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SUMMARY REPORT

AN EVALUATION OF THE SUBSTITUTION OF ZIRCALOY FOR STAINLESS STEEL IN N.S. SAVANNAH FUEL-ELEMENT CONTAINERS

T. D. Anderson
E. E. Gross
H. C. McCurdy
L. D. Schaffer
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C. L. Whitmarsh

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Printed in USA. Price: $2.00 Available from the
Office of Technical Services
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ORNL-3476

Contract No. W-7405-eng-26

Reactor Division

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ABSTRACT

The possibility of reducing fuel-cycle costs for the N.S. SAVANNAH by replacing the stainless steel fuel-element containers in the permanent reactor core structure by similar containers of a zirconium alloy was investigated. These containers, although not integral parts of the fuel-bearing components, are located within the active core and divide the core into 32 separate channels into which the fuel elements are placed. Areas of investigation included reactor physics, fuel-cycle economics, materials compatibility, structural design, and reactor hazards. A summary of the method of analysis and results is given for each area of investigation.

Calculations indicated that the substitution of Zircaloy containers would increase core reactivity about 6% Δk and control-rod worth about 4% Δk. Fuel-cycle costs would be reduced about 26%. Zircaloy-4 appears to be compatible with the reactor system, except for some uncertainty with respect to fretting corrosion, which can be resolved only by tests. The substitution of cold-worked Zircaloy for stainless steel in the container assembly would necessitate only minor design modifications. Although this evaluation is strictly applicable only to the N.S. SAVANNAH reactor, the results demonstrate the feasibility and advantages of using zirconium alloys for in-core capital-cost components.

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1. INTRODUCTION

The N.S. SAVANNAH is the world's first nuclear-powered merchant ship. This combination passenger and cargo vessel is powered by a pressurized-water reactor that was designed and built by the Babcock & Wilcox Company. The reactor has a normal operating power rating of 63 Mw (thermal).

From the standpoint of economics, the inherent features of a nuclear propulsion system are high capital costs and potentially low fuel costs. Although the N.S. SAVANNAH is not competitive, in an economic sense, with conventional vessels, it does represent a substantial first step on which a sound nuclear-powered ship program could be based. Economic nuclear power for ships can be attained only by reducing capital costs and, at the same time, exploiting the potential of low fuel-cycle costs. The work summarized here was directed toward reducing fuel-cycle costs for the N.S. SAVANNAH by replacing part of the permanent stainless steel reactor core structure by a similar structure made of a zirconium alloy. The parts of the reactor core structure considered were the fuel-element containers, which are shown in Fig. 1. These containers, although not integral parts of the fuel-bearing components, are located within the active core. They provide 21 control-rod channels and 32 separate channels into which fuel elements are placed. To give a clearer understanding of the makeup of the core, a brief description of the reactor is given in the Appendix. A more comprehensive description was presented in the Final Safeguards Report for the N.S. SAVANNAH.[1]

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[1]Babcock & Wilcox Company, "Nuclear Merchant Ship Reactor Final Safeguards Report - Volume I, Description of the N.S. SAVANNAH," USAEC Report BAW-1164, June 1960.

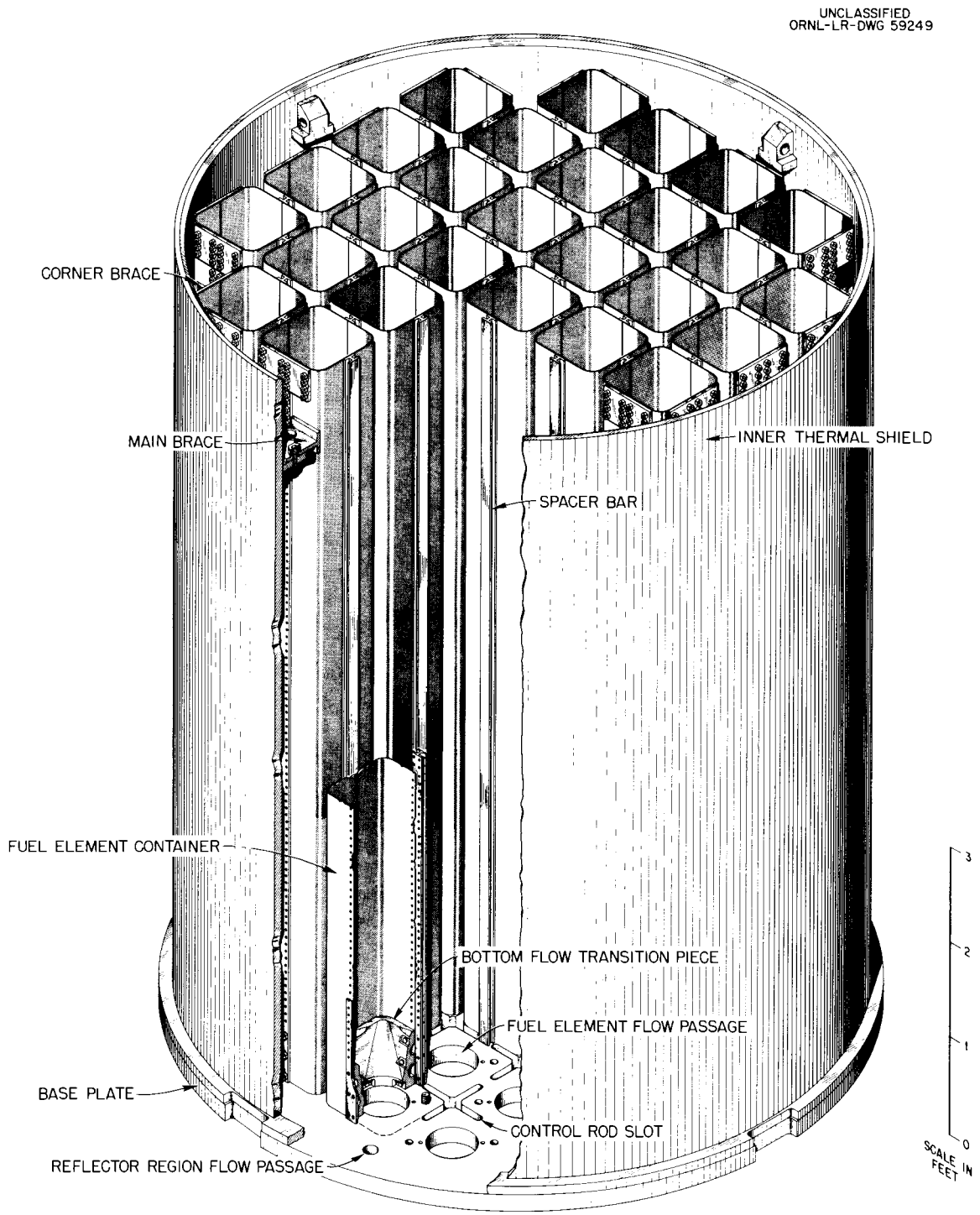

Fig. 1. Fuel-Element Container Assembly.

The present fuel-element containers, which are located in regions of high neutron flux, account for about one-third the total quantity of stainless steel in the active portion of the core. Thus the substitution of Zircaloy for stainless steel in the fuel-element container assembly was expected to result in a lower U^{235} inventory for a given reactivity lifetime. Another expected change in the core nuclear characteristics was an increase in control-rod worth, inasmuch as the dependence of control-rod worth on fuel-element container material had been demonstrated during the critical experiments. The physics and economic implications of this dual effect of the Zircaloy substitution made it necessary to examine more than one Zircaloy-core case.

The evaluation of the Zircaloy substitution encompassed several areas of investigation, as shown in Fig. 2. Physics calculations were made to determine the nuclear characteristics of the reactor using core I-type fuel elements and Zircaloy fuel-element containers. Each type of physics calculation made for the Zircaloy containers was also performed for stainless steel containers. In addition, the calculational method was checked by comparison with experimental results obtained during the critical experiments. Thus, some confidence was established that the estimated changes resulting from the Zircaloy substitution were real rather than merely a result of the calculational method. The evaluation of the economic incentive for changing to Zircaloy was based on the results of the physics calculations and a reference fuel cycle established for core I-type fuel elements with stainless steel containers.

A rather broad investigation was made to determine the compatibility of the Zircaloy fuel-element containers with the reactor system. This work included (1) consideration of the effects of corrosion, wear, and hydrogen pickup on the useful life of the Zircaloy structure, and (2) an evaluation of possible changes in the amount of crud activity as a result of replacing stainless steel components with Zircaloy. A structural analysis of the Zircaloy assembly was performed to determine the design modifications required to withstand the various forces imposed by ship motion, pressure forces, and temperature gradients. Further, consideration was given to the reactor safety implications of the Zircaloy substitution.

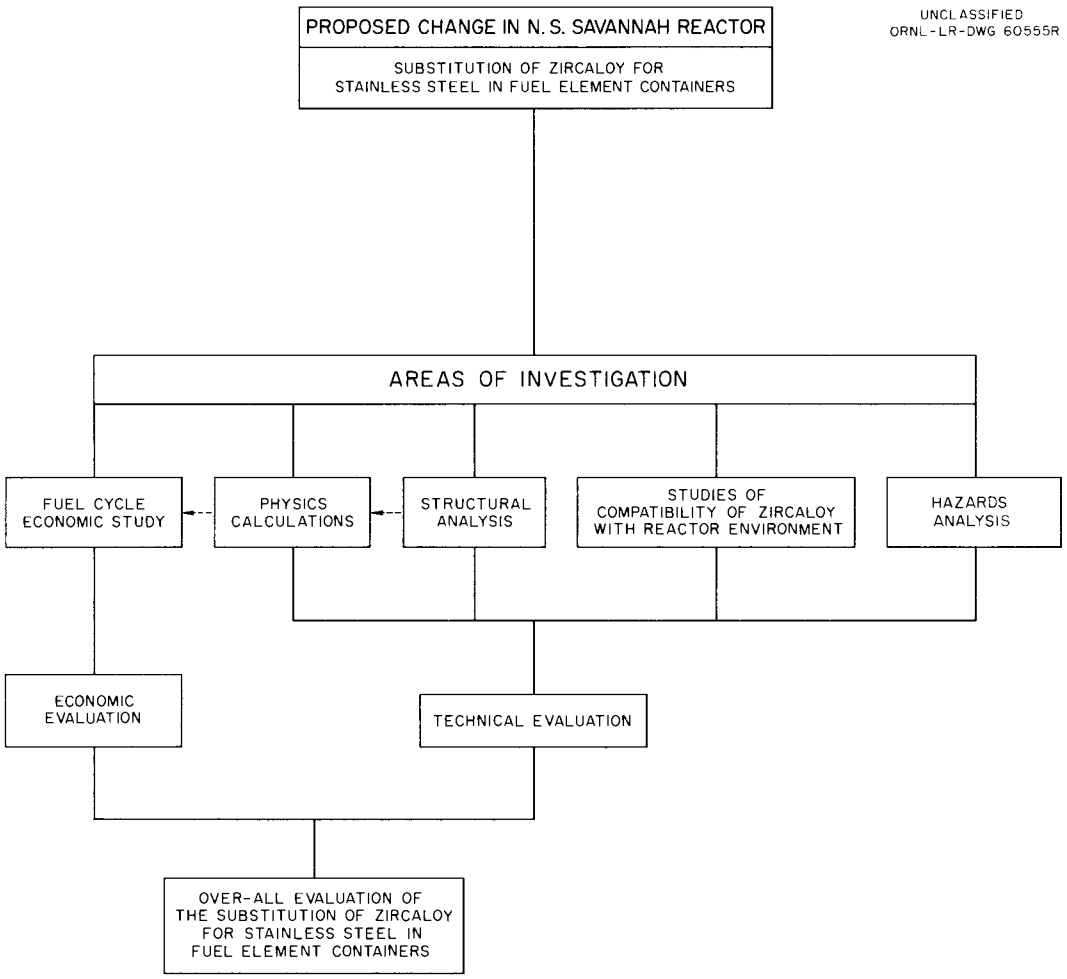

Fig. 2. Diagram Showing Scope of Zircaloy Evaluation.

A general ground rule of the evaluation was that no changes in the reactor system should be considered except those absolutely essential for the Zircaloy substitution. On this basis the only method of reactor control allowed was by control rods of the existing type. Similarly, for the physics and economic portions of the evaluation, the only fuel elements considered were those of the core-I configuration. In the structural investigation, however, consideration was given to requirements that might be imposed by future fuel-element designs. Both Zircaloy-2 and Zircaloy-4 (low-nickel-content Zircaloy-2) were considered in the compatibility portion of the evaluation. For the other parts of the evaluation, it was assumed that the properties of these two zirconium alloys are identical.

2. SUMMARY, CONCLUSIONS, AND RECOMMENDATIONS

Results of Investigations

Physics Calculations

The nuclear characteristics of a reference core made up of stainless steel fuel-element containers and core I-type fuel elements were computed to serve as a basis for judging the changes associated with the use of Zircaloy fuel-element containers. It was, of course, intended that the reference core be representative of core I. The main difference was that the reference core had a single-zone enrichment of 4.2 wt %, whereas core I has two enrichment zones (4.2 and 4.6 wt %). The difference between the two was shown, however, to be inconsequential. A summary of the important nuclear characteristics of the various cases studied is given in Table 1.

The reference core (case S-420), composed of stainless steel fuel-element containers and 4.2 wt % enriched fuel, gave a reactivity lifetime of about 1.63 full-power years (full thermal power, 63 Mw). The worth of all control rods in the reference core was about 16% Δk, giving a shutdown margin of a clean, cold core of something over 2% Δk. Since it was felt that a more sensible shutdown criterion would be based on the most important rod stuck out of the core, the shutdown margin with the central rod out was computed; the one-stuck-rod shutdown margin was 1.4% Δk.

The effect of substituting Zircaloy for stainless steel in the fuel-element containers was well demonstrated by case Z-420. This core was identical with the reference core, except for the use of the Zircaloy containers. It may be noted that the reactivity lifetime increased from 1.63 full-power years (fpy) to 3.20 fpy as a result of the Zircaloy substitution. In addition to the increase in core reactivity, the control-rod worth increased from 16 to 20% Δk. The increase in rod worth was not sufficient, however, to offset the 6% Δk increase in core reactivity. The shutdown margin was less than 1% Δk, and the cold, clean core could not be shut down with the central rod out. Thus case Z-420 would not be practical using the present control rods.

A case of more practical interest is case Z-386. The enrichment of this core, which had Zircaloy containers, was reduced to 3.86 wt % so that

Table 1. Summary of Calculated Nuclear Characteristics

Case S-420: Reference case, core I-type fuel elements (uniform enrichment of 4.20 wt %) in stainless steel containers

Case Z-420: Same as reference case except Zircaloy containers

Case Z-386: Core I-type fuel elements (uniform enrichment of 3.86 wt %) in Zircaloy containers; enrichment adjusted to give same shutdown margin with central rod out as reference case

Case Z-338: Core-I type fuel elements (uniform enrichment of 3.38 wt %) in Zircaloy containers; enrichment adjusted to give same lifetime as reference case

	Stainless Steel Containers, Reference Case S-420	Zircaloy Fuel-Element Containers		
		Case Z-420	Case Z-386	Case Z-338
One-zone enrichment, wt % U^{235}	4.20	4.20	3.86	3.38
Multiplication factor of clean core with rods out (followers in)				
Cold	1.132	1.195	1.172	1.135
Hot	1.079	1.138	1.114	1.079
Multiplication factor of clean core with rods in				
Cold	0.972	0.991	0.969	0.934
Hot	0.882	0.910	0.886	0.848
Cold worth of all control rods, % Δk	16	20	20	20
Shutdown margin of cold, clean core with central rod out, % Δk	1.4	Supercritical	1.4	4.9
Lifetime, years at thermal power of 63 Mw	1.63	3.20	2.52	1.63

the shutdown margin with the central rod out was the same as for the reference case, that is, 1.4% Δk. The computed reactivity lifetime of this core was 2.52 fpy. Thus, using the same control rods as for the reference case, an 8% reduction in enrichment gave the same shutdown margin (with the central rod out) and a 55% increase in life relative to that of the reference core.

Case Z-338 demonstrated the effect of the Zircaloy substitution on enrichment. In this case it was possible to reduce the enrichment to 3.38 wt %, a reduction of nearly 20%, and yet maintain the reference core lifetime of 1.63 fpy. Also, this core was just as reactive as the reference core, and yet the shutdown margin was 4.9% Δk, as compared with the 1.4% Δk of the reference core. This again demonstrated the significant increase in control-rod worth resulting from the Zircaloy substitution.

A calculation in x-y geometry of the power distribution for the reference core with the control rods out (followers in) gave an overall radial peak-to-average power ratio of approximately 2. Similar calculations for the core with Zircaloy containers showed that the local peaking was more severe but the gross radial distribution was more uniform than in the reference core. These two effects combined to give almost identically the same overall radial peak-to-average power ratio for the cores with Zircaloy containers as for the reference case.

Economic Evaluation

A steady-state fuel-cycle cost was calculated for the reference core (case S-420) to provide a basis for comparing fuel-cycle cost reductions resulting from the use of Zircaloy fuel-element containers. The reference core was selected for this purpose in preference to core I to maintain consistency with the physics analysis. It was found, however, that the calculated fuel-cycle cost savings resulting from the Zircaloy substitution were little affected by the choice between core I and the one-zone reference core. A summary of the calculated cost reductions for the cores with Zircaloy fuel-element containers is given in Table 2.

The cost savings for case Z-420 show the effects of the direct substitution of Zircaloy for stainless steel. The results indicate that a

Table 2. Cost Reductions Resulting from Substituting Zircaloy for Stainless Steel in Fuel-Element Containers

Cost Component	Cost Reduction (%) Based on Cost for Reference Case (case S-420)[a]		
	Case Z-420 (lifetime, 3.20 fpy)	Case Z-386 (lifetime, 2.52 fpy)	Case Z-338 (lifetime, 1.63 fpy)
Fabrication	49	38	9
Net burnup	<1	4	2
Reprocessing	49	37	7
Uranium use charge	7	15	26
Working capital	2	5	9
Total fuel cycle	32	26	10

[a]See Table 1 for description of cases.

reduction in total fuel-cycle cost of about 32% could be realized for this case. As pointed out before, however, case Z-420 is marginal with respect to shutdown capability. Thus, it appears that different control rods or a burnable poison would be necessary before the indicated cost saving could be achieved in practice.

Case Z-386 could be controlled with the present control rods and without the use of burnable poisons. The estimated fuel-cycle cost reduction for this case is 26%. This cost reduction is brought about by both an increase in lifetime and a reduction in enrichment relative to the reference core. Extension of the lifetime is, however, the more important effect.

The economic effect of utilizing Zircaloy containers to reduce enrichment (keeping lifetime constant) is illustrated by case Z-338. Decreasing the enrichment from 4.2 to 3.38 wt % resulted in a fuel-cycle cost saving of 10%. Although substantial, this cost reduction does not compare favorably with the saving obtainable by utilizing the Zircaloy containers to give increased reactivity lifetime.

It should be noted that most previous studies comparing Zircaloy and stainless steel as core structural materials were of a rather general

nature and, consequently, were incapable of demonstrating the influence of structural materials on control-rod worth. If the more general approach had been used in this study, case Z-338 would have been the only one considered; and, as a result, the economic potential of the Zircaloy substitution would have been underestimated.

Compatibility Studies

Zircaloy-2 and Zircaloy-4 were evaluated for long-term service in the core on the basis of a literature survey of basic properties of the materials, an evaluation of the reactor water chemistry, and operating experience with other pressurized-water reactors. It should be emphasized, however, that existing data were obtained in relatively short-duration tests, and uncertainties inherent in extrapolation of the data are present. For 20-year service, the following conclusions were reached:

1. Corrosion resistance is excellent, with an indicated maximum penetration of less than 2 mils on each exposed surface.

2. Based on rather limited data, no major wear problem is expected as a result of using Zircaloy fuel-element containers.

3. The susceptibility of Zircaloy to fretting corrosion is uncertain, and to resolve this problem, a proof test is needed.

4. Zircaloy-4 is recommended over Zircaloy-2 because of a lesser tendency to pick up hydrogen. Hydrogen concentrations at 20 years of 261 and 104 ppm for Zircaloy-2 and Zircaloy-4, respectively, were calculated for 140-mil-thick plates.

5. Long-lived crud activity would not exceed the activity resulting from the stainless steel components for which the primary coolant decontamination system was designed.

Structural Analysis

It was found that only minor design modifications of the fuel-element container assembly would be required as a result of the Zircaloy substitution if 15% cold-worked Zircaloy were used. The most significant design change would be an approximately 30% increase in thickness of the container walls.

Hazards Analysis

The only aspect of the Zircaloy substitution that might add to the operational hazards would be the possibility of a zirconium-water reaction during the maximum credible accident (mca). It was conservatively calculated that about 21% of the total zirconium in the core would be oxidized during the mca. The energy released from the reaction would not, however, contribute to the peak containment-vessel pressure because of the time delay between the start of the mca and the initiation of the zirconium-water reaction. It is concluded, therefore, that the Zircaloy substitution would not significantly alter the consequences of the maximum credible accident.

Conclusions and Recommendations

It is apparent from the results of this study that the use of Zircaloy fuel-element containers would significantly reduce the fuel-cycle cost of the reactor. It must, of course, be recognized that there are some uncertainties in the evaluation given here. Recognition of these uncertainties forms the basis for additional work which should be accomplished prior to the use of Zircaloy containers. The two major areas in which additional work is required are discussed below.

1. Although the physics calculations were quite detailed, there are uncertainties in such calculations. In order to obtain the maximum benefit from the Zircaloy containers with the first new core loading, it is thought that a critical experiment should be performed. The critical experiment would be useful as a design tool to specify the maximum enrichment, and therefore lifetime, consistent with the minimum acceptable shutdown margin. Nevertheless, it should be noted that a critical experiment would not be essential if something other than the optimum enrichment were acceptable. Using the results of the physics calculations given here, it should be possible to construct an acceptable core. Of course, if the zero-power test of the core should indicate an insufficient shutdown margin, reliance would have to be placed on reactivity shimming by means of lumped poison.

2. With respect to the compatibility of Zircaloy with the system, it was found that there was a possibility of the occurrence of fretting corrosion between the fuel-element bundles and fuel-element containers. It would be necessary to perform a loop test using a full-scale mockup of a fuel-element container and a fuel bundle to resolve this uncertainty.

3. REACTOR PHYSICS

There is another important impetus, aside from the expected reduction in U^{235} inventory, for investigating the substitution of Zircaloy for stainless steel in the fuel-element container structure. Because of the proximity of the containers to the control rods, substituting Zircaloy for stainless steel in this region should result in an increase in control-rod worth, since the flux in this region is not as depressed by Zircaloy containers as by stainless steel containers. This effect was demonstrated earlier with aluminum containers in the critical experiments.[2]

An increase in control-rod worth would permit an increase in core reactivity and consequently an increase in core reactivity lifetime. Thus, a core with Zircaloy containers offers the attractive possibility of reducing fuel-cycle costs by simply increasing core lifetime for a given control rod shutdown requirement. It is to be emphasized that the magnitude of this effect (i.e., control-rod worth dependence on container material) depends very much on the details of this very heterogeneous core and cannot be evaluated by a general type of study.

These considerations led to analytical methods which were somewhat detailed in both physics and geometry and which made extensive use of available machine codes. Descriptions of the physical and geometrical models employed in the nuclear evaluation of the container materials are given below. The results obtained by using these models are then compared with critical experiment data, and, finally, results comparing Zircaloy and stainless steel as container materials are presented.

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[2]R. M. Ball, A. L. MacKinney, and J. H. Mortenson, "Marty Critical Experiments - Summary of 4% Enriched UO_2 Cores Studied for NMSR," USAEC Report BAW-1216, Babcock & Wilcox Co., May 1961.

Analytical Model

Treatment of Cross Sections in Analytical Model

In order to consider the two-dimensional aspects of the core design and to adequately include the effects of the control rods, a four-energy-group method was used, since the available two-dimensional codes had a four-energy-group limit. (This limitation of two-dimensional codes has recently been alleviated by the 18-energy-group ANGIE code.[3]) The group cross sections for each material were flux-averaged cross sections. The thermal group cross sections were flux averaged over a hardened Maxwell-Boltzmann distribution using an effective neutron temperature based on the work of Brown.[4] The water and control rod followers in the channels between the fuel-element containers were assumed to be in an unhardened Maxwell-Boltzmann flux distribution when the followers were present in the channels. The dependence of the average thermal cross sections on the moderator temperature was based on the work of Petrie, Storm, and Zweifel.[5]

A multigroup slowing-down calculation using the GNU code[6] was used to obtain flux-averaged cross sections over the three nonthermal groups. A typical slowing-down flux versus lethargy distribution giving the limits of the three nonthermal groups is shown in Fig. 3. In the slowing-down calculation, the core was assumed to be homogeneous with respect to the nonthermal neutrons. The very heterogeneous effect of resonance absorption in U^{238} was accounted for by generating multigroup U^{238}

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[3]Stuart P. Stone, "9-ANGIE – A Two-Dimensional Multigroup Neutron Diffusion Theory Reactor Core for the IBM 709 or 7090," USAEC Report UCRL-6076, University of California Radiation Laboratory, Oct. 28, 1960.

[4]H. D. Brown, "Neutron Energy Spectra in Water," USAEC Report DP-64, Du Pont de Nemours (E.I.) & Co., February 1956.

[5]C. D. Petrie, M. L. Storm, and P. F. Zweifel, "Calculation of Thermal Group Constants for Mixtures Containing Hydrogen," Nuclear Sci. Eng., 2: 728 (1957).

[6]J. M. Bookston, C. L. Davis, and B. E. Smith, "GNU, A Multigroup One-Dimensional Diffusion Program for the IBM-704," General Motors Corporation Report, April 8, 1957.

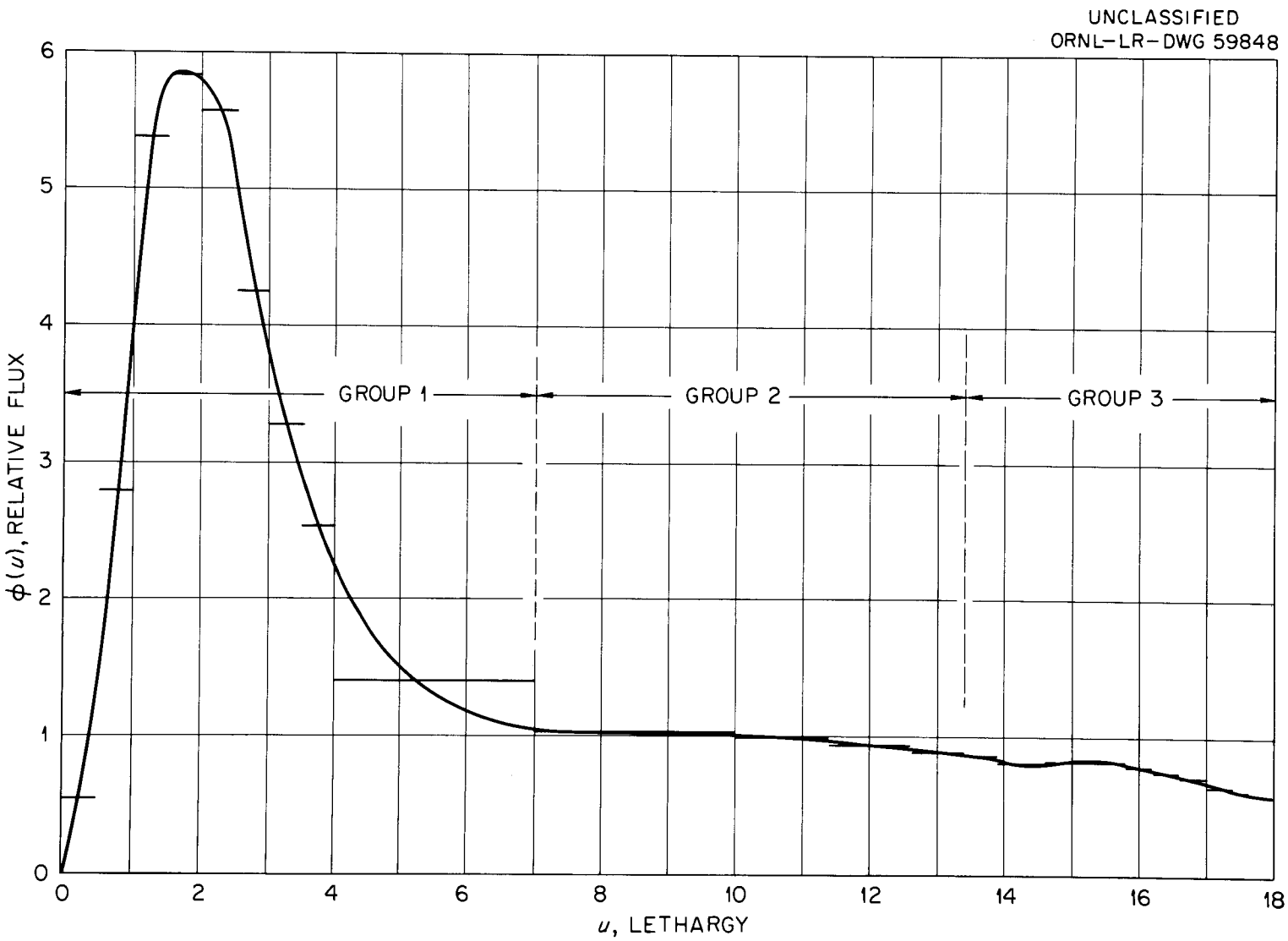

Fig. 3. Slowing-Down Flux Versus Lethargy Distribution for Core I at 68°F.

absorption cross sections that were consistent with a Dancoff-corrected[7] resonance integral[8] appropriate to the N.S. SAVANNAH reactor geometry. Other important physical effects accounted for in the determination of the slowing-down flux versus lethargy distribution (Fig. 3) were (1) Goertzel-Selengut hydrogen slowing down, (2) inelastic-scattering slowing down, (3) leakage from the core, (4) absorptions in all materials on the basis of available resonance integrals,[9] and (5) nonthermal fissions in U^{235} consistent with differential cross-section data and consistent with the measured fission resonance integral.[9]

Treatment of Geometry in Analytical Model

The four-group cross sections so obtained were used to determine flux distributions throughout the reactor. For purposes of the physics calculations, the nature of the core geometry is well illustrated by the horizontal section through a ferrule plane shown in Fig. 4. The geometry of Fig. 4 was, of course, much too complicated and needed to be greatly simplified before gross flux distributions throughout a full-size core could be computed.

The first step in simplification was to determine flux distributions across a so-called typical fuel-pin cell. The objective here was to reduce the complexity of the core by replacing the fuel-pin region by an equivalent homogeneous medium. The repetitive arrangement of fuel pins suggested defining a unit fuel-pin cell whose area was the fuel-pin pitch squared. For an infinite array of fuel pins the net current across a cell boundary should be zero, and a reasonable approximation to this boundary was the surface of a cylinder whose axis was coincident with the fuel-pin axis. The geometry of a fuel-pin cell giving details of the pellet, gap, and cladding sizes for a core I fuel pin is shown in Fig. 5. The

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[7]J. A. Thie, "A Simple Analytical Formulation of the Dancoff Correction," Nuclear Sci. Eng., 5: 75 (1959).

[8]E. Hellstrand, "Measurements of the Effective Resonance Integral in Uranium Metal and Oxide in Different Geometries," J. Appl. Phys., 28: 1943 (1957).

[9]R. L. Macklin and H. S. Pomerance, "Resonance Capture Integrals," Progr. in Nuclear Energy, Vol. 1, p. 179, McGraw-Hill, New York, 1956.

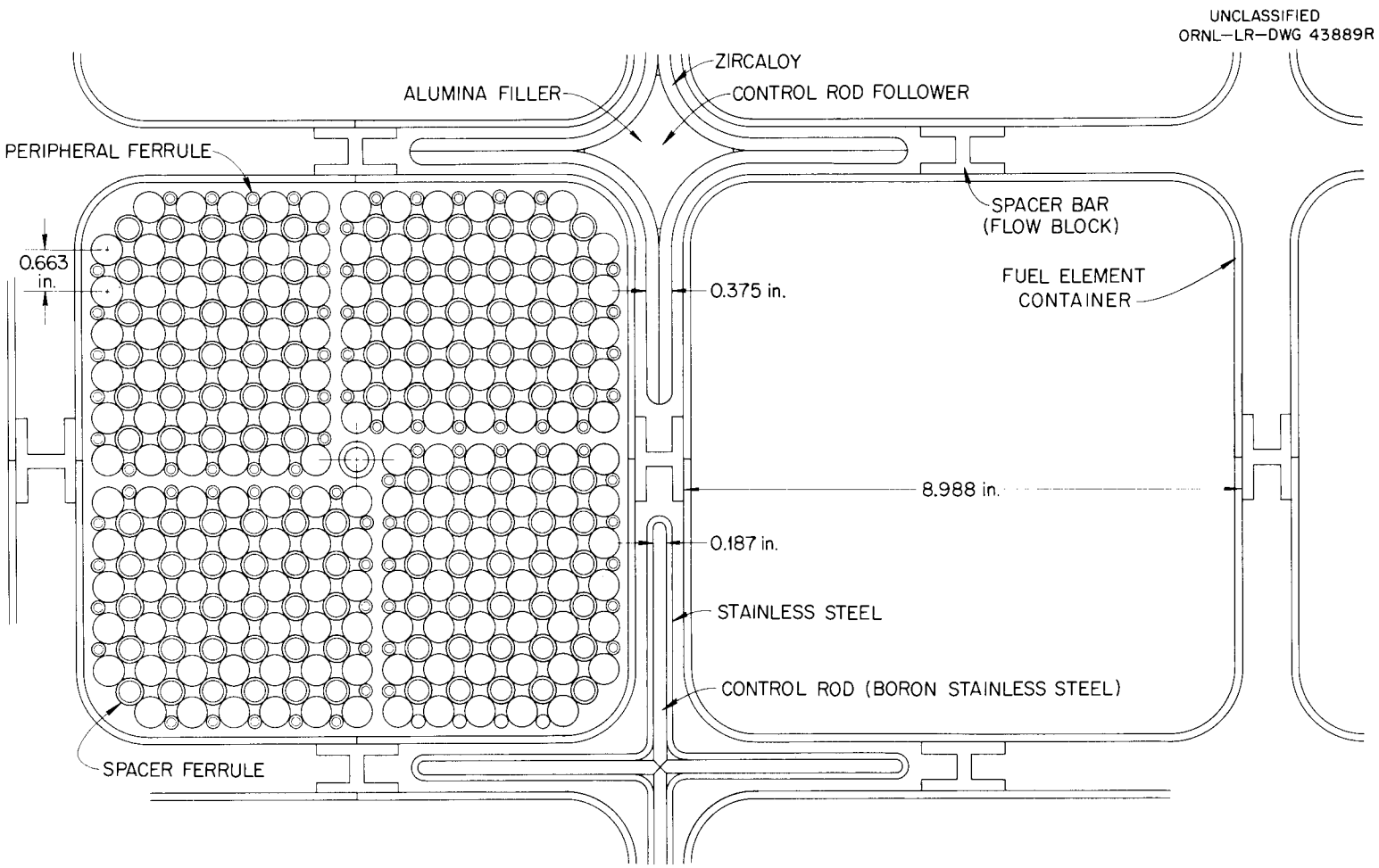

Fig. 4. Details of Fuel Cell.

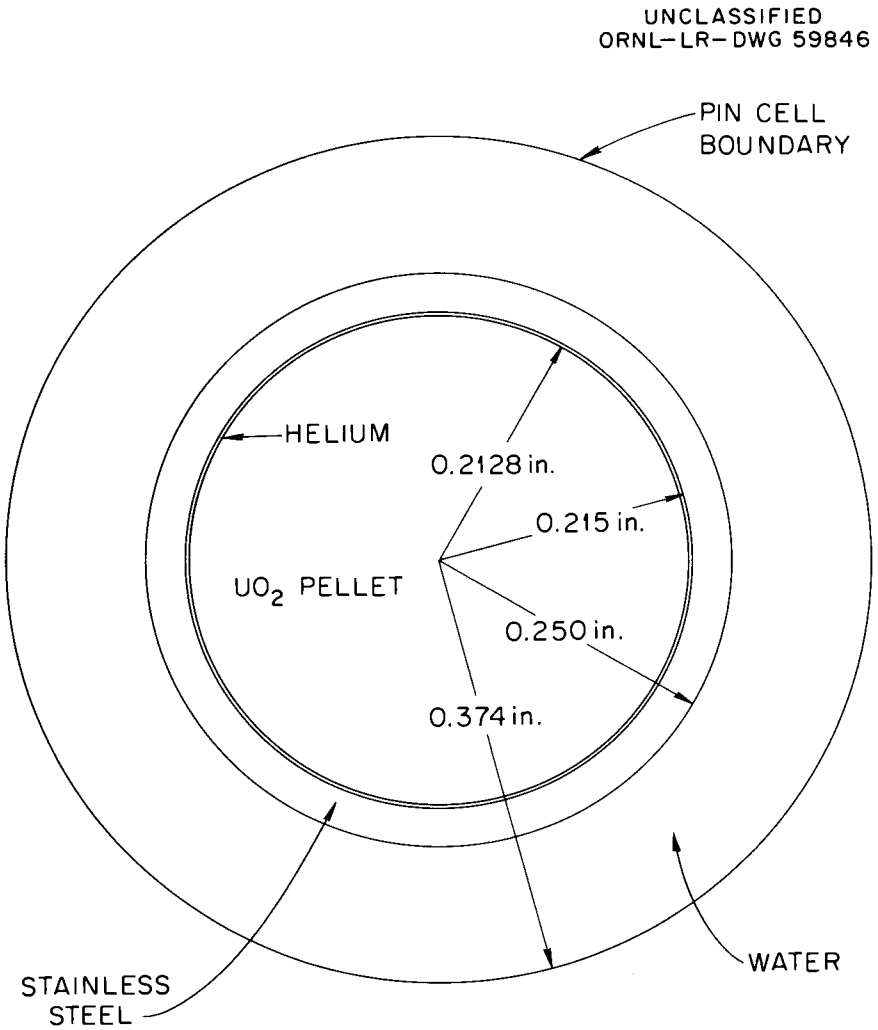

Fig. 5. Details of Geometrical Model Used for Fuel-Pin Cell.

four-group flux distributions satisfying the above assumptions are shown in Fig. 6. The data presented were obtained using the SNG transport theory code[10,11] in the S-4 approximation, since diffusion theory was not expected to apply well in the strongly absorbing UO_2 region.

The data of Fig. 6 indicate that the fuel-pin region can be replaced by a single homogeneous region. A similar calculation with an additional

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[10]B. G. Carlson, "The Sn Method and the SNG and SNK Codes," USAEC Report LASL T-1-159, Los Alamos Scientific Laboratory, Jan. 26, 1958.

[11]B. G. Carlson and G. I. Bell, "Solution of the Transport Equation by the Sn Method," Proceedings of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy, Geneva, 1958, Vol. 16, p. 535, United Nations, New York, 1958.

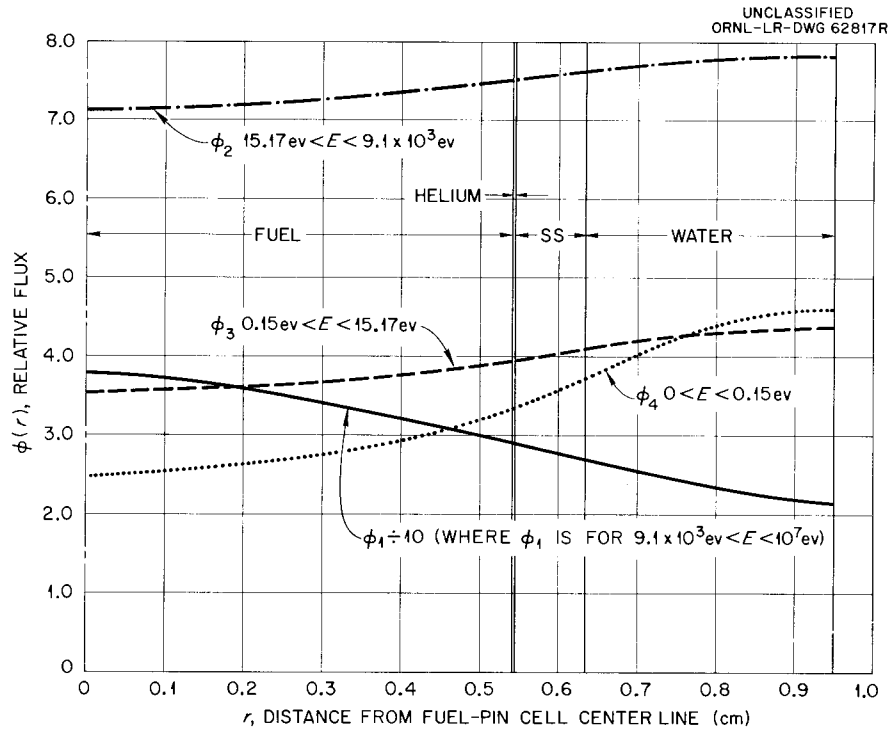

Fig. 6. S-4 Flux Distributions in Fuel-Pin Cell at 68°F.

annular region representing the stainless steel ferrules was used to homogenize the fuel-pin region of the core containing ferrules. The homogenized fuel-pin region without ferrules was then combined with the homogenized fuel-pin region with ferrules in slab geometry to obtain homogenized cross sections representing the entire fuel-pin region. In addition to reducing the complexity of the core, this series of calculations revealed that elimination of the stainless steel ferrules (less than 0.3% of the core volume) would increase core reactivity by 1% Δk.

Although homogenization of the fuel-pin region greatly reduced the complexity of the core geometry, there was still so much remaining detail that the full core geometry could not be handled within the space-point limitations of the available two-dimensional PDQ code.[12] The next step in simplification was to further reduce the complexity of an individual fuel-element container cell, which was the next largest repetitive unit

[12]G. G. Bilodeau et al., "PDQ, An IBM-704 Code to Solve the Two-Dimensional Four-Group Neutron Diffusion Equation," USAEC Report WAPD-TM-70, Westinghouse Atomic Power Division, 1957.

in the core. Although a fuel-element container cell was obviously two dimensional, many features of this cell are well illustrated by the results obtained from a one-dimensional slab geometry treatment. Again using symmetry arguments, solutions can be obtained for the flux distributions in an infinite array of slab fuel elements. The resulting S-5 flux distributions for the case when control rods were present in the channels between containers is shown along with the slab geometry in Fig. 7. These slab geometry results should be representative of the flux distributions several mean free paths (thermal mean free path, ~1 cm) from a corner, and the effect of the control rod on the four-group fluxes is well demonstrated. The strong nonthermal absorption properties of the control rods revealed by Fig. 7 may also be illustrated by the energy distribution of neutron absorptions in the control rod as shown in Table 3. It is to be noted that, according to these results, only about one-fourth of the neutrons absorbed by the control rods are thermal neutrons.

Aside from describing the general characteristics of the control rods, the data of Fig. 7 served the important function of providing

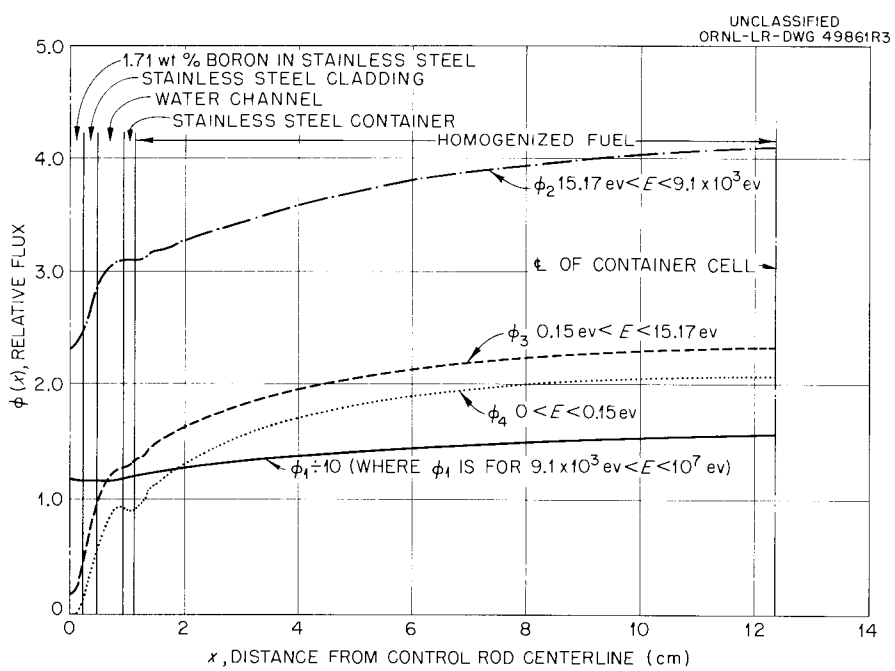

Fig. 7. S-5 Flux Distributions at 68°F with Control Rods Present in Channels Between Fuel-Element Containers.

Table 3. Energy Distribution of Control-Rod Absorptions

Group	Energy Interval	Control-Rod Absorptions (%)
1	9.1×10^3 ev < E < 10^7 ev	8
2	$15.17 \times$ ev < E < 9.1×10^3 ev	29
3	0.15 ev < E < 15.17 ev	37
4	0 < E < 0.15 ev	26

transport-theory boundary conditions at the surface of the control rod.[13] These boundary conditions were then used to replace the control rods in a two-dimensional diffusion-theory calculation of a fuel-element-container cell (an x-y geometry transport theory code was not available for these calculations). This procedure provided a consistent method for treating the transport properties of control rods, with due respect for the heterogeneities in the control-rod neighborhood.

Similar results for Zircaloy followers in the channels between containers are shown in Fig. 8. These data illustrate the severe thermal flux peaking caused by the water channels, and the important effect of this peaking on the flux level in the container material is evident. The use of thermal cross sections averaged over a Maxwell-Boltzman distribution for the channel region and thermal cross sections averaged over a hardened Maxwell-Boltzman distribution for the container and fuel regions, as described above, may not give the best results for the thermal flux distributions in the channel area.[14,15] This method was consistently applied, however, and the effect on thermal flux and reactivity of

[13]B. W. Colston, E. E. Gross, and M. L. Winton, "Heterogeneous Control Rod Studies," Proc. ANPP Reactor Analysis Seminar, October 11-12, 1960, USAEC Report MND-C-2487, Martin Co., Nuclear Division, January 1961.

[14]W. B. Wright and F. Fenier, "Note on Position Dependent Spectra," Nucl. Sci. Eng., 6: 81 (1959).

[15]G. P. Calame, "A Few-Group Theory of Water Gap Peaking," Nucl. Sci. Eng., 8: 400 (1960).

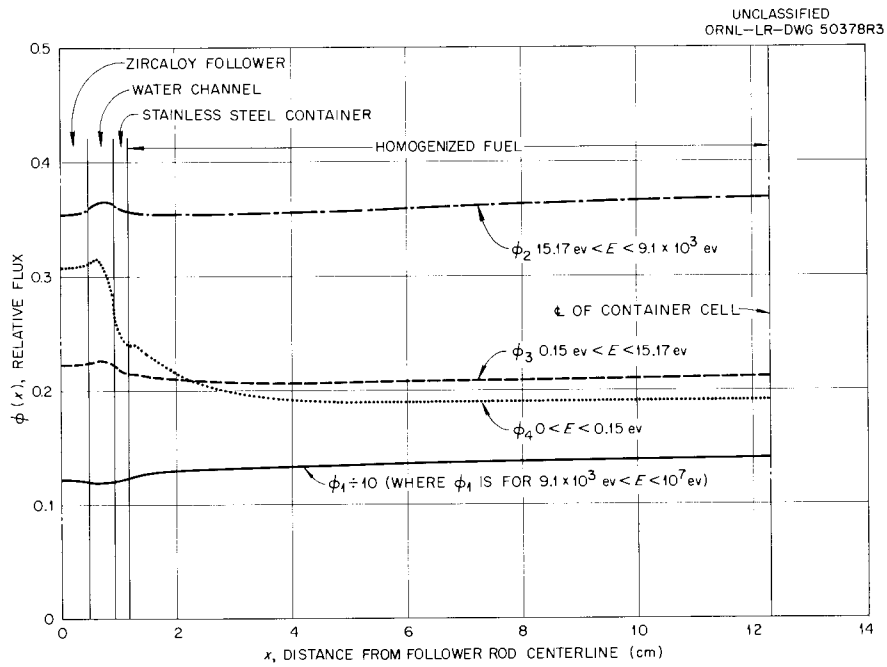

Fig. 8. S-5 Flux Distributions at 68°F with Control-Rod Followers Present in Channels Between Fuel-Element Containers.

Zircaloy containers relative to stainless steel containers should be fairly represented.

The two-dimensional aspects of the container were investigated in the geometry of Fig. 9, which shows one quadrant of a fuel-element container cell with control-rod followers in the channel region. The compromises in geometry necessary to obtain a practical result are evident upon comparison of Fig. 9 with the actual fuel-element container and control-rod follower geometries shown in Fig. 4. The purpose of the fuel-element-container cell calculation was to further reduce the complexity of the core so that the full-size core (32 fuel elements and 21 control rods or followers) could be handled within the limitations of the PDQ code.[12] A satisfactory simplification in container cell geometry was found to be (1) homogenization of the fuel-pin region with the central stainless steel support tube and the water filling this tube, (2) homogenization of the container material with the channel water, and (3) homogenization of the Zircaloy follower and its alumina filler.

With the above simplification in container cell geometry, the full-sized core could be handled in the geometry of Fig. 10. The leakage of

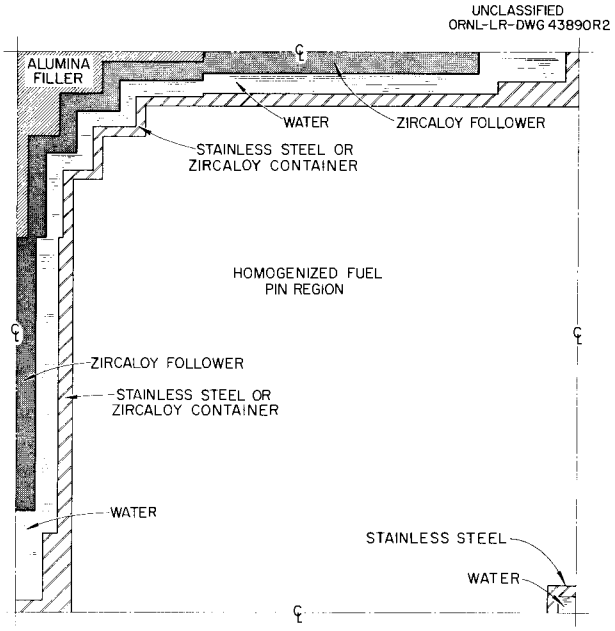

Fig. 9. Geometrical Model for One Quadrant of a Fuel-Element Container Cell with Control-Rod Followers in the Channel Region.

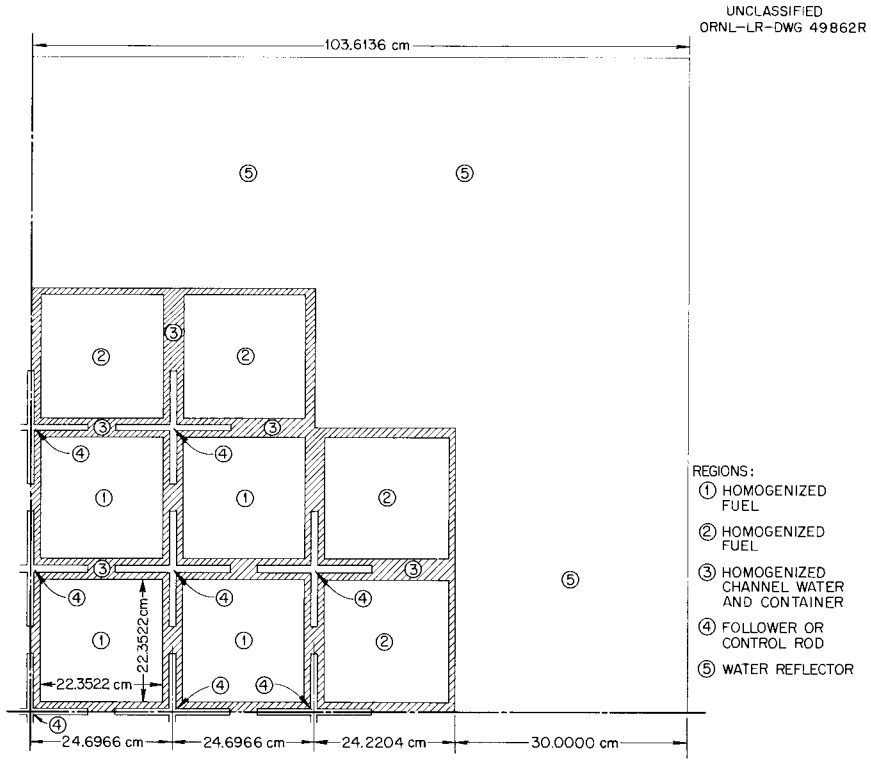

Fig. 10. Geometrical Model for One Quadrant of Core I.

neutrons in the axial direction was accounted for by specifying a group-independent axial buckling, which was obtained from the critical experiments.[16] The results obtained from a calculation in the geometry of Fig. 10 were used to homogenize the entire core, and a calculation of the full-sized core in r-z geometry was then performed. In this way leakage from all sides of the core was treated explicitly.

One way of illustrating the detail contained in the method just outlined is shown in Fig. 11, which presents the average four-group flux levels in each of the regions of core I at the operating temperature. All fluxes were normalized to unity in the UO_2 region. It is to be noted that the thermal flux level in the stainless steel container is

[16] "NMSR Quarterly Technical Report, July–September 1959," BAW-1180, pp. 23 ff.

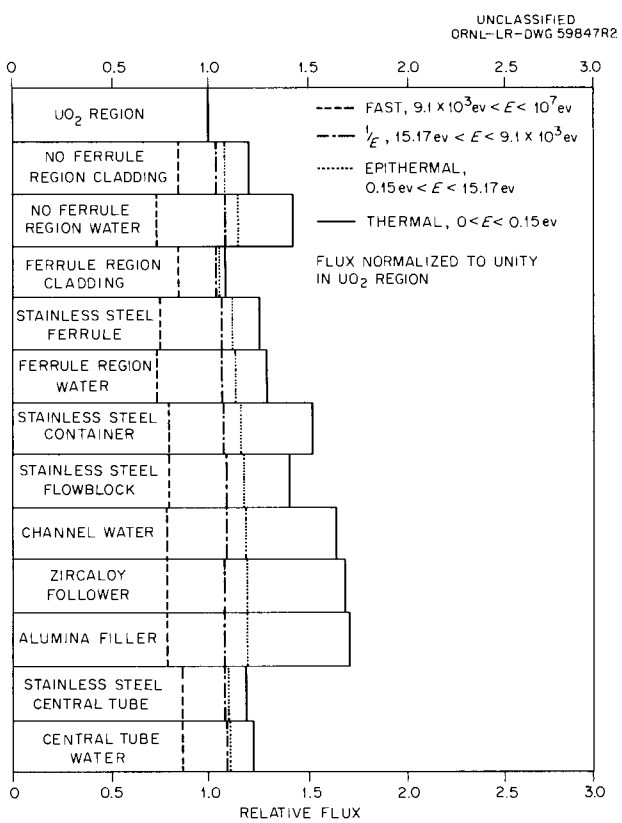

Fig. 11. Relative Four-Group Fluxes in Core I Components at 508°F.

considerably higher than the thermal flux level in the stainless steel cladding.

Comparison of Model Results with Critical Experiment Data

The four-group method, which is described in more detail elsewhere,[17] was applied to some critical experiment data[2] that compared aluminum fuel-element containers with stainless steel containers in a core with 16 identical fuel elements. A comparison of measured and calculated multiplication factors for various control-rod patterns in Marty II (a core with stainless steel containers) and Marty III (a core with aluminum containers) is shown in Table 4. The comparison shows the sizeable increase in both core reactivity and control-rod worth resulting from the substitution of aluminum for stainless steel. The ability of the analytical method to estimate the effects of material exchange is also evident from Table 4.

Analytical Results

Beginning-of-Life Multiplication Factors

Application of the four-group method to core I fuel elements within stainless steel containers and within Zircaloy containers yielded the multiplication factors shown in Fig. 12. These data are for single-fuel-zone cores at 68°F. The effect of two-fuel-zone loading was not important in this evaluation and is discussed later. Calculations of two enrichments (3.0 and 4.2 wt % U^{235}) were made for the Zircaloy containers. The shape of the curve between 3.0 and 4.2 wt % U^{235} enrichments was taken from the results of a previous survey[18] using a somewhat simpler

[17]E. E. Gross, B. W. Colston, and M. L. Winton, "Nuclear Analyses of the N.S. SAVANNAH Reactor with Zircaloy or Stainless Steel as Fuel-Element Containers," USAEC Report ORNL-3261, Oak Ridge National Laboratory, April 24, 1962.

[18]Memo from J. A. Bast et al., Massachusetts Institute of Technology Engineering Practice School, to E. E. Gross, Oak Ridge National Laboratory, "Comparison of Zircaloy and Stainless Steel as the Structural Materials in the Core of the N.S. SAVANNAH," Nov. 9, 1959.

Table 4. Comparison of Calculated and Experimental Multiplication Factors for Marty II and Marty III Critical Experiments

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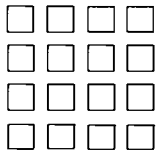
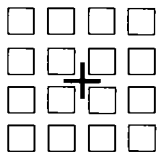
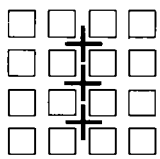
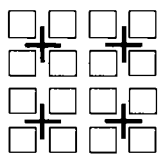
	MARTY II		MARTY III	
ROD PATTERN	k_d EXP	k_s CALC	k_d EXP	k_s CALC
	1.043 ± 0.001	1.055	1.16 ± 0.02	1.180
	1.024 ± 0.001	1.030	1.13 ± 0.02	1.148
	1.0011 ± 0.0001	1.000	1.09 ± 0.01	1.099
	1.013 ± 0.001	1.021	1.11 ± 0.01	1.126

analytical method. It is apparent from Fig. 12 that substitution of Zircaloy containers for the present stainless steel containers would increase the reactivity of 4.2 wt % U^{235} core I fuel elements by about 6% Δk.

The worth of the control rods may be obtained from Fig. 12 by subtracting the multiplication factor of the core with control rods from the multiplication factor of the core with control-rod followers. The rod worth obtained in this way is remarkably independent of enrichment in a core with Zircaloy containers, at least in the enrichment range of 3.0 to 4.2 wt % U^{235}. Although the 21 control rods appear to be worth

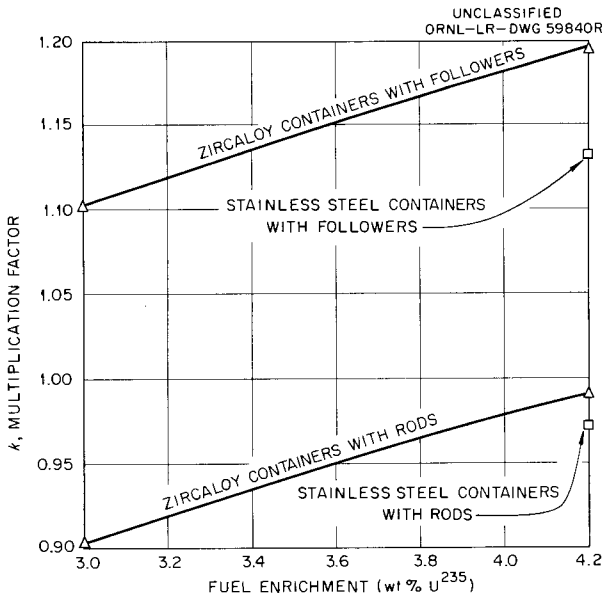


Fig. 12. Calculated Multiplication Factors at 68°F for Core I Type Fuel Elements in Zircaloy and in Type 304 Stainless Steel Containers at Beginning of Life.

4% Δk more in the core with Zircaloy containers than in the core with stainless steel containers, this rod worth is barely enough to control 4.2 wt % enriched core I fuel elements. In order to obtain the same shutdown margin as is present in the core with stainless steel containers, the core with Zircaloy containers would require 3.9 wt % enriched fuel elements. Since the control rods are worth more in a core with Zircaloy containers, a better shutdown criterion would be to require the same shutdown margin with the central rod stuck out of the core as there is in the core with stainless steel containers. Based on the data of Table 4 and Fig. 12, the one-stuck-rod criterion for the shutdown margin would be satisfied with an enrichment of about 3.86 wt % U^{235}.

Calculated multiplication factors for full-power conditions at the beginning of core life are shown in Fig. 13. By comparing Fig. 13 with Fig. 12, it may be seen that there is an overall temperature deficit (loss of reactivity in going from room temperature to full power) of 5.3% Δk for the core with stainless steel containers and 5.7% Δk for the core with Zircaloy containers. Also, it is apparent from Fig. 13 that a fuel enrichment of 3.38 wt % U^{235} in a core with Zircaloy containers

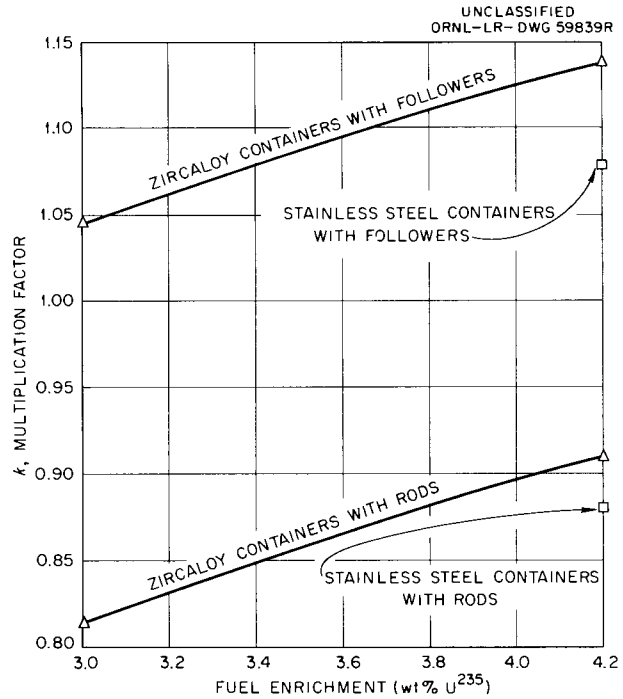


Fig. 13. Calculated Multiplication Factors for Core I Type Fuel Elements in Zircaloy and Type 304 Stainless Steel Containers at Beginning of Life for Full-Power Operation at 508°F Without Xenon.

would have the same initial hot reactivity (and therefore about the same reactivity lifetime) as 4.2 wt % enriched fuel in a core with stainless steel containers.

Power Distributions

The calculated power distributions of the cores with stainless steel and Zircaloy containers are compared in Fig. 14. Since Zircaloy is a much weaker thermal-neutron absorber than stainless steel, the power peak is higher near the water channels for the Zircaloy containers than for the stainless steel containers. The increased local power peaking caused by the Zircaloy containers is offset, however, by a gross radial power flattening. Power profiles along directions other than that shown in Fig. 14 have similar characteristics. Although the results of Fig. 14 are encouraging, the three-dimensional power peaking for partially inserted rods needs to be investigated.

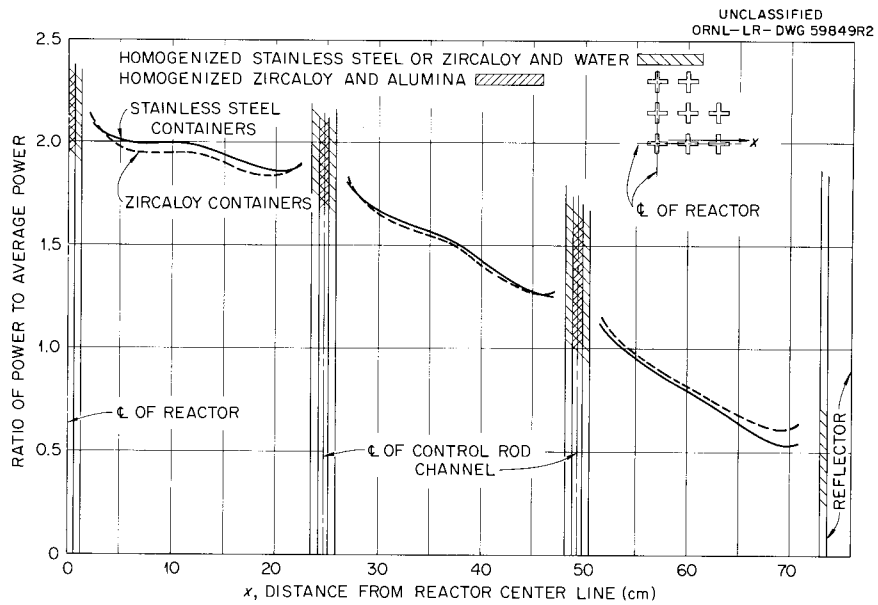


Fig. 14. Comparison of Power Distribution for 4.2 wt % Enriched Core I Fuel Elements in Zircaloy and in Type 304 Stainless Steel Containers at 508°F.

Reactivity Lifetime

Simple one-dimensional radial burnup calculations were performed using the CANDLE 2 code[19] to estimate the reactivity lifetime of a core with Zircaloy containers relative to core I. Four-group cross sections representing a core at the beginning of life were obtained from homogenization of the results for a full-sized core (geometry of Fig. 10) containing only control-rod followers. For the burnup calculations the core was controlled by a uniformly distributed poison, and the microscopic cross sections of the core materials were assumed to be constant in time.

The full-power multiplication factor as a function of time calculated with CANDLE 2 (with the above assumptions) for an initial loading of 4.2 wt % enriched fuel in both stainless steel and Zircaloy containers is shown in Fig. 15. After equilibrium xenon and samarium have built up (~0.3 year), the multiplication factor as a function of time appears to be remarkably linear, with essentially the same slope for both cores.

[19] D. J. Marlowe and P. A. Ombrellaro, "CANDLE — A One-Dimensional Few-Group Depletion Code for the IBM-704," USAEC Report WAPD-TM-53, Westinghouse Atomic Power Division, May 1957.

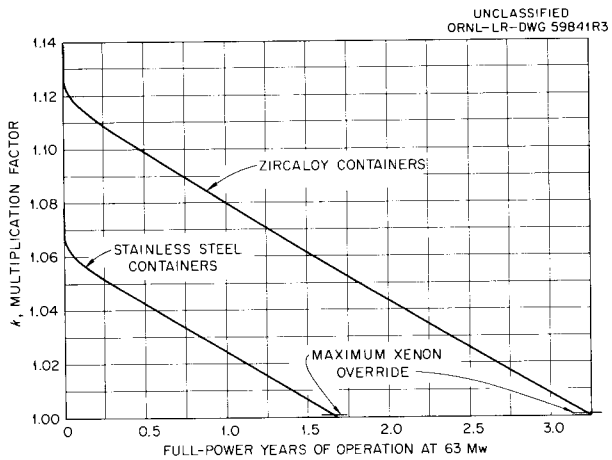


Fig. 15. Calculated Reactivity Lifetime of 4.2 wt % Enriched Core I Fuel in Zircaloy and in Type 304 Stainless Steel Containers at a Power Level of 63 Mw.

After about two years of burnup at full power, the buildup of plutonium begins to affect the burnup curve slightly.

A xenon override option in CANDLE 2 reveals a maximum xenon override of 0.1% Δk over and above the reactivity tied up in equilibrium xenon (1.2% Δk). Assuming that the end of core life occurs when maximum xenon cannot be overridden, the core with stainless steel containers would last 1.63 full-power years, while the core with Zircaloy containers would last 3.2 full-power years, or about twice as long as the stainless steel core.

As pointed out earlier, a core with Zircaloy containers would not have as large a shutdown margin as a core with stainless steel containers with the same fuel enrichment. In order to obtain the same shutdown margin with a central rod removed as is now available in core I would require about 3.86 wt % U^{235} in Zircaloy containers. Based on the data of Fig. 13, which provide an estimate of the initial hot reactivity for this enrichment, and the slope of the burnup curve of Fig. 15, such a core is estimated to have a reactivity lifetime about 0.9 full-power years longer than core I.

Burnup of U^{235} and Plutonium Production

The burnup of U^{235} and the production of plutonium are also important considerations in fuel-cycle economics. The U^{235} inventory as a

function of core life (full-power years) is shown in Fig. 16. These data apply to an initial one-zone loading of 4.2 wt % enriched fuel in either stainless steel or Zircaloy containers. The burnup of U^{235} is fairly linear, with a slope of about 25 kg per year. The buildup of Pu^{239} and Pu^{241} for both cores is shown in Figs. 17 and 18. The buildup of Pu^{239} occurs at a rate of about 8 kg per year during the first year and at the rate of about 6 kg per year during the second year of full-power operation. The buildup of Pu^{241} is quadratic, depending as it does on the buildup of intermediate isotopes of plutonium.

The estimate of the buildup of plutonium depends directly on the resonance integral employed for U^{238} and therefore contains the usual uncertainties encountered when evaluating the resonance integral of lumps

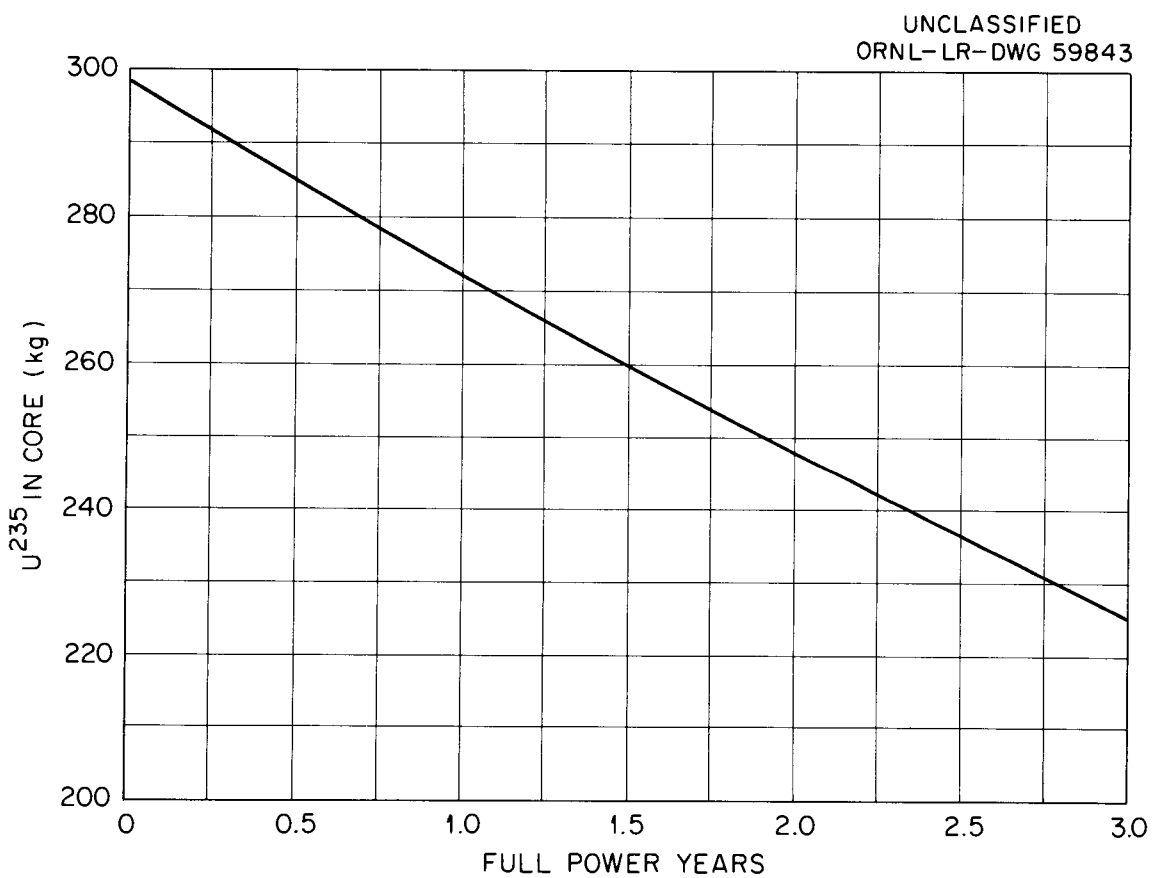


Fig. 16. Burnup of U^{235} in 4.2 wt % Enriched Core I Fuel Elements at a Power Level of 63 Mw.

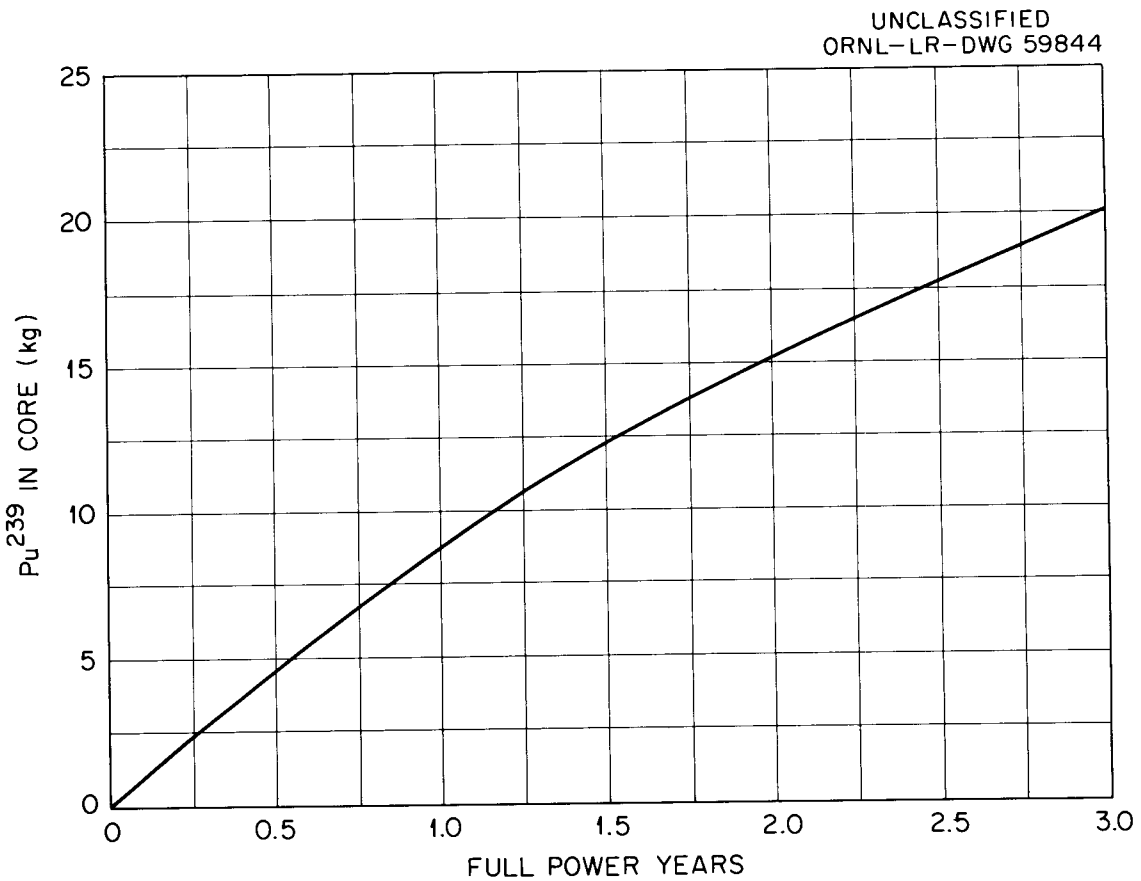


Fig. 17. Buildup of Pu^{239} in 4.2 wt % Enriched Core I Fuel Elements at a Power Level of 63 Mw.

of U^{238} with nonuniform temperature distributions in a heterogeneous lattice. The same resonance integral (20.35 barns at full power) was used in both burnup calculations, however, and therefore the production rates of plutonium in both cores are on the same basis.

Two-Fuel-Zone Loading

The calculations described above were based on cores initially containing fuel elements with a single fuel enrichment. In the actual core I, the inner 16 fuel elements initially contained 4.2 wt % U^{235}, whereas the outer 16 fuel elements contained 4.6 wt % U^{235}. This two-zone loading arrangement helped to flatten the radial power distribution and provided a method of trimming core reactivity upward, if necessary.

By repeating some of the calculations for two-zone loading, it was found that the cold reactivity of the core with stainless steel containers

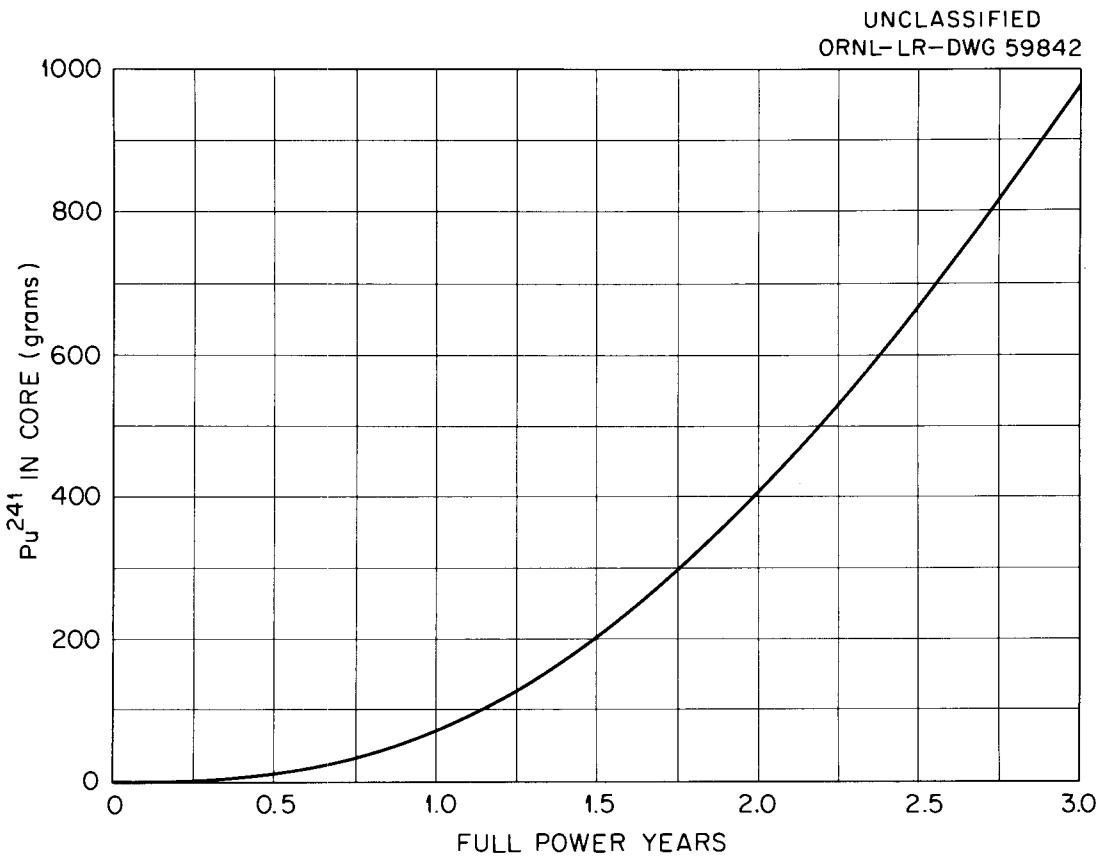


Fig. 18. Buildup of Pu^{241} in 4.2 wt % Enriched Core I Fuel Elements at a Power Level of 63 Mw.

was increased by about $\Delta k = 0.003$ when compared with one-zone loading. Thus the reactivity data for uniform fuel loading would be little affected by consideration of two-zone fuel loading. Consequently, since two-zone loading would not play an important role in evaluating container materials, evaluation on the basis of a uniform fuel loading appeared to be justified.

4. FUEL-CYCLE ECONOMICS

In order to evaluate the saving that would result from the Zircaloy substitution,[20] it was first necessary to establish a steady-state fuel-

[20] C. L. Whitmarsh, "Potential Fuel-Cycle Cost Savings Resulting from Technological Changes in the N.S. SAVANNAH Reactor," USAEC Report ORNL TM-144, Oak Ridge National Laboratory, April 6, 1962.

cycle cost for a reference core. The reference core, designated case S-420, consisted of core I-type fuel elements (one-zone enrichment of 4.2 wt %) in stainless steel fuel-element containers. This core was selected in preference to core I (two fuel zones) to maintain consistency with the physics analysis. Calculations showed, however, that the estimated fuel-cycle cost savings resulting from the Zircaloy substitution were little affected by whether the one-zone core or core I was chosen as the reference core.

Zircaloy containers can be utilized to reduce fuel-cycle costs by either increasing the core lifetime or decreasing enrichment (keeping lifetime constant). In order to examine the relative merits of these two approaches, a cost analysis of three different cores with Zircaloy containers was necessary. The results of the fuel-cycle cost analysis for these three cases are presented below. Following the presentation of results is a discussion of the effects of various cost variables and uncertainties. Finally, the basis on which each cost component was obtained is given.

Results of Cost Analysis

Economics of Extended Core Life

Results of the physics calculations (see Section 3) indicated that core lifetime would be extended from 1.63 to 3.20 full-power years if no changes were made in the core other than the substitution of Zircaloy for stainless steel. This material change, in addition to increasing excess reactivity, also increased control-rod worth; however, the increase in control-rod worth was not sufficient to give the same degree of shutdown safety as the reference core. In order to give the same shutdown margin with the central control rod out as would obtain with the reference core, the fuel enrichment to be used with the Zircaloy containers should be reduced to 3.86%. For this case, the core lifetime would be 2.52 full-power years.

The fuel-cycle cost savings are presented in Table 5. If the total excess reactivity is utilized for lifetime extension (case Z-420), a reduction of 32% in fuel-cycle cost may be realized. If the shutdown margin

Table 5. Effect of Extended Lifetime on Fuel-Cycle Costs[a]

	Cost Reduction (%) Based on Cost for Reference Case (Case S-420)[b]	
	Case Z-420[c] (lifetime, 3.20 fpy)	Case Z-386[d] (lifetime, 2.52 fpy)
Fabrication	49	38
Net burnup	<1	4
Reprocessing	49	37
Uranium use charge	7	15
Working capital	2	5
Total fuel cycle	32	26

[a]Assumed load factor of 0.67.

[b]Case S-420: Reference case, core I-type fuel elements (uniform enrichment of 4.20 wt %) in stainless steel containers.

[c]Case Z-420: Same as reference case except Zircaloy containers.

[d]Case Z-386: Core I-type fuel elements (uniform enrichment of 3.86 wt %) in Zircaloy containers; enrichment adjusted to give same shutdown margin with central rod out as reference case.

(with central rod out) of the core utilizing Zircaloy containers is adjusted to be equivalent to the shutdown margin of the reference core, the cost saving is 26%. The additional cost of Zircaloy containers compared with stainless steel containers is not included in the above analysis. A cost estimate indicated that the Zircaloy containers would cost about $130,000 more than the stainless steel containers. This differential is attributable to higher material and labor costs. When amortized over the 20-year service life, however, this additional cost for Zircaloy appears insignificant compared with annual fuel-cycle costs of $700,000 to $1,000,000.

Economics of Reduced Fuel Enrichment

As an alternative approach, the initial fuel enrichment may be reduced and still maintain the reference core lifetime. The reduced enrichment, as indicated in Section 3, was calculated to be 3.38%. As shown in Table 6, the resulting reduction in fuel-cycle cost was 10% of the reference fuel-cycle cost. As in the previous analysis, the additional cost of the Zircaloy containers was not included.

Table 6. Effect of Reduced Enrichment on Fuel-Cycle Costs[a]

	Cost Reduction (%) for Case Z-338[b] Based on Cost for Reference Case
Fabrication	9
Net burnup	2
Reprocessing	7
Uranium use charge	26
Working capital	9
Total fuel cycle	10

[a]Assumed load factor of 0.67

[b]Case Z-338: Core I-type fuel elements (uniform enrichment of 3.38 wt %) in Zircaloy containers; enrichment adjusted to give same lifetime (1.63 fpy) as reference case.

Effect of Cost Variables

The relative importance of each cost component to the total fuel-cycle cost for the reference core is presented in Table 7. It is to be noted that greater than 60% of the reference fuel-cycle cost is represented by components with costs that are inversely proportional to core lifetime, that is, fabrication and reprocessing. Thus it appears that lifetime extension represents the greatest potential of any single change

Table 7. Estimated Relative Importance of Fuel-Cycle Cost Components

	Cost (% of total fuel cycle)			
	Reference Case S-420	Zircaloy Fuel-Element Container Cases		
		Case Z-420	Case Z-386	Case Z-338
Fabrication	39	29	33	39
Net burnup	16	23	21	18
Reprocessing	23	17	20	24
Uranium use charge	15	21	17	12
Working capital	7	10	9	7

for reducing the reference fuel-cycle cost. Use charge, an enrichment-dependent cost, represents 15% of the total cost; burnup and working capital are virtually independent of any changes under consideration.

The load factor, fabrication costs, and uranium burnup represent the greatest uncertainties in fuel-cycle cost calculations. The load factors can only be estimated on the basis of past experience with similar situations. In many respects the SAVANNAH is unique; and, consequently, any estimated load factor will be primarily a guess. The effect of the load factor on cost reduction is shown in Fig. 19 to be small; for example, a reduction of the load factor from 0.67 to 0.3 reduces the Zircaloy core fuel-cycle saving from 32 to 26%. Fabrication costs will become more definitive as industry gains experience in this field. At present, data must be extrapolated from similar operations to obtain unit costs, which are then combined to obtain total core cost. These costs are sensitive to many factors, such as dimensional tolerances, chemical specifications, and hardware complexity. As also shown in Fig. 19, the effect of core procurement cost is small; for example, halving the fabrication cost reduces the Zircaloy core fuel-cycle saving from 32 to 29%.

Burnup values may be refined by improved calculational techniques but can be determined precisely only by actual reactor operation. Increased burnup will decrease the relative importance of the lifetime-dependent cost components — fabrication and reprocessing. Lifetime calculations are complex, however, and the resulting change in fuel-cycle costs cannot be evaluated quantitatively without extensive additional work.

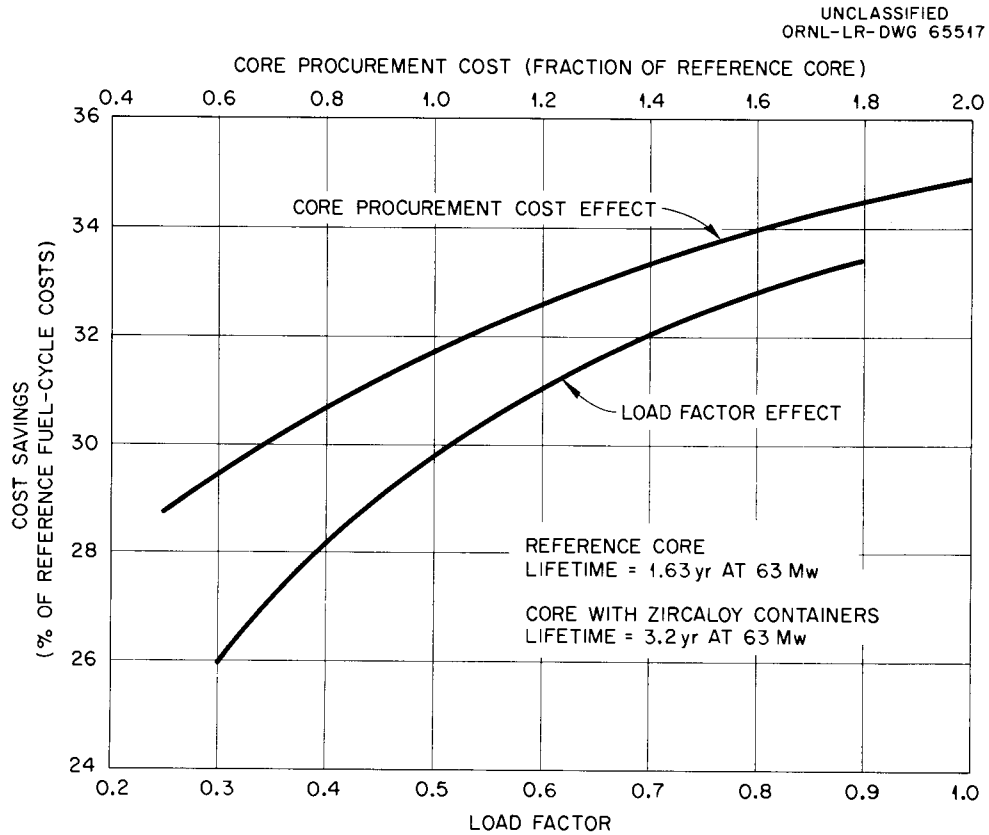


Fig. 19. Effects of Load Factor and Fabrication Cost Estimates on Fuel-Cycle Cost Savings Resulting from Use of Zircaloy Fuel-Element Containers.

Indications are that, for longer reference core lifetimes (>1.63 full-power years), the estimated relative full-cycle cost saving for the Zircaloy cases would be reduced; conversely, for shorter reference core lifetimes, the relative cost saving would be increased.

Fuel-Cycle Cost Components

In general, core data from the preliminary safeguards report[21] were used to establish an equilibrium fuel cycle. Information on reactivity lifetime, U^{235} burnup, and plutonium production was obtained, however,

[21] Babcock & Wilcox Co., "Nuclear Merchant Ship Reactor Project, Preliminary Safeguards Report," USAEC Report BAW-1117, Vol. I (Rev. 1), December 1958.

from the burnup calculation outlined in Section 3. Components considered in a typical fuel cycle are shown in Fig. 20. The ground rules and assumptions used to evaluate the economics are summarized below. All calculations are based on one core loading plus four spare fuel elements. For calculational purposes, it was considered that the spare fuel was kept on standby during reactor operation and then reprocessed with the spent core.

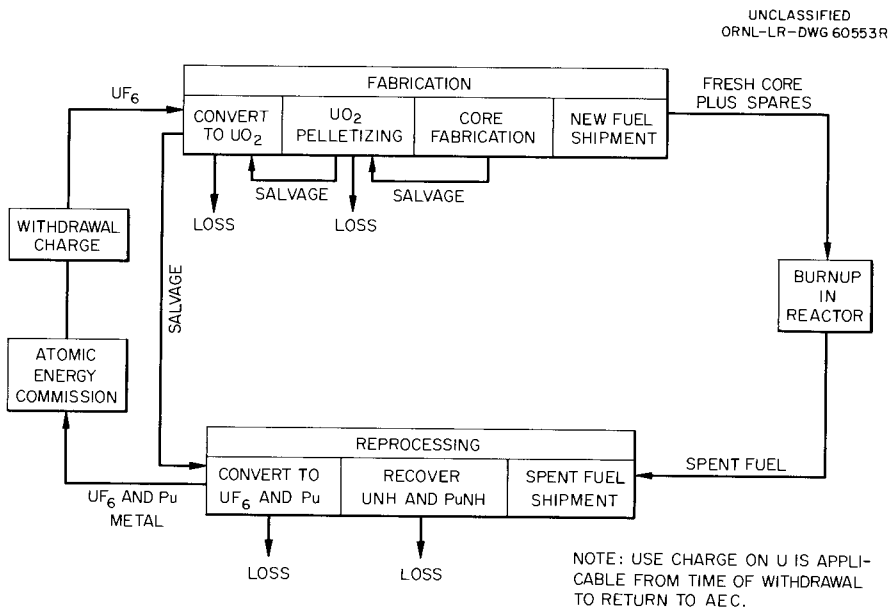


Fig. 20. Fuel Cycle Flow Chart.

Fabrication

Unit costs considered included UF_6 withdrawal from the AEC, conversion of UF_6 to UO_2, pelletizing, fabrication of core, uranium losses, recovery of salvageable losses, use charge on the uranium, and shipment to reactor site. Cost data were obtained from averaged unit costs from AEC sources, suppliers, and manufacturers,[22] since specific data on core I were not available. Core fabrication, shipment, and storage at reactor site

[22]Kaiser Engineers, "Cost Evaluation Handbook, Nuclear Power Plants, Volume 4. Nuclear Fuel Cycle Costs," USAEC Report (unnumbered) Oct. 1, 1960.

prior to use were assumed to occur during the last year of the preceding fuel cycle.

Net Burnup

The difference between the fuel value of a new core and a spent core is the net burnup. This accounts for uranium depletion and plutonium buildup. The AEC price schedule for enriched uranium[23] was used in conjunction with burnup calculations presented in Section 3 to determine the burnup cost. Plutonium was credited at $9.50 per gram.

Reprocessing

The reprocessing cost includes shipment of the spent core to the reprocessing plant, recovery of fuel as uranium and plutonium nitrates, conversion of recovered material to UF_6 and plutonium, processing losses, and use charge on uranium during this time. A cost of $16 per kilogram of uranium was assumed[22] for fuel shipment, since neither the distance involved, insurance rates, or shipping rates and regulations have yet been defined. The cost per day for reprocessing plant operation was determined from an AEC-established formula[22] based on plant depreciation, operational costs, waste storage, and overhead. This cost is subject to escalation[24] and, as of August 1960, was $16,988 per day. The costs of conversion to UF_6 and plutonium for return to the AEC were determined from the AEC-established prices of $5.60 per kilogram of uranium at $1.50 per gram of plutonium. Processing losses of 1% uranium and 1% plutonium during reprocessing and 0.3% uranium and 1% plutonium during conversion were regarded as acceptable for these operations.[22] A period of 120 days was allowed for cooling and shipment prior to reprocessing.

Use Charge

The AEC has established a lease charge on all enriched uranium at the rate of 4.75% per year. During reactor operation, in lieu of other

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[23] Federal Register, 26: 103 (May 30, 1961).

[24] "Costs of Nuclear Power," USAEC Report TID-8531, January 1961.

data, the average inventory value was considered as the arithmetical average of the initial and final values.

Working Capital

Based on conventional power plant operation, the cost of working capital was estimated at 12.7% per year.[22] This includes taxes, insurance, interest on borrowed money, and profit. That portion of the working capital attributable to the fuel cycle was considered to be the average inventory plus 3% of the annual fuel-cycle cost. The average inventory, in lieu of specific data, was estimated as 60% of the cost of a new core.

5. COMPATIBILITY OF ZIRCALOY CONTAINERS WITH REACTOR SYSTEM

Zircaloy-2 and Zircaloy-4 were evaluated for long-term (20-year) service in the core on the basis of a literature survey of basic properties of the materials,[25] an evaluation of the reactor water chemistry,[26] and operating experience with other pressurized-water reactors. The compatibility problems considered in this evaluation were: corrosion, wear, fretting corrosion, hydrogen embrittlement, and the accumulation of long-lived activity in the primary system.

Corrosion

Corrosion rates of Zircaloy-2 applicable to exposure conditions in the core are listed in Table 8. Since the corrosion film on Zircaloy-2 is adherent, specimen weight gain was considered indicative of the corrosion rate. Since the core inlet and outlet water temperatures are 495 and 520°F, respectively, the corrosion calculations were based on a bulk water temperature of 550°F for conservative design purposes. Experimental data indicate that exposure to fast neutrons will increase corrosion, but

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[25] C. L. Whitmarsh, "Review of Zircaloy-2 and Zircaloy-4 Properties Relevant to N.S. SAVANNAH Reactor Design," USAEC Report ORNL-3281, Oak Ridge National Laboratory, July 9, 1962.

[26] C. L. Whitmarsh, "Water Chemistry in the N.S. SAVANNAH Reactor — Effect of a Core Component Change from 304 Stainless Steel to Zircaloy-4," USAEC Report ORNL CF-61-1-109, Oak Ridge National Laboratory, Jan. 20, 1961.

Table 8. Corrosion Rate of Zircaloy-2 in Water at 550°F

Source	Weight Gain[a] (mg/dm^2 per day)	Comments
Lustman and Kerze (see ref. 27)	$0.95\ t^{-0.70}$ (t in days)	Pretransition rate; tests of up to 1000 days did not show transition
Riedinger (see ref. 28)	<0.02	Tests at 600°F showed a weight gain of 0.02 mg/dm^2 per day; lower temperature tests not reported
Perryman (see ref. 29)	<0.03	Tests at 572°F showed a weight gain of 0.03 mg/dm^2 per day; lower temperature tests not reported
Whitmarsh (see ref. 25)	0.015	Extrapolation of averaged data from several sources correlated in a 1/T plot

[a]Rates were determined from tests in static, neutral water.

the magnitude of the effect is uncertain; increases up to a factor of 5 have been obtained, with 2 or 3 being more probable values.

Predictions of long-term corrosion show considerable spread of the data when the rates from Table 8 are plotted as in Fig. 21; however, the highest rate indicated would result in a weight gain of only 220 mg/dm^2 in 20 years, or a metal penetration of 0.4 mils on each exposed surface. If a factor of 5 were applied to this to account for radiation effects, the maximum total corrosion would be 1200 mg/dm^2 or 2 mils on each exposed surface. These figures indicate satisfactory corrosion resistance.

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[27]B. Lustman and F. Kerze, The Metallurgy of Zirconium, p. 622, McGraw-Hill Book Company, New York, 1955.

[28]A. B. Riedinger, Corrosion Behavior of Zircaloy, Thesis, Union College, May 1958.

[29]E. C. W. Perryman, "A Review of Zircaloy-2 and Zircaloy-4 Properties Relating to the Design Stress of CANDU Pressure Tubes," Canadian Report CRMet-937, June 1960.

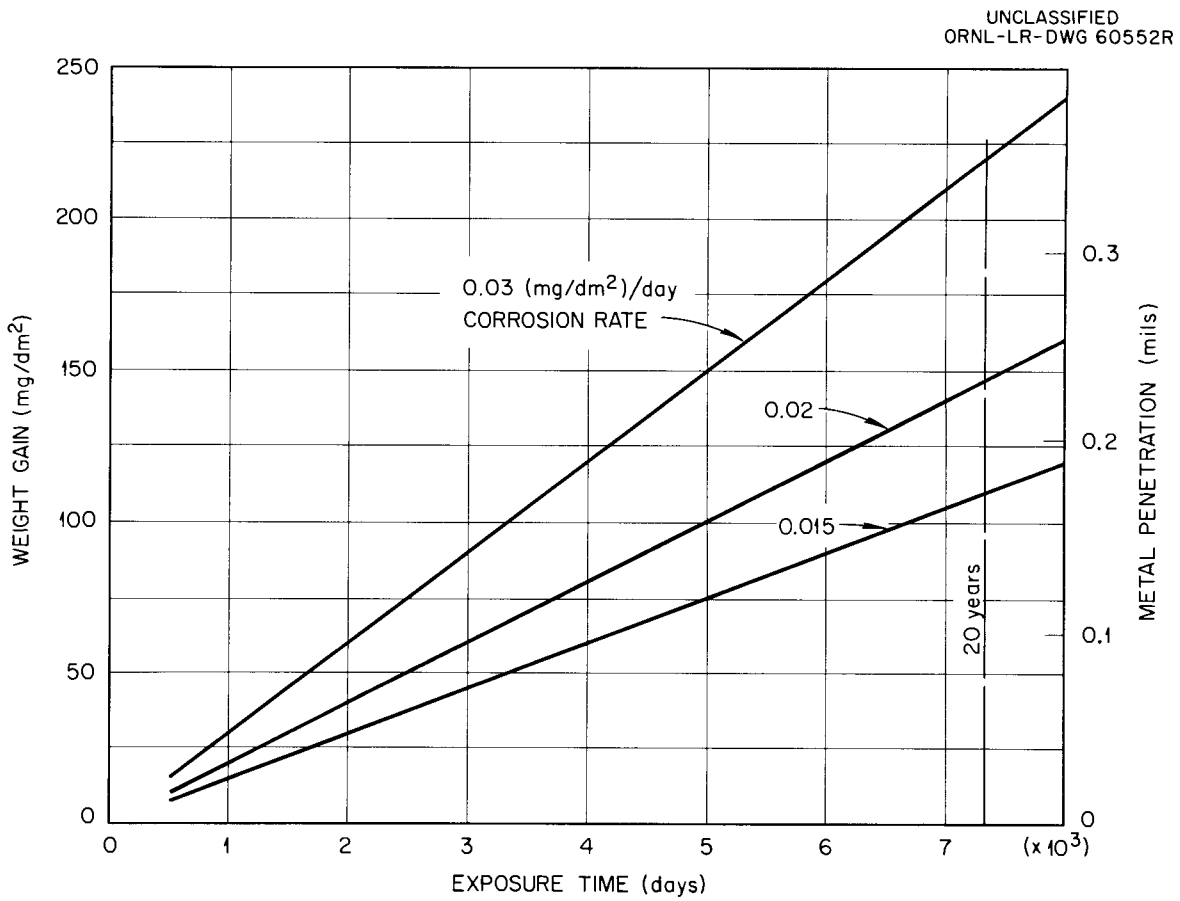

Fig. 21. Predicted Long-Term Corrosion Behavior of Zircaloy-2 in Water at 550°F.

The predictions of corrosion behavior indicated above are based on the absence of external effects such as fretting corrosion and wear. If the protective oxide film were mechanically removed or broken as a result of fretting or wear, it would be replenished each time by reaction between the water and the exposed surface of the Zircaloy, thereby causing an accelerated corrosion rate.

Wear

Since the control-rod assembly is inserted in channels defined by the container walls, a potential wear problem exists between the control rod and container walls. Wear of the control rods is considered less

serious than wear of the containers, since periodic replacement of the control rods is necessary for reasons not associated with wear. In the existing system, sliding contact occurs between the following materials: Zircaloy (rod follower) and stainless steel, stainless steel and stainless steel, and stainless steel and chromium-plated stainless steel. With Zircaloy containers and the present control rods, the wear pairs would be Zircaloy-Zircaloy, stainless steel-Zircaloy, and chromium-plated stainless steel-Zircaloy.

Ideally, an evaluation of wear should be divided into two problems. First, the conditions of service (bearing loads, rate of relative motion, etc.) would be defined; and, second, the wear properties of the particular set of materials would be used to assess the suitability of the materials for the application. In this particular case, however, the service conditions are not easily defined because of the uncertainty regarding the influence of ship motion and because of the nonuniformity of mating of the wear surfaces. In addition, little information is available on the wear characteristics of Zircaloy against any other materials. In view of these difficulties, the evaluation of wear in the present problem must be of a rather qualitative nature, but a review of the information obtained from tests by Babcock & Wilcox[30] on a mockup of the existing core components is of interest.

A prototype control-rod drive and control-rod assembly were tested in a high-temperature, high-pressure test vessel that contained a mockup of a control-rod channel. Stainless steel fuel-element container angles were assembled to form the control-rod channel. The prototype control-rod assembly, which included a Zircaloy-2 follower section, was tested for a period corresponding to about one-half of the design life of the drive mechanism. Part of the testing was performed with the control rod and control-rod slot misaligned the maximum amount expected in the reactor. In addition, some of the testing was done with the whole test setup inclined at 30° from vertical (simulating maximum ship roll).

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[30]D. E. Thoren and D. K. Gestson, "NMSR Control Rod Driveline Tests," USAEC Report BAW-1176, Babcock & Wilcox Co., November 1959.

After completion of the tests the various parts were examined for wear and corrosion. The control-rod assembly showed signs of wear in the form of polishing of the stainless steel and very light scratching on the Zircaloy. None of the wear was deep enough to measure by conventional methods. Wear on the stainless steel containers was primarily on either end and was characterized by polished areas with very fine scratches. As in the case of the control rods, the wear on the stainless steel containers was not enough to measure. No corrosion was noted on any of the parts.

Of course, the information given above is not directly applicable to the case of Zircaloy fuel-element containers. Nevertheless, the results indicate very light service conditions with respect to wear. The wear-pairs not covered in the mockup test which are of interest when considering Zircaloy containers are Zircaloy-Zircaloy and Zircaloy chromium-plated stainless steel. One test is described in the literature[31] that gives an indication of the relative wear properties of Zircaloy-Zircaloy as compared with those of Zircaloy-stainless steel. In the test a rotating disk of Zircaloy-2 contacted a stationary weighted rod of either Zircaloy-2 or type 304L stainless steel to produce a wiping action; the test was run in deionized water pressurized to 1500 psia at temperatures ranging from 100 to 400°C. Wear rates of both the rod and disk were measured as a function of rod material, temperature, contact pressure, autoclave pretreatment of the Zircaloy, and rotational velocity. For the purposes of the present study, the significant result of the test was that the disk wear was independent of whether Zircaloy-2 or stainless steel contact rods were used. Based on this result, it appears that the contact points between the Zircaloy containers and the Zircaloy followers will exhibit about the same wear characteristics as the present stainless steel containers and Zircaloy followers. There appears to be no information which can be used to evaluate the wear properties of Zircaloy against chromium-plated stainless steel. The value of the chromium plate on the control-

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[31] R. J. Lobsinger, "Fretting Corrosion of Zircaloy-2 — Progress Report," USAEC Report HW-66969, Hanford Atomic Products Operation, Sept. 30, 1960.

rod blades is questionable, however, even with the stainless steel containers. Without the use of the chromium plate, the stainless steel-to-Zircaloy surface contacts should perform satisfactorily, as demonstrated in the prototype control-rod-drive tests.

In summary, no major wear problem is expected as a result of using Zircaloy fuel-element containers. This conclusion is, however, based on rather limited data.

Fretting Corrosion

Fretting corrosion may be roughly defined as a surface damage that occurs when two solid surfaces are in contact under moderate to high loads and with small relative motion. Since Zircaloy derives it corrosion resistance from an adherent oxide film it would be expected that Zircaloy would be somewhat susceptible to fretting corrosion. A few cases of fretting corrosion of Zircaloy components have been observed in practice.[32-34] It should be noted, however, that no serious fretting corrosion problem has arisen in the various reactors that use Zircaloy components.

Although a number of places exist in the fuel-element container assembly where fretting corrosion could occur, it appears that the most likely areas of concern are the contacts between (1) the fuel bundle and container wall and (2) the control-rod assembly and container wall. The material pair of interest at the fuel-bundle to container-wall contact is Zircaloy—type 304 stainless steel. Material pairs associated with the control-rod to container-wall contacts are the same as those described in the discussion of wear.

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[32] F. H. Krenz et al., "Some Preliminary Measurements with an Apparatus for Studying the Fretting Corrosion of Zircaloy-2," Canadian Report CRMet-857, January 1960.

[33] R. T. Pennington, "Nuclear Superheat Project Seventh Quarterly Progress Report, January—March 1961," USAEC Report GEAP-3724, General Electric Atomic Products Division.

[34] R. G. Gray and F. P. Mrazik, "Examination of an In-Pile Failure of a PWR Core I Type Fuel Rod," p. 53, USAEC Report WAPD-BT-23, Westinghouse Atomic Power Division, August 1961.

The evaluation of fretting corrosion is difficult because of the lack of understanding of the mechanisms and important parameters involved. As a rule, the only method of determining whether a fretting corrosion problem exists in a particular system is to test the actual system or a mockup of the system. Nevertheless, the few fretting corrosion tests on Zircaloy that have been performed[32,35-38] are, as discussed below, of value in making a judgment as to whether mockup tests are justified.

Bettis Tests

Fretting corrosion studies[35] of Zircaloy-2 have been performed at the Bettis Atomic Power Laboratory. The tests were performed in room temperature water, and the fretting couples used in the investigation were Zircaloy-2 against Zircaloy-2 and Zircaloy-2 against type 304 stainless steel. Two of the significant results were:

1. Fretting weight losses between Zircaloy surfaces were less than those between mild steel surfaces. This result is of little consolation, however, since it is known that mild steel is rather poor with respect to fretting corrosion.

2. Fretting weight losses of Zircaloy against stainless steel were greater by a factor of 10 than for Zircaloy-Zircaloy couples.

Chalk River Tests

Fretting corrosion tests of Zircaloy-2 against Zircaloy-2 were conducted in an autoclave containing degassed, deionized water at 280°C.[32] The bearing pressures between the Zircaloy surfaces ranged from about 500 to 4000 psi. The amplitude (or slip) was 0.005 in., with a frequency of

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[35]W. B. McMullen, "Fretting Corrosion of Zircaloy," p. 75, USAEC Report WAPD-BT-15, Westinghouse Atomic Power Division, September 1959.

[36]R. J. Lobsinger, "Evaluation of Fretting Corrosion of Zircaloy-2," USAEC Report HW-61915, Hanford Atomic Products Operation, Feb. 5, 1960.

[37]R. J. Lobsinger, "Fretting Corrosion of Zircaloy-2, Progress Report," USAEC Report HW-66969, Hanford Atomic Products Operation, Sept. 30, 1960.

[38]A. P. Larrick, "Fretting Corrosion of PRTR Fuel Elements and Process Tubes," USAEC Report HW-68613, Hanford Atomic Products Operation, March 14, 1961.

1600 cycles per minute. The tests indicated that Zircaloy-2 subjected to fretting corrosion conditions is rapidly penetrated under bearing loads of 500 to 4000 psi. The penetration rate was about 1 mil per each 24-hour period.

Hanford Tests

Several investigations of fretting corrosion of Zircaloy have been made at Hanford.[36-38] In general, the results have indicated that fretting corrosion of Zircaloy can result in very rapid penetration. One test[37] in which Zircaloy-2 wire was allowed to vibrate against both Zircaloy-2 and type 304 stainless steel indicated that the fretting corrosion penetration of Zircaloy was greater when contacting stainless steel than when contacting Zircaloy. This test was conducted in water at 300°C, and the result is consistent with the Bettis room-temperature results described above.

A phenomenon of particular interest is the "impact corrosion" of Zircaloy which was observed at Hanford. Two tests were run to determine the effects of multiple impacts on Zircaloy-2.[37] These tests employed a solenoid-actuated plunger tipped with Zircaloy-2 which beat against a rigid plane surface of a Zircaloy-2 anvil. Wear was determined by measuring the plunger lengths before and after the test. Both tests were of 3 weeks duration with an impact rate of 550 impacts per minute and an impact energy of about 2 in.-lb. In deionized water at 300°C, the plunger wear was 2.5 mils. In water at 400°C, the plunger wear was 7 mils.

Additional testing on a mockup of a fuel element and process tube for the Plutonium Recycle Test Reactor (PRTR) confirmed the effect of impact on corrosion.[38] A 19-rod cluster was suspended from the top in a vertical Zircaloy-2 process tube. Six Zircaloy-2 spacer lugs were attached near the bottom of the fuel element. These lugs could contact the process tube. The assembly was tested in a high-pressure, high-temperature (300°C) loop with the flow rate adjusted to that of the PRTR. After two weeks of exposure, the process tube showed penetrations up to 5 mils at each of the six points corresponding to the six support lugs on the fuel element.

Conclusions

The test results on fretting corrosion of Zircaloy-2 seem to indicate that rapid penetration of Zircaloy surfaces can occur under certain conditions. In particular, the susceptibility of Zircaloy to "impact corrosion" has important implications with respect to the contact areas between fuel elements and container walls. Because of flow-induced vibration, machinery-induced vibration, and gross motion of the ship, it is expected that repeated impacts between the fuel bundles and containers could occur. To resolve this problem, testing of a full-size fuel element and fuel-element container would be desirable. This test should be run in a loop with water conditions (flow, temperature, pH, etc.) corresponding to those in the reactor. Fuel-element movements arising from ship motion are also important. Although this motion could be simulated by actually moving the loop, it would appear that a less-complicated method could be devised. For example, an electro-magnet could be used to induce movement of the fuel element. This method has been used to test small-scale fuel elements at Hanford.[37]

Hydrogen Embrittlement

Hydrogen absorption is the limiting service condition for Zircaloy in some reactor applications. Long-term accumulation of hydrogen in Zircaloy plates exposed to the core conditions in the N.S. SAVANNAH reactor was calculated, and the results are presented in Table 9 and Fig. 22. The assumptions and ground rules on which the calculations were based were the following:

1. The average temperature of the Zircaloy would be in the range 500 to 550°F.

2. The initial hydrogen concentration in the Zircaloy plate would be 25 ppm.

3. All the Zircaloy specimens would be corrosion tested prior to use, with resulting corrosion weight gains of 10 mg/dm^2.

4. The hydrogen pickup of Zircaloy-2 would be 30% of theoretical during corrosion testing and 100% during reactor service.[29]

Table 9. Predicted Hydrogen Pickup of Zircaloy Plates in Water

Average Zircaloy plate temperature: 550°F
Average water temperature: 508°F
Average corrosion rate: 0.03 mg/dm^2 per day on each exposed surface
Water pH: 6.5 to 7.5
Hydrogen content of water: 3.6 ppm
Initial hydrogen in Zircaloy plate: 25 ppm

Zircaloy Plate Thickness		Zircaloy Plate Surface Density (mg/dm^2)	Hydrogen Added to Zircaloy During Corrosion Testing (ppm)	Total Initial Hydrogen in Zircaloy (ppm)	Estimated Zircaloy Hydrogen Pickup in Service (ppm)			Total Zircaloy Hydrogen Content (ppm)		
In Mils	In Centimeters				5 Years	10 Years	20 Years	5 Years	10 Years	20 Years
				Zircaloy-2						
		$\times 10^6$								
100	0.254	0.167	2	27	82	164	328	109	191	355
120	0.305	0.20	2	27	69	137	274	96	164	301
140	0.356	0.234	2	27	59	117	234	86	144	261
160	0.406	0.267	1	26	51	103	205	77	129	231
180	0.458	0.301	1	26	46	91	182	72	117	208
				Zircaloy-4						
100	0.254	0.167	1	26	27	55	109	53	81	135
120	0.305	0.20	1	26	23	46	91	49	72	117
140	0.356	0.234	1	26	20	39	78	46	65	104
160	0.406	0.267		25	17	34	68	42	59	93
180	0.458	0.301		25	15	30	61	40	55	86
100			1[a]	26	14	27	55	40	53	81
120			1	26	11	23	46	37	49	72
140			1	26	10	19	39	36	45	65

[a]Corrosion rate of 0.015 mg/dm^2 per day.

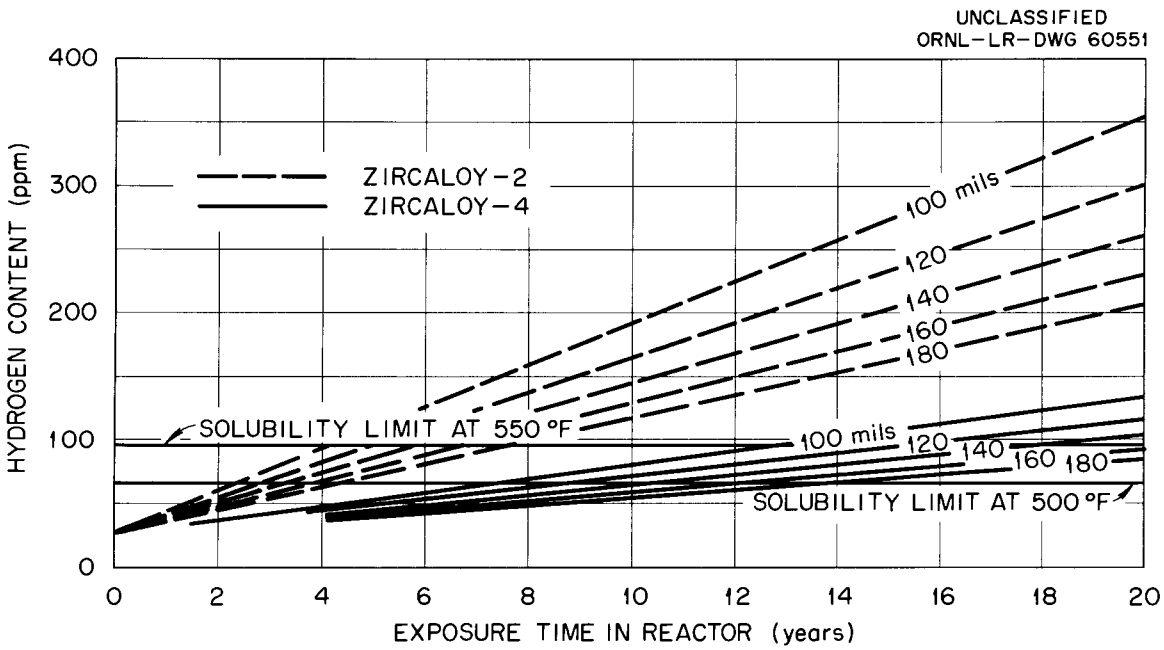

Fig. 22. Predicted Hydrogen Pickup of Zircaloy Plates in the Reactor Core.

5. The hydrogen pickup of Zircaloy-4 would be one-third that of Zircaloy-2.[29]

6. The average corrosion rate would be 0.03 mg/dm^2 per day during reactor service and would occur on both surfaces of the plates.

7. The effect of dissolved hydrogen in the water would be negligible.

8. The redistribution effects of thermal and stress gradients would be negligible.

The data of Fig. 22 indicate that the solubility limit for hydrogen in Zircaloy at 550°F would be exceeded in 20 years for Zircaloy-4 plates less than 160 mils thick. Thus, if the limiting concentration is the solubility limit, only plates of Zircaloy-4 that were greater than 160 mils thick would be acceptable for 20-year service. For both Zircaloy-2 and Zircaloy-4 at 500°F none of the thicknesses considered would be adequate; however, it should be noted that these calculations were conservative from the standpoint of corrosion rate, maximum initial hydrogen content, and hydrogen absorption rate. If only one of these factors, for example, corrosion rate, were assumed to have a more realistic value (0.015 mg/dm^2 per day), the Zircaloy-4 plate thickness could be reduced

to 100 mils and still be well below the solubility limit at 550°F for 20-year service, and plates greater than 140 mils thick would be below the 500°F solubility limit. It is doubtful that the use of Zircaloy-2 could be justified with respect to the solubility limit acceptance criterion within the range of conditions and plate thicknesses studied.

When evaluating the above information, the substantial degree of uncertainty involved must be considered. Investigators still do not have a fundamental understanding of the mechanism of hydrogen pickup; and, in addition, available data are by necessity obtained from relatively short-term tests. Long-term experience in the operation of pressurized-water reactors is, of course, nonexistent. Another problem is that of establishing a maximum permissible level for hydrogen concentration. The solubility limit is probably a lower limit for an acceptance criterion, since effects of hydrogen on mechanical properties of Zircaloy are apparently not prohibitive at concentrations up to 500 ppm. Based on the available information, it seems likely that, relative to hydrogen absorption, Zircaloy-4 fuel-element containers would be suitable for long-term service in the N.S. SAVANNAH reactor.

Long-Lived Activity

The primary coolant purification system is designed to remove crud by ion exchange and filtration in a resin bed. Action of the resin bed maintains the pH of the water between 6.5 and 8.5. Other primary system water specifications are: dissolved hydrogen, 1.8 to 3.6 ppm; hydrazine, 20 to 25 ppm at startup and zero during operation; dissolved oxygen, ≤ 10 ppb; total solids, ≤ 1 ppm; and chlorine, ≤ 0.1 ppm, with up to 1 ppm allowed for periods of less than one day.

In any pressurized-water reactor, an accumulation of corrosion products (crud) occurs in the primary system. Deposition of this crud in low-velocity regions creates problems of low heat transfer and high radiation levels. The crud problem is dependent on the formation of corrosion products, transport of these corrosion products, and activation of certain isotopes. Experience with existing reactors has shown that most of the long-lived activity in crud can be attributed to Co^{58} and Co^{60},

which are produced by the reactions $Ni^{58}(n,p)Co^{58}$ and $Co^{59}(n,\gamma)Co^{60}$, respectively.

Radiochemical analyses of crud samples from two pressurized-water reactors are listed in Table 10. Even though the PWR contains Zircaloy-clad fuel elements, greater than 56% of its crud activity is from cobalt

Table 10. Relative Crud Activities in Pressurized-Water Reactors

Nuclide	Long-Lived Activity (% of total after 1440 EFPH)	
	SM-1	PWR
Co^{60}	10.8	39.2
Co^{58}	58.0	16.9
Cr^{51}	13.5	14.8
Zr^{95}		6.5
Fe^{59}	13.8	12.1
$Hf^{181(175)}$		5.1
Mn^{54}	3.9	5.4
Total	100.0	100.0

isotopes. Compared with stainless steel, Zircaloy is not a significant source of Co^{58} or Co^{60}; reactor-grade type 304 stainless steel contains 8 to 12% Ni and 50 to 100 ppm Co. Chemical analyses of crud samples showed the major component (80 to 95%) to be iron as Fe_3O_4. On the basis of this information, it appears that increasing the amount of Zircaloy exposed to the primary coolant (with consequent reduction of stainless steel) would tend to decrease the primary system activity resulting from crud; however, since the fuel-element containers do not constitute the major portion of the primary system surface area, changing this component to Zircaloy is not likely to have a pronounced effect on crud activity.

6. STRUCTURAL ANALYSIS

The fuel-element containers perform three distinct functions; they hold the fuel elements in a fixed geometry, separate the reactor vessel flow passes, and define channels in which the control rods operate. Various forces operate to displace the containers from their nominal locations. These forces arise mainly from ship motion, pressure differences on opposite sides of a container wall, and temperature differences within a part and between parts of the assembly. Excessive deflections of the fuel-element container walls could impair the operation of control rods or reduce the coolant flow area associated with the outer fuel tubes of a fuel element. It is clear that the thickness of the container walls must be made such that the deflections will be within tolerable limits. Analyses were made previously of the present stainless steel container assembly both with core I fuel elements[39,40] and core II fuel elements[41] installed.

The purpose of the analysis[42] summarized here was to determine the Zircaloy fuel-element container design modifications required to maintain structural integrity and limit deflections of the container walls so as not to interfere with control-rod motion or coolant flow around the fuel elements. Since the fuel-element container assembly is a permanent part of the reactor, consideration was given to container requirements that might be imposed by future reactor fuel elements. It was, of course, impossible to guess the form of future fuel elements; however, it appeared that the two types of fuel elements already in existence (core I and

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[39]T. A. Hughes and D. L. Mayer, "The Stress and Delection Characteristics of N.S. SAVANNAH Fuel Element Containers Due to Internal-to-External Pressure Differentials," Trans. Am. Nuclear Soc., 4(1): 15 (1961).

[40]Ebasco Services, Inc., Structural Analysis of Fuel Container Assembly - First Core, ESI-3 (a)(b)-1 (April 1960).

[41]Ebasco Services, Inc., Structural Analysis of Fuel Container Assembly - Second Core, ESI-3 (a)(b)-2 (May 1960).

[42]L. R. Shobe, "Structural Analysis of Zircaloy Fuel-Element Containers for the N.S. SAVANNAH Reactor," USAEC Report ORNL-TM-483, Oak Ridge National Laboratory, Aug. 19, 1963.

core II) were sufficiently different to be representative of possible future designs. Core I fuel elements can be characterized by a relatively high pressure drop and high rigidity. Core II fuel elements have a relatively small pressure drop and are very flexible. Core I elements would be expected to give the largest pressure forces on the containers. The weight loads on the containers imposed by the two types of fuel elements would be distributed differently because of the difference in flexibility. Thus the containers should be designed to withstand the distributed pressure load from core I elements and, at the same time, be suitable for the weight loading imposed by either core I or core II elements. The structural analysis was therefore performed using core I pressure-drop characteristics combined with both core I and core II weight loadings. In addition, the pressure loads were increased 30% over those used in the design of the stainless steel fuel-element containers so that, if desirable, it would be possible at some future date to take advantage of the maximum flow capability of the primary system. This can be done by plugging the bypass holes in the upper grid plate which are now used to divert some of the flow around the core.[43]

The following sections summarize the structural investigation. The conditions under which the fuel-element containers must operate are described, and the design criteria used in the analysis are given. Finally, an outline of the method of analysis and the results are presented.

Operating Conditions

Flow and Pressure Conditions

The main paths taken by the coolant in going from reactor inlet to exit are shown schematically in Fig. 23. Briefly, the coolant enters through two nozzles at the bottom of the reactor vessel and flows through the annular space between the inner thermal shield and the vessel wall to the upper grid plate assembly; this is called the first coolant pass.

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[43]Babcock & Wilcox Company, letter from R. F. Ryan to C. R. McFarland, "Nuclear Merchant Ship Reactor Core Flow - Pressure Drop Analysis," Dec. 8, 1959.

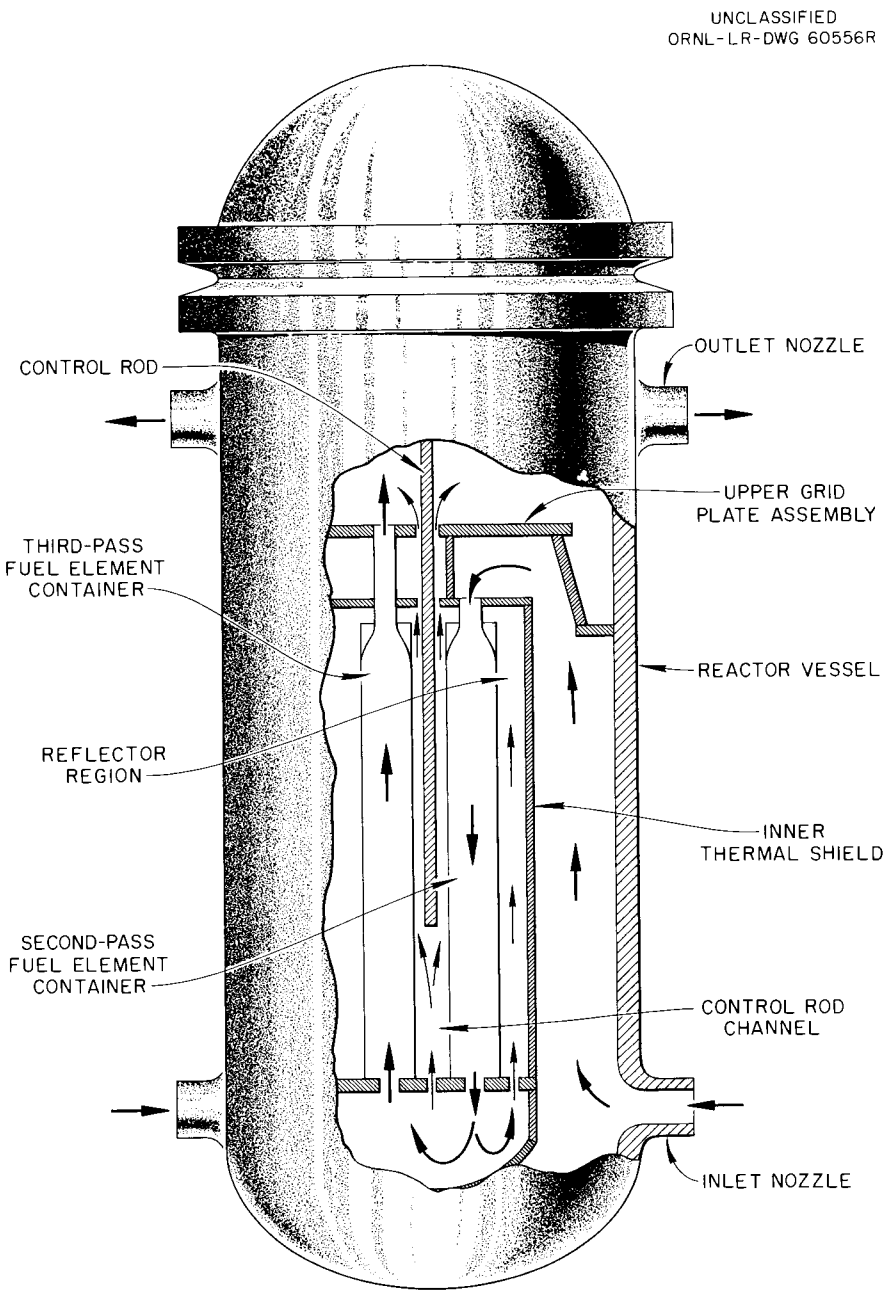

Fig. 23. Schematic Diagram of Reactor Vessel Showing Main Flow Paths.

The upper grid plate serves, among other things, to separate the second and third coolant passes. From the upper grid plate, the coolant flows downward through the outer 16 (second pass) fuel elements to the lower plenum chamber. From the lower plenum, the coolant takes three parallel paths. One path is through the inner 16 (third pass) fuel elements. The other two paths are through the control-rod channels and the reflector

region between the outer container walls and the inner thermal shield. Finally, the coolant goes to the upper plenum and then exits through the two reactor vessel outlet nozzles.

The diagram of Fig. 24 shows in more detail the flow paths through the reactor, including the leakage flow. The bypass flow path established for the purpose of reducing the pressure differential on the existing stainless steel fuel-element containers is included.

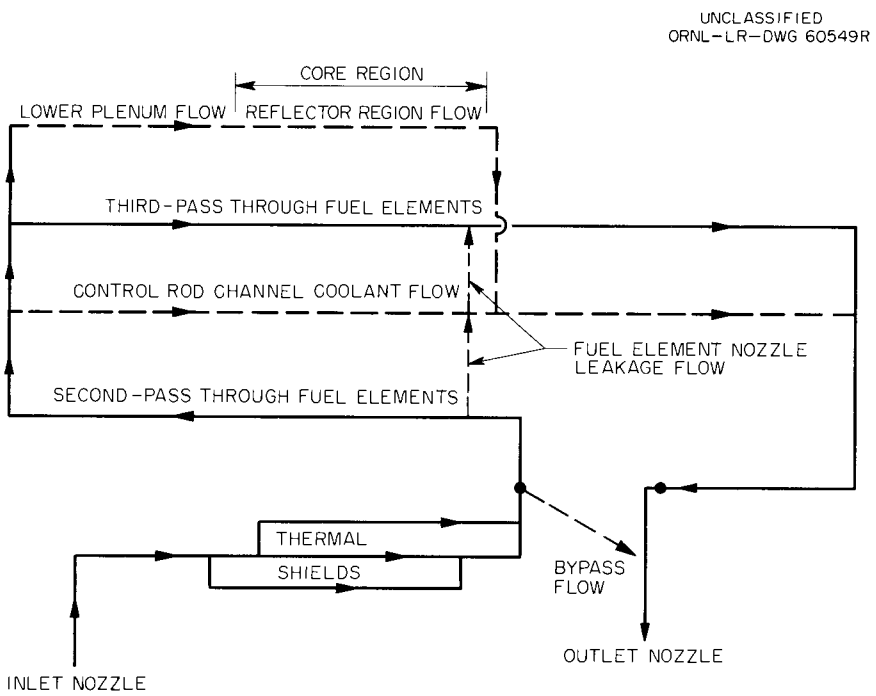

Fig. 24. Diagram of Reactor Flow Paths.

Coolant flow through the reactor gives rise to pressure differentials between the inside of the fuel containers and control-rod channels. Pressure and flow information for the reactor has been reported[44,45]

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[44]Don M. Bylund, "Nuclear Merchant Ship Reactor Project, Survey of the Reactor and Control Drive System Research and Development Program," USAEC Report BAW-1153-1, Babcock & Wilcox Co., August 1959.

[45]Babcock & Wilcox Company, letter from R. L. Whitelaw to C. R. McFarland, "Nuclear Merchant Ship Reactor Primary Loop Pressure Gradient," April 30, 1959.

that was obtained from tests on a scale model of the reactor vessel and tests on a full-sized element. This pressure drop information, which was given for the original design flow rate of 8 million pounds per hour, was scaled up by the square of the flow rate to correspond to a reactor flow rate of 9.2 million pounds per hour. The latter figure is the maximum flow rate with no bypass holes in the upper grid plate. The adjusted pressure differential curves are shown in Fig. 25 both for two-pump and four-pump operation.

Ship Motion

The various ship motions that were considered for normal operation of the N.S. SAVANNAH were the following:[46]

1. 30° roll with a 14-sec period,
2. 7° pitch with a 7-sec period,
3. fore and aft acceleration of 0.25 g,
4. lateral heave of 0.7 g,
5. vertical heave of 0.3 g.

Any or all of these motions were considered to occur simultaneously; thus, in combining stresses or deflections, no credit was taken for forces acting in opposite directions. In addition to the normal conditions outlined above, the unusual circumstance of the ship lying on its side with a vertical heave of 2 g was considered. In this extreme situation, the fuel-element containers should not prevent control-rod insertion.

Thermal Conditions

Radial power variations and the multipass arrangement of the reactor core cause nonuniformity of temperature in the container walls and spacer bars. This variation of temperature within a given part and between parts of the container assembly can cause temperature-induced stresses and deflections, which must be added to those caused by other forces. A thermal analysis was performed to obtain the temperature distribution in the container walls and spacer bars.[47]

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[46]USAEC Contract AT(04-3)-189, Project Agreement No. 2.

[47]T. D. Anderson, "A Thermal Analysis of Zircaloy Fuel-Element Containers for the N.S. SAVANNAH Reactor," USAEC Report ORNL-TM-197, Oak Ridge National Laboratory, Sept. 21, 1962.

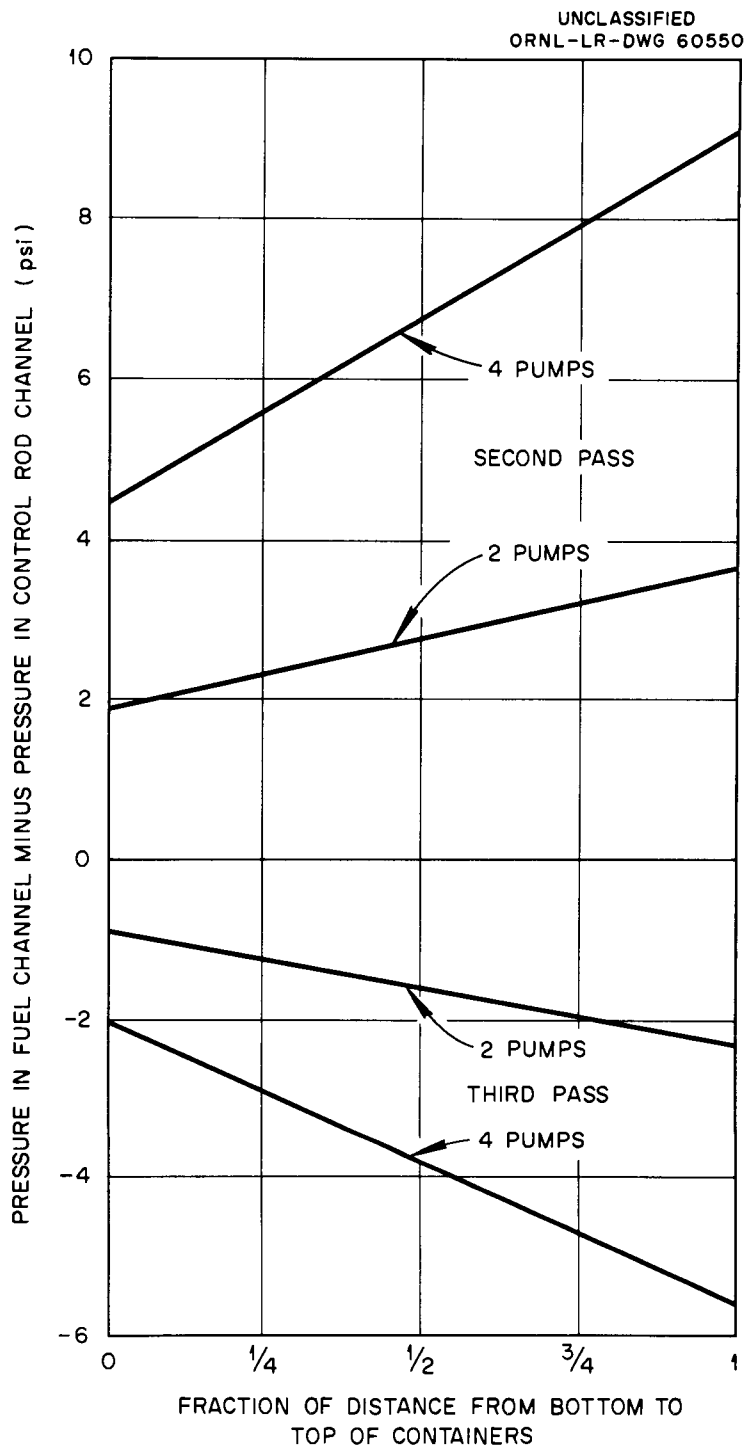

Fig. 25. Maximum Pressure Differential Expected Across Container Walls with No Core Bypass Flow.

Design Criteria

In an analysis of this type, certain ground rules must be established. These rules, expressed as various conditions and limitations, are itemized below:

1. The stresses must be consistent with a conservative design stress, which shall be chosen to ensure a useful lifetime of 20 years for the fuel-element container assembly.

2. The sum of the outward movement of the two fuel-element container walls that partially define a control-rod channel must not exceed the difference between the minimum width of the control-rod channel[48] and the maximum width of a control rod.[49] Outward movement is defined as a movement toward the control rod; inward movement is not to be considered in obtaining the sum. Movement is measured from the nominal drawing location of the container wall. The maximum permissible movement as outlined above is 0.237 in.

3. The sum of the inward movement of two opposing fuel-element container walls must not exceed the difference between the minimum inside dimension of the fuel-element container and the maximum width of a fuel bundle. Based on the outside dimensions of the present containers[48] and the reference envelope dimensions of core I fuel elements,[50] the maximum allowable inward movement is 0.145 in. if the container wall thickness is 0.140 in.

4. For purposes of this analysis, any increase in thickness of the fuel-element container wall necessitated by the material change will be made by decreasing the inside dimension of the fuel-element container; the outside dimension will remain the same as that in the present design.[48]

5. The weight of a fuel element is 760 lb. In any case where the fuel element is either partially or wholly supported by the container

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[48]Babcock & Wilcox Company Drawing No. 10423F-4, "Fuel Container Assembly."

[49]Babcock & Wilcox Company Drawing No. 10441F-9, "NMSR Control and Follower Rod Assembly."

[50]Babcock & Wilcox Company Drawing No. 10424F-10, "4-Bundle Fuel Element Assembly."

walls, both the following types of loads shall be considered: (1) one-half the load acting at the top and one-half at the bottom (corresponding to core I) and (2) one-half the load acting at the center and one-fourth at each end (corresponding to core II).

Method of Analysis and Results

Selection of Important Stress Models

It appeared that the temperature differentials in the assembly would induce bowing and stresses of significant magnitude in the spacer bars. A theoretical analysis in which the spacer bars were treated as beams on elastic foundations showed, however, that the maximum stress was less than 500 psi and that the maximum deflection was extremely small. Therefore temperature-induced stresses and deflections were ignored.

Three different models were considered in determining the stresses and deflections from beam action of the fuel-element containers. First, the entire assembly of 32 containers (with the weight but not the strengthening effect of the spacer bars considered) was treated as a simply supported, builtup beam. Second, a single fuel-element container was considered as a propped cantilever beam. The third model was that of a single fuel-element container acting as a simply supported beam. Since the latter model was the most conservative, the single container treated as a simple beam was chosen for dealing with deflections and stresses from beam action. With the ship on its side and subjected to a vertical heave of 2 g, the following quantities were calculated for 91-mil-thick containers:

Axial bending stress, psi	3000
Maximum deflection, in.	0.030
Shear stress, psi	463

As may be seen, beam action alone is not an important consideration in the fuel-element container design.

In addition to the unimportant cases considered above, stresses and deflections arising from torsion (caused by nonsymmetrical loading) and acceleration (connected with ship rolling) were shown to be negligible. Hence, it was found that the only important stresses and deflections

were those connected with plate action caused by pressure and concentrated loads. For determining the stresses and deflections from plate action in the walls of the containers, two columns of containers were considered. These two columns, along with the assumed attitude of the ship, are shown in Fig. 26. Column 1 consists of second-pass containers only, and column 2 consists of four third-pass containers with second-pass containers at top and bottom. The second-pass containers are all subject to an internal pressure (at mid-height) of 7 psi (see Fig. 25), and their left faces receive no support from an adjacent container. Simple supports at the upper corners were, therefore, assumed for column 1 in order to allow the sides of the containers to deflect freely. The third-pass containers are subject to an internal pressure (at mid-height) of 4 psi, and each is supported by an adjacent container on every side. Furthermore, the supports are such that there should be little, if any, rotation at the supports; thus, fixed supports were assumed on the third-pass containers. The right supports of the second-pass containers at the top and bottom of column 2 are likewise greatly fixed against rotation by adjacent containers, while the left sides are free. The second-pass containers in column 2 were assumed to be simply supported at the midpoints of their sides. All intermediate containers or frames in a given column were assumed to be identical in loading, stresses, and deflections. Based on these considerations, the container assembly models shown in Fig. 27 were chosen for the analysis. Examination of the stresses in the models shown in Fig. 27 required analyses of three basic rigid frames, and these were done by the slope-deflection method. The three basic frames are shown in Fig. 28. In Figs. 27 and 28, the symbols used are defined as follows:

L = cross-sectional dimension of a container,

Q = reaction of fuel element on container wall,

R = reaction of upper spacer bar on container wall,

P = reaction of lower spacer bar on container wall.

The spacer bars were analyzed by treating them as beams on elastic foundations where the fuel-element container walls provided the elastic foundations. The true end condition is difficult to evaluate, but it is intermediate between the fixed and the simply supported conditions.

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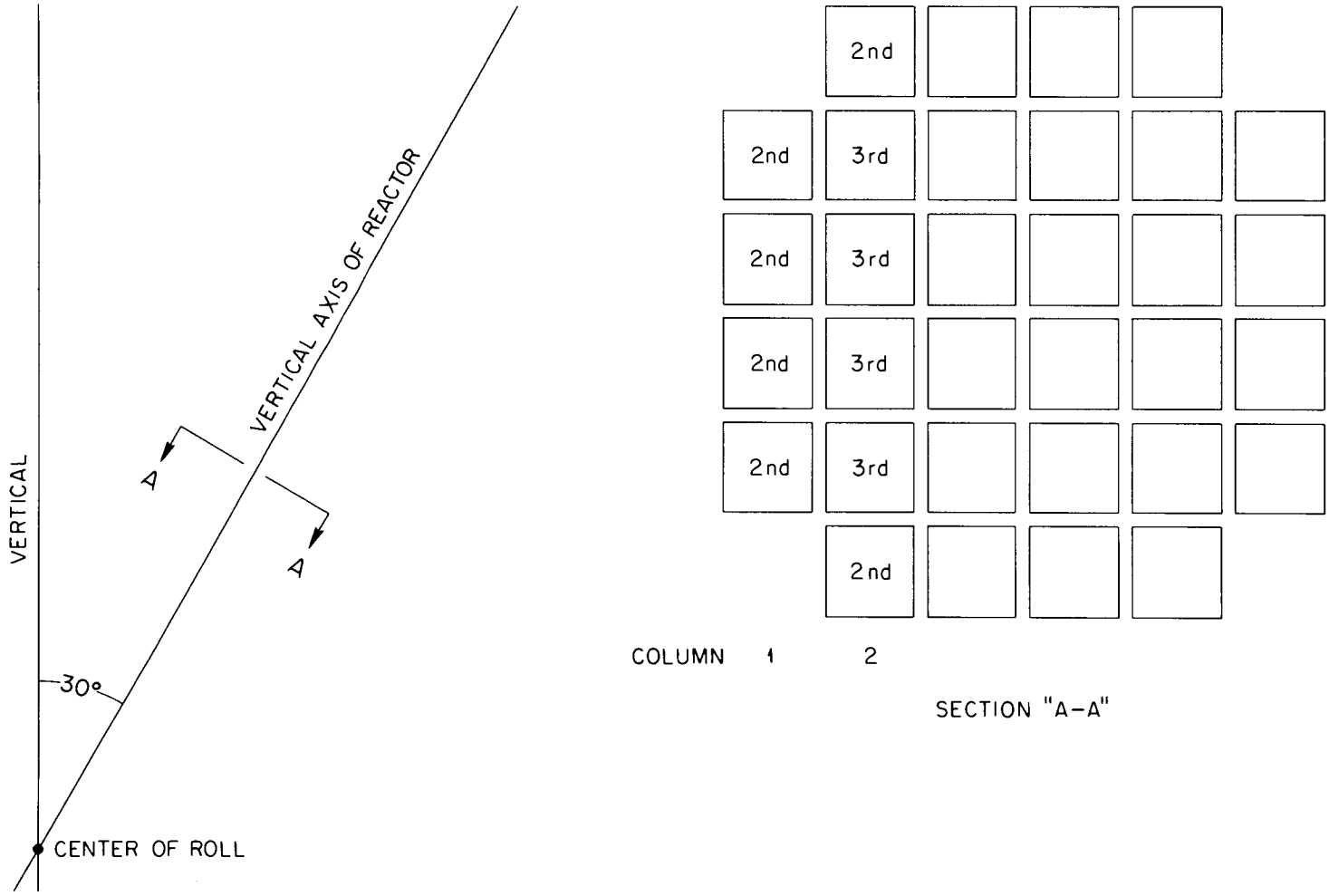


Fig. 26. Orientation of Fuel-Element Containers Used in Analyzing Stresses Arising from Plate Action.

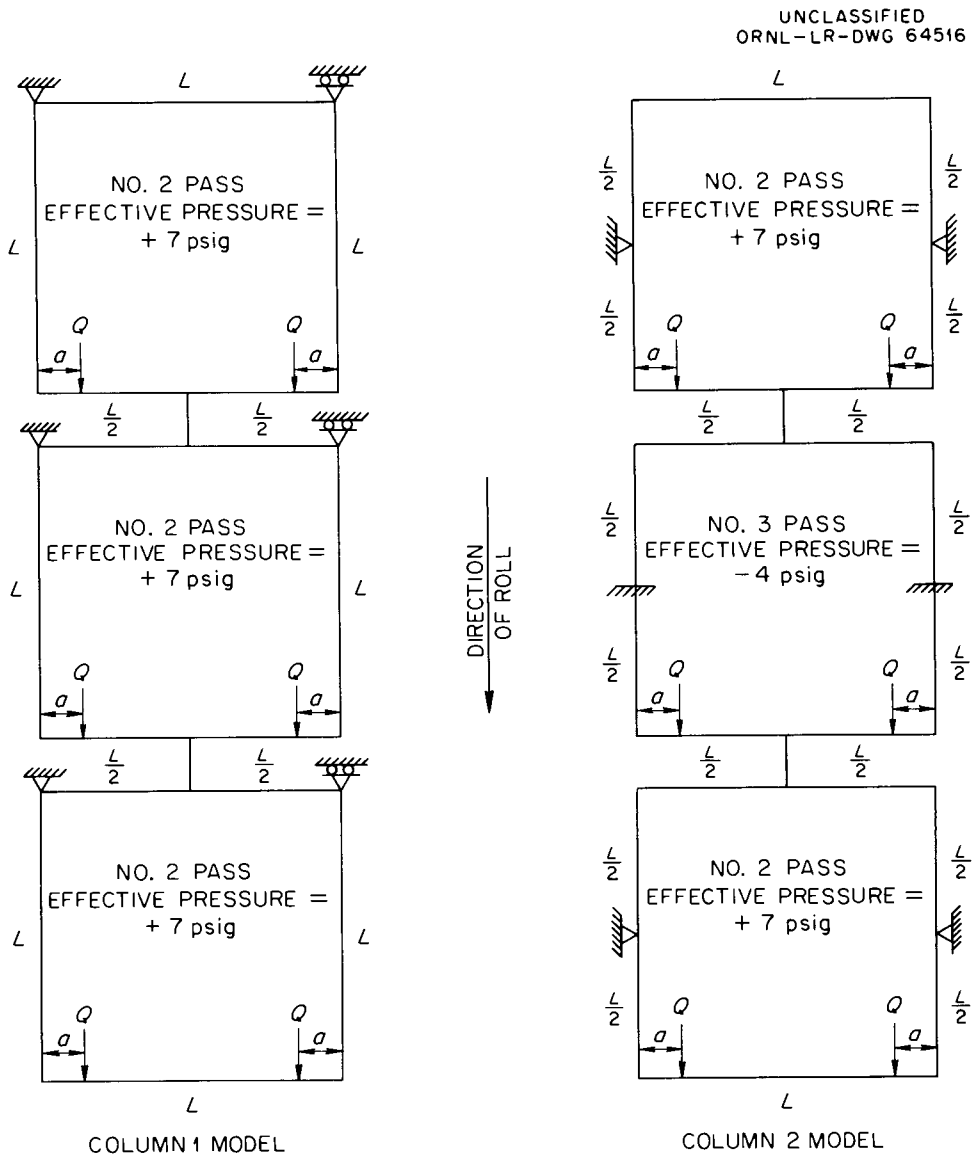


Fig. 27. Container Assembly Models Used in Analyzing Stresses from Plate Action of Fuel-Element Containers.

For this reason two beam models, one simply supported and one with a fixed end, but both semiinfinite in length, were considered.

In considering core I and core II type fuel elements, it was clear that the more critical elevation in the core would be at midheight for core II fuel elements and near the top for core I elements. Thus, the analysis was limited to the critical area for each type of fuel element. Since the analysis indicated that the loads imposed by core II were more severe, only core II results are summarized below.

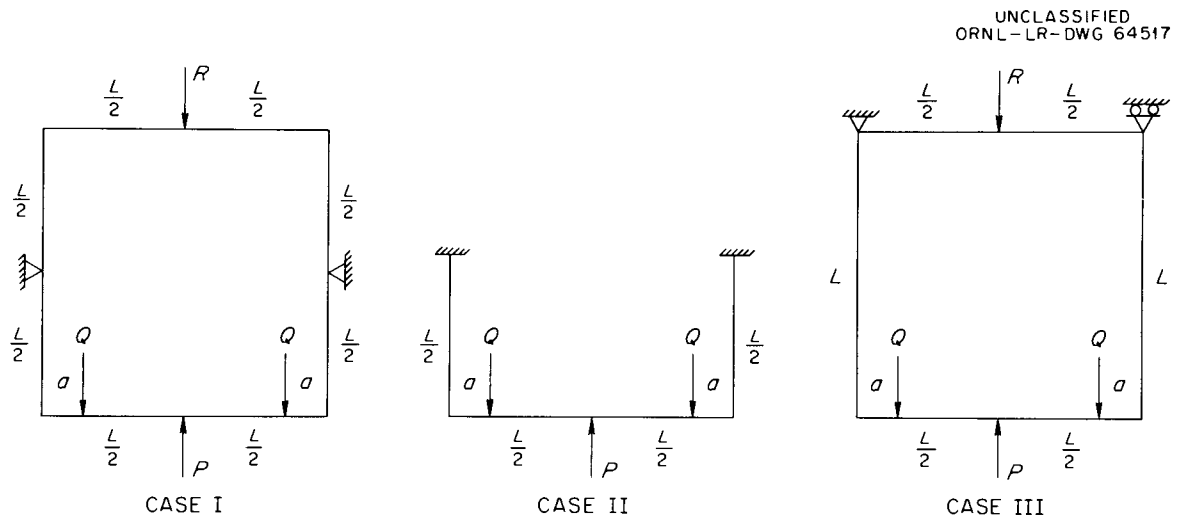


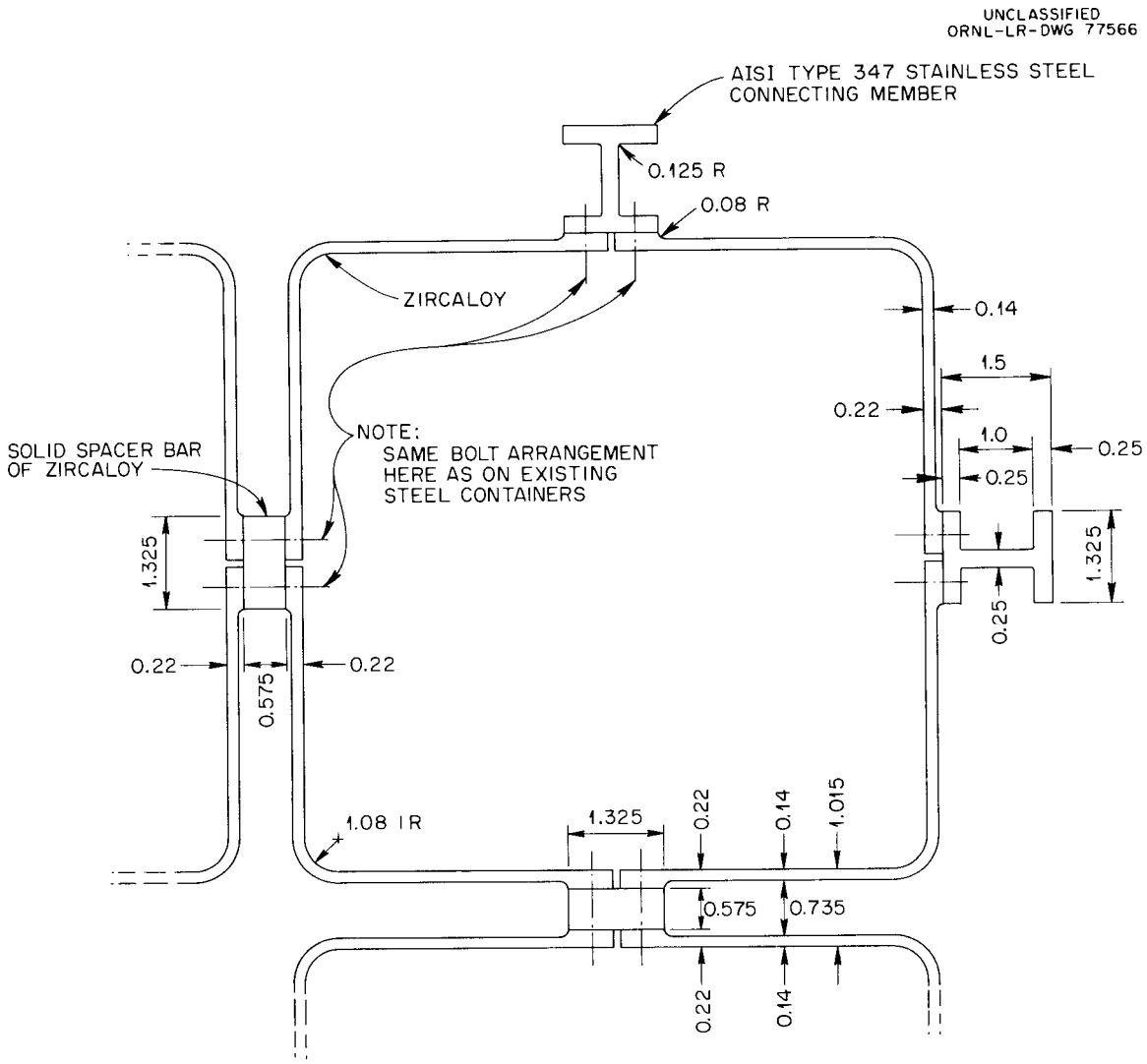
Fig. 28. Basic Rigid-Frame Models Used in Stress Analysis of Fuel-Element Containers.

Results

Using 15% cold-worked Zircaloy with a design stress of 17,600 psi (approximately one-third of the ultimate strength), it was found that a container wall thickness of 0.14 in. would be adequate under the worst normal steady-state load condition. A fatigue analysis based on a 30° ship roll combined with simultaneous 0.7-g lateral and 0.3-g vertical heaves indicated, however, that the 0.14-in. wall thickness was inadequate at the points of attachment of the containers to spacer bars and connecting members. Fatigue considerations therefore dictated an increase in thickness of the containers to 0.22 in. at the midpoints of the container sides. In addition, strenghtening of the spacer bars and connecting members was required. To obtain the required additional strength, the spacer-bar design was modified to a solid rectangular section; and an I-section of AISI type 347 stainless steel was chosen for the connecting members.

The modified fuel-element container design is shown in Fig. 29. The design of Fig. 29 was analyzed for the unusual condition of the ship on its side and subjected to a vertical heave of 2 g. Under these conditions the analysis indicated that the maximum container wall deflection

would be 0.10 in. Since this deflection is within the limits set previously, it appears that the modified container design is adequate under all foreseeable conditions of operation.



NOTE:
ALL DIMENSIONS ARE IN INCHES

Fig. 29. Modified Fuel-Element Container Design Using Zircaloy.

7. HAZARDS ANALYSIS

In considering potential design changes of an existing reactor plant approved for operation, an investigation is necessary to determine whether any of the proposed changes will decrease the safety of the plant. In considering the design modification of substituting Zircaloy for stainless steel in the fuel-element containers, the one new aspect that might add to the operational hazards of the system is the possibility of a chemical reaction between the Zircaloy and the reactor coolant. Such a reaction would occur, however, only during a very serious reactor accident that would involve significant melting of the reactor core. The two general classes of reactor accidents that can result in overheating and meltdown of fuel in a pressurized-water reactor are the rapid-power-rise accident (reactor runaway) and the loss-of-coolant accident. Hazards evaluations by the Babcock and Wilcox Company[51] indicated that loss of coolant is the maximum credible accident (mca). Thus the evaluation of the effect on plant safety of a zirconium-water reaction, which is summarized here, was made under the conditions of the loss-of-coolant accident.

Method of Analysis

The computational method used in this analysis was based on the method devised by Lustman[52] to determine zirconium-water reactions for reactor loss-of-coolant accidents. Lustman applied this analytical approach to the PWR loss-of-coolant accident,[53] and the method has been shown to be in excellent agreement with experimental work by Higgins and Schultz[52] on zirconium-water reactions.

It was conservatively assumed that during the early phase of the loss-of-coolant accident the temperature of all the Zircaloy containers would

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[51] G. R. Thomas, "NMSR Final Safeguards Report — Volume VII. Power Plant Accidents," USAEC Report BAW-1164, Babcock & Wilcox Co., March 1960.

[52] B. Lustman, "Zirconium-Water Reactions," USAEC Report WAPD-137, Westinghouse Atomic Power Division, Dec. 1, 1955.

[53] B. Lustman, "Zirconium-Water Reaction Data and Application to PWR Loss-of-Coolant Accident," USAEC Report WAPD-SC-543, Westinghouse Atomic Power Division, May 1957.

correspond to the temperature of the hottest fuel element. The fuel element temperature as a function of time after the accident was obtained from information in the safeguards report.[51] If the Zircaloy temperature reached about 1000°K, the zirconium-water reaction would become significant, and the heat released from the reaction would contribute to the Zircaloy temperature increase. The reaction would, therefore, become autocatalytic. It was assumed that when the temperature of the Zircaloy reached the melting point, the molten metal would form droplets and fall through steam into water at the bottom of the reactor vessel. During the post meltdown period the metal-water reaction would continue quite rapidly until the metal was cooled to below 1000°K. The effects of the energy produced by the metal-water reaction were added to the previously determined accident conditions, and the adequacy of the containment vessel under these amended accident conditions was determined.

Results

The calculated Zircaloy temperature and reaction data are presented in Table 11. Based on conservative assumptions, it was found that about 21% of the Zircaloy in the core would react with water, and this reaction would release about 1.89×10^6 Btu of energy to the containment. The metal-water reaction would not become significant until about 200 sec after the accident and would be essentially complete at about 270 sec after the accident.

The postaccident containment pressure, excluding the effect of the metal-water reaction, was calculated previously[51] and is shown in Fig. 30. Since the metal-water reaction does not become significant until 200 sec after the accident, the containment pressure curve in Fig. 30 is unaltered by the metal-water reaction for times less than 200 sec after the accident. It may be noted from Fig. 30 that the peak pressure of approximately 173 psig occurs at about 20 sec after the accident, and at 200 sec the pressure has decreased to approximately 140 psig. If it is assumed that all the energy from the zirconium-water reaction is released at 200 sec, the effect is to increase the containment pressure from 140 to 153 psig. If, in addition, all the hydrogen generated in the zirconium-water reaction

Table 11. Summary of Zircaloy Temperature and Reaction Data

Temperature of Zircaloy (°K)	Time After Accident (sec)	Cumulative Total Zirconium Reacted (% of total in core)
1000	31.5	0
1200	112.5	0.54
1400	181.5	1.2
1600	217.5	2.2
1800	231.2	3.4
2000	235.9	4.7
2200	237.5	6.2
2200 (molten)	244.1	10.4
1000	~270	21
<1000	>270	21

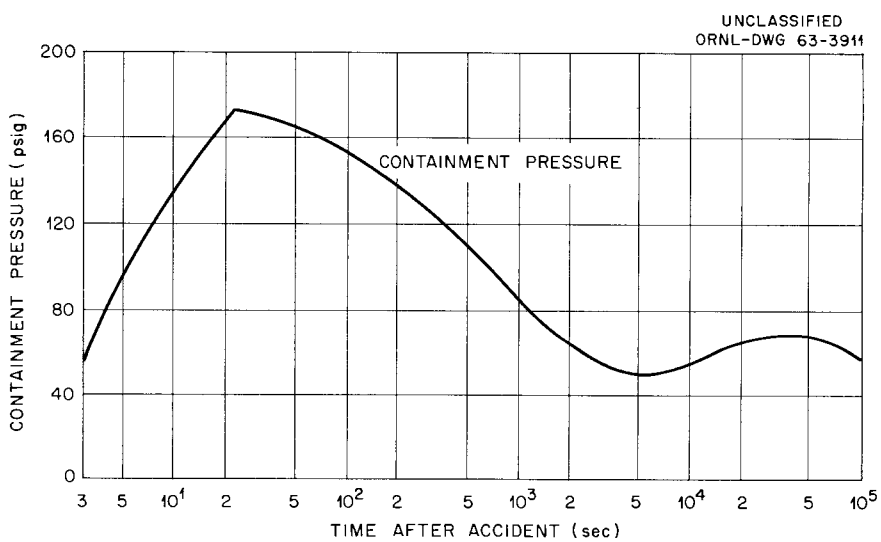

Fig. 30. Containment Pressure After Maximum Credible Accident.

were to react with oxygen at 200 sec after the system failure, the containment pressure would be increased to 163 psig, which is still 10 psi below the peak containment pressure. It appears therefore that the zirconium-water reaction would not significantly alter the previously determined consequences of the maximum credible accident.

ACKNOWLEDGMENT

Valuable contributions to this work by B. W. Colston, J. M. Corum, S. E. Moore, W. G. Stockdale, and M. L. Winton are gratefully acknowledged.

Appendix

DESCRIPTION OF N.S. SAVANNAH REACTOR

The reactor pressure vessel and associated internal components are shown in Fig. A.1; the vessel measures approximately 27 ft in height and 10 ft in diameter. Cooling and moderation are provided by pressurized light water, which makes three passes within the reactor vessel; two of these passes are through the core. A fuel-element container assembly consisting of 32 fuel-element containers and flow transition pieces, an inner thermal shield, and a base plate is a permanent, but removable, part of the reactor core. This assembly, with the fuel elements and control rods inserted, is shown in Fig. A.2. A cutaway view of the container assembly is shown above in Fig. 1. Each fuel-element container is made up of four stainless steel angles approximately 100 mils in thickness. The containers are bolted together to form the 32-container assembly, with spacing between containers maintained by stainless steel spacer bars that run the full length (~80 in.) of the containers. The spaces between containers accommodate the control rods.

The reactor is fueled with 32 separate fuel-element assemblies, each of which contains 164 stainless steel, UO_2-filled, fuel rods. The fuel-element assemblies, shown in Fig. A.3, are about 8 1/2 in. square and 76 1/2 in. long. The fuel rods are held in a square lattice by short stainless steel spacer ferrules brazed to the tube walls. These ferrules are located on nine planes approximately 8 in. apart along the length of the fuel element.

Reactor control is provided by 21 cruciform control rods. As shown in Fig. A.4, the control rods consist of three sections: (1) a poison section containing boron, (2) a Zircaloy-2 follower section, and (3) a stainless steel extension section. All three sections are fastened into a rigid, continuous assembly.

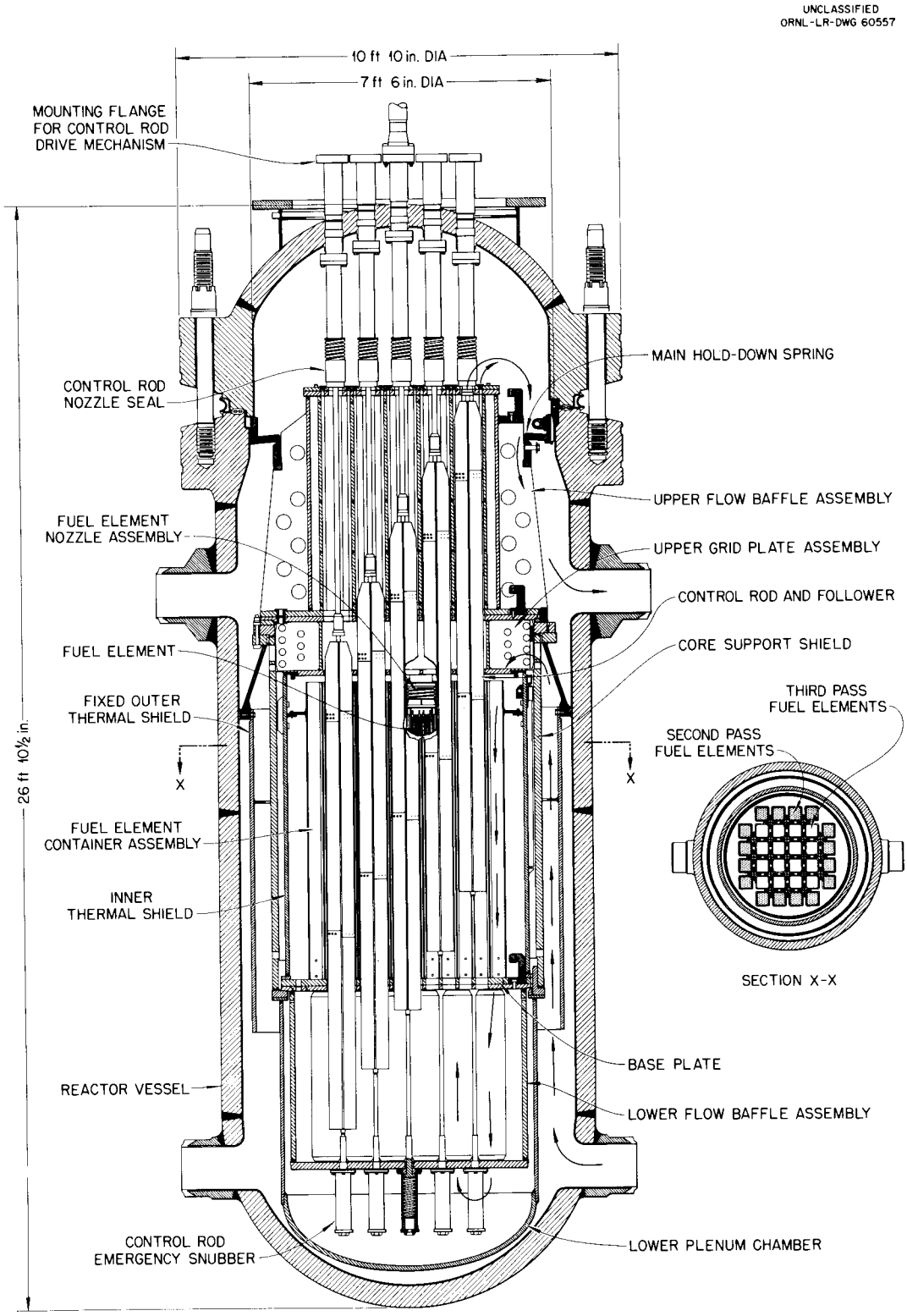

Fig. A.1. Reactor Vessel and Internal Components.

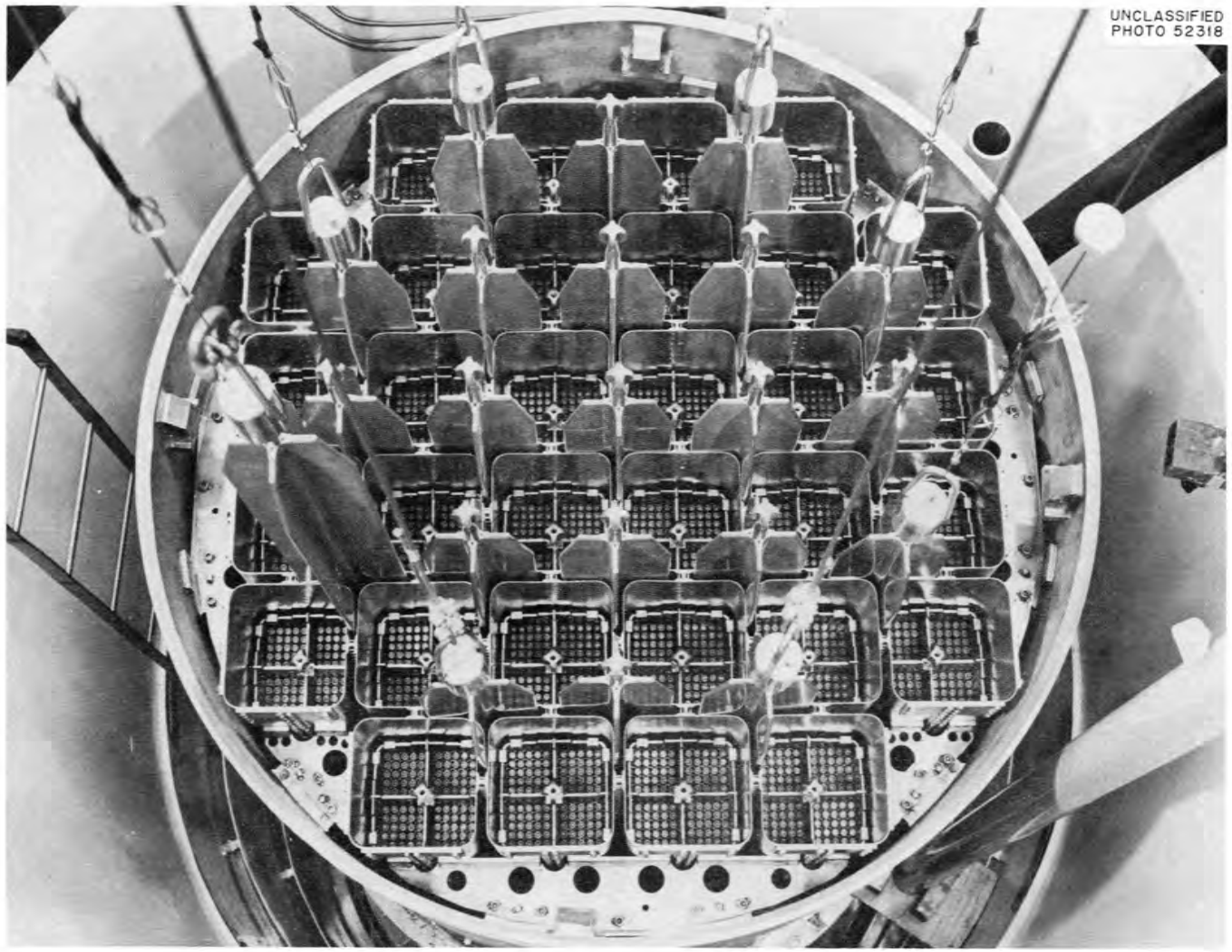

Fig. A.2. Core Assembly.

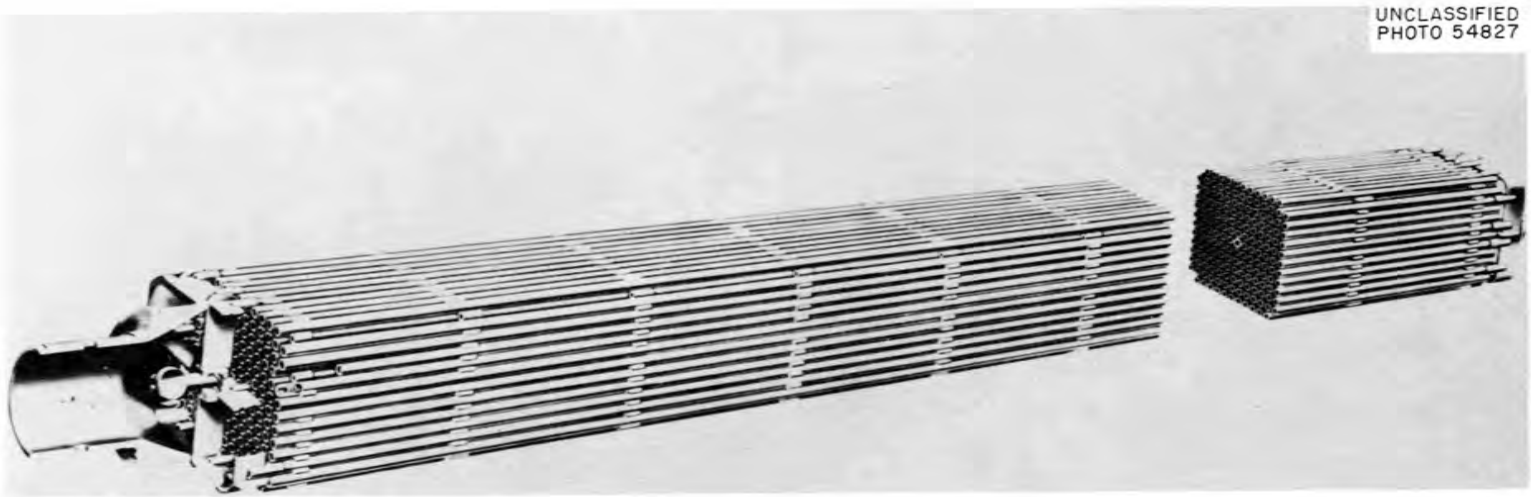

Fig. A.3. Core I Fuel Element.

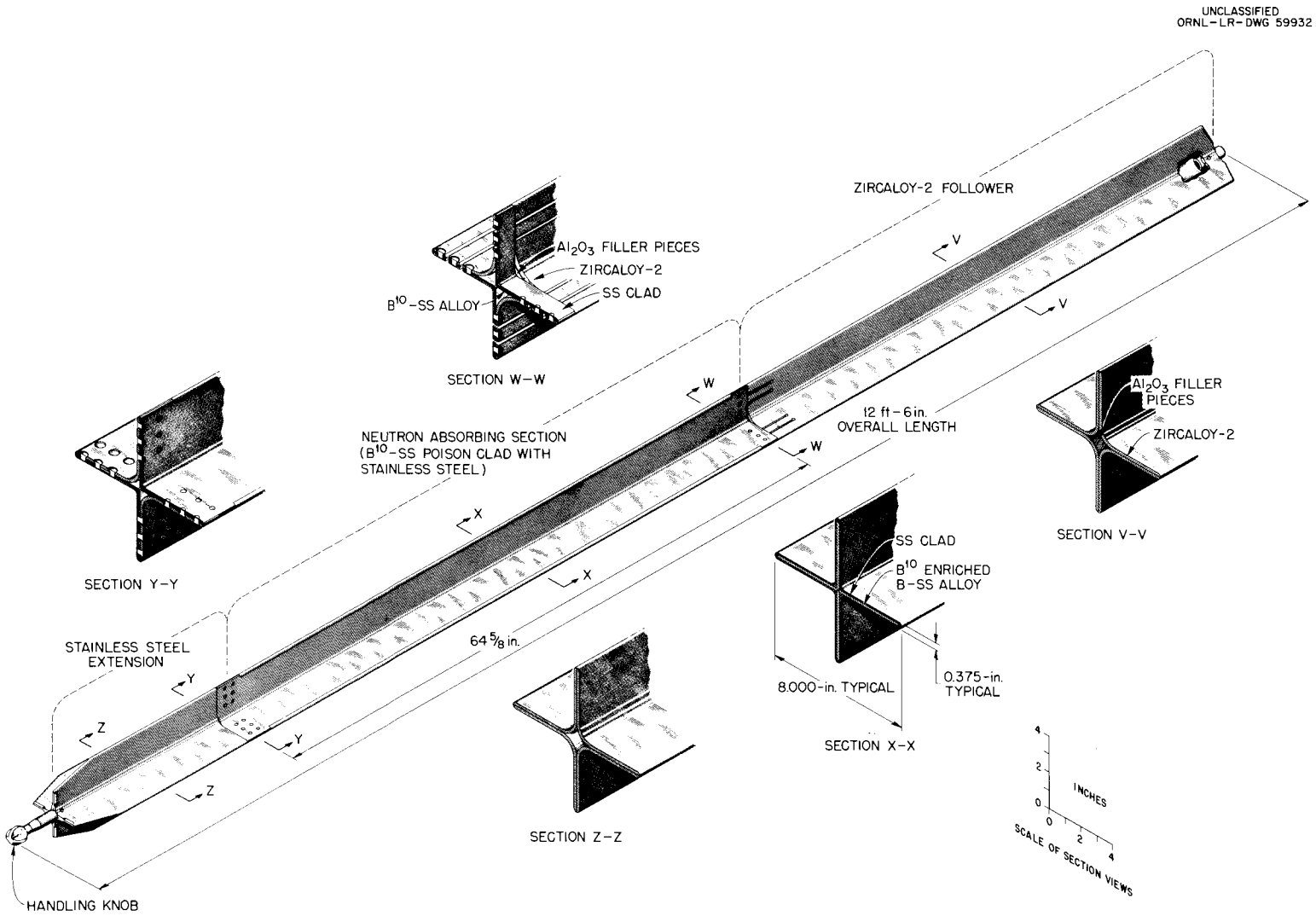

Fig. A.4. Control Rod.

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