN DNBR $\geq$ 1.19	VERY SMALL FRACTION OF 10CFR100	SMALL FRACTION OF 10CFR100	WELL WITHIN 10CFR100	UP TO 10CFR100
ON PRESSURIZER PRESSURE		WITHIN THE DESIGN BASIS OF THE COMPONENTS INVOLVED			

In order to determine the ability of the plant to meet the radiological consequences criteria⁽²⁾, the radiological dose that would result from releases of radioactivity to a hypothetical receptor is calculated and presented in Chapter 15 of a safety analysis report. Various postulated accidents are accompanied by release of steam to the atmosphere. Any radioactivity assumed to be contained in the Main Steam System has potential for entrainment in the steam and subsequent release to atmosphere. In the analysis, a determination of Main Steam System fission product inventories is made. Assumptions basic to the model for the whole body dose due to immersion in a cloud of radioactivity are:

- i. All radioactive releases are treated as ground level releases regardless of the point of discharge.
- ii. The dose receptor is a standard man, as defined by the International Commission on Radiological Protection (ICRP).
- iii. No credit is taken for cloud depletion by ground deposition and radioactive decay during transport to the Exclusion Area Boundary (EAB) or the outer boundary of the low population zone.
- iv. The whole body dose delivered to a dose receptor is obtained by considering the dose receptor to be immersed in a radioactive cloud that is infinite in all directions above the ground; i.e., an infinite hemispherical cloud.

- v. The concentration of radioactive material within this cloud is uniform and equal to the maximum centerline ground level concentration that would exist in the cloud at the appropriate distance from the point of release.

The whole body dose due to gamma radiation, the skin dose due to beta radiation and the thyroid dose for a given time period are calculated according to Ref. (3). The halogen and noble gas radiological concentration are determined using the following assumptions:

- i. An equilibrium reactor coolant system fission product inventory equivalent to that associated with 1 percent failed fuel.
- ii. A continuous 1 gpm primary-to-secondary leak.

At the end of a cycle, the SSCR behaves very much like a PWR, since in both cases the coolant is composed of essentially light water. However, some of the safety related core parameters of the SSCR are different at the beginning of a cycle where the presence of D_2O and the absence of boron in the coolant cause the greatest difference from a PWR. In the SSCR, the beginning-of-cycle Moderator Temperature Coefficient (BOC MTC) is about 50% more negative than that at end-of-cycle (EOC). In the PWR, the BOC MTC is much smaller (less negative) than that at EOC. The impact of these more negative MTCs on safety analysis results and other changes in the physics parameters were assessed. Results of the evaluation of representative limiting events from each effect category are presented. These events were chosen on two bases: because of their effect on System 80TM design and to illustrate the

differences between SSCR and System 80TM. The safety parameters which were used in the evaluation might in some cases be numerically different than described in Section 5.4.1 (DETERMINATION OF CORE PHYSICS PARAMETERS). The values used in the safety assessment are in general more conservative or represent envelope/ upper-limit values.

5.4.3.2 Radiological Consequences

The magnitude of the additional skin dose contribution at the site boundary due to the high equilibrium tritium concentration in the SSCR was calculated. The concentration level is expected to be 1100 $\mu\text{Ci}/\text{mL}$ D_2O (for derivation of tritium activity see Section 5.4.4.1 - Tritium Activity In Coolant). For System 80TM, the steam generator tube rupture accident resulted in the largest predicted mass of steam released to the atmosphere. Using this mass release, it is assumed that all tritium which leaked to the steam generator secondary side is released. The tritium activity released is calculated to be 300 Ci. For this activity level, the tritium beta skin dose is 0.827×10^{-4} rem. This value is small when compared to the total dose resulting from the release of all radioactive isotopes in the coolant. The value is also a small fraction of the Code of Federal Regulation (10 CFR 100) criteria (25 rem for whole body dose, 300 rem for thyroid dose). The relative importance of the tritium contribution would be even smaller for events during which fuel failures are predicted.

5.4.3.3 Accident Analysis

The following accidents are evaluated in this section:

Increase in Heat Removal by the Secondary System

- Increased main steam flow through an atmospheric dump valve;
- Steam line break;

Decrease in Heat Removal by the Secondary System

- Loss of condenser vacuum with fast transfer breaker failure;

Decrease in Reactor Coolant System Flow Rate

- Total loss of forced reactor coolant flow;
- Seizure of a single reactor coolant pump shaft;

Reactivity and Power Distribution Anomalies

- Control Element Assembly Misoperation;
- Control Element Assembly Ejection;

Increase in Reactor Coolant System Inventory

- Inadvertent operation of emergency core cooling system;
- Light Water Dilution Accident;

Decrease in Reactor Coolant System Inventory

(other than Loss-of-Coolant Accident)

- Steam generator tube rupture

5.4.3.3.1 Increased Main Steam Flow Through An Atmospheric Dump Valve

Summary Description of Accident. This transient is initiated by an atmospheric dump valve (or steam generator safety valve) opening inadvertently. In the presence of a negative moderator temperature coefficient, the increased cooldown results in a power increase. If the initial thermal margin is sufficiently low, a Core Protection Calculator (CPC) low DNBR trip will occur. If not, reactor trip may occur on low steam generator level or low condensate tank level, or by operator action. The pressure differential between the two steam generators is such that emergency feedwater will be initiated to both steam generators upon Emergency Feedwater Actuation Signal (EFAS), which will prevent the affected steam generator from emptying. Plant cooldown to shutdown cooling is accomplished via the turbine bypass valves and the condenser.

Significant Differences Between SSCR And System 80TM. In the following tabulation, the differences in Doppler effect, moderator reactivity and shutdown control rod worth are summarized:

<u>Hot Full Power to Hot Zero Power Doppler Defect</u>		
	<u>BOEC*</u>	<u>EOEC**</u>
UO ₂ (System 80)	1.014%Δρ	0.882%Δρ
UO ₂ (SSCR)	1.562%Δρ	0.891%Δρ
ThO ₂ (SSCR)	2.849%Δρ	1.548%Δρ

Moderator Reactivity Variation (592°F to 445°F)

	<u>BOEC*</u>	<u>EOEC**</u>
UO ₂ (System 80)	<3.25%Δρ	3.28%Δρ
UO ₂ (SSCR)	8.50%Δρ	6.03%Δρ
ThO ₂ (SSCR)	7.79%Δρ	6.31%Δρ

Scram Rod Worth (one stuck rod)

	<u>BOEC*</u>	<u>EOEC*</u>
UO ₂ (System 80)	---	-10.00%Δρ
UO ₂ (SSCR)	-14.10%Δρ	-11.44%Δρ
ThO ₂ (SSCR)	-14.91%Δρ	-11.84%Δρ

* Beginning-of-Equilibrium Cycle

** End-of-Equilibrium Cycle

Effect On Results. The difference in Physics parameters are not large enough to change significantly the total mass of steam released to the atmosphere during this event for the SSCR as compared to that for System 80TM. The greater moderator cooldown reactivity contribution indicates that the thermal margin degradation will occur at a more rapid rate for the SSCR design; however, the CPC low DNBR trip will occur earlier and prevent any fuel damage. If fuel damage up to several percent were to occur, the thyroid inhalation dose at the exclusion area boundary would not be significantly different because the largest contribution to the thyroid dose is from the release of radioactivity which is present in the affected steam generator at the initiation of the event.

5.4.3.3.2 Steam Line Break

Summary Description Of Accident. The steam line break (SLB) accident is defined as a rupture in one of the main steam pipes. A rupture in a main steam pipe increases the steam flow rate from the steam generators, resulting in a rapid decrease in the temperature and pressure of both the steam generator and the reactor coolant system. The trips to protect the plant against violating design limits are provided by low steam generator pressure, low steam generator water level, high linear power level, low pressurizer pressure and low DNBR trips. A low steam generator pressure signal initiates a main steam isolation valve signal which begins closure of the main steam isolation valves to protect the intact steam generator from emptying. The decrease of reactor coolant pressure initiates a safety injection actuation signal, causing borated light water to be injected into the core providing long term reactivity control. Additional protection against loss of the secondary sink is provided by automatic initiation of emergency feedwater to steam generators on a low steam generator water level.

Significant Differences Between SSCR and System 80TM. The SLB accident is characterized by a rapid cooldown due to excessive heat removal from the steam generators through the break. The SSCR has a more negative Moderator Temperature Coefficient (MTC) than System 80TM at beginning-of-cycle. The total moderator reactivity feedback due to a cooldown to 400°F at Beginning-of-Cycle (BOC) conditions is 13.0% Δp and 11.8% Δp for the SSCR with UO_2 and ThO_2 , respectively; it is only 8.0% Δp for System 80. The total fuel reactivity feedback is also different for the SSCR design (2.6% Δp , 4.2% Δp , for UO_2 and ThO_2 , respectively, versus 1.8% Δp for System 80).

Effect on Results. The control element assembly (CEA) insertion upon the reactor trip signal brings the reactor to shutdown conditions with no return-to-power if enough CEA shutdown worth is available. A study of the SLB was performed using full load, two loop conditions with a coincident loss of AC electrical power. In addition, the assumptions recommended in Regulating Guide 1.7⁽²⁾ were used to estimate the required CEA worth to prevent return-to-power. The result of the study indicates that enough CEA worth is available for the UO_2 -fueled SSCR at end-of-cycle to prevent (as in System 80TM) return to power after CEA insertion; however, an additional shutdown CEA worth of $-1.5\%\Delta p$ is required for the UO_2 -fueled SSCR at the beginning-of-cycle. The ThO_2 -fueled SSCR requires additional shutdown worth at beginning-of-cycle ($-0.7\%\Delta p$) as well as at end-of-cycle ($-0.3\%\Delta p$). These results were obtained with conservative analysis methods. It is expected that by using improved analysis methods the increase in shutdown worth for the SSCR can be shown to be unnecessary. It should be noted that return-to-power is an acceptable condition from a safety standpoint as long as the Reactor Coolant System is maintained in a safe status (e.g., the coolability of the core is maintained).

5.4.3.3.3 Loss of Condenser Vacuum With Fast Transfer Breaker Failure

Summary Description of Accident. A loss of condenser vacuum may occur due to the failure of the circulating water system to supply cooling water, failure of the main condenser evacuation system to remove noncondensable gases, or excessive leakage of air through a turbine gland. The fast transfer breaker failure is assumed to occur immediately upon the loss of the turbine resulting in failure to achieve fast transfer of one half of

the non-Engineered Safety Features (ESF) loads from the unit auxiliary transformer to the start-up transformer. The sudden reduction of steam flow caused by the loss of condenser vacuum leads to a reduction of the primary-to-secondary heat transfer. Concurrently, the failure to achieve fast transfer of one half of the non-ESF load causes two reactor coolant pumps to coast down which further reduces the heat transfer capability of the Reactor Coolant System (RCS). These occurrences lead to a rapidly increasing pressurizer pressure giving a high pressurizer pressure trip. Steam is released by the steam generator safety valves until the operator opens the atmospheric dump valves which is assumed to occur thirty minutes after the start of the event.

Significant Differences Between SSCR and System 80TM. The differences in safety analysis parameters which significantly impact the Loss of Condenser Vacuum accident are: a more negative Moderator Temperature Coefficient (MTC) at the beginning-of-cycle in the SSCR than in the System 80TM and higher values of certain fission product specific activities in the primary coolant for the ThO₂-fueled SSCR (e.g., Krypton).

Effect On Results. The effect of the more negative MTC is to mitigate the pressure increase during the event, because as RCS temperature rises from the initial conditions, more negative reactivity would be added to the SSCR design. This effect is closely coupled with D₂O concentration and diminishes with burnup.

The ThO₂ fuel cycle of the SSCR has a different fission product specific activity in the fuel than System 80; greater quantities of Br, Kr, Sr, Y,

Zr, Te-129, I-129, Cs-134,-136, and Pr. However, the more important nuclides in determining the whole body and thyroid radiological doses (I and Xe) are found in similar quantities as in System 80TM design. The results of this event are expected to be less severe in the SSCR than in System 80TM.

5.4.3.3.4 Total Loss of Forced Reactor Coolant Flow

Summary Description of Accident. A total loss of forced reactor coolant flow (TLOF) is caused by a simultaneous loss of electric power to all four Reactor Coolant Pumps (RCP). A TLOF will occur as part of the sequence of events following a loss of all AC power (LOAC). The postulated LOAC to all Reactor Coolant Pumps produces a reduction of coolant flow through the core. The reduced coolant flow rate causes an increase in core average coolant temperature with a concurrent decrease in margin to DNB. A low DNBR reactor trip is generated by the core protection calculators, and the insertion of the scram rods prevents the minimum DNBR from falling below the Specified Acceptable Fuel Design Limit on DNBR.

Significant Differences Between SSCR and System 80TM. The parameters that have the largest impact on the TLOF analysis and are different for the SSCR design are the scram rod worth and the Moderator Temperature Coefficient (MTC). For the SSCR the MTC is more negative and the scram rod worth is greater than for the System 80TM reactor. The difference between delayed neutron fractions, decay constants, neutron lifetimes, and Doppler coefficients do not have a significant impact.

Effect On Results. The combination of increased scram rod worth and more negative MTC for the SSCR would result in a more rapid turn around of the DNBR transient following a TLOF. If the TLOF is used as a Design Basis Event (DBE) for the Core Operating Limit Supervisory Systems (COLSS) thermal margin requirements, then the event response for the SSCR will be essentially the same as that for System 80. The difference will be that the SSCR will require less thermal margin, resulting in increased operating flexibility.

5.4.3.3.5 Seizure of a Single Reactor Coolant Pump Shaft

Summary Description Of Accident. Seizure of a reactor coolant pump shaft can be caused by seizure of the upper or lower thrust bearings or by expansion of the rotor into the stator. The former can occur as a result of a loss of component cooling water to the upper or lower lube oil coolers or by direct loss of lube oil. The latter can occur as the result of a loss of component cooling water to the air cooled rotor and stator heat exchanger. The most adverse single failure which can accompany the single RCP shaft seizure is failure of the turbine bypass valves to open on manual request. If this should occur plant cooldown must be accomplished by relieving steam directly to the atmosphere via the atmospheric dump valves.

Significant Differences Between SSCR And System 80TM. The differences in safety analysis parameters for the SSCR and the System 80TM design which impact the consequences of the accident are:

- 1) Small values for neutron kinetics parameters in the SSCR design (i.e., delayed neutron fraction, delayed neutron decay constants, and prompt neutron life time).

- 2) Greater total shutdown reactivity worth in the SSCR design.
- 3) Higher values of specific activities in the primary coolant for the isotopes of Krypton in the ThO_2 -fueled SSCR design.

Effect On Results. The smaller values for neutron kinetics parameters and the greater total shutdown worth indicates that the SSCR design will yield a more rapid reduction in core power after reactor trip. This will result in a minimum DNBR during the seized pump shaft accident which is higher than that for System 80TM. Core conditions during the transient are not expected to result in the release of additional activity from the fuel to the primary coolant over the amount nominally assumed (1% failed fuel). Activity releases to the atmosphere via the steam generator safety valves and the atmospheric dump valves are expected to result in doses which are small fractions of the Code Of Federal Regulations Guidelines (10CFR100), which is also the case for System 80TM.

5.4.3.3.6 Control Element Assembly Misoperation

Summary Description Of Accident. A Control Element Assembly (CEA) misoperation is defined as any event which could result from a single malfunction in the reactivity control system, with the exception of sequential group withdrawals. Inadvertent withdrawal of a single CEA may be caused by a single malfunction of the reactor regulating system, or the Control Element Drive Mechanism Control System (CEDMCS), or by operator error. The effect of a CEA withdrawal is a positive reactivity insertion and subsequent increase in core power and heat flux.

Significant Differences Between SSCR and System 80TM. The lower values of the kinetics parameters for the SSCR combined with more shutdown worth available on trip would allow for an overall response quicker than that for System 80TM design. This will result in less of a pressure overshoot after trip.

Effect On Results. The increasing magnitude of Doppler and moderator feedback would help this transient by increasing the amount of negative reactivity per degree of heatup. The SSCR analysis results for this event would be similar to but less severe than those for the System 80TM design.

5.4.3.3.7 Control Element Assembly Ejection

Summary Description Of Accident. This transient is initiated by the postulated rupture of a Control Element Drive Mechanism (CEDM) housing or nozzle resulting in the rapid ejection of a Control Element Assembly (CEA) and the subsequent addition of positive reactivity to the core. The reactor core power increases rapidly for a short period of time until terminated by a combination of Doppler feedback and delayed neutron effects. Following this, a reactor shutdown is initiated on high neutron power and the transient is terminated approximately 5 seconds following the start of the event. The potential for fuel damage from a CEA ejection is minimized by restrictions on CEA patterns and/or Power Dependent Insertion Limits (PDIL) during operation, and by a reactor trip, all of which combine to limit the fuel enthalpy, fuel and clad temperatures, and Reactor Coolant System (RCS) pressure during the transient to acceptable values.

Significant Differences Between SSCR And System 80TM. The difference in safety analysis parameters for the SSCR design and the System 80TM design which will affect the CEA ejection accident are:

- 1) Smaller values for neutron kinetics parameters in the SSCR design;
- 2) More negative Doppler feedback;
- 3) More negative moderator feedback, and
- 4) Higher values of specific activities for Krypton in the primary coolant of the ThO₂-fueled SSCR.

Effect On Results. The lower values for the kinetics parameters combined with more shutdown worth available on trip would allow for a quicker power response in the SSCR than for System 80TM. This would result in a smaller pressure overshoot after trip. The larger Doppler and moderator feedback in the SSCR would make the consequences of this transient less severe by limiting the overpower transient. The higher specific activities for Krypton would not significantly increase site boundary doses for this event. Due to the harder neutron spectrum resulting from the presence of D₂O in the coolant, a reduction of the effectiveness of the control rods might be anticipated in the SSCR design. However, the thermal diffusion length is longer in D₂O and the control rods are seen by neutrons from greater distance. As a consequence of these competing effects, the ejected control rod worth in the SSCR is comparable to that in the standard System 80TM design. Overall, the transient would not be as severe in the SSCR as in the standard System 80TM design.

5.4.3.3.8 Inadvertent Operation of the Emergency Core Cooling System

Summary Description Of Accident. The consequences of inadvertent operation of the emergency core cooling system are the same as in the System 80TM design. During power operation, the Core Protection Calculator (CPC) low pressure trip ensures that the reactor coolant system pressure will always be above the shut-off head of the safety injection pumps and the opening pressure of the safety tanks. Therefore, a spurious safety injection actuation signal will not cause injection of emergency cooling fluid into the RCS during power operation.

5.4.3.3.9 Light Water Dilution Accident

Summary Description Of Accident. The reactor core is initially subcritical with all control element assemblies inserted and with the shutdown margin greater than the minimum value consistent with the technical specification limit for cold shutdown ($5.0\% \Delta p$). A CVCS malfunction occurs which causes unborated water to be pumped into the RCS.

For the SSCR design, the resulting decrease in RCS boron and heavy water concentrations cause core reactivity to increase. Sufficient time must exist for detection and termination of the event by an operator before the reactor goes critical.

Significant Differences Between SSCR and System 80TM. The light water dilution accident for the SSCR involves dilution of boron as well as D₂O by light water. The System 80TM incident involves dilution of only boron by light water. The soluble boron worth is insensitive to boron concentration

and burnup. D_2O worth is sensitive to D_2O concentration but is insensitive to burnup. Another significant difference is the absence of boron during hot clean zero and full power conditions (CEAs withdrawn) in the SSCR design. The D_2O concentration varies throughout a cycle in the SSCR design as boron concentration varies in the standard System 80 design.

Effect On Results. The reactivity addition rate used in the dilution incident is based on the maximum time rate of change of the boron and heavy water concentration and the reactivity worth of the soluble boron and D_2O , respectively. Assuming complete mixing of boron or heavy water in the Reactor Coolant System (RCS), the rate of change of boron or heavy water concentration during dilution is described by the following equation:

$$M \frac{dC^a}{dt} = -GC^a$$

where, M = RCS mass

C^a = Concentration of heavy water ($a = D$); concentration of boron ($a = B$)

G = Charging mass flow rate of unborated water.

The magnitude of $\frac{dC}{dt}$ ^a is maximized by assuming:

G = Constant equal to the maximum possible value,

M = Constant equal to the minimum value occurring during the dilution incident.

For the System 80TM design, the maximum reactivity insertion rate is given by:

$$\frac{d\rho}{dt} = - \frac{1}{\tau} C_0^B w^B$$

where, $\tau = M/G$ = Dilution time constant

C_0^B = Initial concentration of boron

w^B = Reactivity worth of boron

For the SSCR design, the maximum reactivity insertion rate is given by:

$$\frac{d\rho}{dt} = - \frac{1}{\tau} (C_0^B w^B + C_0^D w^D)$$

The above relationship was used to compare the reactivity insertion rates during various plant operational modes for both the System 80TM and the SSCR design. Assuming shutdown margin requirements (usually specified in the NSSS Technical Specification) are the same, a faster rate of dilution results in less available time before the core reaches criticality.

The reactivity insertion rates for System 80TM as well as for the UO₂-fueled and ThO₂-fueled SSCR were compared for beginning-of-cycle (BOC) and end-of-cycle (EOC) conditions. The worst case of the System 80 boron dilution analysis was found to occur during refueling where the least amount of time is available for detection of a dilution incident. However, the fastest rate of reactivity insertion for the SSCR will occur during cold shutdown since two absorbers (boron and heavy water) are involved in the dilution.

The comparison revealed that the rate of reactivity insertion is approximately 1.6 times faster for the UO₂-SSCR and approximately 1.8 times faster for the ThO₂-SSCR than for the System 80 dilution (all at BOC conditions).

The time to reach criticality for System 80 during refueling is calculated to be 100.5 minutes. For the SSCR design, the calculated times to reach criticality for the UO₂ and ThO₂ fuel cycles are 62.8 minutes and 55.8 minutes, respectively. After subtracting 36 minutes detection time (the response time for the System 80TM Boronometer was assumed to be representative of the response time of a D₂O concentration measuring unit also) and the 15 minutes NRC requirement⁽⁴⁾ for termination of the incident, a margin of 11.8 minutes and 4.8 minutes remains for the UO₂ and ThO₂ fuel cycles, respectively. This indicates sufficient time exists in the event of a dilution accident for its detection and termination.

If the times to reach criticality due to light water dilution should decrease as a result of redesign of the SSCR, the required 15-minute time period for termination of the incident might exceed the time available. In this case

several possible modifications of design or operating limits would be available to compensate for or limit the increased rates of reactivity change:

- 1) an increased CEA worth for regulating large reactivity changes,
- 2) a greater shutdown worth,
- 3) reduction in charging flow during cold shutdown by operating procedure or design change which would increase $\tau(=M/G)$ and thus increase the time to reach criticality, and/or
- 4) more restrictive controls on CVCS operation.

5.4.3.3.10 Steam Generator Tube Rupture

Summary Description Of Accident. The Steam Generator (SG) tube rupture accident involves a penetration of the barrier between the Reactor Coolant System (RCS) and the Main Steam System, which results from a failure of a steam generator U-tube. In terms of break size, the worst case postulated is a double-ended tube rupture. Following the tube rupture, the RCS pressure would gradually decrease. The primary-to-secondary leak rate and drop in RCS pressure would cause Chemical and Volume Control System (CVCS) charging pumps to start and a reactor trip to occur due to low pressurizer pressure. Following reactor trip, the main steam system pressure would increase to the point where the turbine bypass valves would open and control the main steam system pressure. If turbine bypass is unavailable, the steam generator safety valves would open and control the main steam system pressure. The

operator can isolate the damaged steam generator and cool the NSSS using manual operation of the auxiliary feedwater and the atmospheric steam dump valves associated with the unaffected steam generator any time after reactor trip occurs. The loss of integrity of the barrier between the RCS and main steam system is significant from a radiological standpoint, since a leaking steam generator tube would allow transport of reactor coolant into the main steam system. Radioactivity contained in the reactor coolant would mix with shell-side water in the affected steam generator. During plant operation, some of this radioactivity would be transported through the turbine to the condenser where the noncondensable radioactive materials would be released to the atmosphere via the condenser air ejectors.

Significant Differences Between SSCR And System 80TM. The differences in safety analysis parameters for the SSCR and System 80TM which will affect the consequences of the SG tube rupture accident are:

- 1) Smaller values for neutron kinetics parameters in the SSCR design,
- 2) More negative Doppler feedback,
- 3) More negative moderator feedback, and
- 4) Higher values of specific activities in the primary coolant for Krypton in the ThO₂-fueled SSCR.

Effect On Results. The smaller values for neutron kinetics parameters and the greater total shutdown worth indicate the SSCR design will yield a more rapid reduction in core power after reactor trip. Parametric studies indicated that the more negative moderator temperature coefficient and Doppler effect, will result in very small changes in the total leakage through the break. For the SSCR design, the activity release to the atmo-

sphere is expected to result in doses which are a small fraction of the Code of Federal Regulations Guidelines (10CFR100), which was also the case for System 80TM.

5.4.3.4 Effect Of High Fuel Burnup On Accident Analysis

The results presented in Section 5.4.3.3 are based on physics data for a SSCR employing a three-batch fuel management scheme with a batch-average fuel burnup of 30,000 MWd/t. As described in Section 5.2 (RESOURCE UTILIZATION AND ECONOMICS OF SSCR REFERENCE DESIGN), considerations of uranium utilization as well as power cost economics make an improved fuel management scheme more attractive. This improved fuel management uses five fuel batches and a batch-average fuel burnup in the 50,000 MWd/t range. It results in higher fissile enrichment and, in all probability, will require modifications of the fuel rod design, e.g., longer gas plena and/or annular fuel pellets. The effect of the higher enrichment on safety-related physics parameters and their effect on accident analyses must, of course, be determined specifically for detailed safety analyses; however the results presented in the preceding section are not expected to be strongly effected by the change in physics parameters. The anticipated modifications to the fuel rod design might require some modifications to fuel behavior analysis methods; however, the results of the safety evaluation is not expected to be significantly affected.

5.4.3.5 Summary Of Accident Analysis

Of the accidents which were evaluated in Section 5.4.3.3 only the Steam Line Break accident may need special attention.

The evaluation of the Steam Line Break Accident indicated that in the present core configuration of the SSCR enough control rod worth is available for the UO_2 -fueled core to prevent return-to-power at the end-of-cycle. At the beginning-of-cycle, however, the need for additional control rod worth is indicated to avoid return-to-power. For a ThO_2 -fueled core more control rod worth would be required over the entire cycle. These results were obtained with conservative analysis methods and it is expected that by using improved analysis methods it can be shown that an increase in shutdown worth for the SSCR can be avoided. It should be noted that return-to-power is an acceptable condition from a safety standpoint as long as the Reactor Coolant System is maintained in a safe status (e.g., the coolability of the core is maintained).

The evaluation of the light water dilution accident indicates that, although the safety criteria are met, the margin to the safety criteria is smaller for the SSCR than for System 80TM. If in the evolution of the SSCR design the safety criteria were exceeded, several design or operational measures are available to compensate for the increased rates of reactivity change of the SSCR.

REFERENCES FOR SECTION 5.4.3

"Accident Analysis Excluding Loss-of-Coolant Accident"

1. System 80 CESSAR-F, to be published.
2. "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plant, LWR Edition; US-NRC Regulatory Guide 1.7, Revision 2, September 1975.
3. "Standard for Protection Against Radiation", US-NRC, Title 10, Code of Federal Regulations - Energy, Part 20; May 1977.
4. NRC Standard Review Plan; "Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant (PWR)" - Section 15.4.6; April 1975.

5.4.4

TRITIUM EXPOSURE DURING NORMAL OPERATION

5.4.4.1

Tritium Activity In Coolant

An upper-limit calculation was performed to estimate the concentration of tritium in the coolant of the SSCR. The tritium inventory is determined as the excess of the tritium production in the core over the sum of tritium losses due to radioactive decay as well as losses due to coolant leakage and evaporation losses from the refueling pool during refueling. For the present analysis, however, no leakage and evaporation losses were assumed. In addition, when determining the tritium concentration, only the volume of the primary coolant system ($12,803 \text{ ft}^3$ with 60% D_2O at beginning of cycle) was considered instead of all D_2O containing fluid systems. Under these assumptions, the total tritium activity and tritium concentration in the coolant builds up during the lifetime of the plant and approaches asymptotically an equilibrium value of $2.4 \times 10^5 \text{ Ci}$ or $1100 \text{ } \mu\text{Ci/ml } \text{D}_2\text{O}$ (see Table 5.4.4-1). This equilibrium value which was determined for UO_2 cores (the value for the ThO_2 cores is only about 1/2 of this value due to lower D_2O concentrations and neutron fluxes in the core) was taken as representative of the tritium concentration in the reactor coolant system at the end of the 40-year plant life and was used for all tritium radiological hazards calculations.

When considering the results of the radiological hazards calculations which are presented in the following sections, it should be remembered that the tritium concentrations involved occur only at the end of the plant life and that during the earlier portion of the plant life any tritium exposure

is considerably lower. During the plant life it will be possible to monitor the gradual build-up of the tritium dose and, if necessary, to correct or improve radiation protection measures.

In general, the tritium production in a SSCR is about 20 times higher than in a PWR. Using the equilibrium value of 1100 $\mu\text{Ci}/\text{m}\ell$ D_2O results in coolant activities in the SSCR which range from 660 $\mu\text{Ci}/\text{m}\ell$ liquid at the beginning of a cycle to 22 $\mu\text{Ci}/\text{m}\ell$ liquid at the end of a cycle. These values are one to two orders of magnitude higher than the tritium concentrations for a PWR which are typically in the 1 $\mu\text{Ci}/\text{m}\ell$ liquid range.

TABLE 5.4.4-1
TRITIUM BUILD-UP IN SSCR

Cycle	Total Activity [Curies]	Coolant Activity [$\mu\text{Ci}/\text{m}\ell \text{ D}_2\text{O}$]
1	2.46×10^4	115
2	3.38×10^4	159
3	4.40×10^4	206
4	5.37×10^4	252
5	6.34×10^4	298
6	7.28×10^4	341
10	1.04×10^5	488
15	1.37×10^5	642
20	1.62×10^5	760
25	1.82×10^5	854
30	1.95×10^5	920
∞	2.40×10^5	1100

- Basis:
1. UO_2 core
 2. Coolant System Volume : $12,803 \text{ ft}^3$
 3. D_2O concentration at beginning of cycle : 60%
 4. Assuming no losses other than radioactive decay

5.4.4.2

General Design Approach To Minimize Tritium Exposure

In the design of the D_2O -carrying fluid systems and their respective buildings the reduction of radiological hazards due to tritium was considered. The potential for exposure to tritium during normal operation is minimized by equipment arrangement, leakage control, preventive maintenance and administrative controls.

Equipment Arrangement

Most of the equipment containing D_2O has been clustered in one area of either the Annulus Building or the Solid Waste and Decontamination Building for better operational control, lower D_2O inventory and to consolidate the radiation exposure restricted area. In addition, remote operating mechanisms are employed wherever possible to permit operation of isolation valves from outside the equipment cubicles.

Leakage Control

Most of the D_2O containing systems operate at low pressure and temperatures (0-150 psig @ ambient $\leq 212^\circ F$). To minimize water leakage, the D_2O containing systems use, where possible, diaphragm valves, double packing with leak-off collection systems for control valves and large diameter valves which cannot be replaced by diaphragm type design, and canned rotor pumps (see also Section 5.1.10). Leakage is further reduced by employing all welded connections where possible, and seal welds on flanges that cannot be eliminated (e.g., manholes on tanks).

Preventive Maintenance

A regular preventive maintenance program for keeping seals in leak-tight conditions and for cleaning spills as soon as discovered is an important part of minimizing tritium exposure. These procedures also minimize losses of expensive D_2O .

Administrative Controls

The areas containing D_2O equipment should be classified as Restricted Areas. Before entering, radiological surveys, particularly for tritium, should be performed. When excessive concentrations of tritium are found, means are provided to reduce the tritium concentration in the atmosphere using dilution steam as described in Section 5.4.4.3. In the event that dilution steam does not reduce tritium levels to an acceptable level, or time is not available to reduce the tritium concentration, plastic protective clothing and fresh air breathing equipment must be used to minimize exposure to the maintenance and operations personnel.

5.4.4.3 Tritium Exposure In Containment Building During Power Operation

During normal power operations, the containment building is isolated from the outside environment; therefore, any heavy water/tritium leakage within the containment will have no effect on the offsite dose. Heavy water leakage within the containment, however, will affect the radiation dose inside the containment. In the following sections the effect of uncollected reactor coolant system (RCS) leakage which evaporates into the containment

atmosphere is evaluated. All other leakage from the RCS (i.e. collected leakage) has no effect on the containment's airborne tritium concentration, because it is collected and contained in the Reactor Plant Gaseous Vents and Drains System.

The containment ventilation system for a conventional PWR (see SWESSAR/CESSAR) is designed to maintain the containment atmosphere at 85°F (dry bulb) with a relative humidity between 35 and 40 percent (wet bulb temperatures corresponding to 65°F and 67°F, respectively). The maximum design (upset) condition for the system during normal operation is 105°F (dry bulb) at a relative humidity of 30 percent (wet bulb temperature of 78°F). This upset condition occurs only when one of the three normally operating unit coolers is not in operation due to mechanical failure and is expected to exist only for a short duration since a fourth (spare) unit cooler exists in the system. The moisture in the containment atmosphere stems from D₂O-containing (which also means tritium-containing) leakage from the RCS as well as from leakage of light water from other sources in the containment (e.g., Main Steam System, Reactor Plant Component Cooling Water System).

In order to evaluate the tritium concentration in the containment building air and compare it with the Maximum Permissible Concentration (MPC) for unrestricted access of 5×10^{-6} $\mu\text{Ci/cc}$, calculations were performed for two values of uncollected leakage from the RCS:

1. 65 lb liquid/day; this value is typical for conventional PWR's and
2. 6.5 lb liquid/day; this value is a factor of 10 lower than (1) and is considered typical for the generally more leaktight RCS of a SSCR.

The heavy water concentration of the RCS was assumed to be 60 wt percent at the beginning of the fuel cycle and to decrease linearly to 2 wt percent at the end of a 300 day full power cycle. The tritium concentration of the D_2O was conservatively assumed to be 1100 $\mu Ci/cc$ (the equilibrium tritium concentration achieved after 40 years of operation). The conditions in the containment are considered to remain constant through the entire fuel cycle. The upset design conditions is the worst case because it yields the largest quantity of water vapor in the containment atmosphere.

The containment ventilation system for a conventional PWR is designed to remove 24000 lb/day of water vapor. The condensed water vapor from the ventilation system is directed to the Reactor Plant Nonrecoverable Aerated Vents and Drains System for disposal by the Radioactive Liquid Waste System. Therefore, a total allowable leakage rate within the containment of 24000 lb/day was assumed for the calculations. This means that to reduce the tritium concentration in the containment, light water dilution steam (extraction steam or steam from the auxiliary boilers) may be bled into the containment at a rate which produces a maximum total humidity input to the containment atmosphere of 24000 lb/day. This dilution steam is injected into the containment so as to create a uniform air-water mixture.

The results of these analyses are presented in Figure 5.4.4.-1. With a RCS leakage rate of 65 lb/day, the Maximum Permissible Concentration (MPC) for tritium ($5 \times 10^6 \mu\text{Ci/cc}$) is exceeded during approximately the first 250 days of the cycle. The maximum tritium concentration is approximately $2.6 \times 10^{-5} \mu\text{Ci/cc}$ and occurs at the beginning of a cycle under upset containment atmosphere conditions. If access time in the containment is limited during this period in order that exposure limits are not exceeded, no respiratory equipment or special protective clothing is required. With a RCS leakage rate of 6.5 lb/day, the tritium concentration in the containment remains less than MPC throughout the cycle and unlimited access time to the containment can be maintained throughout. By interpolating between the curves on Figure 5.4.4.-1, it is seen that if the RCS leakage rate is less than 12.5 lb liquid/day, access to the containment building is always unlimited.

Details of the tritium concentration analysis are presented in the APPENDIX.

5.4.4.4 Tritium Exposure in Annulus Building and Solid Waste And Decontamination Building

The interior of the Annulus and Solid Waste and Decontamination Buildings has been arranged so that components containing heavy water with the potential for leakage (i.e. pumps, valves, etc.) are located in cubicles. During normal operation, these cubicles are isolated so that any D_2O /tritium leakage within the cubicle has a negligible effect on the radiation dose in the remaining parts of the building or at the site boundary.

Figure 5.4.4-1
TRITIUM CONCENTRATION IN CONTAINMENT BUILDING

The cubicles are provided with unit coolers consisting of a fan, cooling coil, and ductwork, to recirculate air within the cubicle to maintain component temperatures within design limits. Also, the capability is provided to sample the cubicle atmosphere for tritium from outside the cubicle. If access to the cubicle is required, and the tritium concentration in the atmosphere is too high, means are provided to inject light water dilution steam into the cubicle to reduce the tritium concentration to an acceptable level prior to entry.

In order to estimate the amount of required light water dilution steam, it was assumed that all moisture in the atmosphere of a typical cubicle of 4000 ft³ free volume stems from the leakage of the fluid containing 60 wt% D₂O with a tritium concentration of 1100 µCi/ml D₂O. Assuming further that the cubicle atmosphere is at 105°F and 30% humidity and that 8 hours is a reasonable time period to apply dilution steam, about 30 lb/hr of dilution steam is needed to reduce the tritium concentration in the cubicle to the Maximum Permissible Concentration (5×10^{-6} µCi/cc). This amount of steam can easily be provided from available steam sources (extraction steam or auxiliary boilers.) Details of this calculation are presented in Section 7.2 - CALCULATION OF DILUTION STEAM REQUIREMENTS.

5.4.3.5 Tritium Exposure In Containment Building During Refueling

Water from the warm (between 120°F and 135°F) Refueling Cavity evaporates into the containment atmosphere during refueling. This water contains small concentrations of D₂O; thus tritium will be added to the containment atmosphere.

In order to reduce the D_2O concentration and the resulting tritium concentration in the refueling cavity, the Reactor Cooling System (RCS) is flushed prior to refueling. This operation is described in detail in Section 5.1.12. In summary, the operation proceeds as follows:

1. After cooldown, the RCS is drained into the RCS Loop Drain Tank to the centerline of reactor vessel nozzles.
2. The remaining rest volume in the vessel is flushed to the Reactor Vessel Flush Tank using water from the Refueling Water Storage Tank (RWST). A total flush volume of 90,000 gallons has been assumed.
3. The RCS and the Refueling Cavity are then filled with water from the RWST.
4. After refueling is complete, the water from refueling cavity is pumped back to the RWST.
5. The RCS is then drained to the centerline of reactor vessel nozzles. This water is also pumped back to the RWST.
6. The RCS is then filled from the Primary Grade Water System (PGWS), after which heatup and return to normal operation can commence.
7. During power operation, the RWST is replenished with 90,000 gallons of water from the D_2O upgrader overhead stream with a maximum D_2O concentration of 0.1 wt%.

8. Subsequent refuelings proceed in the similar manner.

The refueling described above yields D_2O concentrations in the refueling cavity of approximately 0.0032% at the first refueling increasing asymptotically to approximately 0.064% at the end of plant life of 40 years. In the above analysis it was conservatively assumed that no dilution from the fuel pool occurs during transfer of fuel assemblies from the fuel pool to the refueling cavity.

Figure 5.4.4.-2 plots the maximum allowable D_2O concentration in the refueling cavity ($C_{R_{max}}$) as a function of relative humidity in the containment for purge exhaust rates of 15,000 and 30,000 cfm (the standard capacities of the containment purge fans for a conventional PWR; see SWESSAR/CESSAR) and refueling cavity temperatures of 120°F and 135°F. For conservatism, the minimum design temperature of the containment atmosphere for a conventional PWR (see SWESSAR/CESSAR) of 65°F was chosen because this condition provides the greatest evaporation rate. Details of this calculation are presented in Section 7.3 - CALCULATION OF EVAPORATION FROM REFUELING CAVITY.

Figure 5.4.4-2 also shows a comparison between the expected D_2O concentration in the refueling cavity with the maximum allowable concentration. At the beginning of plant life, the expected D_2O concentration in the refueling cavity is well below the allowable concentration even with the low containment purge of 15,000 cfm. Thus, the tritium concentration in the containment atmosphere is well below the MPC-value. At the end of plant life, the high containment purge rate of 30,000 cfm is needed to keep the tritium concentration in the containment below the MPC-value. The overall conclusion from

Figure 5.4.4-2
 MAXIMUM ALLOWABLE D₂O CONCENTRATION IN REFUELING CAVITY
 WITHOUT EXCEEDING TRITIUM MPC IN CONTAINMENT

the curves in Figure 5.4.4-2 is that with the containment purge fan capacity available for the conventional PWR the tritium level can be maintained low enough so that access to the containment building during refueling is unrestricted.

5.4.4.6 Tritium Exposure In The Spent Fuel Pool Area

During refueling, fuel assemblies are transferred from the refueling cavity to the spent fuel pool. As part of the transfer process also, some water from the refueling cavity is transferred into the spent fuel pool. Thus, the spent fuel pool water will pick up small quantities of D_2O , and evaporation from the fuel pool surface will carry tritium into the atmosphere of the spent fuel pool building.

An analysis for the spent fuel pool area similar to that described for the containment during refueling (Section 5.4.4.5) was performed. For this analysis, the following parameters from a conventional PWR (See SWESSAR/CESSAR) were used:

1. The maximum long term fuel pool temperature of 120F.
2. The exposed surface area for the fuel pool equal to 1538 ft².
3. The maximum fuel building ventilation purge rate of 32,000 cfm.
4. Minimum design air temperature in the fuel building (65°F) and 0% relative humidity; this maximizes the evaporation rate.

The equilibrium tritium concentration of 1100 $\mu\text{Ci/cc } D_2O$ was assumed to exist in the spent fuel pool.

Using these parameters a concentration of 0.127 wt% D₂O can exist in the fuel pool before the Maximum Permissible Concentration (MPC) for tritium in air (5×10^{-6} μ Ci/cc) is reached at equilibrium in the fuel building atmosphere. Because D₂O is transferred from the refueling cavity to the fuel pool only during refueling, the D₂O concentration in the fuel pool cannot exceed the maximum D₂O concentration in the refueling cavity (0.064 wt% D₂O at end of plant life). Therefore, the tritium concentration in the fuel building will not reach MPC. However, as will be described in the following section, a covering for the spent fuel pool surface might be required in order to not exceed the allowable site boundary dose.

5.4.5 NEUTRON FLUENCE ON REACTOR VESSEL

Spectral shift operation results in a longer mean path length of neutrons in the D₂O-containing coolant and causes a harder neutron spectrum than in a poison-control System 80TM PWR. This increases the neutron fluence at the reactor vessel wall which, in turn, affects the brittle fracture characteristics of the reactor vessel material.

The maximum fluence at the reactor vessel inside surface for neutron energies greater than 1 MeV is 4.08×10^{19} n/cm². This value is reached after 32 Effective Full Power Years at a core power of 3800 MW(th). This fluence value is about 30% higher than the corresponding value for the System 80TM vessel which is 3.1×10^{19} n/cm². When assessing the effect of the higher fluence of the SSCR it is assumed that the axial and azimuthal fluence profile is the same as for the System 80TM core.

One of the criteria for evaluating the effect of the higher neutron fluence on the brittle fracture behavior is the increase in the Nil-Ductility Transition Temperature (T_{NDT} -shift). The maximum change in T_{NDT} -shift is predicted to be about 20°F, from 138°F (System 80TM value) to 158°F. With the initial Reference Nil-Ductility Transition Temperature (RT_{NDT}) for System 80TM of 40°F, the end-of-life RT_{NDT} for the SSCR would then be 198°F versus 178°F for System 80TM. This increase in end-of-life RT_{NDT} for the SSCR appears to be, in general, tolerable. It does, however, reduce the degree of conservatism associated with the System 80TM vessel design and a more detailed analysis would be required to confirm that the increase in fluence can be accommodated if spectral shift control operation would be

considered for a reactor with an existing System 80TM vessel. If spectral shift control is considered for a new reactor, then the effects of the increased fluence can be counteracted by tighter controls on material chemistry (primarily copper and phosphorous content) and fracture toughness of reactor vessel beltline plates and welds. With respect to licensing aspects, exceeding an end-of-life RT_{NDT} of 200°F would introduce consideration involving vessel annealing and surveillance capsule withdrawal schedule according to the Code of Federal Regulations (10 CFR 50, Appendices G and H).

5.4.6

LONG-TERM COOLING

The long-term cooling (LTC) performance of the SSCR following a loss-of-Coolant Accident (LOCA) was evaluated. The objective of LTC is to assure core cooling and avoid blockage of core coolant passages by precipitation of boron from the coolant. This objective is achieved by providing sufficient safety injection flow to the core in order to a) remove fission product decay heat for the long term after a LOCA and b) maintain the boric acid concentration in the reactor vessel below the precipitation level. This is accomplished by providing simultaneous hot and cold leg safety injection to cool and flush the reactor core. The simultaneous hot and cold leg injection has to be started before the boric acid concentration in the reactor vessel has risen so high that precipitation of boron would take place.

The LTC assessment of the SSCR consisted of comparing the pertinent parameters for the SSCR with those for System 80TM. The comparison shows that all of the SSCR parameters are bounded by the System 80TM parameters with the exception of some of the containment characteristics (See Table 5.4.6-1. NOTE: The containment characteristics of the SSCR used in this report are somewhat different from those of the C-E System 80TM PWR because of the differences which exist between CESSAR and SWESSAR and not because of any characteristics inherent in the SSCR itself.) Containment characteristics, in general, affect LTC by way of the containment pressure. The containment pressure determines (via the corresponding boiling temperature) the boron concentration in the reactor vessel beyond which precipitation of boron occurs. In the absence of a specific calculation it is difficult to predict if the long-term containment pressure for the SSCR would be lower than for

System 80TM. However, even if the SSCR long-term containment pressure is conservatively assumed to be atmospheric, a time period of about 3.5 hours after LOCA is available before simultaneous injection, which is started 2 hours after LOCA, is required. Thus, even under pessimistic assumptions, the long-term cooling capability of the SSCR is well within acceptable limits.

TABLE 5.4.6-1

CONTAINMENT DATA COMPARISONS

	<u>SYSTEM 80TM</u>	<u>SSCR</u>
	<u>(CESSAR Containment)</u>	<u>(SWESSAR Containment)</u>
Net Free Volume (ft ³)	3.707×10^6	3.488×10^6
T _{air} initial (°F)	50	85
Spray Flow (gpm)	11,000	7,000
Refueling Water Storage Tank Temperature (°F)	60	40
No. of Fan Coolers On	0	4
Containment Heat Sinks		
Metal (ft ²)	1.3×10^5 ; 1.65 in. thick	1.15×10^5 ; 0.5 in. and 0.375 in. thick
Concrete Walls (ft ²)	6.25×10^5	5.35×10^5

Based on the general similarity of a SSCR plant with a conventional PWR plant, the environmental impact of a SSCR is expected to be in general similar to that of a PWR. However, the SSCR is different from a PWR with respect to release of tritium, addition of the D₂O upgrader facility and cooling water requirements. Their environmental impact is discussed in the following sections.

5.5.1 EFFECT OF TRITIUM RELEASE ON ALLOWABLE SITE BOUNDARY DOSE

5.5.1.1 Comparison Between Conventional PWR and SSCR

When analyzing the effect of tritium on the allowable site boundary dose for a conventional PWR, it is assumed that all tritium produced or entering into the coolant in one year is released to the environment. The total tritium activity released is calculated according to Ref. (1) which states that the tritium produced and hence released may be calculated based on the thermal output of the core using the ratio of .4 Ci (tritium)/MW(th)-yr. Therefore, for a 3800 MW(th) core, the total tritium activity assumed to be released annually for a conventional PWR is 1520 curies. Of these 1520 curies, 50% of the total release (760 curies) are considered to be airborne, and the remaining 50% (760 curies) are considered to be liquid (ratio established in Ref. (1)). The assumption that the tritium produced annually is discharged to the environment is appropriate because the tritium concentration in the Reactor Coolant System (RCS) of a conventional PWR is controlled by programmed releases of reactor coolant (normally by the disposal of boric acid).

The above assumptions are inappropriate for an SSCR because: 1) the annual production of tritium for a SSCR is greater than for a conventional PWR; and 2) due to the high cost of D_2O , programmed releases of reactor coolant cannot be tolerated to control the tritium concentration. Therefore, the RCS tritium concentration increases with plant life. To evaluate the effect of tritium release for a SSCR, the total tritium activity discharged annually for a conventional PWR was compared to the expected release from a SSCR using the equilibrium tritium concentration of $1100 \mu Ci/cc D_2O$ which is a conservatively high value representing conditions at the end of the 40 years plant life. Because the tritium concentration in the SSCR builds up during the plant life, the tritium release prior to the fortieth year is less.

5.5.1.2 Sources of Tritium Release From the SSCR

To evaluate the effect of tritium on the allowable site dose, the sources of tritium release are summarized in the following section. The derivation of the leakage values is discussed in detail in Section 5.1.10.

RCS Leakage. The total estimated annual loss of D_2O due to RCS Leakage is 205 lb. This leakage is assumed to be removed from the containment as liquid from the containment sump during normal operation (i.e., condensed vapor from the containment ventilation coolers) and processed and discharged via the Radioactive Liquid Waste System (RLWS). Because the containment is isolated during normal operation, no airborne releases were assumed.

Auxiliary System and Upgrader Leakage. The total estimated annual loss of D_2O from the Auxiliary Systems and Upgrader is 322 lb. Because the cubicles containing equipment with D_2O are isolated during normal operation, all leakage has been assumed to be removed as condensate via the unit coolers and discharged via the RLWS as liquid.

Leakage to Secondary System Via The Steam Generators. The assumed annual loss of D_2O to the secondary system via leakage in the steam generators is 2790 lb. Using the methodology recommended by Ref. (1), the airborne and liquid releases from the secondary system are estimated to be 930 lb and 1860 lb respectively.

Evaporation During Refueling. The estimated annual loss of D_2O due to evaporation from the refueling cavity is 45 lb. This D_2O is assumed to be discharged as an airborne release via the containment purge during refueling.

Spent Fuel Pool Evaporation. To estimate the amount of D_2O discharged to the environment due to evaporation, it was assumed that the D_2O concentration in the spent fuel pool equals the refueling cavity concentration at the 40th refueling (0.064 wt%). Assuming the worst conditions for evaporation (reference Section 5.4.3.6), it is calculated that the spent fuel pool inventory evaporates at a maximum rate of 466 lb/hr, or 0.298 lb/hr of D_2O . To minimize the loss of D_2O and the tritium releases due to evaporation, it was assumed that the pool is covered during normal operation in order to preclude evaporation. During refueling (7 days) it was assumed that the pool surface is totally exposed and is evaporating at the rate of 0.298

1b/hr D_2O , yielding an annual D_2O loss of 50 lb due to evaporation. This D_2O is assumed to be an airborne release from the fuel building via the ventilation system.

Degasifier. The estimated annual loss of D_2O via the degasifier (assuming continuous degasification) is 135 lb. This D_2O is assumed to be an airborne release from the Annulus Building.

Upgrader Vent. The estimated annual loss of D_2O from the Upgrader Vent is 80 lb. This D_2O is assumed to be an airborne release from the Upgrader.

Sampling. The estimated annual loss of D_2O due to sampling is 65 lb. This D_2O is assumed to be a liquid release from the Radioactive Liquid Waste System (RLWS).

Ion Exchanger Resin Disposal. The estimated annual loss of D_2O from ion exchanger disposal is 168 lb. Most of the D_2O is disposed as solid waste during resin disposal; however, for conservatism, it was assumed that this D_2O is discharged as liquid via the RLWS.

The above annual releases are summarized in Table 5.5.1-1. Using the equilibrium tritium concentration of 1100 $\mu Ci/cc$ D_2O , the estimated annual releases of tritium is 1853 curies total, 595 curies airborne, and 1258 curies liquid.

TABLE 5.5.1-1
ESTIMATED ANNUAL LOSSES OF D₂O
TO THE ENVIRONMENT

<u>Source</u>	<u>Total Release (lb. D₂O)</u>	<u>Airborne Release (lb. D₂O)</u>	<u>Liquid Release (lb. D₂O)</u>
RCS Leakage	205	0	205
Auxiliary System Leakage	300	0	300
Upgrader System Leakage	22	0	22
Primary to Secondary Leakage Via Steam Generators	2790	930	1860
Evaporation During Refueling	45	45	0
Spent Fuel Pool Evaporation	50	50	0
Degasifier	135	135	0
Upgrader Vent	80	80	0
Sampling	65	0	65
Ion Exchanger Resin Disposal	168	0	168
 Total	 3860	 1240	 2620

5.5.1.3

Site Boundary Dose

A comparison of the annual tritium activity release from a PWR and the SSCR results in the following tabulation:

<u>Annual Tritium Release, Ci</u>	<u>PWR</u>	<u>SSCR</u>
Airborne	760	595
Liquid	<u>760</u>	<u>1258</u>
Total	1520	1853

These values indicate that at the 40th year of SSCR operation, the total activity of tritium discharged from the SSCR is approximately 22 percent higher than the total activity assumed to be discharged each year from a conventional PWR. The actual effect of this increase in tritium discharge on the site boundary dose can only be evaluated when specific site characteristics are known. From the available activity data, however, the following qualitative estimate can be made. The calculated airborne release of tritium for a SSCR is less than the releases assumed for a conventional PWR. Therefore, if a site is acceptable for a conventional PWR, airborne releases from a SSCR will be acceptable also. The liquid releases of tritium for a SSCR are approximately 65% greater than that postulated for a conventional PWR. If it is assumed as a worst case that the calculated site boundary dose for a conventional PWR is primarily from tritium, the effect of tritium would be to increase the site boundary dose by a maximum of 65 percent.

It is further assumed that the site boundary dose limit as described in the Code of Federal Regulations (10 CFR 50 Appendix I) is to be met. This limit is 3 mrem/year total body exposure from all pathways for radioactive liquid effluents. Under these conditions, a plant site, for which the liquid release site boundary dose from a PWR is calculated as 1.81 mrem/year, could accept a SSCR also. However, the liquid release dose for a PWR, when evaluated for a specific site, is typically less than 1 mrem/year. Thus, the liquid release dose for a SSCR, when evaluated for a specific site, is expected to be less than 1.65 mrem/year. Therefore, it is not anticipated that the tritium release via liquid effluents will cause significant siting restrictions for a SSCR.

In this context, it is pointed out that, as shown in Table 5.5.1-1, about 70% of the liquid release comes from leakage in the steam generators which was assumed as 30 lb liquid per day. Plant experience shows, however, that the majority of PWR steam generators operate without detectable leakage. Thus, the liquid release of a SSCR is expected to be less than assumed in the analysis which makes siting considerations even more favorable.

REFERENCES FOR SECTION 5.5.1

"Effect of Tritium Release on Allowable Site Boundary Dose"

1. Calculation of Releases of Radioactive Material in Gaseous and Liquid Effluents for PWR's; NUREG-0017, April 1976.

5.5.2

D₂O UPGRADER FACILITY

5.5.2.1

Release of Airborne Tritium

The release of airborne tritium from the D₂O upgrader facility during normal operation is included in the discussion of the tritium release from the power plant in Section 5.5.1 (EFFECT OF TRITIUM RELEASE ON ALLOWABLE SITE BOUNDARY DOSE).

With respect to abnormal releases, an instantaneous airborne release of 5000 gallons of pure D₂O can be tolerated before the whole body dose limit of 0.5 rem (according to the Code of Federal Regulations - 10CFR20) is reached at the site boundary, assuming an atmospheric dispersion factor of $3.8 \times 10^{-4} \text{ sec/m}^3$ (according to SWESSAR). The maximum D₂O inventory in the upgrader facility, assuming maximum liquid levels in storage tanks, is 4900 gallons. Thus, the dose limit will not be exceeded and specific design measures are not required.

5.5.2.2

Release of Liquid Tritium

The release of liquid tritium from the D₂O upgrader facility during normal operation is also included in the discussion of the tritium release from the power plant in Section 5.5.1 (EFFECT OF TRITIUM RELEASE ON ALLOWABLE SITE BOUNDARY DOSE).

With respect to abnormal releases, the limits for liquid release to the ground water are in general dependent on the geology and hydrology of a specific site. In order to facilitate compliance with ground water release

limits the upgrader structures are designed with a "bath tub" - like foundation configuration capable of retaining the liquid inventory of the facility. For some sites, the design of the retention area may require the addition of a steel liner to prevent the release of radioactive liquids through cracks which may develop in the concrete foundation.

5.5.2.3 Upgrader Towers

The upgrader facility contains three distillation towers which stand next to each other and adjacent to the upgrader building. The top of the towers is about 225 ft above ground level, the outside diameter of the towers including insulation is in the 10 to 15 ft range. For comparison, the only other structure of similar height on the site (assuming mechanical draft cooling towers) is the containment building with a height between 220 and 245 ft depending on the specific containment design. The distance between the upgrader building and the containment building is about 350 ft.

Apart from possible considerations of aesthetics which depend on specific site conditions, the distillation towers might create some obstruction in the air space, similar to radio transmission towers. As with other similarly tall structures, sufficient conventional warning and protective measures are available to minimize any hazards associated with the distillation towers so that their environmental impact becomes negligible.

5.5.3

COOLING WATER REQUIREMENTS

The cooling water requirements for the SSCR are only slightly different from the requirements of a conventional PWR. This is the case whether cooling towers are used or a once-through cooling system. The total cooling water requirements for the SSCR is approximately 2% higher than for a PWR. It is expected that this increase will have no detrimental environmental effects or affect the licensability of the SSCR.

SECTION 6 - COMMERCIAL INTRODUCTION OF THE SSCR

6.1 POSSIBLE SCENARIOS FOR INTRODUCTION OF THE SSCR

6.1.1 GENERAL

The spectral shift controlled reactor is unique among the advanced converter systems being considered in NASAP because it is founded on PWR technology and built of PWR components. This relationship with the PWR brings to the SSCR not only marked advantages, but some disadvantages as well. For example, the fuel resource requirements of the SSCR, although significantly below those of the conventional PWR, are not as low as achievable in some other alternatives, e.g., in the heavy water reactor (HWR). Both the basic nuclear properties of light water and the lack of on-line refueling limit the fuel utilization efficiency of the SSCR to below that of the HWR. On the other hand, the SSCR is essentially a current-design PWR to which is added a D_2O upgrader. The major plant components, e.g., reactor vessel, steam generators, primary coolant system, and secondary steam plant are the same as in a PWR. They represent present technology for which manufacturing techniques and capacity already exist. Even the D_2O upgrader is similar to designs already in operation at the Canadian HWR power stations.

Since the proof of the spectral shift control principle has already been demonstrated by the operation of the BR-3 reactor in the late 1960's, and since the components required for heavy water containment and handling are well established by HWR operating experience, the SSCR is considered to be at a stage of development where either a prototype or a large power plant demonstration is required. In the following sections, several scenarios will be presented which represent possible routes for introducing the SSCR into the U.S. electrical power generation mix. Commercial introduction dates which reflect the timeliness of each scenario will be postulated for use in the systems analysis being performed as part of the NASAP evaluation.

6.1.2 ROUTES FOR INTRODUCING THE SSCR

For most alternative reactor concepts at this stage of development, i.e., where the proof of principle has been demonstrated but where there has been no prototype demonstration, a prototype program would be necessary because of the capital cost and high risk associated with bypassing the prototype stage and constructing a large power reactor demonstration. Such a prototype program may also be desirable for the SSCR if the program involved the modification of an existing PWR for spectral shift control. This strategy has the advantage of placing on-line a SSCR in the shortest possible time. The feasibility of this introduction route was demonstrated by the BR-3 conversion from a conventional PWR to a SSCR. The disadvantage of this route is the high cost of downtime and conversion of an operating plant which has a much larger electrical output than that of the 11 MW(e) BR-3.

The primary reason for proceeding through the prototype stage for a new reactor concept is cost and the associated risk. With the cost of nuclear plants running at about \$700/kWe (in constant 1978 dollars), it is indeed difficult to justify the construction of large units without a sound basis in past performance to justify the anticipation of satisfactory performance; utilities might also have difficulty in justifying the incorporation of reactors with uncertain availability on their grids. As a result, logic dictates that small prototype units should be operated to gain experience and confidence in a concept prior to the commitment of large power demonstration units. However, it is possible to design the SSCR so that the plant can be operated in either the conventional poison control mode or in the spectral shift control mode by a simple realignment of valves and changes in operating procedures. As a result, a great majority of the capital investment in the plant and the power output of the plant itself is not at risk, for should spectral shift control fail, the reactor could always be operated as a conventional poison-controlled PWR. Likewise, the potential for serious licensing delays are largely mitigated, since the reactor could initially be operated as a poison-controlled PWR and could easily be reconfigured for spectral shift control once the licensing approvals were obtained. Consequently, the capital at risk is limited to the additional expenditures required to realize spectral shift control, roughly \$26 million in equipment costs, plus rental charges on the heavy water inventory. Since these costs are rather modest compared to the cost of a large power plant, skipping the prototype stage and going to a large power plant demonstration is considered feasible.

The large plant demonstration can be pursued via two routes. The most expeditious route is by taking a reactor plant which is already in the construction phase (most practically in the early stages of construction) and installing the equipment required for spectral shift control operations. In this route, the point of departure for the design of the SSCR systems would be the conventional PWR design. Systems which would not be affected by the conversion to spectral shift control would be left unmodified. No attempt would be made to redesign (e.g., optimize) the PWR systems for spectral shift control.

The major advantage of this large plant demonstration route is that it may permit the early years (about four) of plant procurement, which are spent preparing the environmental impact statement, obtaining the construction permit, and preparing the site for plant construction, to be bypassed. Admittedly, some rework would be required on these early phases of plant procurement, but it may be possible to accomplish this through amendments or modifications to the original work (i.e. the construction permit and the environmental report) while construction is continued. Since the plant can always be operated in the poison-control mode, there is no risk of prolonged delays in plant startup. There is, however, a certain amount of financial risk associated with this procedure (about a 4% capital cost penalty) if an adverse finding/ruling were subsequently made on spectral shift control. This is not anticipated, however, because of the strong similarity between the SSCR and the PWR.

The other route to a large plant demonstration would be the design and construction of a totally new plant. This approach has the advantage of eliminating duplication of effort but has the disadvantage of taking the longest time. The NSSS would still be designed such that the plant could be operated in either the spectral shift control or the poison control mode. However, a number of features would probably be included to minimize the impact of spectral shift control operation. For example, excess reactor coolant system volume could be eliminated to reduced D_2O inventory costs.

This last route to commercialization of the SSCR is the most likely for all plants other than possibly the initial demonstration plant. It employs the most thorough approach and entails the least overall long term cost. The other two routes which are probably more favorable for the initial demonstration plant, will be discussed in more detail in Section 6.6 -

BACKFITTING POTENTIAL.

6.1.3 POTENTIAL INTRODUCTION DATES FOR THE SSCR

The potential introduction dates for the SSCR are dependent on the route taken for introducing this concept. The shortest route would be modification of an existing plant while the longest would be redesign of the entire plant for spectral shift operation. Figure 6.1.3-1 shows an estimated schedule for commercial availability using the three introduction

Figure 6.1.3-1

ESTIMATED SCHEDULE FOR COMMERCIAL AVAILABILITY OF THE SSCR

Figure 6.1.3-1 (Continued)

routes. By backfitting an operating reactor with the modifications necessary for spectral shift operation, a prototype reactor could be placed on-line about seven years after a decision was made to develop the concept. This time would include four years for R&D work and the preparation of a preliminary safety analysis report, and approximately three years to convert an operating reactor to spectral shift control.

By modifying a new reactor under construction, this initial operations date would be increased to about twelve years. For this route to commercialization, the plant conversion time is increased to about eight years versus three years for the backfit route. This time estimate assumes that the total plant procurement time, including licensing and construction, is twelve years and that the conversion starts during the initial stages of the construction time. Thus, about four years are saved compared to the normal plant procurement; the time saved includes that associated with site selection, preparation of the environmental impact statement and the licensing required for a construction permit.

Finally, by designing a reactor specifically for spectral shift operation, initial power operations would be delayed to about eighteen years. This commercialization route includes three years for R&D work, three years for plant design, and the full twelve years for plant procurement (i.e., for licensing and construction).

These introduction dates assume that at each stage in the development process, the necessary approvals/reviews are readily made. For example, it is assumed that once an Environmental Impact Statement is prepared, no

significant adverse findings are obtained which would delay plant construction. If review of the various safety, licensing, and environmental documents raised serious questions regarding the operation of the SSCR, then the introduction dates shown in Figure 6.1.3-1 could be delayed substantially.

In the sections which follow, a number of important aspects relating to the commercial introduction of the SSCR will be presented. In particular, the technical problem areas and the research and development requirements will be discussed with the aim of identifying programs which would facilitate introduction of the spectral shift concept. In addition, the commercial viability of the SSCR is considered to ascertain the incentives for adopting this new concept. Finally, the potential for backfitting spectral shift control into existing PWR plants and plants under construction is investigated in sufficient detail to quantify the costs of these introduction routes.

6.2

TECHNICAL PROBLEM AREAS

6.2.1

GENERAL

In this section, the technical problem areas associated with commercialization of the SSCR are discussed. The cost and time required to perform the necessary engineering R&D to allay concerns in these areas are summarized in Section 6.5 - REQUIRED RESEARCH AND DEVELOPMENT.

During the SSCR design study no technical areas were uncovered where major technological advances are required before spectral shift control becomes technically feasible. This is seen in Table 6.2.1-1 which shows a survey of technology requirements for major plant components. The format of Table 6.2.1-1 is geared to the comparison of various reactor concepts evaluated within the frame work of NASAP (Nonproliferation Alternative Systems Assessment Program) and was taken from Ref. (1). In general, all equipment modifications or additions for spectral shift control can be accomplished by employing contemporary technology or, at most, modest extentions from it. Technologies and design concepts which are in use in the Light Water Reactor or Heavy Water Reactor field can be successfully applied to realize spectral shift operation.

The proof of principle of spectral shift operation was demonstrated more than 10 years ago by the Vulcain experiment in the BR-3 nuclear plant at Mol, Belgium. The BR-3 plant, after two years of operation as a conventional

TABLE 6.2.1-1

SPECTRAL SHIFT CONTROL - SURVEY OF TECHNOLOGY REQUIREMENTS

	No new knowledge required	Contemporary technology with modified configuration/application	Modest improvement in performance or size from present knowledge	Modest improvement in performance or size and modified configuration/application	Major improvement in performance or size from present knowledge	Major improvement in performance or size & modified configuration/application	New technology (e.g. materials) required to meet system requirements
a. Nuclear Fuel (see footnote a.)	X						
b. Reactivity Control Systems		X					
c. Reactor Vessel (see footnote b.)		X					
d. Core Support Structure	X						
e. Reactor Vessel Internals including Shielding, Ducting, Control Rod Guides, Baffles, etc. (see footnote b.)		X					
f. Primary Coolant Pumps and Auxiliary Systems		X					
g. Primary coolant chemistry/Radio-chemistry control		X					
h. Primary system heat exchangers	X						
i. Reactor instrumentation		X					
j. Emergency core cooling/safe shutdown systems	X						
k. Containment, containment cleanup systems and effluent control systems					X		
l. Other accident mitigating systems. i.e. plant protection systems		X					

TABLE 6.2.1-1

SPECTRAL SHIFT CONTROL - SURVEY OF TECHNOLOGY REQUIREMENTS

	No new knowledge required	Contemporary technology with modified configuration/application	Modest improvement in performance or size from present knowledge	Modest improvement in performance or size and modified configuration/application	Major improvement in performance or size from present knowledge	Major improvement in performance or size & modified configuration/application	New technology (e.g. materials) required to meet system requirements
m. On-site fuel handling, storage/shipping equipment		X					
n. Main turbine	X						
o. Other critical components, if any			X				
p. Balance of plant components		X					

- Note: a. Only items specific to spectral shift operation are listed. Development of high-burnup fuel was assumed to be part of PWR Improvement Program.
- b. Some analyses might be required because of the higher fluence in the SSCR compared to the PWR.

11 MW(e) PWR, was modified for spectral shift control operation and successfully operated with this mode of control between 1966 and 1968. Subsequently, the BR-3 reactor was returned to conventional PWR operation.

With respect to the applicability of PWR technology to heavy water reactor systems, the successful operation of the Argentinian Atucha reactor is extremely encouraging. The Atucha Plant is a heavy water reactor fueled with natural uranium. It was designed and built by the German company Kraftwerk Union, and is a 340 Mw(e) Plant using a PWR-type pressure vessel operating at 1700 psi. In the design of the plant, maximum use of PWR technology was made by using PWR design concepts and PWR components. By relatively minor adaptations of PWR components, excellent leak-tightness was achieved. Irrecoverable losses of D_2O per year amount to about 0.4% of the total D_2O inventory. The operating experience of Atucha shows that large heavy water systems with a system pressure close to a PWR can be built and operated to extremely high standards of leak-tightness.

The development of high burnup fuel certainly would have to be classified a major technical problem area. The development work requires extensive efforts by analytical, design and material-science disciplines and must include inpile irradiations. This fuel development, however, is not a specific prerequisite for spectral shift control. Instead it is seen as a generic PWR development effort which is equally desirable for conventional poison control PWR's as well as for spectral shift control PWRs. For the following discussion it is assumed that the high burnup fuel development would proceed as part of a generic PWR improvement program and need not be considered as a specific development item associated with spectral shift control.

6.2.2

SSCR-FEATURES REQUIRING SPECIAL ATTENTION

Although no major technical problem areas are associated with spectral shift control operation, there are several technical items which are specific features of the SSCR and require special attention. These features are discussed in the following sections.

6.2.2.1 Refueling-Coolant Exchange

As described in Section 5.1.11-REFUELING, the tritium level in the coolant at the end of plant life is estimated to be high enough so that the PWR refueling procedure requires modification. The idea is to replace the D_2O (and tritium) - containing water volume, which remains in the reactor vessel after the draining of the reactor coolant system, with light water, before the vessel head is removed. It is difficult to quantify the effectiveness of this exchange or flushing process without experimental verification. Although the uncertainty associated with the flushing process will not affect the technical feasibility of the basic exchange concept, it will affect the size of the tankage which need be provided to receive the flush water. The tankage size, in turn, affects the degree of modification to the Annulus Building to accommodate the tanks. For sizing the tanks in the present study, the coolant flushing process was conservatively assumed to be similar to a bleed/feed operation which results in large flush volumes.

6.2.2.2 Radioactivity of Liquid Plant Effluents

Under conservative assumptions, the tritium activity due to the liquid effluents from the SSCR at the end of plant life is estimated to be up to about 65% higher than for a PWR because of the higher tritium activity in

the coolant. (See Section 5.5.1 - EFFECT OF TRITIUM RELEASE ON ALLOWABLE SITE BOUNDARY DOSE). Although this estimate is considered conservatively high and the actual tritium activity of the liquid effluents is expected to be lower, it is anticipated that more detailed analyses of possible release paths to the environment and of the release quantities will be required for the SSCR than for the PWR.

6.2.2.3 D₂O Concentration Measurements

During a core cycle, the D₂O concentrations in the SSCR range from as high as 85 wt% (bottoms of the D₂O upgrader) to as low as 0.1 wt% (overhead of the D₂O upgrader during special operation). Although there are on-line D₂O-concentration measurement units commercially available to measure the high or low concentrations, there are at present no on-line instruments which cover the entire range of concentrations. The general measurement principle (measurement of infra-red absorption spectra) can be used over the entire range of D₂O concentrations of the SSCR.

In developing the modifications for the plant systems, determination of D₂O concentrations by sampling was generally assumed and appears to be feasible. However, it is expected that for more convenient plant operation reliable, accurate on-line D₂O concentrations measurements will be highly desirable. The measurements units could be used for monitoring as well as for control purposes.

6.2.2.4 Cubicle Tanks

To optimize space utilization in the Annulus Building, it is planned to equip several building cubicles with stainless steel liners and use the cubicles for coolant storage (e.g. Refueling Water Storage Tank, Letdown Tanks, Reactor Vessel Flush Tank). This arrangement has the added benefit of minimizing D_2O inventory by minimizing interconnecting piping. It also places the tanks in a seismically protected building.

The lining concept for these cubicle tanks is the same as the one employed in the refueling cavity or in the spent fuel pool. The storage cubicles, however, are rectangular or trapezoidal in plan. Thus, from a material strength standpoint, they are an extremely inefficient shape to hold liquids. Further development of the layout would be required to verify that the lined cubicle concept is an optimum solution for liquid storage in the SSCR, particularly for the large Refueling Water Storage Tank.

6.2.2.5 Effect of Load Change Capability on Fuel Duty

As described in Section 5.1.13 - LOAD CHANGE CAPABILITY, the SSCR requires substantially more frequent and more extensive insertion of CEA's (Control Element Assemblies) in order to provide load change capabilities similar to those of the System 80 PWR. Significant short or long-term CEA insertions cause distortions of the local power shape. Thus, more analysis is needed to ascertain whether the impact on the thermal margin of the core due to the CEA insertion near full power can be tolerated. The local power shape perturbations also may effect the pellet-cladding interaction (PCI) behavior

of the fuel rods possibly making the fuel rods more susceptible to failure due to PCI. Thus it is expected, that fuel which is highly resistant to PCI will be required for the SSCR. In this aspect, the SSCR will benefit from ongoing PWR fuel development programs which are designed to investigate the PCI behavior of PWR fuel.

In general, the fuel duty can be decreased by decreasing the severity of load changes. Thus, load change capability can be traded against fuel duty. For a more detailed SSCR design, studies will be required which determine the balance between load change capability reduction and fuel duty reduction.

6.2.2.6 Verification of Physics Parameters

In support of licensing analyses, the verification of physics analytical methods and safety-related parameters is needed. Although some of these data have been developed for UO_2 fuel during the initial SSCR work performed by the USAEC in the 1960's, additional work is still required. For the denatured thorium fuels or for plutonium/thorium fuel no previous physics verification has been performed.

6.2.2.7 Thermal-Hydraulic Data For D_2O/H_2O Mixtures

The thermodynamic and physical property data of D_2O/H_2O mixtures must be established for detailed steady state and transient (safety analysis) calculations. Some data, primarily for reactor grade (essentially pure) D_2O are available from Canadian and German sources but their applicability

to SSCR-type D_2O concentrations must be determined. Critical heat flux data for PWR fuel geometries and D_2O concentrations typical of the SSCR are essentially non-existent. Also, the effects of the D_2O/H_2O mixtures on the safety-related operation of reactor components, such as pump coastdown and control element drop times must be evaluated.

REFERENCES FOR SECTION 6.2

"Technical Problem Areas"

1. "System Data Requirements for NASAP Scoping Studies," U.S. Department of Energy, January 1978.

The SSCR is basically a modified PWR. The differences between the SSCR and the PWR from a licensability standpoint are primarily associated with differences in core characteristics which are most pronounced at the beginning of a core cycle due to the presence of D_2O and absence of boron in the coolant of a SSCR. These differences essentially disappear toward the end of a core cycle. Due to the relatively close similarity between SSCR and PWR no significant difference in plant response has been uncovered in the safety evaluation as described in Section 5.4 - SAFETY ASSESSMENT. No major changes in pressure boundaries, heat transfer equipment or containment arrangements from the PWR are necessary for safety reasons. Thus, it is concluded that with respect to reactor safety the SSCR is fundamentally licensable.

The differences between SSCR and PWR which do exist result in some differences in safety-related parameters, e.g., more negative moderator temperature coefficient of reactivity at the beginning of cycle. This causes the response of the SSCR to certain accidents, in particular to the steam line break, to be less desirable than for the PWR. Also, reactivity addition via accidental dilution of coolant by unborated light water is faster than for the PWR. Furthermore, the fast neutron fluence on the reactor vessel wall is about 30% higher than for the PWR. The most significant difference to the PWR is probably the tritium activity in the coolant of the SSCR which, depending on the D_2O concentration in the coolant during the cycle, is between one and two orders of magnitude higher at the end of a forty-year plant life.

The close similarity between SSCR and PWR in terms of design concepts, plant components and overall technology implies that the General Design Criteria and specific safety criteria which are applicable for PWRs can be equally well applied to the SSCR. The only major plant system which is specifically needed for spectral shift operation but not for conventional PWR poison control is the D₂O Upgrader Facility. The function of this facility is not safety-related. However, the liquid contained in the facility becomes radioactive because of the gradual buildup of tritium in the reactor coolant system of a SSCR. At present, there are no specific safety-related design criteria for a D₂O-Upgrader Facility. For the present study, guidelines by the US Nuclear Regulatory Commission for the design of structures containing radioactive waste management systems⁽¹⁾ were used for the design of the D₂O Upgrader Facility, which we believe to be adequate. However, the regulatory authorities might find it necessary to develop specific criteria for D₂O Upgrader Facilities.

The safety evaluation of the SSCR performed in this study indicated that all safety criteria applicable to the PWR are met for the conceptual SSCR design. For licensing a SSCR, the specific spectral shift related items mentioned above will, of course, have to undergo the customary NRC safety review. This will result in an additional safety analysis effort for the SSCR which is in excess of the analyses provided in the Standard Safety Analysis Reports (e.g., CESSAR or SWESSAR) of a PWR. Although the safety analysis effort of the SSCR is likely to be more extensive than for a PWR, it is expected that after submittal of a safety analysis which is based on a detailed design, the SSCR can be licensed within a reasonable time period.

REFERENCES FOR SECTION

"Licensability"

1. Design Guidance for Radioactive Waste Management Systems Installed in Light-Water-Cooled Nuclear Reactor Power Plants, Branch Technical Position - ETSB No. 11-1 (Rev. 01); November 1975.

6.4

COMMERCIAL VIABILITY OF THE SSCR

6.4.1

UTILITY PERCEPTION OF THE NEED FOR AN ADVANCED CONVERTER

In a recent survey of utility management,⁽¹⁾ the consensus opinion was that at the present time there is no need for a new reactor technology to extend the supply of uranium. The present estimated resource base is felt to be adequate for the time frame of utility decisions - 20-30 years. Also, in the past, utilities have been able to negotiate long-term uranium supply contracts which guaranteed sufficient fuel for each of their reactors. Thus, there appears to be little incentive from a fuel supply standpoint for a utility to adopt an unproven technology.

The development of technologies which exhibit improved proliferation resistance is also felt to be outside the domain of utility decisions. Threats of nuclear/special nuclear material diversion are very small in the U.S. at the present time. It is difficult, therefore, to place substantial importance on these topics when deciding whether to order/develop one reactor type versus another. Policy actions to reduce the threat of weapons proliferation are viewed as the responsibility of the federal government.

The present status of the fast breeder reactor development program is also a deterrent for utility management to become seriously involved in developing a new reactor technology. The breeder is the only near term option which can free the nuclear economy from a limited resource base. However,

a definite, permanent policy on the development of this technology does not exist in the U.S. The breeder program stands out as an example of what might happen to any new and as yet unproven concept. Furthermore, the timeframe during which an advanced converter reactor can play a significant role in contributing to uranium resource conservation and overall power generation economics is dictated by how fast the breeder is developed. For example, the sooner the breeder is ready for deployment, the shorter the time available for the advanced converter to repay the investment required to develop it.

6.4.2 POTENTIAL BENEFITS OF THE SSCR - ARE THEY SUFFICIENT?

The spectral shift controlled reactor is a more fuel efficient converter reactor than the conventional pressurized water reactor as discussed in detail in Section 5.2 - RESOURCE UTILIZATION AND ECONOMICS OF THE SSCR REFERENCE DESIGN. On the UO_2 once-through cycle, the SSCR requires about 11% less U_3O_8 than the PWR over a 30-year plant lifetime. On the thorium fuel cycles, where the bred and residual fissile material is recycled, a reduction in fissile material consumption up to 27% can be obtained by using spectral shift control instead of poison control. It is questionable whether the magnitude of these savings is sufficient to warrant the development of an alternate reactor concept, particularly since the overall savings in U_3O_8 is obtained by weighting the individual SSCR uranium savings by the fraction of the total installed electric capacity which the SSCR constitutes. Even with a massive effort to deploy the SSCR,

less than 50% of the total nuclear capacity could be operated in this control mode by the year 2000, the remainder consisting of conventional light water reactors. By this time, a major portion of the U.S. estimated uranium reserves will have been committed to operating reactors. Furthermore, if one considers the earliest possible introduction dates for the thorium fuel cycles (early to mid 1990's) and the likelihood of these dates coming to pass, it appears that the overall impact on uranium conservation of full deployment of the spectral shift concept would be at most about 5%.

The most advantageous use of the spectral shift controlled reactor may be in a symbiotic nuclear economy in which plutonium fueled breeder reactors produce U-233 for use in converter reactors. The lower consumption of U-233 and the higher production of plutonium makes the SSCR a better choice than the LWR for this role. Operating on the denatured U-233/ThO₂ cycle, the SSCR requires about 17% less U-233 than the PWR while producing about 16% more fissile plutonium. Even this application of the SSCR may be unjustifiable when one considers the costs of development and the impact on power generation costs of spectral shift control.

6.4.3 THE COSTS OF SPECTRAL SHIFT CONTROL - WHO WILL PAY?

The capital costs of the SSCR (excluding the costs of heavy water) are about \$26 million greater than those of the conventional PWR. When the costs of heavy water are included, a power cost penalty of approximately 1.1 mills/kWh occurs. To make the SSCR competitive with the PWR, the improved fuel utilization of the SSCR must result in a fuel cost savings of at least this amount. As shown in Section 5.2 - RESOURCE UTILIZATION

AND ECONOMICS OF THE SSCR REFERENCE DESIGN, only when the price of U_3O_8 reaches \$100/lb or more does the savings in fuel cycle cost reach this size. This implies that in the near term, when the price of uranium is relatively low, there is no economic incentive for a utility to adopt the spectral shift controlled reactor.

This lack of a near-term economic incentive is a major obstacle to the development of the SSCR. There is no substantive reason for either a utility or a reactor manufacturer to pursue the concept. The return on their investments would hinge on the future price of uranium which is uncertain at best. Future discoveries of economically recoverable U_3O_8 and/or improved methods of mining presently uneconomical grades of ore, may forestall the SSCR uranium breakeven price for an unforeseen number of years. It would therefore be very difficult for management of either industry to favorably embrace the spectral shift control concept. Only with strong governmental support would the concept advance beyond the conceptual design stage.

Since the SSCR can always operate in the poison control mode, the financial risk to a utility is limited to the extra capital equipment required to realize the concept and to rental/use charges on the heavy water inventory. Because of the lack of economic incentive to the utility, the government would have to purchase the extra equipment required for spectral shift control for the early units until the power generation costs of the SSCR are competitive with those of the LWR. This timeframe may be well beyond that of the initial reactor demonstration. Because of the long lead times on installation of nuclear plants and because the heavy water inventory can be purchased a relatively short time prior to a reactor's commercial operation

date, heavy water inventory subsidies may only be required on the initial few plants. In any event, it is clear that the federal government will have to bear the costs of developing the SSCR. The proper economic incentives do not exist for the nuclear industry to commit a significant effort to this endeavor.

6.4.4 PROSPECTS OF SUCCESS

There is a high probability that the SSCR can be successfully developed, licensed, constructed, and operated in the U.S. because of its close resemblance to the conventional PWR. Only a limited amount of research and development (R&D) work is required, primarily for experimental verification of procedures and extrapolation of data. This engineering R&D, which is discussed in Section 6.5 - REQUIRED RESEARCH AND DEVELOPMENT, can most likely be completed in two to four years for a few million dollars. The small development program required for this reactor concept enhances the prospects of its adoption.

The possibility of delays and unforeseen costs associated with the licensing process are one of the largest unknowns and therefore one of the major risks associated with any new reactor technology. The licensing review cannot be conducted until detailed designs, tests, and complicated analyses methods exist. In general, industry is reluctant to invest its resources in these programs when there are no assurances that they will be acceptable. In the past, the regulatory agency has had an essentially moving target for its review process. A series of changing guidelines with

ratcheting criteria and sometimes non-definitive specifications, such as "low as practicable," have been imposed on the nuclear industry. LWR technology has "broken ground" in these areas and has met the new criteria in each instance. Reasonably well-defined procedures and precedents now exist to help buoy a new applicant through the review process.

The existing LWR licensing procedures and guidelines would be, to a major extent, directly applicable to the SSCR, giving a potential reactor operator a reasonable assurance of a favorable review by the regulating agency. The compatibility of the present licensing process with the spectral shift control concept thus enhances its prospects for early acceptance.

Since the equipment and layout of the SSCR are nearly identical to that of the conventional PWR, current manufacturing and construction practices can be used with only slight modifications. Even the D₂O upgrader facility, which is the only major component unique to the SSCR, is essentially identical (except for size) to the upgraders presently is used in the Canadian heavy water reactor program. Thus, unforeseen manufacturing and construction problems are not expected for the SSCR, not even for the initial demonstration plant(s). This reliance on past experience will help hold the costs and construction schedules of the early plants to the design estimates and thus enhance the acceptability of the concept.

The excellent operating experience of the BR-3 and Atucha reactors is indicative of the potential operating success of the SSCR. The availability of the reactor plant should be essentially equal to that of the

conventional PWR since the majority of the plant systems are identical to those of the PWR. Malfunctions of the D₂O upgrader will, in general, not affect reactor operations unless these malfunctions persist over an extended period. The reactor can operate independently of the upgrader by storing letdown coolant for future processing while the upgrader can be run if necessary with auxiliary site steam when the reactor is shutdown. There are no plant operations associated with spectral shift control which are anticipated to have an adverse effect on the operating performance or availability of the reactor.

The aforementioned items all point to the successful implementation of the spectral shift control concept. What remains the over-riding consideration, however, is whether/when the SSCR can compete economically with the conventional LWR.

Unless some intangible or external factor influences the decision process, the SSCR will not be adopted in the near term because of its higher power generation costs. In the long term, when the price of uranium becomes high enough the spectral shift control concept will become attractive. However, if the necessary development and demonstrations have not been completed, the concept will not be available for deployment at that time.

The SSCR has a number of distinguishing features which affect its potential for commercialization. Some of these make the SSCR more attractive from a commercialization standpoint than other advanced converter reactors.

However, the true measure of its commercial viability must be assessed with respect to the conventional light water reactor (LWR) since the SSCR must compete against this reactor type for a share of the nuclear market. The list of features given below summarizes the advantages and disadvantages of the SSCR relative to other reactor concepts.

6.4.5.1

Advantages of the SSCR

- The SSCR is based on the well established LWR technology.
 - This results in a low financial risk and no risk to station generating capacity.
 - The manufacturing technology and capability already exist except for D₂O production capacity.
 - The development effort is modest and is similar to that of other LWR improvements presently under consideration by DOE.
 - The licensing criteria and standards are essentially the same as for the LWR and do not, in general, require establishing new procedures and precedents.

- Uranium requirements of the SSCR are significantly below those of the LWR.
 - On the UO_2 once-through cycle, 11% less U_3O_8 is required by the SSCR.
 - On the thorium cycles, up to 27% less fissile material is required by the SSCR than by the LWR on similar cycles.
 - The lower U_3O_8 usage reduces the requirements for expanding the uranium mining and milling industries.
 - In the long term, the lower uranium consumption of the SSCR will help hold down the price of U_3O_8 because it will extend the time over which the more easily recovered ore will be available.
- Separative work requirements of the SSCR are significantly below those of the LWR.
 - On the UO_2 once-through cycle, 15% less separative work is required by the SSCR.
 - On the denatured U-235/ ThO_2 cycle with storage of plutonium, the separative work requirements are 12% less than those of the LWR.

6.4.5.2. Disadvantages of the SSCR

- The power costs of the SSCR are higher than those of the LWR until the price of uranium reaches about \$100/lb U_3O_8 .

- This is a disincentive for developing the SSCR in the near term, although development in the long term may be economical.

- Adequate heavy water production capacity does not presently exist in the U.S.

- This lack of adequate capacity could be a pacing factor for the rate of deployment of the SSCR if it were adopted.

6.4.5.3 Conclusions Regarding the Commercial Viability of the SSCR

In order for the SSCR to be adopted by an electric utility, the power costs of the SSCR would have to be lower than those of the LWR or any other practical alternative. It appears unlikely that this will occur in the near term. However, if the cost of heavy water were to decrease substantially, e.g., by the development of a low cost isotopic separation technique, then this situation could change dramatically. At present uranium prices, the SSCR would be competitive with the LWR if the price of heavy water were \$20/lb (versus \$80/lb used in this study). As the price of uranium increases, the breakeven heavy water price also increase so that at \$100/lb U_3O_8 the breakeven heavy water cost is \$80/lb D_2O . The feasibility and cost of an improved heavy water separation process are beyond the scope of this report. It is felt, however, that effort should be directed in this area if the SSCR is judged a desirable alternative to the LWR.

From a commercialization standpoint, the SSCR should probably be viewed as a major advancement in the evolution of the improved PWR. A number of modifications/operating strategies, which show considerably less potential than spectral shift control, are currently being investigated for the PWR. For other than the five-batch, extended burnup fuel management scheme, it appears that only small improvements in fuel resource utilization can be attained with these modifications. The R&D requirements and licensing effort to commercialize these improvements will probably be comparable to that needed to develop the SSCR. It seems logical therefore to take the step required to pursue spectral shift control as long as the resource savings of this concept are in the range of interest for LWR improvements. A larger benefit can be obtained in a shorter time than by trying to combine a number of the marginal improvement techniques presently being investigated for the LWR.

REFERENCES FOR SECTION 6.4

"Common Viability"

1. The Economics and Utilization of Thorium in Nuclear Reactors - ORNL/TM-6331, Oak Ridge National Laboratory, May 1978.

Although no areas were identified in the SSCR study where major advances in technology are required to realize spectral shift operation, the deployment of SSCR plants will require some engineering development work, primarily for experimental verification. The following description of research and development items is based on and related to the discussion in Section 6.2 TECHNICAL PROBLEM AREAS.

6.5.1

REFUELING COOLANT EXCHANGE

The objective of this development is to quantify the effectiveness of the coolant exchange operation in the reactor vessel prior to refueling. In particular, experimental data are required to determine flush water volumes and the obtainable final D_2O concentration of the residual water volume in the reactor vessel.

The tests could be performed with a 1/5-scale model of a reactor vessel which is available at Combustion Engineering. Although this scale model is limited to a maximum water temperature of 80°F, it could be utilized for a simplified test program. If water temperatures more similar to the SSCR (about 170°F) are desired a new vessel would need to be fabricated. For cost estimating purposes, tests with a new 1/5 and a new 1/2 scale model were assumed. The tests would be initially conducted with dyes for flow visualization. For the subsequent generation of quantitative data, the use

of salt traces and conductivity probes appears feasible. Conductivity probes will be placed in the simulated core region and at selected locations in the reactor vessel wall, outlet and inlet pipes. Utilization of twenty-four conductivity probes was assumed. The overall duration of the test programs is estimated to be between 9 and 12 months with the actual testing (about 30 tests) taking up about two months.

The cost of the tests is estimated to be \$125,000 for the simplified 1/5-scale test, \$205,000 for the 1/5-scale test and \$330,000 for the 1/2-scale test. These estimates include a 3 man-months effort for data analysis and a 4.5 man-months effort for test data documentation and for the preparation of a test report.

6.5.2 D₂O CONCENTRATION MEASUREMENTS

The objective of this development is to provide reliable, accurate, on-line measurement units for determining the D₂O content of D₂O/H₂O mixtures in the range occurring during spectral shift operation (0.1 wt% D₂O to 85 wt% D₂O.)

At Canadian CANDU power plants and heavy water plants associated with the CANDU power plants, heavy water liquid analyzers of the type commercially available from BARRINGER RESEARCH LIMITED; Rexdale Ontario, Canada, are in regular use. The working principle of these instruments is the measurement of the absorption of infrared radiation by the three stable water species H₂O, D₂O and HDO. The instruments are fully automatic and are developed for continuous, real time analysis.

In the Canadian application, high D_2O -concentrations (>99 wt%) and low D_2O concentrations (<2 wt%) are measured with two separate unit types which are designed for the specific concentration ranges. At present, there is no single instrument commercially available which would cover the entire range of D_2O concentrations occurring in the SSCR. However, the successful development of such an instrument appears quite feasible. The measurement principle as used in the present analyzer would be retained and a measurement resolution of 0.1% $D/(D+H)$ appears achievable. The approximate cost for such a development is \$100,000 and it is estimated that the effort would require about a one year's time.

6.5.3 CRITICAL HEAT FLUX DATA

Critical heat flux data for PWR fuel geometrics and D_2O concentrations typical of the SSCR are essentially nonexistent. These data are needed for detailed thermal-hydraulic design calculations and safety analyses.

The first step in developing the data would be to determine whether critical heat flux correlations which were developed for light or heavy water can be applied to or corrected for heavy water/light water mixtures. It is expected that as a minimum small-scale, bench mark heat transfer tests are required for confirmation of applicability or for generation of correction factors.

If this approach proves impractical or provides results with excessively large error bands, specific heat transfer tests might become necessary to develop the required correlations for heavy/light water mixtures. The cost of a test program is expected to range between \$100,000 and \$250,000, depending on whether bench mark tests are sufficient or a more extensive test program is required.

6.5.4 PHYSICS VERIFICATION

This effort consists of verification of physics analytical methods and safety-related parameters. This effort would help provide the data base required to support the licensing process. For UO_2 fuel, this data base has been partially developed in the initial SSCR development work performed by the USAEC in the 1960s. However, additional work, primarily in the area which establishes reactivity predictions, power distribution, D_2O worths, and control rod worths is still required for uranium fuel. No previous physics verification has been performed for the denatured thorium fuels or for plutonium/thorium fuel. The cost of the R&D effort could range from several million to well above ten million dollars, depending on the fuel systems to be investigated and on how much work from PWR improvement programs can be applied to the SSCR.

6.5.5

TRITIUM RELEASES

Essential to any D_2O containing system is the minimization of leakage from plant components so as to control the radiological hazard of tritium; for the SSCR, this is perhaps the major licensability question. It is felt therefore, that an in-depth evaluation of pathways for tritium release to the environment and of methods for reducing these releases would be an essential R&D program for development of the spectral shift control concept. The cost of this evaluation is estimated to be \$50,000 and would take approximately six months.

Low leakage valve and seal designs have already been developed and employed on heavy water reactors. However, some further development will probably be required because the operating pressure of the SSCR is higher than that of current heavy water reactors. R&D programs here would be devoted to developing valve and seal designs to minimize D_2O leakage, with subsequent proof testing of these devices. A one year development program will be generally required, although the development of seals for the reactor coolant pumps may take somewhat longer, perhaps two years. The total estimated cost of the valve and seal development program is two million dollars.

6.5.6

TRITIUM SEPARATION

If it becomes necessary to reduce the release of tritium to the environment, it is conceivable that the technology for separating tritium from D_2O

may need to be developed further. Distillation and electrolysis appear to be the most likely candidates from implementation of a commercial tritium separation unit. Units are available based on cryogenic distillation for use with heavy water reactors, where the problem of tritium production is somewhat more severe. The cost of an initial feasibility study encompassing process evaluation and selection, as well as initial design development may be desirable and is estimated to cost about \$200,000.

6.5.7 COST OF D_2O

The cost of the D_2O inventory (approx. 54 million dollars) represents about 2/3 of the capital outlay required to realize spectral shift control. The present method of producing heavy water by a dual temperature isotopic exchange (GS process) followed by vacuum distillation relies on a proven, but expensive, technology.

Because of the significant impact of the D_2O inventory cost on the overall economics of the SSCR, a study of the economics of D_2O production should be performed. In this study, a survey of the D_2O cost from world wide sources would be performed. In addition, the feasibility of lowering the production cost using present technology would be investigated. Furthermore, the prospects of using different separation techniques (e.g. lasers) would be explored. Work is being carried out at the present time on the feasibility of D_2O production by advanced isotopic separation techniques. The cost of these and possible future programs is large because of the technology involved.

The D₂O upgrader which is employed on the SSCR to reconcentrate heavy water is similar to the upgraders employed on the Canadian HWRs and in the last stage of the heavy water production facilities. However, the sizing of this upgrader is different because of the volume and range of D₂O concentrations which the upgrader must accept at various times in the SSCR operating cycle. The majority of the work required on the SSCR upgrader is that necessary to establish a preliminary design of sufficient detail for incorporation into a preliminary safety analysis report. It is estimated that the required engineering work to develop the upgrader design would take about 1 year and cost approximately \$500,000.

In order for a reactor system to be commercially available, a high degree of assurance of its licensability must be established. This assurance is obtained by the development of a preliminary system design and the performance of a preliminary safety analysis. The culmination of this effort is the preparation of a preliminary SAR which is submitted to the regulatory agency for review.

Since the SSCR plant is very similar to that of the PWR, preliminary design of components and systems would be limited to those areas where changes would be required to achieve comparable consequences of postulated accidents and anticipated operational occurrences with those of the conventional

PWR. The main areas which are felt to require attention are the implications of the reactivity coefficients on accidents which result in primary plant cooldown and the implications of the higher tritium production in the SSCR. The implications of spectral shift control on plant operations and load change performance should also be addressed as part of this preliminary design evaluation. The cost of this SAR related development is estimated to be about 10 million dollars and to take approximately two years to perform.

6.6

BACKFITTING POTENTIAL

6.6.1

GENERAL

The close similarity of the SSCR with a PWR invites considerations of the potential for backfitting a PWR for spectral shift control operation. In the following section two major backfit variations are discussed in some detail:

- a) backfitting an existing PWR and
- b) backfitting or modifying a PWR during the construction phase of the plant for spectral shift control operation.

In this context, it is mentioned again that the conceptual design of the SSCR of this present study was developed from a CESSAR/SWESSAR PWR on the basis that a switch-back from spectral shift control to conventional poison control operation is possible without hardware changes. However, the plant design of the present study was geared to and optimized for spectral shift operation. For instance, modifications to the interior of the Annulus Building were made and the Solid Waste and Decontamination Building was enlarged specifically to accommodate equipment for spectral shift operation. Some of this equipment as well as the D₂O upgrader would stand idle if the SSCR were operated as conventional PWR.

6.6.2 BACKFITTING EXISTING PWR FOR SPECTRAL SHIFT CONTROL OPERATION

6.6.2.1 Additional Equipment For Spectral Shift Operation

When backfitting an existing PWR for spectral shift control operation the following major plant equipment is needed: the D_2O upgrader facility, boron evaporators with a combined capacity of 135 gpm, additional tankage with a combined capacity of about 450,000 gallons, zero-leakage valves and pumps as well as connecting piping.

The D_2O upgrader is a self-contained facility which can be, although this is not necessary, housed in its separate building. Part of the upgrader facility are three 225 ft. tall distillation towers. The design of the facility requires that these towers be located in the immediate vicinity of the remaining equipment of the D_2O upgrader facility (e.g. condensers, reboilers, vacuum system). The height of the distillation towers puts some restrictions on the choice for their location at the plant site because of considerations of their possible interference with other structures in the case of tornado damage. The fabrication and construction time for the D_2O upgrader is estimated to be about 2.5 years.

The capacity of the boron evaporators for a conventional PWR is about 20 gpm. Spectral shift operation requires a significant extension of that capacity to 135 gpm. When backfitting an existing PWR, the new evaporators will have to be located in a separate building. Thus, probably the most

practical approach is to provide a new set of three 45 gpm boron evaporators together with the BDSU (boron/heavy water separation unit) rather than to incorporate the existing small evaporator. The existing evaporator could then be used as a fall-back reserve. The time for manufacturing and installation of the boron separators is estimated to be about two years.

The additional tankage is needed for storing liquid of various D_2O concentrations for subsequent reuse or processing. There are few stringent functional restrictions for selecting the size and arrangement of the various tanks. In general, more smaller tanks provide more operational flexibility than fewer larger ones. The only major restriction is that the Reactor Vessel Drain Tank, or an intermediate buffer tank with forwarding pumps, be placed on the lowest floor (building mat elevation) of the Annulus Building to receive gravity-drainage from the reactor coolant system during the refueling coolant exchange. In general, when selecting the size of the individual tanks, a balance between equipment cost and operational flexibility has to be worked out. The lead time for manufacturing and installing the tanks is estimated to be between 1 and 1.5 years.

The D_2O -containing fluid systems of a SSCR, in general, need to be more leak tight than the ones of a PWR to minimize tritium radiological hazards and economic losses associated with unrecoverable leakage. Systematic replacement of valves and pump seals of a conventional PWR with zero-leak designs is a very costly and time-consuming proposition. However, with an operating PWR this approach might be only partially necessary. First of all, it should

be possible to quantify on the basis of actual field data with reasonable accuracy the amount of recoverable and irrecoverable leakage. As pointed out before, the recoverable leakage can be thought of as part of the bleed flow for the normal D_2O dilution over a cycle and, unless it exceeds the minimum bleed flow rate at the beginning of a cycle (about 250 gal/day) or poses a radiological hazard, the recoverable leakage is not a major problem. In this context, it might be mentioned that many of the valves and pumps of a conventional PWR are equipped with leakage collection piping anyway. With respect to the tritium exposure, the tritium concentration in the reactor coolant builds up gradually and a converted PWR which is to be operated as a SSCR only during a part of its plant life will have a smaller tritium concentration at the end of plant life than a SSCR. Thus, the degree of system leak tightness required for spectral shift operation becomes a cost-balance decision with cost of the system modifications on one side and cost of D_2O losses and radiation protection measures on the other.

6.6.2.2 Arrangement of Additional Equipment

When converting an existing PWR for spectral shift operation, some of the additional equipment will be arranged differently than for the SSCR design as described in Section 5.1.2-DESCRIPTION OF PLANT LAYOUT. Rearrangement of equipment in the Annulus or Solid Waste and Decontamination building will be kept to a minimum to minimize perturbations of the PWR operation of the plant. The additional boron separators and storage tanks will probably

be placed in a separate, new building or into an extension of the Solid Waste and Decontamination Building. In general, a new site layout needs to be developed for placing the additional boron separators and storage tanks as well as the D₂O upgrader facility. Overall, it is expected that the cost for additional structures and foundations will be higher when converting a PWR than for an initial SSCR.

6.6.2.3 Construction Approach

The D₂O Upgrader Facility as well as the additional building(s) and systems for spectral shift operation can be installed during the normal operation of the PWR. During the PWR operation and prior to establishing the final piping cross ties with the new SSCR equipment, a large part of the checkout and startup operations can be performed. However, measures to control the construction activities will have to be taken to assure the continued safe operation of the PWR during the construction period. This will add to the overall construction costs. Installation of the piping cross-ties to the PWR fluid systems can be begun during refueling outages. For installation of the final connections and for startup testing, the duration of a refueling outage is probably too short and some additional plant shutdown is required.

Based on the longest equipment lead time (2.5 years) it is expected that from technical considerations a time span in the order of 3 to 4 years will be required for the conversion of a PWR to a SSCR.

Parallel to the construction activities, the licensing effort will be pursued and could become the controlling factor for the conversion schedule. The existence of a Design Approval for SSCR would be desirable before the conversion is started. Otherwise, Safety Analysis Report and Environmental Report revisions or supplements for spectral shift will have to be provided during the construction period.

6.6.2.4 Summary

The technical potential for backfitting an existing PWR for spectral shift operations is very good. However, backfitting results in equipment arrangements which are less optimum than for an original SSCR. Thus, the cost for structures and civil engineering work will be greater. Also, the D_2O inventory will increase because of the less compact equipment arrangement. Installation of the final piping cross-ties will require some plant shutdown. The total plant downtime and the overall cost of backfitting will depend strongly on the extent of equipment replacement (e.g., valve and pump seal replacement). A determination of the extent of replacement requires a cost effectiveness study of the actual plant to be backfit. The cost of replacement must be weighed against the costs of added D_2O leakage and higher releases of tritium. Therefore, the overall cost increase for spectral shift operation over the cost for a conventional PWR plant will be higher when backfitting a PWR than for an originally designed SSCR. Thus, backfitting PWRs for spectral shift operation does not appear to be an economically attractive generic concept. Backfitting, however, might be practical as well as economical to get a SSCR-Prototype reactor on line within a relative short leadtime.

6.6.3 BACKFITTING PWR FOR SPECTRAL SHIFT CONTROL OPERATION DURING CONSTRUCTION OF PLANT

6.6.3.1 General

The following sections address the subject of backfitting or more correctly changing, a poison control PWR design for spectral shift operation while the PWR is in the design or construction phase. It is obvious, that any modifications of the PWR design will result in some additional engineering and construction costs as well as in delays of achieving commercial operation. Thus, when considering engineering/design, procurement, construction, and licensing for a plant, it is most desirable to establish whether the unit is to be a PWR or SSCR during its lifetime before engineering authorization.

6.6.3.2 PWR Construction Schedule

In order to establish a frame of reference, the construction schedule of a PWR is described briefly. A major milestone date in the engineering and procurement efforts for a nuclear plant is the Authorization ("A") date. "A" - date is defined here as the date on which an utility-applicant authorizes the architect-engineer to proceed with engineering and design efforts utilizing a specific nuclear steam supply system. Assuming that the plant site and the plant design were pre-approved, and that no unforeseen delays due to licensing would occur, the schedule for the Commercial Operation ("CO") date is 96 months (8 years) after "A" date.

The critical path during the early construction phase is the placement of the building material for the containment/annulus building which is scheduled to start 36 months after "A". In preparation for this date, the detailed reinforcing steel (rebar) drawings must start 7 months after "A" in order that the fabrication of the mat (foundation) rebar for the containment/annulus building can begin, the start of which is scheduled nineteen (19) months after "A". During the first 7 months after "A", general arrangements drawings and system layouts are completed such that design details suitable for equipment procurement can be started. It should also be noted, that, although rebar is identified as the critical path, other engineering and design activities are proceeding during this time in order to levelize manpower and procure long lead-time equipment.

6.6.3.3 Operation Of SSCR As Conventional PWR

If spectral shift operation is seriously considered but a final decision has not been made by the time design for the plant starts, then one possible approach is to design a SSCR and operate it later as SSCR or as conventional PWR. The plant would be licensed basically as a SSCR with the option of operating it in the conventional poison control operation mode.

If the SSCR were operated exclusively in the poison control mode, some of the SSCR equipment would be superfluous and idle (e.g., tanks, D₂O upgrader) while some SSCR equipment might be used for more convenient (more leaktight systems) or more flexible (bigger boron separators) poison control operation.

Nonetheless, a SSCR operated as PWR has higher capital cost than an original PWR. However, the cost penalty for switching to poison control can be reduced if a decision is made early during the construction time of the plant. Obviously, a huge savings is made (about 54 million dollars) if the D_2O inventory is not acquired. A further sizable reduction (about 10 million dollars) is achieved if the D_2O Upgrader Facility is not built. This decision would have to be made about 3 to 4 years prior to Commercial Operation ("CO"), or by just past the half-way mark of the construction period. Not procuring the high capacity boron evaporators would also result in a reduction (about 1 million dollars). This decision would have to be made probably 3 years prior to "CO" at the latest. Finally, several hundred thousand dollars can be saved by not fabricating and installing storage tanks unnecessary for poison control operation. This decision would have to be made about two years prior to "CO".

6.6.3.4 Modification Of PWR For Spectral Shift Operation

When contemplating the modification of a PWR for spectral shift operation during the construction period of the PWR, two variants can be distinguished a) retaining the PWR construction schedule ("CO" date remains firm) and b) accepting slippage of the PWR "CO" date. In the following sections, the two alternatives are discussed in some detail.

6.6.3.4.1 "CO"-Date Firm

For the SSCR design of the present study, changes relative to the SWESSAR PWR in the cubicle arrangement of the Annulus Building become necessary for the installation of SSCR equipment. Because the construction of the Annulus Building is one of the first construction activities, the decision to change from a PWR to a SSCR has to be made early in the construction time of the plant. If the "CO"-date is to remain unchanged, the decision can not be delayed beyond 19 months after "A" (authorization)-date, because at this time the fabrication of the containment/annulus building mat reinforcing steel (rebar) begins. If design flexibility is to be retained up to this date, a parallel engineering/design and licensing effort for a PWR and a SSCR building design must be carried on. The cost penalty associated with this parallel effort is estimated to amount to about 18 million dollars. If the PWR/SSCR decision can be made prior to month 7 after "A"-date (this is when rebar drawings are started), then the parallel effort up to this point amounts to only about 2.8 million dollars.

The existence of a pre-approved licensing document for the SSCR would be a pre-requisite for the above-described approach. The SSCR could then be incorporated into a base PWR docket by reference. However, additional costs will occur because the site release portions of the PWR Safety Analysis Report and Environmental Report would have to be updated for SSCR application.

If the safety and licensing review of the PWR is completed before the SSCR review, an additional licensing fee of a maximum of \$850,000 might be required.

This sum is the approximate maximum cost for a single unit application for a Construction Permit and could be incurred if two separate construction permits for PWR and SSCR were obtained.

In this context it might be mentioned that the SSCR design of the present study does not lend itself easily for later incorporation into a plant design which was originally started as a SWESSAR PWR. In the interest of developing a cost-effective SSCR design, equipment rearrangements and, subsequently, annulus building changes to the SWESSAR design were made to arrive at a compact design of D_2O -carrying systems and to place D_2O -carrying tanks into the seismically-protected annulus building. If retaining the flexibility to delay a PWR/SSCR decision becomes an overriding factor in the development of the SSCR, then it would be worthwhile to investigate alternate layouts and develop cost comparisons with the present SSCR layout.

6.6.3.4.2 "CO"-Date Slippage

If slippage of the "CO"-Date is acceptable then the PWR/SSCR decision can be delayed until the very end of the PWR construction period. However, the cost penalty becomes larger, the longer this decision is delayed. Even for this variant, the start of the building mat rebar fabrication at 19 months after "A"-date is a significant decision point. Prior to this date only engineering/design work has to be repeated if a switch to SSCR is decided upon. Thereafter, building structure hardware alterations or new fabrication are required also. However, it is more likely, that the SWESSAR PWR containment/annulus building design would be retained, if the 19 months decision point had passed. A SSCR plant layout would be adopted which is different from

the one of the present study and would be similar to the one for converting an operating PWR to spectral shift operation, as described in Section 6.6.2. Based on the longest equipment lead times, the slippage in "CO" could amount to four years if the switch-over decision is delayed to the end of the PWR construction time.

6.6.3.5 Summary

Converting a PWR to a SSCR design during the construction of the PWR is technically feasible but not advisable economically. One approach would be to build a SSCR which leaves the option open to operate the plant as PWR or SSCR.

If the plant design is started as PWR, major cost penalties occur if the PWR/SSCR decision is delayed beyond 19 months of the authorization date because fabrication of steel reinforcements for the containment/annulus building starts at this time. If the PWR commercial operation date is to be retained, then a parallel engineering effort for a PWR and a SSCR design needs to be carried out which costs about 18 million dollars. Making the PWR/SSCR decision by month 7 after the authorization date reduces the parallel effort to about 2.8 million dollars. If the flexibility to defer a decision on PWR or SSCR operating modes until the construction of the plant is well under way is a significant consideration, investigation of alternate, more easily adjustable plant layouts would be advisable.

SECTION 7 - APPENDICES

7.1

CALCULATION OF TRITIUM CONCENTRATION IN CONTAINMENT BUILDING

In the following section the calculation of the tritium concentration in the containment building is presented. The results of this calculation were discussed in Section 5.4.4.3 Tritium Exposure In the Containment Building During Power Operation.

The amount of D_2O in the containment atmosphere at any time, t , may be expressed by the following mass balance:

$$\text{Change} = \text{Input} - \text{Output}$$

$$\frac{dY}{dt} = C X(t) - \frac{C}{W} Y \quad (1)$$

where: $Y = D_2O$ in the containment atmosphere; lb
 $C =$ Total leak rate to the atmosphere; lb/day
 $X(t) = D_2O$ concentration of the total leakage; wt fraction
 $W =$ Total water content in the containment atmosphere; lb

Holding both C and W constant, equation (1) reduces to a linear first order differential equation. Assuming all D_2O comes from the RCS:

$$X(t) = \frac{C'}{C} Z(t) \quad (2)$$

where: C' = Leakage rate from the RCS; lb/day

$Z(t)$ = D_2O concentration of the RCS; wt fraction

The RCS D_2O concentration decreases approximately linearly with full

power days of operation; therefore:

$$Z(t) = B(1-\alpha t) \quad (3)$$

where: B = initial RCS D_2O concentration; wt fraction

α = linear concentration coefficient with time; day⁻¹

Substituting equation (3) into equation (2), and setting B' equal to

$\left(\frac{C'}{C}\right) B$:

$$X(t) = B'(1-\alpha t) \quad (4)$$

Substituting equation (4) into equation (1) and rearranging terms:

$$\frac{dY}{dt} + \left(\frac{C}{W}\right)Y = CB'(1-\alpha t)$$

which has the solution:

$$Y(t) = B'W[(1 + \frac{\alpha W}{C})(1 - e^{-\frac{C}{W}t}) - \alpha t] + Y_0 e^{-\frac{C}{W}t} \quad (5)$$

where: Y_0 = the initial mass of D_2O in the containment atmosphere; lb

Relating to tritium concentration:

$$T_c(t) = \left[\frac{T_{D_2O} (454 \text{ gm/lb})}{V_c \rho_{D_2O}} \right] Y(t)$$

where: $T_c(t)$ = tritium concentration in the containment; $\mu\text{Ci/cc}$

ρ_{D_2O} = density of D_2O ; 1.1 gm/cc

T_{D_2O} = tritium concentration of D_2O ; 1100 $\mu\text{Ci/cc}$

V_c = Volume of the containment; $9.69 \times 10^{10} \text{ cc}$ ($3.42 \times 10^6 \text{ ft}^3$)

Substituting into equation (5):

$$T_c(t) = (4.685 \times 10^{-6}) \{ B'W[(1 + \frac{\alpha W}{C})(1 - e^{-\frac{C}{W}t}) - \alpha t] + Y_0 e^{-\frac{C}{W}t} \} \quad (6)$$

where: $C = 24,000 \text{ lb/day}$

$$B' = .00162 \text{ (65 lb/day RCS leakage)}$$

$$B' = .000162 \text{ (6.5 lb/day RCS leakage)}$$

$$W = 2493 \text{ lb (85F dry bulb/67°F wet bulb)}$$

$$W = 3438 \text{ lb (105F dry bulb/78°F wet bulb)}$$

$$\alpha = .00322 \text{ day}^{-1}$$

The quantity $\left(\frac{C}{W}\right)$ is large; therefore, the term $e^{-\frac{C}{W}}$ in equation (6)

rapidly approaches zero, and equation (6) can be approximated by:

$$T_C(t) \cong (4.685 \times 10^{-6}) \{B'W[(1 + \frac{\alpha W}{C})^{-\alpha t}]\}$$

In the following section the calculation of the amount of steam required for tritium dilution in building cubicles is presented. The results of this calculation were discussed in Section 5.4.4.4- Tritium Exposure In Annulus Building And Solid Waste And Decontamination Building.

The calculation of the required dilution steam uses a method similar to the one described in Section 7.1. The amount of heavy water per ft^3 in the cubicle atmosphere at any time, t , may be represented by the following mass balance:

$$\frac{dY}{dt} = CX(t) - \frac{C}{W} Y \quad (1)$$

Y = Pounds of D_2O per ft^3 of the cubicle atmosphere; lb/ft^3

C = Total rate of water vapor added per ft^3 of the cubicle atmosphere; $\text{lb}/\text{hr}\text{-ft}^3$

$X(t)$ = D_2O concentration of C above, wt fraction

W = Total water content per ft^3 of the containment atmosphere; lb/ft^3

Because the systems containing D_2O are low temperature (i.e., normally less than 200°F), the evaporation rate of this leakage is assumed to be negligible compared to the dilution steam flow rate. Therefore, C is set equal to the dilution steam flow rate only. Because the dilution steam contains no D_2O , $X(t) = 0$, yielding from equation (1):

$$\frac{dY}{dt} = - \frac{C}{W} Y$$

which has the solution:

$$Y(t) = Y_0 e^{-\frac{C}{W} t}$$

where:

Y_0 = the initial mass of D_2O per ft^3 of the cubicle atmosphere; lb/ft^3

Relating to tritium concentration:

$$Ta(t) = \left[\frac{T_{D_2O} (3.532 \times 10^{-5} ft^3/cc)}{(2.205 \times 10^{-3} lb/gm) \rho_{D_2O}} \right] Y(t)$$

with:

$Ta(t)$ = Tritium concentration in the cubicle atmosphere; $\mu Ci/cc$

ρ_{D_2O} = Density of D_2O ; $1.1 gm/cc$

T_{D_2O} = Tritium concentration in D_2O ; $1100 \mu Ci/cc$

or

$$T_a(t) = 16 \left[\frac{\mu C_i \times \text{ft}^3}{\text{cc} \times \text{lb } D_2O} \right] Y_0 e^{-\frac{C}{W} t} \quad (2)$$

Assuming the maximum design conditions in the cubicle (105°F dry bulb 78°F wet bulb), $W = .001 \frac{\text{lb moisture}}{\text{ft}^3 \text{ air}}$, and assuming that the total moisture in the atmosphere was derived from system leakage at 60 wt percent D_2O :

$$Y_0 = \left[\frac{0.6 \text{ lb } D_2O}{\text{lb moisture}} \right] \left[0.001 \frac{\text{lb moisture}}{\text{ft}^3 \text{ air}} \right]$$

$$Y_0 = 6 \times 10^{-4} \frac{\text{lb } D_2O}{\text{ft}^3 \text{ air}}$$

Substituting into Equation (2):

$$T_a(t) = (9.65 \times 10^{-3} \frac{\mu C_i}{\text{cc}}) e^{-\frac{C}{.001} t}$$

To determine the required dilution steam flow rate, it is assumed that 8 hrs. is a reasonable time period to reduce the tritium concentration in the cubicle to the Maximum Permissible Concentration (5×10^{-6} $\mu\text{Ci/cc}$ air).

$$\text{Therefore, } C = -0.001 \ln \left[\frac{5 \times 10^{-6}}{9.65 \times 10^{-3}} \right]$$

$$C = 7.56 \times 10^{-3} \frac{\text{lb steam}}{\text{ft}^3 \text{ of cubicle atmosphere}}$$

Assuming a typical cubicle of 4000 ft^3 free volume, the total dilution steam required is:

$$C_{\text{total}} = (7.56 \times 10^{-3}) (4000)$$

$$C_{\text{total}} = 30.2 \frac{\text{lb}}{\text{hr}} (\text{steam})$$

In the following section the calculation of the amount of water evaporation from the refueling cavity is presented. The results of this calculation were discussed in Section 5.4.4.5- Tritium Exposure In Containment Building During Refueling.

To calculate the tritium concentration in the containment during refueling, the following conservative equation for evaporation rate from the refueling cavity was used (reference: "Heat Losses from Tanks, Vats, and Vessels", by S. J. Friedman), together with the equilibrium tritium concentration of 1100 $\mu\text{Ci}/\text{mL D}_2\text{O}$.

$$E = 0.067 (P_v - P_a)^{1.22}$$

where: E = evaporation rate per ft^2 of surface area; lb/hr-ft^2

P_v = vapor pressure of the evaporating liquid at the temperature of the liquid; in. Hg

P_a = vapor pressure in the atmosphere; in. Hg.

Relating to relative humidity:

$$E = 0.067 (P_v - R P_a)^{1.22} \quad (1)$$

where: R = relative humidity; fraction

$P_{a'}$ = vapor pressure of atmosphere at 100% relative humidity

The total evaporation rate from the refueling cavity is equation (1) times the exposed surface of the refueling cavity, or:

$$E' = E \times A$$

$$E' = 128.6 (P_v - R P_{a'})^{1.22} \quad (2)$$

where: E' = Total evaporation rate; lb/hr

A = exposed surface area of the refueling cavity; 1919 ft² (same as a conventional PWR; see SWESSAR/CESSAR)

At equilibrium in the containment atmosphere, the following applies:

$$\begin{array}{ccc} \text{rate of tritium addition} & = & \text{rate of tritium removed} \\ \text{by evaporation} & & \text{by containment purge} \end{array}$$

or

$$E' \times C_R \times \frac{454g}{lb} \times \bar{V}_{D_2O} \times \text{Act.} = ER \times \frac{60 \text{ min}}{hr} \times \frac{28,320 \text{ cc}}{ft^3} \times T_a$$

where: C_R = wt. fraction of D_2O in refueling cavity

$\bar{V}_{D_2O}$ = specific volume of D_2O ; .91 cc/gm

Act. = tritium activity of D_2O ; 1100 $\mu\text{Ci/cc}$ (40th year tritium activity in the RCS)

ER = containment purge exhaust rate; ft^3/min

Ta = tritium concentration in the containment atmosphere; $\mu\text{Ci/cc}$

Solving for C_R :

$$C_R = \frac{3.74 (ER) (Ta)}{E'} \quad (3)$$

Substituting equation (2) into equation (3)

$$C_R = \frac{3.74 (ER) Ta}{128.6 (P_v - RPa')^{1.22}}$$

or

$$C_R = \frac{0.029 (ER) Ta}{(P_v - RPa')^{1.22}} \quad (4)$$

By substituting the Maximum Permissible Concentration (MPC) of tritium in air ($5 \times 10^{-6} \mu\text{Ci/cc}$) for Ta in equation (4), the maximum D_2O concentration in the refueling cavity which causes the containment atmosphere to reach MPC can be calculated:

$$C_{R_{\max}} = \frac{1.45 \times 10^{-7} (ER)}{(P_v - RPa')^{1.22}} \quad (5)$$

7.4

SPECIFICATIONS FOR D₂O UPGRADER PLANT

7.4.1 FUNCTION, PROCESSING SCHEME, AND OPERATING CONDITIONS

The D₂O Upgrading Plant is designed to process a heavy water feed solution of varying concentration and deliver products with the qualities described in the following section.

The processing scheme and operating conditions for the Plant are generally as shown on Process Flow Diagrams Figure 5.1.8-1 and 5.1.8-2 of Section 5.1.8-D₂O UPGRADER DESIGN.

7.4.2 CHARGE STOCKS AND THEIR SPECIFICATIONS

Heavy Water Feed

D ₂ O	2.0 wt. percent, 60.0 wt. percent,
pH	7.5, min. 8.0, max.
NH ₃	10 wt. ppm, max.
Non-volatile impurities (as Fe ₂ O ₃)	1 wt. ppm, max.
KMnO ₄ demand	100 wt. ppm, max.
Radioactivity	0.5 curies/lb D ₂ O, max.

7.4.3 CHARGE RATE

The D₂O Upgrading Plant is designed to process the following non-simultaneous quantities of design charge stock while yielding products as specified in Sections 7.4.4 and 7.4.5.

<u>Heavy Water Feed Concentration, wt. percent D₂O</u>	<u>Gallons per day</u>
2.0	2,361
5.0	1,725
10.0	1,416
20.0	1,299
40.0	1,454
60.0	2,066

7.4.4 PRODUCTS AND THEIR SPECIFICATIONS

The D₂O Upgrading Plant is designed to deliver Concentrated and Dilute Heavy Water Products having the following characteristics:

Concentrated Heavy Water Product

D ₂ O	85.0 wt. percent min.
Chloride (Cl)	0-0.15 ppm
Silica (SiO ₂)	0-0.02 ppm

Concentrated Heavy Water Product (Continued)

Conductivity	0-2 micromhos/cm
pH	7.0-8.0
Fluoride (F)	0-0.1 ppm
Suspended Solids	0-0.5 ppm

Dilute Heavy Water Product

D ₂ O	0.5 wt. percent, max.
Chloride (Cl)	0-0.15 ppm
Silica (SiO ₂)	0-.02 ppm
Conductivity	0-2 micromhos/cm
pH	7.0-8.0
Fluoride (F)	0-0.1
Suspended Solids	0-0.5 ppm

7.4.5 YIELDS OF PRODUCTS

The D₂O Upgrading Plant is designed to deliver the following quantities of products when charged with non-simultaneous feeds of the compositions and quantities as indicated in the following table:

Heavy Water Feed Concentration, wt. percent D ₂ O	Heavy Water Feed Rate, Gallons per day	Concentrated Heavy Water Product, Gallons per day	Dilute Heavy Water Product, Gallons per day
2.0	2,361.5	41.4	2,320.1
5.0	1,725.0	91.4	1,633.6
10.0	1,416.2	158.7	1,257.5
20.0	1,298.9	300.1	998.8
40.0	1,454.1	679.6	774.5
60.0	2,006.5	1,455.9	610.6

7.4.6 BATTERY LIMIT CONDITIONS

The D₂O Upgrading Plant is designed to receive the Charge Stock and to deliver Products at the Battery Limit at the temperatures, pressures and states as follows:

Heavy Water Feed

Temperature, °F	120 max. 65 min.
Pressure, psig	15 min.
State	Liquid

Concentrated Heavy Water Product

Temperature, °F	120 max. 65 min.
Pressure, psig	50 min.
State	Liquid

Dilute Heavy Water Product

Temperature, °F	120 max. 65 min.
Pressure, psig	50 min.
State	Liquid

7.4.7 UTILITIES AND THEIR SPECIFICATIONS

The utilities which are to be supplied to (or delivered from) the D₂O Upgrading Plant and their characteristics at the Battery Limit will be as follows:

Recirculating Cooling Water

Pressure, psig	
Supply	60, min.
Return	20, min.
Temperature, °F	
Supply	105, max.
Return	115, max.
Specification	Treated water suitable for use in shell and tube process heat exchange equipment

Once-Through Cooling Water (Cooling Tower Blowdown)*

Pressure, psig	60, min.
Temperature, °F	105, max.
Specification	Treated water suitable for use in direct contact heat exchange (spray type) equipment

*Once-through cooling water is used in the barometric condensers of the vacuum chilling system.

Steam

Pressure, psig 150, min.
Temperature, °C Sat'd.

Condensate

Pressure, psig 50, min.
Temperature, °F 190, min.
212, max.

Electricity

	<u>Motors Less than 250 HP</u>	<u>Small Motors and Instruments</u>
Voltage	480	120
Phase	3	1
Hertz	60	60

125 and 250 volt DC is also available

Plant Air

Temperature, °F Ambient
Pressure, psig 80, min.
100, max.
Specification No particles greater than
3 microns.

Instrument Air

Temperature, °F Ambient
Pressure, psig 80, min.
100, max.
Specification Oil-free. No particles
greater than 3 microns.
Atmospheric dew point to be
less than -40°F

Nitrogen

Temperature, °F	Ambient
Pressure, psig	50, min.
Specification	
Nitrogen	99.9 vol. percent min.
Hydrocarbons	1 vol. ppm, max.

7.4.8 ESTIMATED UTILITIES CONSUMPTIONS

At design operating conditions, the estimated consumption of utilities for the D₂O Upgrading Plant will be as follows:

Recirculating Cooling Water, U.S. GPM ⁽¹⁾	15,400
Once-Through Cooling Water, U.S. GPM	200
Steam, lbs/hr ⁽²⁾	74,000
Condensate, lbs/hr	(73,000)
Electricity, KW ⁽³⁾	100
Plant Air, SCFM	40
Instrument Air, SCFM	30
Nitrogen, SCFM	10

() signifies export

(1) At 10°F temperature rise

(2) Excluding building heating and steam tracing.

(3) Excluding lighting and instrumentation.

The following equipment is provided in the D₂O Upgrader Plant. The process and physical interrelationships of the equipment are indicated in Process Flow Diagrams, Figures 5.1.8-1 and 5.1.8-2, Plot Plan and General Arrangement, Figures 5.1.2-2 and 5.1.2-3, and Steam Balance Diagram, Figure 7.6-1.

<u>Equipment No.</u>	<u>Description</u>
AD-101X	Hotwell
BH-101	Steam Desuperheater
CB-101X	First Stage Barometric Condenser
CB-102X	Second Stage Barometric Condenser
DA-101	Distillation Column No. I
DA-102	Distillation Column No. II
DA-103	Distillation Column No. III
EA-101	Column No. I Reboiler
EA-102	Column No. I Condenser
EA-103	Column No. II Reboiler
EA-104	Column No. II Condenser
EA-105	Column No. III Reboiler
EA-106	Column No. III Condenser
EA-107	Overhead Product Vaporizer
EA-108	Overhead Product Condenser/Cooler
EA-109	Bottoms Product Vaporizer
EA-110	Bottoms Product Condenser/Cooler
EA-111X	Primary Recovery Chiller
EA-112X	Secondary Recovery Chiller
EA-113X	Interstage Chiller
EA-114	Vent Condenser
EE-101X	First Stage Ejector
EE-102X	Second Stage Ejector
EE-103X	Third Stage Ejector
FA-101	Column No. I Reflux Drum
FA-105X	Chilled Water Tank
FA-111X	Compressor Suction Drum
FA-112X	Interstage Knockout Drum
FA-113X	Vent Gas Knockout Drum
FA-102	Condensate Tank

Equipment List - Continued

<u>Equipment No.</u>	<u>Description</u>
FB-101	Upgrader Feed Tank
FB-102	Overhead Product Storage Tank
FB-103	Bottoms Product Storage Tank
FB-104	Process Drain Tank
GA-101 & S	Column No. I Reflux Pump
GA-102 & S	Column No. I Bottoms Pump
GA-103 & S	Column No. II Bottoms Pump
GA-104 & S	Column No. III Bottoms Pump
GA-106 & S	Overhead Product Pump
GA-107 & S	Bottoms Product Pump
GA-108	Process Drain Tank Pump
GA-109	Process Drain Pit Sump Pump
GA-105 & S	Upgrader Feed Pump
GA-110 & S	Chilled Water Circulating Pump
GA-120 & S	Condensate Pump
GB-101 & S	Vacuum Pump Package ⁽¹⁾
GC-101	Water Chiller Package ⁽²⁾

(1) Component Items:

EA-111X, EA-112X, EA-113X,
FA-111X, FA-112X, FA-113X,
GB-101 & S.

(2) Component Items:

AD-101X, CB-101X, CB-102X,
EE-101X, EE-102X, EE-103X,
FA-105X.

The Upgrader requires steam for the distillation towers, product vaporizers, and chilled water unit. The various design requirements are indicated schematically in Figure 7.6-1. Approximately 74,200 lbs/hr of 150 psig saturated steam is imported from the battery limit. The water chilling unit (GC-101) ejectors require about 900 lbs/hr of 150 psig steam. The remaining 73,300 lbs/hr are let down to 45 psig and desuperheated in Desuperheater BH-101 using 1300 lbs/hr of condensate. The resulting 74,600 lbs/hr of saturated 45 psig steam are then used to supply reboilers EA-101, EA-103, and EA-105, and product vaporizers EA-107 and EA-109.

Condensate from the various exchangers is flashed to condensate tank FA-120 which operates at essentially atmospheric pressure. Water cooled vent condenser EA-114 recovers condensate from the flash vapors. The condensate is returned by gravity to FA-120. Condensate pumps GA-120&S deliver condensate to the desuperheater and to the water chiller package to make up for flash and blowdown losses. The balance of approximately 73,000 lbs/hr is returned to the battery limit.

Figure 7.6-1
D₂O UPGRADER PLANT
STEAM BALANCE DIAGRAM

Recirculating cooling water is supplied to the column condensers EA-102, EA-104, and EA-106, to the product condenser/coolers EA-108 and EA-110, to Vent Condenser EA-114, and to Vacuum Pump Package, GB-101&S. Due to the high design supply temperature of 105°F, a 10°F rise has been used to limit the design return temperature to 115°F.

Once-through cooling water (cooling tower blowdown) is supplied to Barometric Condensers CB-101X and CB-102X (part of Water Chiller Package GC-101). This cooling water, plus a small amount of condensate, are sent to sewer.

The individual duties and flows are summarized in Table 7.6.2-1. The total cooling water requirements are approximately 15,400 gpm of recirculating cooling water and 200 gpm of once-through cooling water. The Upgrader liquid effluent will consist of the 200 gpm of once-through cooling water mixed with approximately 3 gpm of condensate (a total of 203 gpm), and will be at a temperature of approximately 115°F.

TABLE 7.6.2-1

COOLING WATER BALANCE

<u>RECIRCULATING COOLING WATER USERS</u>	DUTY <u>MM BTU/HR</u>	GPM @ <u>10°F RISE</u>
Column No. I Condenser, EA-102	21.6	4320
Column No. II Condenser, EA-104	23.3	4660
Column No. III Condenser, EA-106	23.5	4700
Overhead Product Condenser/Cooler, EA-108	0.86	172
Bottoms Product Condenser/Cooler, EA-110	0.57	114
Vent Condenser, EA-114	6.8	1360
Vacuum Pump Package, GB-101&S	----	<u>15</u>
		15,341
 <u>Once-Through Cooling Water Users</u>		
Water Chiller Package, GC-101	----	200

7.7

PHYSICS ANALYSIS METHODS

7.7.1

GENERAL

To predict the performance of the SSCR accurately, a realistic representation of spectral shift control had to be developed and incorporated into existing PWR methodology. A number of items, such as deuterium cross sections, D_2O concentration variation with burnup, etc., had to be included in the computational capabilities of Combustion Engineering's design methodology to provide a means of analyzing the SSCR. The principal analytical tools chosen for this study were the C-E lattice cell code-CEPAK and the few-group spatial diffusion code-PDQ. A description of CEPAK and the changes made to represent spectral shift control are discussed in Section 7.7.2, while the PDQ model developed for spatial calculations is described in Section 7.7.3. A more complete discussion of the methods developed to analyze the SSCR is given in the complementary Electric Power Research Institute (EPRI) report.⁽¹⁾

7.7.2

CEPAK LATTICE CELL CODE

7.7.2.1

Description of the C-E Design Code CEPAK

The CEPAK lattice cell code is a combination of several integrated program modules including THERMOS⁽²⁾, FORM⁽³⁾, and CINDER⁽⁴⁾. The CEPAK input consists of differential cross section libraries, reactor core and lattice geometry data, and parameters describing the operating conditions of the reactor. The energy spectrum is partitioned into 83 groups covering the

energy range from 0 to 10 MeV. The thermal flux calculation is performed in THERMOS with 29 energy groups and a one-dimensional model comprised of fuel, clad, and coolant. The epithermal calculation is carried out in FORM for a homogenized cell utilizing 54 energy groups. The B-1 approximation is used to account for leakage effects throughout the entire energy range. Fuel depletion calculations are performed in CINDER using the few group cross sections and fluxes obtained from the thermal and epithermal spectrum calculations. The output of CEPAC consists of material inventory data, fuel exposure information, and reactivity as a function of fuel depletion. In addition, burnup-dependent, few-group microscopic cross sections and diffusion constants are generated for use in core spatial calculations.

7.7.2.2 Modifications to CEPAC to Analyze the SSCR

The design code CEPAC was developed primarily for conventional poison controlled light water reactors. As a result, several modifications were necessary to extend its capabilities to include spectral shift controlled reactors. The major changes to the CEPAC lattice cell code are as follows:

Cross Sections and Depletion Chains - Libraries updated to include ENDF/B-IV cross sections, including deuterium, fission product yields and fission energies.

Operational Changes - Input modified to accept D_2O concentrations as a function of burnup and to alter moderator compositions accordingly.

Resonance Absorption Treatment - Dancoff factor modified to include deuterium

interactions and input self-shielding factors updated to reflect harder neutron spectrum of SSCR.

7.7.3 DEVELOPMENT OF SPATIAL DIFFUSION MODEL

Although CEPAC can be used to predict core related quantities such as fissile loadings, cycle lengths, and material inventories, spatially dependent parameters must be determined by multi-dimensional methods. Therefore, information about power distributions, core operating characteristics, and safety-related physics parameters was obtained using the few-group diffusion code PDQ⁽⁵⁾.

7.7.3.1 Sensitivity of Cross Sections

The spectral shift controlled reactor employs a wide range of heavy water concentrations during a reactor cycle to control the excess reactivity. Due to the poorer moderating properties of heavy water, the neutron energy spectrum becomes more thermal as the D₂O concentration is reduced in the coolant. For a conventional PWR, the cross sections vary as the fuel depletes due primarily to the changing isotopic concentrations and resulting neutron spectrum. Hence, few group cross sections can be expressed, to a large extent, as exclusive functions of burnup for PWR spatial calculations. For the SSCR, another variable has been introduced, D₂O concentration, which strongly affects the neutron spectrum. As a result, it is necessary to formulate cross sections as functions of both fuel exposure and deuterium concentration for the multi-dimensional SSCR calculations.

Cross sections were investigated to determine their dependence on these two variables. Some cross sections were found to be only a function of fuel exposure, while others were found to depend only on D_2O concentration. Those cross sections which displayed a strong dependence on both variables were represented by two-dimensional function tables.

7.7.3.2 Cross Section Representation

The method adopted to represent the dependence of the few group microscopic cross sections on both fuel exposure and D_2O concentration was the multi-dimensional interpolating tables in the Harmony cross section formalism⁽⁶⁾. Cross sections are represented in a matrix where the first dimension is the burnup and the second dimension is the D_2O concentration. The main diagonal of the table defines the most probable combination of these two variables whereas the off-diagonal elements define less likely occurrences.

7.7.3.3 PDQ Model for Core Calculations

To assess the feasibility and consequences of employing spectral shift control in current design PWR's, detailed reactor core calculations were necessary. A coarse mesh PDQ core model was constructed in which the fuel assembly was homogenized in a 4x4 node array. A coarse mesh model was utilized because of the large number of spatial depletion calculations required and the survey nature of these calculations. The 4x4 node array consisted of uneven spacings of 2, 6, 6, and 2 cell pitches. This arrangement has been shown to more accurately reproduce the inter-assembly flux and

power distributions than an evenly spaced coarse mesh representation. Each fuel assembly was explicitly modeled in the core along with the shroud and reflector. It has been found that this model adequately predicts core reactivity, nuclide concentrations, and power peaking as compared to more detailed representations, while significantly reducing the computer time required to perform the calculations.

7.7.4 PROCEDURE FOR ANALYZING SSCR FUEL CYCLES

7.7.4.1 Early Cycles

Two-dimensional core calculations were performed for the early cycles of each fuel cycle. These calculations were required, not only to obtain estimates of the fuel performance during the early cycles, but also to obtain representative operating history, in terms of D_2O rundown and inter-batch neutron leakages, for use in the analysis of later cycles with a less detailed point reactor model. Use of a point reactor model was necessary for all but the early cycles because of the large number of calculations required to obtain 30-year fuel inventories for each fuel cycle. A complete analysis using core spatial calculations would have been prohibitive.

The number of cycles analyzed with the two-dimensional core model depended on the fuel cycle being analyzed. In general, spatial calculations were performed until the D_2O rundown and inter-batch leakages became relatively constant as a function of cycle number. This insured that approximately correct neutron balances were maintained during the depletion calculations.

Depletion calculations to determine the fuel loading requirements of a reactor must be performed in a critical spectrum; failure to do so will result in either a surplus or a deficiency of neutrons during the depletion, and will lead to an erroneous conversion ratio and associated mass flow data. In the conventional PWR, excess neutrons are lost either through leakage or through capture in the soluble boron poison introduced for this purpose; in either case, the effect is much the same, since neutrons are lost from the system. Consequently, conventional PWR point reactor calculations can employ approximate representations of core leakage and utilize a search for critical boron concentration, or alternately use approximate boron concentrations and search for critical neutron leakage in order to achieve a critical spectrum during depletion. The situation is different, however, for the SSCR, since the leakage represents a loss of neutrons from the system while changes in D_2O concentration adjust the capture in fertile material and hence the amount of fissile material bred. Thus, an accurate knowledge of either the neutron leakage or of the evolution of D_2O concentration with depletion is necessary in order to perform accurate point reactor calculations for the SSCR.

7.7.4.2 Later Cycles

Once representative operating histories were obtained from the early cycle spatial calculations, point reactor calculations were performed to obtain mass flow data for 30 years of operation. The lattice cell code CEPAK was the primary analytical tool used to perform these fuel cycle analyses. In CEPAK, a single enrichment fuel batch is analyzed using the appropriate

cycle length and batch discharge exposure to attain the specified lifetime. To simulate spectral shift control, the heavy water concentration is varied from about 50 wt% to 60 wt% at the beginning of cycle to 2 wt% at the end of cycle (EOC) for each cycle that the fuel is resident in the core. The functional relationship between D_2O concentration and burnup, i.e., the D_2O rundown, is established through the early cycle spatial diffusion calculations. To maintain a critical spectrum in the CEPAC depletion calculations, the buckling search option is used which adjusts the leakage to obtain a unity eigenvalue.

To compare the CEPAC results for various fuel cycles, a definition of batch lifetimes must be derived which is consistent with the lifetime that is achieved in an operating core. The lifetime of a core composed of several batches is typically defined as an EOC eigenvalue of one which is equal to the weighted batch eigenvalues. For this study, the batch lifetime is defined as

$$\bar{k}_{\infty} = \frac{\sum_{j=1}^J f_j k_{\infty}(E_j)}{J}$$

where f_j is the fraction of fuel resident in the reactor for j cycles and $k_{\infty}(E_j)$ is the EOC infinite multiplication factor of a fuel batch with a burnup of E_j at end of cycle j . The target multiplication factor, $k_{\infty T}$, for the fuel batch is obtained by using the predicted fuel enrichment and D_2O rundown from the reference spatial calculations in a CEPAC depletion. This target value is then used for subsequent calculations by requiring that $\bar{k}_{\infty} = k_{\infty T}$.

REFERENCES FOR SECTION 7.7

"Physics Analysis Methods"

1. "Evaluation of Spectral Shift Controlled Reactors Operating on the Uranium Fuel Cycle," EPRI Research Project 978-1 Final Report, Draft Version December 1978.
2. H. C. Honeck, "THERMOS, A Thermalization Transport Theory Code for Reactor Lattice Calculations," BNL-5826, Brookhaven National Laboratory (1962).
3. D. J. McGoff, "FORM, A Fourier Transform Fast Spectrum Code for the IBM-709," NAA-SR-Memo-5766, September 1960.
4. T. R. England, "CINDER-A One Point Depletion and Fission Product Program," WAPD-TM-334 (revised), Bettis Atomic Power Laboratory, June 1964.
5. W. R. Cadwell, "PDQ-7 Reference Manual," WAPD-TM-678, Bettis Atomic Power Laboratory, January 1967.

Capital cost estimates for the SSCR were presented in Section 5.2.4-ECONOMIC EVALUATION. These cost estimates were divided into two categories: the capital costs of a typical large current design PWR and the additional costs required to realize spectral shift control. The PWR cost estimates are based on cost development by United Engineers and Constructors (UE&C) and are detailed in Ref. (1). The add-on costs for spectral shift control operation which were developed as part of this study are given in detail in this section. These cost estimates are organized to correspond to the uniform system of accounts as presented for the PWR in Ref. (1).

TABLE 7.8.1-1

1260 MW(e) SSCR-

COST INCREASE RELATIVE TO PWR

Account Number	Account Description	FACTORY		SITE			TOTAL COST		
		Quantity	Cost \$	Quantity	Labor Hours	Labor Cost \$	Material Cost \$	Total Site Cost \$	\$
2.	Total Direct Cost								
21.	Structures & Improve- ments								
215.	Primary Auxiliary Building & Terminals								
215.1	Annulus Building Structure			LS	52,132	644,600	1,357,000	2,001,600	2,001,600
216.	Waste Process Building								
216.1	Evaporator Area Structure			LS	142,636	1,637,000	1,188,000	2,825,000	2,825,000
216.2	Evaporator Area Services			LS	14,385	211,000	431,650	642,650	642,650
21.	Structure & Improve- ments				209,603	2,492,600	2,976,650	5,976,650	5,469,250

TABLE 7.8.1-1 (Continued)

1260 MW(e) SSCR-

COST INCREASE RELATIVE TO PWR

Account Number	Account Description	FACTORY		SITE			TOTAL COST	
		Quantity	Cost \$	Quantity	Labor Hours	Labor Cost \$	Material Cost \$	Total Site Cost \$
22.	Reactor Plant Equipment							
220A.221	Main Coolant Pumps							
220A.2211	Shaft Seal Assembly	4	16,500					16,500
224.12	Gaseous Vents & Drains							
224.121	Rotating Machinery	LS	38,000	LT	360	4,200		4,200
224.122	Tanks & Pressure Vessels	LS	18,800	LT	172	2,000		2,000
224.123	Miscellaneous	LS	36,000	LT	160	1,800		1,800
224.12	Gaseous Vents & Drains		92,800		692	8,000		8,000

TABLE 7.8.1-1 (Continued)

1260 MW(e) SSCR-

COST INCREASE RELATIVE TO PWR

Account Number	Account Description	FACTORY		SITE			TOTAL COST		
		Quantity	Cost \$	Quantity	Labor Hours	Labor Cost \$	Material Cost \$	Total Site Cost \$	\$
226.3	Primary Grade Water System								
226.31	Rotating Machinery	LS	44,500	LT	286	3,300		3,300	47,800
226.32	Liners of Cubicle Tanks	LS	372,600	LT	7,835	94,000	141,200	235,200	607,800
226.33	Valves, Piping, Fittings & Hangers	LS	92,000	LT	11,679	135,600		135,600	227,600
226.34	Insulation				355	5,000	7,300	12,300	12,300
226.3	Primary Grade Water System		509,100		20,155	237,900	148,500	386,400	895,500

TABLE 7.8.1-1 (Continued)

1260 MW(e) SSCR-

COST INCREASE RELATIVE TO PWR

Account Number	Account Description	FACTORY				SITE		TOTAL COST	
		Quantity	Cost \$	Quantity	Labor Hours	Labor Cost \$	Material Cost \$	Total Site Cost \$	\$
226.41	Chemical & Volume Control System								
226.411	Tanks & Pressure Vessels	LS	44,750	LT	1,325	16,000	23,750	39,750	84,500
226.412	Mixing Tank	LS	13,140	LT	2,491	28,910		28,910	42,050
226.413	Insulation				70	1,400	1,000	2,400	2,400
226.41	Chemical & Volume Control System		57,890		3,886	46,310	24,750	71,060	128,950

TABLE 7.8.1-1 (Continued)

1260 MW(e) SSCR-

COST INCREASE RELATIVE TO PWR

		FACTORY				SITE		TOTAL COST	
Account Number	Account Description	Quantity	Cost \$	Quantity	Labor Hours	Labor Cost \$	Material Cost \$	Total Site Cost \$	\$
226.42	Boron Recovery System								
226.421	Rotating Machinery	LS	40,800	LT	360	4,176		4,176	44,976
226.422	Tanks	LS	61,330	LT	798	9,600	11,264	20,864	82,194
226.423	Boron Evaporators	LS	682,460	LT	2,778	32,225		32,225	714,685
226.424	BDSU-Unit	LS	85,000	LT	732	8,500		8,500	93,500
226.425	Valves, Piping,								
	Fittings & Hangers	LS	114,960	LT	16,686	193,985		193,985	308,945
226.426	Insulation				540	10,800	8,000	18,800	18,800
226.42	Boron Recovery System		1,055,580		21,894	259,286	19,264	278,550	1,263,100

TABLE 7.8.1-1 (Continued)

1260 MW(e) SSCR-

COST INCREASE RELATIVE TO PWR

Account Number	Account Description	FACTORY		SITE			TOTAL COST	
		Quantity	Cost \$	Quantity	Labor Hours	Labor Cost \$	Material Cost \$	Total Site Cost \$
226.43	Containment Spray Syst.							
226.431	Liner for RWST	LS	98,450	LT	1,586	20,000	27,600	47,600
								146,050
226.43	Containment Spray Syst.		98,450		1,586	20,000	27,600	47,600
								146,050
22.	Reactor Plant Equipment, excluding D ₂ O Upgrader		1,830,320		48,213	571,496	220,114	791,610
								2,550,900
228.	D ₂ O Upgrader							
228.1	Process Equipment							
228.11	Distillation Towers	3	1,006,000					180,000
								1,186,000
228.111	Distillation Tower Internals		155,000					
								155,000

TABLE 7.8.1-1 (Continued)

1260 MW(e) SSCR-

COST INCREASE RELATIVE TO PWR

Account Number	Account Description	FACTORY		SITE			TOTAL COST	
		Quantity	Cost \$	Quantity	Labor Hours	Labor Cost \$	Material Cost \$	Total Site Cost \$
228.12	Superheater		2,000					2,000
228.13	Heat Exchangers		569,000					569,000
228.14	Process Tanks		20,000					20,000
228.15	Storage Tanks		52,000					52,000
228.16	Pumps		108,000					108,000
228.17	Vacuum Pumps		192,000					192,000
228.18	Water Chiller		83,000					83,000
228.19	Instrumentation	LS	410,000					410,000
228.110	Piping and Valves	LS	200,000					1,460,000
228.1	Process Equipment		2,797,000					1,640,000
								4,437,000

TABLE 7.8.1-1 (Continued)

1260 MW(e) SSCR-

COST INCREASE RELATIVE TO PWR

Account Number	Account Description	FACTORY		SITE			TOTAL COST		
		Quantity	Cost \$	Quantity	Labor Hours	Labor Cost \$	Material Cost \$	Total Site Cost \$	\$
228.2	Facility Construction								
228.21	Electrical							595,000	595,000
228.22	Function							305,000	305,000
228.23	Painting							90,000	90,000
228.2	Facility Construction							990,000	990,000
228.3	Upgrader Building								
228.31	Foundations							435,000	435,000
228.32	Structures							865,000	865,000
228.3	Upgrader Building							1,300,000	1,300,000

TABLE 7.8.1-1 (Continued)

1260 MW(e) SSCR-

COST INCREASE RELATIVE TO PWR

		FACTORY		SITE			TOTAL COST		
Account Number	Account Description	Quantity	Cost \$	Quantity	Labor Hours	Labor Cost \$	Material Cost \$	Total Site Cost \$	\$
228.1	D ₂ O Upgrader		2,797,000					3,930,000	6,727,000
23	Turbine Plant Equipment								
235.3	Closed Cooling Water Systems								
235.31	Rotating Machinery	LS	31,800	150	1,600			1,600	33,400
235.32	Cooling System for								
	D ₂ O Upgrader	LS	595,500	1,072	12,400			12,400	607,900
235.32	Valves, Piping,								
	Fittings & Hangers	LS	159,100	7,900	92,000			92,000	251,100
23.	Turbine Plant Equipment		786,400	9,122	106,000			106,000	892,400

TABLE 7.8.1-1 (Continued)

1260 MW(e) SSCR-

COST INCREASE RELATIVE TO PWR

Account Number	Account Description	FACTORY				SITE			TOTAL COST
		Quantity	Cost \$	Quantity	Labor Hours	Labor Cost \$	Material Cost \$	Total Site Cost \$	
24	Electric Plant								
	Equipment								
241.1	Generator Equipment								
	Switchgear	LS	20,450	14	1,700			1,700	1,700
241.22	Station Motor Control								
	Center	LS	7,300	80	1,000			1,000	8,300
245.2	Cable Tray	LS	17,400	3,600	43,100			43,100	60,500
246.2	Station Service PWR								
	Wiring	LS	33,850	2,011	24,200			24,200	58,050
246.2121	15 kV Cable	LS	40,500	288	3,500			3,500	44,000
246.2122	15 kV Cable	LS	3,500	272	3,300			3,300	6,800
246.3	Control Cable	LS	27,000	1,596	19,200			19,200	46,200
24.	Electric Plant Equipment		150,000	7,988	96,000			96,000	246,000

TABLE 7.8.1-1 (Continued)

1260 MW(e) SSCR-

COST INCREASE RELATIVE TO PWR

Account Number	Account Description	FACTORY				SITE		TOTAL COST	
		Quantity	Cost \$	Quantity	Labor Hours	Labor Cost \$	Material Cost \$	Total Site Cost \$	\$
25	Miscellaneous Plant								
	Equipment								
252.2	Water Systems								
252.21	Service Water Systems								
252.211	Rotating Machinery	LS	141,000	LT	285	3,300		3,300	144,300
252.212	Piping, Fittings &								
	Hangers	LS	20,000	LT	2,155	24,000		25,000	45,000
252.3	Auxiliary Steam System								
252.31	Auxiliary Boilers	LS	140,000	LT	2,600	30,300		30,300	170,300
252.32	Piping, Fittings,								
	Hangers & Insulation	LS	41,600	LT	4,109	47,900		47,900	89,500
252.36	Aux. Steam & Condensate								
	Return	LS	18,600	LT	560	6,500		6,500	25,100
25	Miscellaneous Plant								
	Equipment		361,200		9,709	113,000		113,000	474,200

TABLE 7.8.1-2.

1260 MW(e) SSCR-

COST INCREASE RELATIVE TO PWR

Account Number	Account Description	FACTORY		SITE			TOTAL COST	
		Quantity	Cost \$	Quantity	Labor Hours	Labor Cost \$	Material Cost \$	Total Site Cost \$
9	Total Indirect Cost							
91	Construction Services							
911	Construction Services							
	A/E							3,100,000
912	Construction Services							
	D ₂ O Upgrader							1,985,000
91	Construction Services							5,085,000
92	Home Office Engineering Service							1,564,000

REFERENCES FOR SECTION 7.8

"Plant Cost Estimates"

1. "Capital Cost: Pressurized Water Reactor Plant," NUREG-0241, June 1977.

In the following section, the 30-year charge and discharge data for the spectral shift controlled reactor is presented. In addition, the isotopic composition of the various fuel batches is given for the beginning and the end of the equilibrium cycle. A summary of the fuel management and resource requirements is also presented for each fuel cycle. Data for five fuel cycles is included in these tables:

- a. UO_2 once-through, three-batch (30,400 MWd/t) cycle;
- b. UO_2 once-through, five-batch (50,650 MWd/t) cycle;
- c. Denatured U-235/ ThO_2 cycle with storage or sale of plutonium;
- d. Denatured U-233/ ThO_2 cycle with storage or sale of plutonium; and
- e. Plutonium/ ThO_2 cycle with sale of U-233.

This data has been presented in a format which is consistent with that of the Nonproliferation Alternative Systems Assessment Program (NASAP) so that it can be used directly by the program's assessment contractors. All the reactor charge/discharge data and the fuel inventory data is given in kilograms for the conceptual reactor design, that is, for a 1260 MWe SSCR. The only data normalized to a GWe plant is that presented in the Fuel Management Information tables. Empty spaces on the reactor charge and discharge data sheets assume values last stated in years one through thirty.

TABLE 7.9-1

FUEL MANAGEMENT INFORMATION
UO₂ Once-Through Three-Batch SSCR

Average Capacity Factor -	75%		
Approximate Fraction of Core Replaced -	one-third annually		
Fissile Material Fabrication Loss Fraction -	1%		
UF ₆ Conversion Loss -	0.5%		
U ₃ O ₈ Requirements (short tons/GWe)			
initial core -	386		
annual equilibrium reload requirement -	175		
30-year cumulative requirements -	5461		
Separative Work Requirements (metric tons/GWe)			
initial core -	191		
Equilibrium reload -	100		
30-year cumulative requirement -	3091		
Heavy Metal in Spent Fuel (kg/GWe) ^a		<u>plutonium</u>	
	U-235	fissile	total
annual equilibrium discharge	186	202	284
30-year cumulative discharge	5526	5986	8394

^aDoes not include credit for two partially burnt fuel batches in final core. Credit for U-235, total plutonium and fissile plutonium in two partially burnt batches is 779, 350 and 279 kg/GWe, respectively.

TABLE 7.9-2

REACTOR CHARGE DATA FOR THE UO_2 ONCE THROUGH THREE-BATCH SSCR

Year	Th-232	Pa-233	U-232	U-233	U-234	U-235	U-236	(kg)							Np-237	Total Heavy Metal	Fission Product
								U-238	Pu-238	Pu-239	Pu-240	Pu-241	Pu-242				
1					2086.5			100700.0								102786.5	
2					921.2			33198.8								34120.0	
3																	
4																	
5																	
6																	
7																	
8																	
9																	
10																	
11																	
12																	
13																	
14																	
15																	
16																	
17																	
18																	
19																	
20																	
21																	
22																	
23																	
24																	
25																	
26																	
27																	
28																	
29																	
30																	
31																	

TABLE 7.9-3

REACTOR DISCHARGE DATA FOR THE UO_2 ONCE THROUGH THREE-BATCH SSCR

Year	Th-232	Pa-233	U-232	U-233	U-234	U-235	U-236	U-238	Pu-238	Pu-239	Pu-240	Pu-241	Pu-242	Np-237	Total Heavy Metal	Fission Product
						(kg)										
1						207.23	48.10	33177.68	.46	158.69	48.44	24.69	4.63	3.86	33674.25	438.14
2						183.91	77.84	32756.87	2.69	183.94	71.20	40.25	12.43	9.99	33340.49	768.55
3						241.17	120.32	32291.25	5.45	204.19	82.37	50.21	16.85	16.70	33029.80	1072.56
4						239.10	115.78	32337.94	5.45	203.93	80.39	48.48	15.41	16.70	33066.62	1035.39
5						234.37	116.56	32316.05	5.45	205.42	81.86	49.48	16.08	16.70	33045.32	1054.73
6																
7																
8																
9																
10																
11																
12																
13																
14																
15																
16																
17																
18																
19																
20																
21																
22																
23																
24																
25																
26																
27																
28																
29																
30																

TABLE 7.9-4

FUEL INVENTORY - BEGINNING OF EQUILIBRIUM CYCLE UO_2 ONCE THROUGH THREE-BATCH SSCR

<u>Isotope</u>	(kg)		
	<u>Fresh Fuel</u>	<u>One Cycle Exposure Fuel</u>	<u>Two Cycles Exposure Fuel</u>
Th-232			
Pa-233			
U-232			
U-233			
U-234			
U-235	921.2	595.9	385.3
U-236		60.5	96.7
U-238	33198.8	32924.4	32634.3
Pu-238		.308	1.72
Pu-239		133.4	180.4
Pu-240		24.9	55.3
Pu-241		11.2	31.8
Pu-242		1.10	6.72
Fission Products		359.8	711.8
Other Isotopics			
Am-241		.106	.534
Cm-242			
Np-237		3.13	8.46

TABLE 7.9-5

FUEL INVENTORY - END OF EQUILIBRIUM CYCLE UO_2 ONCE THROUGH THREE-BATCH SSCR

Isotope	(kg)		
	One Cycle Exposure Fuel	Two Cycle Exposure Fuel	Three Cycle Exposure Fuel
Th-232			
Pa-233			
U-232			
U-233			
U-234			
U-235	595.9	385.3	234.3
U-236	60.5	96.7	116.6
U-238	32924.4	32634.3	32316.0
Pu-238	.308	1.72	5.45
Pu-239	130.6	177.4	205.4
Pu-240	24.9	55.3	81.9
Pu-241	11.2	31.8	49.5
Pu-242	1.10	6.72	16.1
Fission Products	359.8	711.8	1054.63
Other Isotopics			
Am-241	.106	.534	.964
Cm-242	-	-	-
Np-237	3.13	8.46	16.70

TABLE 7.9-6

FUEL MANAGEMENT INFORMATION

UO₂ ONCE-THROUGH, FIVE-BATCH SSCR

Average Capacity Factor -	75%		
Approximate Fraction of Core Replaced -	one-fifth annually		
Fissile Material Fabrication Loss Fraction -	1%		
UF ₆ Conversion Loss -	0.5%		
U ₃ O ₈ Requirements (short tons/GWe)			
initial core -	390		
Annual equilibrium reload requirement -	151		
30-year cumulative requirements -	4769		
Separative Work Requirements (metric tons/GWe)			
initial core -	193		
equilibrium reload -	100		
30-year cumulative requirement -	3093		
Heavy Metal in Spent Fuel (kg/GWe) ^a			
		<u>plutonium</u>	
	U-235	fissile	total
annual equilibrium discharge -	98	137	209
30-year cumulative discharge -	2804	3994	6010

^a Does not include credit for four partially burnt fuel batches in final core. Credit for U-235, total plutonium and fissile plutonium in two partially burnt batches is 1143, 548 and 418 kg/GWe, respectively.

TABLE 7.9-7

REACTOR CHARGE DATA FOR THE UD_2 ONCE-THROUGH, FIVE-BATCH SSCR

Year	<u>Th-232</u>	<u>Pa-233</u>	<u>U-232</u>	<u>U-233</u>	<u>U-234</u>	<u>U-235</u>	<u>U-236</u>	(kg)	<u>Pu-238</u>	<u>Pu-239</u>	<u>Pu-240</u>	<u>Pu-241</u>	<u>Pu-242</u>	<u>Np-237</u>	Total	Fission
															Heavy	Product
															Metal	
1						2104.49			100681.52						102786.5	
2						777.94			19694.06						20472.0	
3																
4																
5																
6																
7																
8																
9																
10																
11																
12																
13																
14																
15																
16																
17																
18																
19																
20																
21																
22																
23																
24																
25																
26																
27																
28																
29																
30																

TABLE 7.9-8

REACTOR DISCHARGE DATA FOR THE UO_2 ONCE-THROUGH, FIVE-BATCH SSCR

Year	(kg)											Total Heavy Metal	Fission Product			
	Th-232	Pa-233	U-232	U-233	U-234	U-235	U-236	U-238	Pu-238	Pu-239	Pu-240			Pu-241	Pu-242	Np-237
1						130.88	29.40	19906.67	0.23	91.45	28.60	14.25	2.95	2.37	20206.8	258.65
2						100.42	40.78	19722.23	0.75	102.87	41.70	22.42	7.06	4.74	20043.0	407.95
3						68.44	47.80	19535.70	2.61	108.87	51.61	29.66	13.06	9.39	19867.1	662.15
4						53.31	57.06	19090.10	5.00	111.96	58.49	35.17	17.67	13.19	19442.0	864.31
5						101.68	102.16	18974.43	8.22	124.00	63.38	38.94	19.28	16.78	19448.9	1074.97
6						123.11	109.71	18877.75	8.22	131.64	63.42	40.82	18.80	16.78	19390.3	1060.01
7																
8																
9																
10																
11																
12																
13																
14																
15																
16																
17																
18																
19																
20																
21																
22																
23																
24																
25																
26																
27																
28																
29																
30																

TABLE 7.9-9

FUEL INVENTORY - BEGINNING OF EQUILIBRIUM CYCLE, UO_2 ONCE-THROUGH, FIVE-BATCH SSCR

Isotope	(kg)				
	Fresh Fuel	Once Cycle Exposure Fuel	Two Cycles Exposure Fuel	Three Cycles Exposure Fuel	Four Cycles Exposure Fuel
Th-232					
Pa-233					
U-232					
U-233					
U-234					
U-235	779.94	564.32	404.00	281.56	189.69
U-236		41.26	70.62	90.79	103.38
U-238	19694.06	19553.50	19401.56	19238.66	19063.87
Pu-238		.145	.877	2.47	4.98
Pu-239		76.95	112.45	127.77	133.28
Pu-240		11.04	26.82	41.74	54.11
Pu-241		4.76	16.00	26.82	35.18
Pu-242		.329	2.41	6.51	12.17
Fission Products		215.74	426.87	638.58	844.60
Other Isotopics					
Am-241		.044	.237	.556	.854
Cm-242		-	-	-	-
Np-237		1.80	5.17	9.15	13.16

TABLE 7.9-10

FUEL INVENTORY - END OF EQUILIBRIUM CYCLE, UO_2 ONCE-THROUGH, FIVE-BATCH SSCR

Isotope	Once Cycle Exposure Fuel	(kg) Two Cycles Exposure Fuel	Three Cycles Exposure Fuel	Four Cycles Exposure Fuel	Five Cycles Exposure Fuel
Th-232					
Pa-233					
U-232					
U-233					
U-234					
U-235	564.32	404.00	281.56	189.69	123.11
U-236	41.26	70.62	90.79	103.38	109.71
U-238	19553.50	19401.56	19238.66	19063.87	18877.75
Pu-238	.145	.877	2.47	4.98	8.22
Pu-239	75.50	110.88	126.05	131.43	131.64
Pu-240	11.04	26.82	41.74	54.11	63.42
Pu-241	4.76	16.00	26.82	35.18	40.82
Pu-242	.329	2.41	6.51	12.17	18.80
Fission Products	215.74	426.87	638.58	849.60	1060.01
Other Isotopics					
Am-241	0.044	.237	.556	.854	1.09
Cm-242	-	-	-	-	-
Cm-237	1.80	5.17	9.15	13.16	16.78

TABLE 7.9-11

FUEL MANAGEMENT INFORMATION

DENATURED U-235/ThO₂ SSCR WITH SALE OR STORAGE OF PLUTONIUM

Average Capacity Factor -	75%
Approximate Fraction of Core Replaced -	one-third annually
Lag Time Assumed Between Fuel Discharge and Recycle reload -	two years
Fissile Material Fabrication Loss Fraction -	1%
UF ₆ Conversion Loss -	0.5%
Fissile Material Reprocessing Loss Fraction -	0.5%
ThO ₂ and U ₃ O ₈ Core Requirements	U ₃ O ₈ ^a ThO ₂ ^b
initial core -	650 71
annual equilibrium reload requirement -	93 21
30-year cumulative requirement -	3703 699
Separative Work Requirements (metric tons/GWe)	
initial core	587
annual equilibrium reload	91
30-year cumulative requirement	3515
Annual Equilibrium Fissile Plutonium Discharge (kg/GWe) -	67
30-Year Cumulative Fissile Plutonium Discharge (kg/GWe) ^c -	1868

a Short tons/GWe

b Tonnes/GWe

c Does not include credit for two partially burnt fuel batches in final core. Additional credit is equal to 94 kg/GWe

TABLE 7.9-12

MAKEUP FUEL CHARGE DATA FOR THE DENATURED U-235/ThO₂ SSCR

Year	(kg)												Total Heavy Metal	Fission Product
	Th-232	Pa-233	U-232	U-233	U-234	U-235	U-236	U-238	Pu-238	Pu-239	Pu-240	Pu-241	Pu-242	Np-237
1	77526.4				25.8	3202.1		12808.4						93562.7
2	24673.6				10.1	1276.0		5093.9						31053.6
3	24673.6				10.1	1276.0		5093.9						31053.6
4	10053.3				4.15	519.9		2079.7						12657.1
5	6175.3				2.55	319.3		1277.5						7774.7
6	3683.2				1.52	190.5		761.9						4637.1
7	2622.5				1.08	135.6		542.5						3301.7
8	2622.5				1.08	135.6		542.5						3301.7
9	2015.6				.830	104.2		417.0						2537.6
10														
11														
12														
13														
14	1579.9				.650	81.7		326.8						1989.7
15														
16														
17														
18														
19	1575.7				.649	81.5		326.0						1983.8
20														
21														
22														
23														
24	1411.7				.581	73.0		292.0						1777.3
25														
26														
27														
28														
29														
30														

TABLE 7.9-13

RECYCLE FUEL CHARGE DATA FOR THE DENATURED U-235/ThO₂ SSCR

Year	(kg)												Total Heavy Metal	Fission Product
	Th-232	Pa-233	U-232	U-233	U-234	U-235	U-236	U-238	Pu-238	Pu-239	Pu-240	Pu-241	Pu-242	Np-237
1	0		0	0	0	0	0	0						0
2														
3														
4	14517.3		0.1	221.9	18.5	473.3	59.6	3110.0						18400.7
5	15384.8		0.1	325.1	36.1	547.4	116.2	4095.1						20391.1
6	20269.4		0.1	356.6	46.1	630.6	164.4	5125.4						26417.9
7	21694.2		0.83	352.1	43.3	696.3	152.3	4810.9						27749.4
8														
9	22054.2		1.28	441.9	95.0	638.6	229.0	5053.5						28512.9
10														
11														
12														
13														
14	22345.2		1.51	458.6	136.4	665.7	285.6	5169.5						29060.9
15														
16														
17														
18														
19	22224.3		1.66	472.6	167.2	669.1	335.6	5195.6						29066.2
20														
21														
22														
23														
24	21992.2		1.74	486.7	202.5	672.4	396.8	5313.6						29065.9
25														
26														
27														
28														
29														
30														

TABLE 7.9-14

MAKEUP FUEL DISCHARGE DATA FOR THE DENATURED U-235/ThO₂ SSCR

Year	(kg)														Total Heavy Metal	Fission Product
	Th-232	Pa-233	U-232	U-233	U-234	U-235	U-236	U-238	Pu-238	Pu-239	Pu-240	Pu-241	Pu-242	Np-237		
1	26708.6	38.4	.13	186.9	18.8	480.5	60.5	3157.4	.26	33.3	7.01	4.31	.54	2.56	30699.8	324.0
2	25090.2	34.4	.13	295.7	36.7	440.3	118.0	4157.5	.26	51.8	14.1	12.5	2.95	8.05	30262.6	712.9
3	23554.9	30.9	.13	331.1	46.8	462.9	166.9	5203.9	.26	68.3	20.3	19.6	5.77	14.5	29925.9	1069.3
4	24021.2	29.6	.90	327.9	40.2	418.5	154.6	4866.5	.27	58.0	17.7	17.3	5.25	14.5	29972.4	1069.3
5	24021.2	29.6	.90	327.9	40.2	418.5	154.6	4866.5	.27	58.0	17.7	17.3	5.25	14.5	29972.4	1069.3
6	9788.8	12.1	0.36	133.8	16.4	170.6	62.8	1983.0	.11	23.6	7.21	7.05	2.15	5.90	12213.9	435.9
7	6012.8	7.41	0.22	82.2	10.1	104.8	38.6	1281.1	.07	14.5	4.43	4.33	1.32	3.63	7502.5	267.7
8	3586.3	4.42	0.13	49.0	6.01	62.5	23.0	726.5	.04	8.67	2.64	2.58	.79	2.16	4474.7	159.7
9	2553.5	3.15	0.096	34.9	4.28	44.5	16.4	517.3	.03	6.17	1.80	1.84	.56	1.54	3186.1	113.7
10	2553.5	3.15	0.096	34.9	4.28	44.5	16.4	517.3	.03	6.17	1.88	1.84	.56	1.54	3186.1	113.7
11	1962.9	2.42	.074	26.8	3.29	34.2	12.6	397.7	.02	4.74	1.45	1.41	.43	1.18	2449.2	87.4
12																
13																
14																
15																
16	1538.7	1.90	.058	21.0	2.58	26.7	9.90	311.7	.02	3.72	1.13	1.11	.37	.93	1915.8	68.5
17																
18																
19																
20																
21	1534.6	1.89	.058	20.9	2.57	26.7	9.88	310.9	.02	3.71	1.13	1.10	.34	.93	1915.8	68.3
22																
23																
24																
25																
26	1374.8	1.70	.052	18.8	2.30	24.0	8.85	278.5	.01	3.32	1.01	.99	.30	.83	1715.5	61.2
27																
28																
29																
30																

TABLE 7.9-15

RECYCLE FUEL DISCHARGE DATA FOR THE DENATURED U-235/ThO₂ SSCR

Year	(kg)														Total Heavy Metal	Fission Product
	Th-232	Pa-233	U-232	U-233	U-234	U-235	U-236	U-238	Pu-238	Pu-239	Pu-240	Pu-241	Pu-242	Np-237		
1																
2																
3																
4																
5																
6	14003.8	17.6	.81	254.8	58.0	159.3	143.3	3044.2	4.89	35.7	11.2	10.5	3.16	15.6	17762.9	629.1
7	14763.4	18.6	.85	268.6	61.1	167.9	151.1	3209.3	5.16	37.6	11.8	11.1	3.33	16.5	18726.3	663.2
8	19450.7	24.5	1.12	353.9	80.5	221.2	199.1	4228.2	6.80	49.5	15.5	14.6	4.39	21.7	24671.7	873.8
9	21119.0	26.5	1.22	384.2	87.5	240.2	216.1	4590.9	7.38	53.9	16.9	15.8	4.77	23.6	26788.0	953.6
10	21119.0	26.5	1.22	384.2	87.5	240.2	216.1	4590.9	7.38	53.9	16.9	15.8	4.77	23.6	26788.0	953.6
11	21477.7	26.5	1.48	416.0	130.0	235.5	277.3	4825.4	9.23	57.3	17.6	16.4	4.73	29.4	27524.5	962.9
12																
13																
14																
15																
16	21764.2	26.6	1.63	430.3	162.0	258.3	330.7	4938.2	10.5	59.5	18.1	16.9	4.91	33.7	28055.5	980.5
17																
18																
19																
20																
21	21656.5	26.1	1.66	432.4	185.4	266.8	376.6	4964.1	11.5	60.6	18.1	16.8	4.53	37.3	28058.4	994.8
22																
23																
24																
25																
26	21584.0	25.9	1.73	442.5	214.8	281.3	435.2	5116.6	12.7	62.8	18.5	17.2	4.57	41.4	28259.2	1000.8
27																
28																
29																
30																

TABLE 7.9-16

REENRICHMENT COMPONENT OF RECYCLE FUEL CHARGE DENATURED U-235/ThO₂ SSCR*

Year	Th-232	Pa-233	U-232	U-233	U-234	U-235	U-236	U-238	Pu-238	Pu-239	Pu-240	Pu-241	Pu-242	Np-237	Total Heavy Metal	Fission Product
1																
2																
3																
4					0.0	0.0		0.0							0.0	
5					1.5	113.7		7.1							122.3	
6					2.2	174.7		10.9							187.7	
7					3.7	284.1		17.7							305.5	
8					3.7	284.1		17.7							305.5	
9					4.6	358.2		22.3							385.1	
10																
11																
12																
13																
14					5.2	400.0		24.9							430.1	
15																
16																
17																
18																
19					5.0	388.4		24.2							417.6	
20																
21																
22																
23																
24					4.9	383.3		23.9							412.1	
25																
26																
27																
28																
29																
30																

NOTE: *This material has already been included in the Recycle Fuel Charge Data, Table 7.9-13; it is given separately here to facilitate fuel cycle cost calculations.

TABLE 7.9-17

MAKEUP FUEL INVENTORY - BEGINNING OF EQUILIBRIUM CYCLE DENATURED U-235/ThO₂ SSCR

Isotope	Fresh Fuel	(kg) One Cycle Exposure Fuel	Two Cycles Exposure Fuel
Th-232	1411.7	1397.0	1388.0
Pa-233		.72	.75
U-232		.006	.026
U-233		9.00	15.5
U-234	.581	.455	1.28
U-235	73.0	51.3	35.4
U-236		4.19	7.02
U-238	292.0	287.8	283.3
Pu-238		.01	.08
Pu-239		2.38	3.13
Pu-240		.37	.77
Pu-241		.20	.64
Pu-242		.02	.12
Fission Products		2.04	40.8
Other Isotopics			
Am-241		.002	.011
Cm-242		-	-
Np-237		.146	.461

TABLE 7.9-18

RECYCLE FUEL INVENTORY - BEGINNING OF EQUILIBRIUM CYCLE DENATURED U-235/ThO₂ SSCR

Isotope	Fresh Fuel	(kg)	
		One Cycle Exposure Fuel	Two Cycles Exposure Fuel
Th-232	21992.2	21947.6	21786.5
Pa-233		11.0	11.5
U-232	1.74	1.74	1.73
U-233	486.7	472.1	464.6
U-234	202.5	208.0	211.6
U-235	672.4	507.3	378.2
U-236	396.8	420.7	431.9
U-238	5313.6	5277.9	5199.6
Pu-238		1.53	6.00
Pu-239		42.7	58.1
Pu-240		6.26	13.6
Pu-241		3.05	10.6
Pu-242		.213	1.68
Fission Products		333.8	666.6
Other Isotopics			
Am-241		.027	.18
Cm-242		-	-
Np-237		15.3	29.4

TABLE 7.9-19

MAKEUP FUEL INVENTORY - END OF EQUILIBRIUM CYCLE DENATURED U-235/ThO₂ SSCR

<u>Isotope</u>	(kg)		
	<u>One Cycle Exposure Fuel</u>	<u>Two Cycles Exposure Fuel</u>	<u>Three Cycles Exposure Fuel</u>
Th-232	1397.0	1388.0	1374.8
Pa-233	1.55	1.61	1.70
U-232	.006	.026	0.052
U-233	8.17	14.6	18.8
U-234	.455	1.28	2.30
U-235	51.3	35.4	24.0
U-236	4.19	7.02	8.85
U-238	287.8	283.3	278.5
Pu-238	.01	.08	.01
Pu-239	2.33	3.08	3.32
Pu-240	.37	.77	1.01
Pu-241	.20	.64	.99
Pu-242	.02	.12	.30
Fission Products	20.4	40.8	61.2
Other Isotopics			
Am-241	.002	.011	.022
Cm-242	-	-	-
Np-237	.146	.461	.829

TABLE 7.9-20

RECYCLE FUEL INVENTORY - END OF EQUILIBRIUM CYCLE DENATURED U-235/ThO₂ SSCR

<u>Isotope</u>	(kg)		
	<u>One Cycle Exposure Fuel</u>	<u>Two Cycles Exposure Fuel</u>	<u>Three Cycles Exposure Fuel</u>
Th-232	21974.6	21786.5	21584.0
Pa-233	23.6	24.6	25.9
U-232	1.74	1.73	1.73
U-233	459.5	451.5	442.5
U-234	208.0	211.6	214.8
U-235	507.3	378.2	281.3
U-236	420.7	431.9	435.2
U-238	5277.9	5199.6	5116.6
Pu-238	1.53	6.00	12.7
Pu-239	41.9	57.2	62.8
Pu-240	6.26	13.6	18.5
Pu-241	3.05	10.6	17.2
Pu-242	.213	1.68	4.57
Fission Products	333.8	666.6	1000.8
Other Isotopics			
Am-241	.027	.18	.38
Cm-242	-	-	-
Np-237	15.3	29.4	41.4

TABLE 7.9-21

FUEL MANAGEMENT INFORMATION

DENATURED U-233/ThO₂ SSCR

Average Capacity Factor -	75%
Approximate Fraction of Core Replaced -	one-third annually
Lag Time Assumed Between Fuel Discharge and Recycle Reload -	two years
Fissile Material Fabrication Loss Fraction -	1%
UF ₆ Conversion Loss -	0.5%
Fissile Material Reprocessing Loss Factor -	0.5%
ThO ₂ Requirements (tonne/GWe)	
initial core -	69
annual equilibrium reload requirement -	21
30-year cumulative requirement -	683
U-233 Requirements (kg/GWe)	
initial core -	1704
annual equilibrium charge -	716
annual equilibrium discharge -	478
30-year cumulative requirement - ^b	9940
Equilibrium Fissile Plutonium Discharge -	76
30-Year Cumulative Fissile Plutonium Discharge - ^a	2145

^a Additional credit (107 kg HM/GWe) available for partially burnt fuel in final core.

^b Additional credit available for ex-core fuel batches (942 kg/GWe) and partially burnt fuel in final core (1594 kg/GWe)

TABLE 7.9-22

REACTOR CHARGE DATA FOR THE DENATURED U-233/ThO₂ SSCR

(kg)															Total Heavy Metal	Fission Product
Year	Th-232	Pa-233	U-232	U-233	U-234	U-235	U-236	U-238	Pu-238	Pu-239	Pu-240	Pu-241	Pu-242	Np-237		
1	75170.0		6.17	2146.6	173.6	58.1	1.09	15983.3								93538.9
2	23261.1		2.05	910.5	73.6	24.7	0.46	6780.3								31052.7
3	23426.7		2.05	891.0	72.1	24.1	0.45	6634.4								31050.8
4	23059.0		2.05	887.5	115.7	27.7	2.32	6963.8								31058.8
5	23470.1		2.05	889.6	171.0	38.9	6.68	6472.7								31051.0
6																
7																
8																
9	23327.9		2.05	891.6	243.8	58.3	15.4	6510.9								31050.0
10																
11																
12																
13																
14	23214.3		2.05	895.8	298.7	74.1	25.4	6539.7								31050.0
15																
16																
17																
18																
19	23123.2		2.05	901.6	339.6	86.4	35.9	6561.2								31050.00
20																
21																
22																
23																
24	23056.6		2.05	901.6	372.9	95.9	46.2	6574.7								31050.0
25																
26																
27																
28																
29																
30																

TABLE 7.9-23

REACTOR DISCHARGE DATA FOR THE DENATURED U-233/ThO₂ SSCR

Year	(kg)														Total Heavy Metal	Fission Product
	Th-232	Pa-233	U-232	U-233	U-234	U-235	U-236	U-238	Pu-238	Pu-239	Pu-240	Pu-241	Pu-242	Np-237		
1	25996.3	40.9	.201	496.0	80.7	16.8	2.02	4021.2	.121	37.9	9.43	5.70	.87	.77	30708.9	341.3
2	24124.6	33.9	.533	531.4	125.8	26.2	4.46	5332.0	.486	55.7	17.8	14.8	3.98	1.87	30273.5	639.2
3	23150.8	31.4	1.53	529.0	148.1	32.3	6.39	5974.5	.560	66.9	22.5	20.7	7.16	1.34	29993.2	1066.8
4	22585.5	31.3	1.11	549.6	155.4	34.1	6.69	6468.7	.560	70.3	24.2	21.5	7.02	1.34	29957.3	1066.8
5	22807.0	30.6	1.11	543.0	149.7	35.6	6.86	6281.4	.560	68.2	23.1	20.6	6.80	1.34	29975.9	1066.8
6	22798.2	30.0	1.96	556.7	221.4	54.4	15.4	6179.9	1.01	69.2	22.1	19.8	6.24	2.74	29979.1	1064.1
7																
8																
9																
10																
11	22671.8	29.2	1.96	559.9	276.0	70.4	25.7	6217.5	1.56	71.1	22.9	20.7	6.24	4.43	29979.4	1061.8
12																
13																
14																
15																
16	22572.5	28.7	1.93	562.8	318.0	82.8	36.3	6246.9	2.12	72.6	22.9	20.7	5.99	6.16	29980.4	1060.4
17																
18																
19																
20																
21	22486.8	28.4	1.93	572.1	350.2	92.2	46.7	6269.8	2.65	73.7	23.0	20.9	5.90	7.75	29982.0	1059.3
22																
23																
24																
25																
26	22425.4	28.1	1.93	574.5	376.2	99.7	56.5	6283.6	3.13	74.5	23.0	20.9	5.84	9.23	29982.5	1058.3
27																
28																
29																
30																

TABLE 7.9-24

FUEL INVENTORY - BEGINNING OF EQUILIBRIUM CYCLE DENATURED U-233/ThO₂ SSCR

<u>Isotope</u>	(kg)		(kg)
	<u>Fresh Fuel</u>	<u>One Cycle Exposure Fuel</u>	<u>Two Cycles Exposure Fuel</u>
Th-232	23056.6	22859.8	22649.4
Pa-233		12.4	12.6
U-232	2.05	2.01	1.97
U-233	901.6	753.3	658.6
U-234	372.9	378.9	379.3
U-235	95.9	96.9	98.5
U-236	46.2	49.8	53.2
U-238	6574.7	6483.3	6386.0
Pu-238		.369	1.45
Pu-239		50.9	68.5
Pu-240		8.22	17.2
Pu-241		4.06	13.2
Pu-242		.319	2.28
Fission Products		353.4	706.0
Other Isotopics			
Am-241		.037	.219
Cm-242		-	-
Np-237		3.36	6.48

TABLE 7.9-25

FUEL INVENTORY - END OF EQUILIBRIUM CYCLE DENATURED U-233/ThO₂ SSCR

<u>Isotope</u>	(kg)		
	<u>One Cycle</u>	<u>Two Cycles</u>	<u>Three Cycles</u>
	<u>Exposure Fuel</u>	<u>Exposure Fuel</u>	<u>Exposure Fuel</u>
Th-232	22859.8	22649.4	22425.4
Pa-233	26.5	27.1	28.1
U-232	2.01	1.97	1.93
U-233	739.2	643.1	574.5
U-234	378.9	379.3	376.2
U-235	96.9	98.5	99.7
U-236	49.8	53.2	56.5
U-238	6483.3	6386.0	6283.6
Pu-238	.369	1.45	3.13
Pu-239	49.9	67.5	74.5
Pu-240	8.22	17.2	23.0
Pu-241	4.06	13.2	20.9
Pu-242	.319	2.28	5.84
Fission Products	353.4	706.0	1058.3
Other Isotopics			
Am-241	0.037	.219	.459
Cm-242	-	-	-
Np-237	3.36	6.48	9.23

TABLE 7.9-26

FUEL MANAGEMENT INFORMATION

Pu/ThO₂ SSCR

Average Capacity Factor -	75%
Approximate Fraction of Core Replaced -	one-third annually
Lag Time Assumed Between Fuel Discharge and Recycle Reload -	two years
Fissile Material Fabrication Loss Fraction -	1%
UF ₆ Conversion Loss -	0.5%
Fissile Material Reprocessing Loss Fractor -	0.5%
ThO ₂ Requirements (tonne/GWe)	
initail core -	81
annual equilibrium reload requirement -	25
30-year cumulative requirement -	820
Fissile Plutonium Requirements (kg HM/GWe)	
initail core -	2410
annual equilibrium charge -	1550
annual equilibrium discharge -	854
30-year cumulative requirement - ^a	26,625
Equilibrium U-233 Discharge (kg/GWe) -	283
30-Year Cumulative U-233 Discharge (kg/GWe)	8642

^a Does not include credit for ex-core fuel batches (1602 kg HM/GWe) or final core (3210 kg HM/GWe).

TABLE 7.9-27

REACTOR CHARGE DATA FOR THE Pu/ThO₂ BURNER SSCR

Year	(kg)											Np-237	Total Heavy Metal	Fission Product
	Th-232	Pa-233	U-232	U-233	U-234	U-235	U-236	U-238	Pu-238	Pu-239	Pu-240	Pu-241	Pu-242	
1	89211.0									2475.7	1038.9	560.3	221.3	93507.2
2	29156.0									1082.3	454.2	244.9	96.7	31034.1
3	29379.0									953.3	400.1	215.7	85.2	21033.1
4	28587.1									1083.1	737.5	432.5	205.4	31045.6
5														
6														
7														
8										1151.5	912.9	494.0	300.2	31047.4
9	28188.8													
10														
11														
12														
13										1231.3	1065.8	545.0	378.2	31049.2
14	27828.9													
15														
16														
17														
18										1285.5	1207.77	584.3	440.9	31050.3
19	27531.8													
20														
21														
22														
23										1338.2	1334.2	614.9	492.6	31051.4
24	27271.5													
25														
26														
27														
28														
29														
30														

TABLE 7.9-28

REACTOR DISCHARGE DATA FOR THE Pu/ThO₂ BURNER SSCR

	(kg)														Total Heavy Metal	Fission Product
Year	Th-232	Pa-233	U-232	U-233	U-234	U-235	U-236	U-238	Pu-238	Pu-239	Pu-240	Pu-241	Pu-242	Np-237		
1	29777.0	32.4	0.16	184.2	13.0	1.02	0.02		0.38	233.1	222.8	141.3	63.8	.00046	30669.2	346.7
2	29176.6	31.7	0.46	307.9	28.7	3.84	0.17		2.33	171.7	263.3	176.8	96.2	.0054	30259.7	697.6
3	28465.9	27.4	0.66	358.8	34.9	5.49	0.28		6.52	252.6	387.4	243.5	130.5	.014	29914.0	1048.7
4																1048.7
5																1048.7
6	28005.1	25.6	0.58	353.4	32.2	5.03	0.23		6.56	310.2	566.4	341.3	228.0	.016	29874.6	1049.4
7																1049.4
8																1049.4
9																1049.4
10																1049.4
11	27628.0	24.4	0.51	348.4	30.2	4.68	0.20		10.44	387.7	718.2	396.8	305.8	.011	29855.3	1049.6
12																1049.6
13																1049.6
14																1049.6
15																1049.6
16	27290.9	23.2	0.46	342.4	28.0	4.27	0.17		11.09	457.5	868.1	444.8	370.6	.0096	29841.5	1049.6
17																1049.6
18																1049.6
19																1049.6
20																1049.6
21	27006.4	22.6	0.43	338.4	26.9	4.06	0.16		11.56	514.2	1000.2	481.3	424.7	.0087	29830.9	1049.7
22																1049.7
23																1049.7
24																1049.7
25																1049.7
26	26757.6	22.0	0.40	334.4	25.8	3.86	0.15		11.89	565.3	1122.0	510.1	469.3	.0079	29822.8	1049.7
27																1049.7
28																1049.7
29																1049.7
30																1049.7

TABLE 7.9-29

FUEL INVENTORY - BEGINNING OF EQUILIBRIUM CYCLE Pu/ThO₂ SSCR

<u>Isotope</u>	(kg)		
	<u>Fresh Fuel</u>	<u>One Cycle Exposure Fuel</u>	<u>Two Cycles Exposure Fuel</u>
Th-232	27271.48	27085.71	26914.89
Pa-233		20.93	21.27
U-232		0.048	0.188
U-233		124.15	240.87
U-234		5.24	14.30
U-235		0.29	1.52
U-236		0.0034	0.037
U-238			
Pu-238		0.897	4.96
Pu-239	1338.19	1045.45	788.67
Pu-240	1334.21	1285.05	1214.37
Pu-241	614.88	584.80	549.87
Pu-242	492.61	486.54	478.68
Fission Products		348.74	698.64
Other Isotopics			
Am-241		17.10	26.68
Cm-242		-	-
Np-237		.00005	.0013

TABLE 7.9-30

FUEL INVENTORY - END OF EQUILIBRIUM CYCLE Pu/ThO₂ SSCR

<u>Isotope</u>	(kg)		
	<u>One Cycle</u> <u>Exposure Fuel</u>	<u>Two Cycles</u> <u>Exposure Fuel</u>	<u>Three Cycles</u> <u>Exposure Fuel</u>
Th-232	27085.71	26914.89	26757.57
Pa-233	20.93	21.27	22.042
U-232	0.048	0.188	0.403
U-233	124.15	240.87	334.41
U-234	5.24	14.30	25.83
U-235	0.29	1.52	3.86
U-236	0.0034	0.037	0.148
U-238			
Pu-238	0.897	4.96	11.89
Pu-239	1045.45	788.67	565.31
Pu-240	1285.05	1214.37	1121.95
Pu-241	584.80	549.87	469.26
Pu-242	486.54	478.68	469.26
Fission Products	348.74	698.64	1049.72
Other Isotopics			
Am-241	17.10	20.68	31.01
Cm-242	-	-	-
Np-237	.00005	.0013	0.0079