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**CORE LENGTH TESTABLE REACTOR
CONCEPT NEUTRONIC ANALYSIS**

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by

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ABSTRACT

Development work on thermionic reactor systems has been ongoing in the U.S. since the early 1950s. While significant successes were achieved, progress has been hampered by frequent changes in direction and funding instabilities (as has been true for many high technology initiatives). The recent Air Force thermionics initiative (1991) represents the latest in thermionics reactor development in the U.S. This Air Force initiative called for the development of thermionics reactors with the output power of about 40 kWe, and which incorporated the features of testability, fabricability, low development cost, high level of safety and reliability, and survivability.

The Russian thermionics program has had a more stable development over the past 30 plus years and has produced two recent flight tested TOPAZ reactor units in addition to a large number of ground test units. In view of the rapidly improving relations between the U.S. and FSR, it was logical to look at the extensions of TOPAZ technology to meet the Air Force requirements. The strong ties between Space Power, Inc. (SPI) and the key Soviet institutions made it possible for SPI to propose such a concept to the Air Force. Argonne National Laboratory served as a subcontractor to SPI in this effort, supporting the prime contractor in the areas of core design and safety analysis.

Several concepts were analyzed to define a design that would meet all the requirements set forth by the Air Force. This report describes the methodology used, the different designs analyzed and reasons for the evolution of the design, and presents the results for the different concepts.

I. INTRODUCTION

A. Background

Development work on thermionic reactor systems has been ongoing in the US since the early 1950s. Progress has been hampered by frequent changes in direction and funding instabilities. The SP-100 power system selection provided the latest opportunity for thermionics space power systems to be considered (in the early 1980s). The decision was made to go with thermoelectric partially because of the perceived advantage in maturity of thermoelectric technology over thermionic technology. A thermionic fuel element (TFE) verification program was initiated in 1980 and is presently in progress, developing the test data for TFEs designed for MW class thermionic systems. The recent Air Force thermionics initiative, for which the present work was performed, represents the latest in thermionics reactor development in the US.

In contrast, thermionics has enjoyed steady support in the Former Soviet Union (FSR) starting from about 1958 when the technology was first introduced to them from the US. The steady development has culminated in the production and testing of a number of thermionics reactors including the space flight tests of multicell thermionic TOPAZ reactor systems in 1987 and 1988. A recent development includes single cell TOPAZ-II reactors that offer the advantage of electrically heated testability. As a general statement, it is fair to say that the thermionics data base in the FSR represents thousands of man years of effort.

The Air Force thermionics initiative called for the development of thermionics reactors with the output power of about 40 KWe, and which incorporated the features of testability, fabricability, low development cost, high level of safety and reliability, and survivability. In view of the rapidly improving relations between the US and FSR, it was logical to look at the extensions of TOPAZ technology to meet the Air Force requirements. The strong ties formed between Space Power, Inc. (SPI) and the key Soviet institutions made it possible for SPI to propose such a concept to the Air Force. Argonne National Laboratory served as a subcontractor to SPI in this effort, supporting the prime contractor in the areas of core design and safety analysis. This report presents results of the work done.

B. Proposed Statement of Work

The statement of work was designed to assist Space Power Inc. (SPI) in the neutronic analysis of their Core Length Testable Reactor Concept. The scope of the work to be performed follows:

1. Assistance in the neutronic design of three reactor cores (40 kWe, 100 kWe and 10 kWe); detailed neutronic analysis for the 40 kWe core and brief analysis for the other two cores. In this area ANL would perform analyses to: a) determine reactivity loss due to burnup of actinides and buildup of fission products over the 10 year life of the reactor; b) confirm safety margins for the designs under postulated accident conditions for the reference design (the postulated configurations would be provided by SPI); c) determine reactivity effect due to moderator temperature coefficient and to determine the Doppler reactivity effect of the fuel and emitter.

2. Assistance in specifying a development plan for technology validation.

3. Participation in two meetings and presentations.

This report addresses only item 1 above, i.e., the neutronic design analysis.

C. Core Concepts Analyzed

Since the main objective of the work was to define a design that would meet all the requirements (not only neutronics) for the 40 kWe system, several concepts were analyzed for this power level. A short description of the concepts analyzed follows, and the results and reasons for going to different designs are presented in section III of this report. The starting point was a design similar to the Soviet TOPAZ reactor, a Zirconium Hydride (ZrH) moderated system in which the TFEs are placed in a matrix of ZrH; even though this core did not meet the requirements all the analysis described in item a) of section I.1 were performed for this core because some qualitative comparisons could be made with results published for the TOPAZ reactor, and this would help in the validation of the methods used in the analysis. Next a concept in which rows of TFE and Be-ZrH rods are alternated in the core was analyzed. Following this concept, a Beryllium-Yttrium Hydride moderated core was analyzed, and since with some modifications this design would be able to meet the requirements, a more complete analysis was then performed for this core. Late in this project some of the modifications to this Be-YH core were implemented and a partial analysis of this latest design was performed.

For the 10 kWe design an abbreviated analysis was performed for one concept which also uses Be-YH as moderator; in this concept drivers are also used in the core.

As for the 100 kWe design a decision was made by SPI to not perform any analysis for this power level. This did not represent a reduction in workscope since the effort on the 40 kWe system more than compensated for it.

The methods used in all these analyses are presented in Section II, and a description of the reactors analyzed together with the results of the analyses are presented in Section III. In Section IV a summary with recommendations for the definition of a final design is presented.

II. CALCULATIONAL METHODOLOGY

The analysis and design data presented below are based on current methods and state-of-the-art codes in the field of reactor physics. A flowchart of the design analysis steps used in the present study is presented in Figure 1, and a short description of the computer codes used in the different steps of the analysis follows.

For general neutronic static analysis the Monte Carlo code MCNP⁽¹⁾ was always used; MCNP was also the code used for control evaluation and safety analysis (submersion). MCNP solves the Boltzmann transport equation for complex geometries using the Monte Carlo method; MCNP has options for continuous and multigroup energy cross section treatment and here the

continuous energy treatment was always used. DIF3D,⁽²⁾ which solves the neutron diffusion equation in one, two or three dimensions, and TWODANT,⁽³⁾ which provides a discrete ordinate multigroup solution for the Boltzmann transport equation in one or two dimensions and includes treatment of anisotropic scattering effects were also used to benchmark against MCNP for other calculations; note that only the one-dimensional option of TWODANT was exercised in the results presented in this report.

For depletion analysis, REBUS-3⁽⁴⁾ which is a code designed for analysis of fuel cycle was used. REBUS-3 solves the non-equilibrium operation of the reactor under the specified set of operating conditions governing power levels and operating time. For the solution of the depletion problem in REBUS-3, the total reactor burn cycle is divided in several subintervals (defined by the user), and an explicit burnup is performed in each region of the reactor over each of these subintervals using the average reaction rates over the subinterval; these average reaction rates are based on fluxes obtained from the diffusion theory solution at both the beginning and end of the subinterval. The transmutation equations are solved by the matrix-exponential technique and the isotopes to be considered in the burnup equations, as well as their transmutation reactions, are specified by the user. Following agreement with SPI, two-dimensional (RZ) models of the reactors analyzed were used in the depletion analyses. Note that REBUS-3 is also the code of choice in the definition of enrichment zones (or moderator-to-fuel ratios) with the objective of flattening the power distribution throughout the core, even though this capability was not used in this phase of the design analysis.

For the determination of kinetics parameters several codes were used. The effects of the moderator temperature were analyzed with MCNP and using a detailed core model. The determination of the Doppler reactivity effects for the fuel and emitter was performed using a unit-cell concept with the one-dimensional option of the TWODANT transport code; this choice for this phase of the design was a decision made in agreement with SPI due mainly to level of effort constraints. For the determination of the neutron generation time, delayed neutron fractions and decay constants the code VARI-3D,⁽⁵⁾ a diffusion theory perturbation theory code, was used. It is important to note that neutronics effects of the photoneutrons (delayed neutrons) emitted as a result of photon (γ, n) interaction were not included in the determination of neutron generation time and delayed neutron fractions; for slow transients, power shaping and effect of control drums motions, the delayed effect of photoneutrons could be significant.

Because the validity of calculations depends strongly on the accuracy of the cross sections on which the calculations are based, special care must be taken in the generation of accurate multigroup cross sections. The following were the methods used in the generation and use of the cross sections:

1. MCNP analysis: For the calculations performed with MCNP, the cross sections used were those obtained from Los Alamos and included with the MCNP code package. These cross sections were basically room temperature cross sections for all isotopes in the different cores and higher temperature cross sections for the ZrH and Be; for these high temperatures, the MCNP cross sections also includes the thermal neutron scattering kernels [$S(\alpha, \beta)$ treatment]. It is important to note that MCNP does not have the $S(\alpha, \beta)$ treatment for the Yttrium Hydride moderator and the $S(\alpha, \beta)$ for the H in ZrH was

then used (the decision to use this as a surrogate for YH was made in agreement with SPI).

Even though the cross sections in MCNP have been widely used by several other groups, and are supposed to be well tested, validation of some of these cross sections was performed in two ways. First, calculations for several criticals using a sphere of highly enriched uranium surrounded by Be reflector were performed and the results are presented in Table I. Also, using a homogeneous model for the SNAP Experimental Reactor (SER) as given in NAA-SR-5619⁽⁶⁾ an MCNP calculation was performed with the critical configuration (55.2 fuel elements) and the result is also presented in Table I.

The results in Table I basically validate the MCNP cross sections (for the materials present in those calculations).

2. Depletion Analysis: Because of funding limitation and the extensive time needed for the generation of fission products cross sections, a decision was made (in agreement with SPI) to use the EPRI-CELL⁽⁷⁾ code for the generation of cross sections for burnup analysis; 97-group EPRI-CELL cross sections for most of the isotopes in the cores to be analyzed were already available at ANL and this would be adequate for this phase of the design. The EPRI-CELL code utilizes a 62-group fast GAM library and a 35-group slow THERMOS library. The cross section library for EPRI-CELL has been processed by the MC2-2⁽⁸⁾ code for the fast library and by the NJOY⁽⁹⁾ code for the thermal library; thermal neutron scattering kernels derived from ENDF/B-V data for H bound in ZrH are present in the ANL EPRI-CELL library. This same process was used to generate cross sections for the materials in the thermionic cores not present in the existing ANL EPRI-CELL library (Y, and W isotopes).

EPRI-CELL is a one-dimensional (cylinder or slab) transport code which determines the neutron spectrum as a function of space, energy and time; for this work, burnup-dependent cross sections were not generated, i.e. cross sections at BOL were used in the burnup calculations. Broad-group cross-sections result from collapsing the GAM/THERMOS fine group cross sections over that spectrum. In this work a 15-group structure (9 fast and 6 slow neutron groups) was generated from EPRI-CELL and then used in the REBUS-3 depletion analyses, and in the VARI-3D calculations. Note that 15-group cross sections were generated for each of the cores analyzed in this work.

The cross sections generated from EPRI-CELL were then validated (for each of the cores analyzed) by comparing the results with those obtained with MCNP; these results are presented in Section III for each of the concepts analyzed, and the agreement with MCNP is very good.

3. Doppler Effect Calculations: For the analysis of the Doppler effect the NJOY code was used to provide a 97-group cross section set for use in the TWODANT transport code; P3 cross sections were generated. One set of NJOY cross sections was generated for room temperature and cross sections at 1000 K were generated for the fuel and emitter/collector materials. This set of cross sections was generated at the beginning of this work and the compositions used for the calculation of the potential cross sections (these are used in the calculation of the self-shielded pointwise cross sections) were those present in the concept defined at that time and shown in Table II. Due to level of effort constraints in this phase of the design, these cross

sections were used for Doppler effect calculations for all the concepts analyzed.

The room temperature cross sections generated by using NJOY were then used in unit cell calculations to compare against MCNP unit cell calculations. The results of these comparisons are presented in Section III for each of the concepts analyzed, and they were considered adequate for this phase of the design.

III. REACTOR MODELS ANALYZED AND RESULTS

A. 40 kWe Concepts

Several concepts were analyzed for this power level. A summary of the concepts analyzed and the reason(s) for going to a different concept is presented in Table III. The specific results obtained for each concept are discussed below.

1. Zirconium Hydride Moderated Core

This concept is similar to the TOPAZ reactor. A 52 cm core containing 151 TFEs with an active length equal to 35 cm, surrounded by a 10 cm Be radial reflector and 12 control drums are the major components of this core. Tables II and IV give the characteristics of the Thermionic Fuel Elements and the core, respectively. The major differences between this core and TOPAZ (based on published information) are the core size (about 30 cm for TOPAZ) and the emitter thickness (1 mm thickness in TOPAZ vs 2 mm in this design).

Multigroup cross sections (15 and 97 groups) were generated for this core and comparisons with MCNP analysis are presented in Table V; differences of less than 2% were obtained, and due to limitations in level of effort the cross sections were considered adequate for this phase of the design.

A burnup calculation was performed (REBUS-3 code) using an RZ model and the 15-group cross section set and a 3.45 percent burnup swing reactivity was calculated for a ten-year operation at a thermal power level equal to 629 kW(th). Next the MCNP code was used to determine the worth of the control drums, since one of the requirements of this design is that the control drums must have enough reactivity to control the reactor. The results of these MCNP calculations, provided in Table VI, indicated a very small worth for the control drums (about 2.2 percent); this small worth is due to the large diameter of the core. Based upon these results it was clear that this concept would not meet the requirements, and it was not considered further.

Since this concept was similar to TOPAZ and there is some information available on several neutronic characteristics of this core, determination of several parameters was performed to confirm (at least qualitatively) some of the results presented in the FSR-published results.

The most unique characteristic of the TOPAZ reactor is its positive moderator temperature reactivity coefficient. A physical explanation for this effect can be described as follows. The temperature effect arises from the temperature dependence of the thermal scattering for the ZrH. The high

temperature hardens the neutron spectrum and as a consequence the absorptions and fissions decrease (reduced cross sections at the higher neutron energies). The decrease in the absorptions is larger than the decrease in the fissions since the fuel is quite black at these neutron energies. The FSR-published results and another study⁽¹⁰⁾ have shown that this temperature effect decreases with the temperature as shown in Figure 2. The MCNP analysis performed for the 40 kWe ZrH moderated core basically confirms the trends shown in the FSR-published results up to a temperature of 800 K; these results are shown in Table VII and plotted in Figure 2. However for the next higher temperature available in MCNP cross sections, i.e. 1200 K, there is a large increase in the temperature effect. Since there is no explanation for this quite large increase, the only possible factor is the validity of the 1200 K cross sections. It is important to note that LANL analysts have also uncovered the same problem with the ZrH high temperature (1200 K) cross sections in the analysis for their reactor design, as discussed in their August 26 presentation to the Air Force. Note also that the absolute values calculated for this concept (up to 800 K) are much smaller than those for TOPAZ; most of this difference can probably be attributed to the different thickness of the Mo in the emitter in the TFE (1 mm in TOPAZ vs 2 mm in this analyzed concept), however no effort was made to explain this difference because of lack of detailed information on the TOPAZ reactor.

An analysis was also performed to compare the moderator temperature effect as calculated with MCNP with that calculated with cross sections generated with NJOY; a 97-group cross section for ZrH at 1200 K was generated with NJOY (methodology described above in item c) of Section II) and a unit cell calculation was performed with TWODANT. This comparison shows that there is only a 10% difference between MCNP and NJOY-TWODANT unit cell (MCNP is larger); since MCNP also uses NJOY for the process of cross section generation, it seems that the problem with the 1200 K ZrH cross section (if any problem does exist) might come from the ENDF/B-V thermal scattering (note that this is speculation, and the effort to try to figure out the reasons for this difference is beyond the scope of this work).

Calculations were also performed to qualitatively compare the Doppler reactivity effects for the fuel and the emitter. The results of these calculations for this Zr-H moderated concept are presented in Table VIII, and the single points calculated for the emitter and fuel are also shown in Figure 2 where it can be seen that as in TOPAZ these effects are negative and small. Again, in figure 2 it can be seen that the emitter Doppler effect is larger (absolute value) in this design, when compared to TOPAZ, and this is probably due to the emitter thickness difference.

One the drawbacks of the TOPAZ design is the large decrease in k-effective due to loss of hydrogen over life (diffusion at elevated temperatures). A calculation was performed for this Zr-H moderated design and for a 10% (weight percent) loss of hydrogen there is a 1.4% decrease in k-effective.

Finally, the neutron generation time, the delayed neutron fractions and decay constants were also calculated for this concept and the results are given in Table IX. It is important to note that as stated in Section II above the effect of photoneutrons from the Be reflector were not taken into account in the results presented above; a delayed neutron fraction

equal to 0.0072 was calculated. As a point of comparison, the delayed neutron fraction extracted from measurements in the SNAP Experimental Reactor (NAA-SR-5619), was equal to 0.0085 with an accuracy no better than 5 to 10%.

2. Rod Moderated Core: Be and ZrH as Moderators

The main problem with the ZrH moderated core was the lack of available control with control drums, which is mainly due to the large diameter of the core. The other negative characteristic of the ZrH moderated core is the large loss of reactivity with loss of hydrogen. The way to try to decrease these two problems was to design a core in which the moderator would be made out of ZrH rods enclosed in Beryllium. This would decrease the hydrogen loss and also make possible a smaller core; studies performed by Los Alamos have indicated that certain ratios of Be and ZrH would permit a smaller core than either moderator by itself.

This Be-ZrH rod moderated core containing 151 TFEs and 120 moderator rods, and a core with an equivalent diameter equal to 44.2 cm was proposed. Tables X, XI and XII present the characteristics for the TFE, Moderator Rod and Core, respectively. For this core, the first analysis performed was a burnup analysis, and the results indicated a 3.83% burnup swing reactivity. Next, the calculation of the worth of the control drums was performed, and as shown in Table XIII, a worth of 6.74% (Δk) was determined; this indicated, in principle, the feasibility of this concept. Then, safety analyses were performed with MCNP to determine the subcriticality of the reactor given a water submersion accident. The results of this analysis are presented in Table XIV, and it was determined that there would be problems with submersion. A first possibility to solve this problem would be to replace the central TFE with a safety rod (with Be follower) that would be withdrawn for reactor startup. Even with the inclusion of this safety rod the reactor would still not meet the criteria for submersion, as seen in Table XIV. Since this concept also had a lack of excess reactivity, sensitivity studies were performed to verify possible means of increasing the beginning of life reactivity; this analysis showed that an increase of about 2.7% in k -effective could be obtained by increasing enrichment and the size of the radial reflector (see Table XV). Finally, calculations were also performed to determine the moderator and radial reflector temperature coefficients; these results, presented in Table XVI, showed a positive moderator feedback (smaller than for the ZrH moderated concept) and apparently a very small positive radial reflector temperature reactivity coefficient (the uncertainty in the calculation was still too high when compared with the difference in k).

There were possible means to modify this design to meet the requirements. However, SPI tests/analysis indicated that even with the Be-enclosed ZrH, there would still be problems of loss of hydrogen to make the design uncertain, and a decision was made to discard this design.

3. Yttrium Hydride-Beryllium Moderated Core

Since the containment of hydrogen was still a problem even with the Be enclosed ZrH, SPI decided to substitute the Zr by Yttrium in the hydride. This substitution "potentially offers a high reliability, and a lower cost solution to the problem of hydrogen containment for the long life of the core; this is due to the fact that the YH dissociation pressure is 2 to 3

orders of magnitude lower than that of the ZrH at the operating temperature." In this concept the YH is also enclosed by Beryllium.

Three versions of this Be-YH moderated core were analyzed; the base case for which detailed analysis was performed, and two modifications to the base case for which, due to the lateness of the information and due to level of effort constraints, only partial analyses were performed.

a. Be-YH Moderated Core: Base Case

The characteristics of this core, of the TFE and of the central safety rod are presented in Tables XVII, XVIII and XIX, respectively. This core contains 150 TFEs in a hexagonal pitch equal to 3.4125 cm; the equivalent core diameter is 45.4 cm and the core active length is equal to 35 cm. Radially the core is enclosed in a 2 mm stainless steel vessel. Surrounding the vessel is the 10 cm Be radial reflector inside which there are 12 control drums containing Be and B_4C . Each drum consists of a 10.4 cm diameter solid block of Be with a 1 cm lens of enriched B_4C along its outer boundary. A ten centimeter axial reflector is placed in each end of the fuel, and at the time of this design the axial reflector geometry had not been completely defined and a homogeneous mixture of Mo, Be, SS316L and NaK was used.

First an MCNP analysis was performed to determine the control drums and safety rod worth for this core; the results of this analysis shown in Table XX, indicated that the drums, with a worth equal to 7.7% (Δk), would probably be able to meet the control requirements of this core (based on the previous concepts analyses).

Figures 3 and 4 present the normalized radial power at the core centerline and the normalized axial power at the TFE closest to the center of the core, respectively; in both cases the power profile is for the case with the control drums in the least reactive position, i.e. control drums out. From Figure 3 it can be clearly seen that there is a need for power flattening, however a decision was made not to work on power flattening in this phase of the design. Figure 5 where the normalized power in each TFE with the "control drums out" and the "control drums in" positions are presented, shows even clearer the need for power flattening.

Figure 6 shows the spectra in the fuel and YH moderators (from MCNP calculations). The peak at 0.1 eV region that has been reported in the work of other groups is clearly seen here in Figure 7 where the YH moderator spectrum is provided for only a small region of the spectrum. Since the spectra plot do not add much information about the characteristics of the reactor a plot of the normalized power and normalized flux in the fuel region is presented in Figure 8. In this figure, it can be clearly seen that about 40% of the fissions in this core occur in the energy above 1 keV, and other 40% occur between 1 eV and 1 keV. This shows that this concept is not a thermal reactor, but it is an intermediate spectrum reactor.

Figure 9 shows the worth of control drums as function of position of the poison section. The shape follows the S-curve form.

For the depletion analysis, EPRI-CELL cross sections were generated. As discussed before in the methodology section, the cross

section validation against MCNP was then performed and the results shown in Table XXI indicate an adequate agreement; note that in Table XXI the transport analysis used in the comparison with MCNP was performed with the 97-group cross section generated for the ZrH moderated core. The depletion analysis was performed with REBUS-3 and a burnup swing reactivity equal to 3.62% was calculated. The burnup reactivity loss as a function of operation time is shown in Figure 10 where a basically linear relationship is seen. The radial power profiles at the beginning and end of life (based on calculations with the control drums out) are shown in Figure 11, where it can be seen that there is almost no shift in power due specifically to burnup; it is clear that there will be shifts in power when the movement of the control drums is simulated in the burnup analysis, a complexity which was not modeled in this phase of the design.

Next, the temperature reactivity coefficients for the moderator, reflector, fuel and emitter were calculated using the methodologies described in section II. The results of these calculations are presented in Table XXII where it can be seen that the fuel and emitter Doppler effects are small and negative and not much different from those calculated for the ZrH moderated core (see Table VIII). However, even though the moderator temperature coefficient is still positive its magnitude is much smaller than that for the ZrH moderated core.

Using the results presented so far, a partial table for the excess reactivity requirements was constructed and is given in Table XXIII. It is seen from this table that the beginning of life core must have an excess reactivity greater than 7 % (Δk). Based on the worth of the control drums (about 7.7%) it is seen that the controllability of this core with control drums only is marginal at best. Also, note that the core as proposed has a k -effective equal to 1.014, and means to increase the BOL excess reactivity were then investigated. Table XXIV shows some of the possible ways to increase the BOL reactivity; it is seen in that table that by increasing the fuel enrichment, by better design of the axial reflector and by increasing the thickness of the radial reflector it is possible to make this core meet the BOL excess reactivity requirements.

Since this concept showed possibility of meeting the requirements, the next step was the safety analysis, i.e analysis of the water submersion accident, for which the results are presented in Table XXV; a plot for k -effective for the control drums and safety rod in different combinations for the operational case and for the submersion case is presented in Figure 11. From this figure and Table XXV, and keeping in mind that the BOL excess reactivity would have to be increased by at least 6%, it is clearly seen that this concept, without modifications, would not meet the submersion subcriticality requirement.

Finally, since a very detailed analysis had already been performed for this concept the determination of the neutron kinetics parameters and the reactivity loss due to possible hydrogen losses were also performed and the results are presented in Tables XXVI and XXVII, respectively. Note that, as in the case for the ZrH moderated core, the effects of photoneutrons from Be were not accounted for in the calculation of the neutron kinetics parameters presented in Table XXVI.

b. Be-YH Moderated Core: Modification 1

This core is basically the same as the base case and the only modifications were:

- increased Fuel Smear Density from 60 to 73 percent, and
- the homogeneous axial reflector used in the base case was replaced by a detailed model of the axial reflector.

Both of these modifications, proposed by SPI, would basically increase the beginning of life reactivity of the core. MCNP calculations were performed to determine the BOL k-effective and the worth of the control drums, and the results are presented in Table XXVIII. From this table, it can be seen that there was an appreciable increase in k-effective (from 1.014 in the base case to 1.056), but there was also a decrease in the worth of the control drums (probably due to shift in importance from the TFEs in the outer rings to those in the inner rings). The excess reactivity requirements presented in Table XXIII indicates that even with those modifications there would not be enough excess reactivity at BOL. A second problem with this modified core was that the control drums did not have enough worth for reactor control.

Since the detailed design of the axial reflector included a void region, a calculation in which Be replaces the void region in the axial reflector was performed and the results indicated a gain of about 2.5%; this means that this core would have enough reactivity at BOL.

However, the other problem with the base case core was related to water submersion, and, as expected, the results of the water submersion case for this modified core, shown in Table XXIX, clearly show that the base core with these modifications would not meet the submersion safety requirements.

c. Be-YH Moderated Core: Modification 2

The changes made to this core were basically directed at addressing the two problems with the design: water submersion subcriticality and controllability with the control drums. The following were the modifications:

- fuel smear density equal to 70 percent;
- fuel enrichment equal to 93 percent;
- detailed axial reflector with void region replaced by beryllium, and
- addition of 90 Gadolinia rods (3 mm diameter) located in the moderator around the first three rings of TFEs.

Clearly the first three modifications would add to the BOL k-effective, and the fourth modification would address the problem of controllability with the control drums. The fourth modification would also help in the water submersion problem. However, in order to be more effective

for the water submersion case, the gadolinia rods should be located in the outer rings of TFEs; this is because in the submersion case, the drums and radial reflector are considered to be lost, and in that case the use of gadolinia in the outer rings would be more effective in absorbing the neutrons reentering the core after thermalization in the surrounding water.

These modifications to the base case were received in the beginning of August and as such only the SPI requested calculations were performed. Calculations were requested for this modified core with and without gadolinia, and the results are described below.

1. No Gadolinia

The results are presented in Table XXX, where it can be seen that even though the excess reactivity meets the requirements, the reactor cannot be controlled with the control drums alone. Also, the water submersion case confirms that these modifications alone do not solve the water submersion problem.

Calculations were also performed, as requested by SPI, for two possible cases of fuel redistribution (shapes defined by SPI), and they are also presented in Table XXX. In both cases there is not much change in reactivity (only about $\pm 0.4\%$); even though no information was provided about these fuel redistribution cases, they seem to apply for normal operation and not for safety related purposes (speculation).

Finally, two calculations, for which the results are also presented in Table XXX, were performed to determine the reactivity effects due to:

(a) Radial and axial expansion (again dimensions provided by SPI): even with a very large number of histories in MCNP, the delta k obtained (very small and negative) is smaller than the uncertainties in the calculation.

(b) Density change for NaK (from room temperature to 800 K): again here the delta k obtained (very small and negative) was smaller than the uncertainties in the calculation.

2. With Gadolinia

The addition of gadolinia has the objectives of providing enough control with the control drums and providing enough subcriticality for the water submersion accident. The results presented in Table XXXI show that there is enough control worth in the control drums, however the reactor is still basically critical when submerged in water.

It is worth to note that this concept with gadolinia can be modified to meet all the operational and safety requirements, however to do this an iterative process will be required. Due to level of effort and time constraints this iterative process was not performed at this phase of the design.

B. 10 kWe Concept

As described in the Introduction, only a partial analysis was performed for one core design at this power level. This reactor contains six driver fuel elements and 48 thermionic fuel elements in a hexagonal pitch equal to 4.5 cm; the central element is a safety rod, and the drivers are located in the first row of fuel elements. The equivalent core diameter is 36.4 cm. The core active length is 35 cm, and the axial and radial reflector are the same as for the 40 kWe defined in Section III.A.3.c above. As in the last 40 kWe described above this core use Be and YH as the moderators. The characteristics of the core, TFE and driver fuel elements are presented in Tables XXXII, XXXIII and XXXIV respectively.

A depletion analysis using REBUS and the cross sections generated for the 40 kWe Be-YH moderated core was performed and a burnup swing equal to 2.5% was determined. The worth of the control drums was calculated and is presented in Table XXXV; it can be seen that controllability with the control drums will not be a problem for this concept.

Table XXXVI presents the results for the calculation of the temperature reactivity feedbacks for the moderator (MCNP analysis) and for the Doppler effects (based on a unit cell for the driver and for the TFE). As seen in that table, the Doppler effects are all negative and not much different from those calculated for the 40 kWe Be-YH design. However, the moderator feedback is positive and large as was the case for the ZrH core; note that the ratio of H to U in this core is much higher than that for the 40 kWe, and the driver elements are not enclosed in the MO-W emitter as is the case for the TFEs.

Finally, a water submersion analysis was performed and the results are presented in Table XXXVII, where it can be seen that this core will not have any problems in meeting the water submersion safety requirement.

In summary, it seems that this core can easily meet all the neutronic and water submersion safety requirements, but a complete analysis was not performed due to level of effort constraints.

IV. SUMMARY AND RECOMMENDATIONS

Analyses were performed for the design of thermionic cores at two different power levels (40 kWe and 10 kWe) and the methodology used and results obtained were presented in detail.

For the 10 kWe core, only a reduced analysis was performed, following the requirements of the statement of the work. This reduced analysis shows the feasibility of the proposed concept.

For the 40 kWe reactor core several concepts were analyzed in detail to provide a final concept that would meet all the design requirements.

The first concept analyzed was a design similar to Russian TOPAZ design, which uses zirconium-hydride as moderator. Even though this concept did not meet the requirements, a detailed analysis was performed to provide some comparison with data published in the literature. Most of the published data was qualitatively confirmed, and the analysis uncovered a possible problem with

the high temperature (1200 K) cross sections for the ZrH; this problem was later confirmed by analysis performed by LANL for a different design.

The next 40 kWe concept design analyzed used ZrH enclosed in Be to decrease the hydrogen loss (diffusion at the high operation temperature); this concept could meet the neutronic requirements, with some modifications, but loss of hydrogen would still be a problem. Thus, Yttrium was used to replace Zr in the next design analyzed; this substitution would provide the solution for the loss of hydrogen problem. Different designs using Be-YH as moderator were analyzed. Due to time/level of effort constraints, the analysis for the final concept for this phase of the project was not performed in detail. However, due to the similarities between the final concept (Be-YH Moderated Core: Configuration 2, described in section III.A.3 above), and the Base Case concept for the Be-YH moderated core it is recommended that the parameters evaluated for the base core be used to construct the characteristics of the final core for this phase of the design; obviously, the parameters will not be the same for both concepts, but the differences should not be large enough to make their use invalid. It is clear that in the next phase of the design, detailed analysis must be performed for what is defined as the "final" core in this report.

The analyses performed during this work have clearly shown that:

- There is a need for both deterministic and Monte Carlo design methods for the preparation of a defensible design.
- Generation of consistent cross sections for the specific designs is very important. For final designs, specific neutron cross-sections must be generated.
- The most appropriate methods for the generation of the temperature effects are dependent upon the magnitude of the parameters being evaluated. For the generation of the moderator temperature effects, the use of MONTE CARLO codes is feasible for designs in which this parameter has a large magnitude; if the reactivity change effect is small, the uncertainties in the Monte Carlo calculations can be larger than the effect itself even if a very large number of histories is used. For the determination of the Doppler reactivity effect, it seems that the use of deterministic codes is a must because of the magnitude of the reactivity changes; as stated in item a) above generation of consistent cross sections is essential.

Finally, it is important to note that for the next phase of the design consistent detailed analyses will be required, and all the required design codes are already available and the design methods are validated.

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Fig. 1 Flowchart of Analytical Approach

Fig. 2 Temperature Effect of Reactivity of the TOPAZ Converter Reactor with Single-Element TFE. 1 - Moderator; 2 - Tube Sheets; 3 - Reflector; 4 - Fuel Element; 5 - Electrodes of TFE; 6 - Total Isothermal Temperature Effect. (Reproduced from: "Choice and Refinement of Neutron-Physical Characteristics of the Converter Reactor for SNPS with Single-Element TIC," by Glushkov et al.).

Fig. 3 Normalized Radial Power for the YH Core (Base Case)

Fig. 4 Normalized Axial Power for the YH Core (Base Case)

Fig. 5 Normalized TFE Power for the YH Core (Base Case)

Fig. 6 Normalized Flux Spectra for the YH Core (Base Case)

Fig. 7 Normalized Flux Spectrum for the Moderator Region at Low Energy: YH Core (Base Case)

Fig. 8 Normalized Power and Flux for One TFE: YH Core (Base Case)

Fig. 9 YH Core (Base Case): Control Drums (12) as a Function of Rotation Angle

Fig. 10 YH Core (Base Case): Burnup Reactivity Loss with Operation Time

Fig. 11 YH Core (Base Case): Normalized Radial Power at BOL and EOL (Control Drums Out)

Fig. 12 YH Core (Case Case): K-Effective for Different Configurations

Table I. MCNP Cross Section Validation

A. Highly enriched U metal spheres reflected by Beryllium (from LA-367-MS, Rev. - page 13)			
Enrichment (% of U ²³⁵)	Reflector Thickness (in.)	Critical Mass (kg of U ²³⁵)	MCNP k-eff
93.9	1.84	22.2 ± .02	0.996540 ± 0.0012* 0.998996 ± 0.0011**
93.6	4.64	13.1 ± .2	0.994496 ± 0.0011* 1.000039 ± 0.0012**
93.9	1.89	21.6	0.996555 ± 0.0010
B. SNAP Homogeneous Model with 55.2 fuel elements (compositions and geometry from NAA-SR-5619)			
k (MCNP) = 0.997072 ± 0.0026			

* Critical Mass = Mean Value

** Critical Mass = Mean Value + One Sigma

Table II. 40 kWe Zirconium Hydride Core -
TFE Cross Section

Region	OR (cm)	Material	Density (g/cm)
Void	0.569	-	-
Fuel	0.90	UO ₂ (90% enr)	10.95
Emitter	1.095	Mo	10.2
Emitter	1.10	Nat. Tungsten	19.1
Gap	1.13	-	-
Collector	1.23	Mo	10.2
Gap	1.27	-	-
Sheath	1.345	Hastelloy w/o Tungsten	9.0
Coolant	1.42	NaK	0.865
Container	1.47	Hastelloy w/o Tungsten	9.0
Getter	1.52	Yttrium	5.51
Container	1.57	Hastelloy w/o Tungsten	9.0

Table III. 40 kWe Design Analysis Summary

Core	Characteristics	Problems
ZrH Moderated	Similar to TOPAZ	<ul style="list-style-type: none"> • Not enough control with drums • H_2 loss a problem • Large positive moderator temperature coefficient
Be-ZrH Rod Moderated	<ul style="list-style-type: none"> • H_2 loss reduction with Be Cladding • Lower Positive Moderator Temp. Coefficient • Smaller Core possible 	<ul style="list-style-type: none"> • Not enough excess reactivity • Submersion problem • H_2 loss still a problem
Be-YH Moderated: Base Case	<ul style="list-style-type: none"> • No H_2 loss problem • Lower positive moderator temp. coefficient 	<ul style="list-style-type: none"> • Not enough reactivity • Submersion problem
Be-YH Moderated: Mod. 1	Same as above, but: <ul style="list-style-type: none"> • Increased Fuel smear density (73%) • Detailed Axial reflector (with void region) 	<ul style="list-style-type: none"> • Not enough reactivity • Submersion problem
Be-YH Moderated: Mod. 2	Same as above, but: <ul style="list-style-type: none"> • 70% fuel smear density • Detailed axial reflector with Be replacing void region • Uses Gadolinia rods 	<ul style="list-style-type: none"> • Submersion problem <p>(solution feasible but requires iterative analysis not performed because of level of effort constraints)</p>

Table IV. 40 kWe Zirconium Hydride Core -
Core Characteristics

Number of TFEs	151
Hexagonal Pitch (cm)	3.85
Equivalent Core Diameter (cm)	52.2
Core active Length (cm)	35.0
U ²³⁵ Fuel Loading (kg)	70.0
U ²³⁵ Enrichment (%)	90
Fuel Smear Density (%)	60
Moderator	Zrh
Radial Reflector	
Material	Be
Thickness (cm)	10
Axial Reflector	
Material	Homogeneous Mixture of: Mo, Be, SS316L, and NaK
Thickness (cm)	10
Control Drums	
Number	12
Diameter (cm)	10.4
Material	Be with 1 cm lens of 90% enriched B ₄ C

Table V. 40 kWe Zirconium Hydride Core -
Cross Section Validation

I.	Core Analysis (with H ₂ loss)	
	MCNP	1.040672 ± 0.0018
	RZ Diffusion/15 Group XS	1.04482
	MCNP/RZ	0.99603
II.	Unit Cell Analysis	
	MCNP	1.420184 ± 0.0012
	Transport (S3P8)/97 Groups	1.402437
	MCNP/Transport	1.01265

Table VI. 40 kWe Zirconium Hydride Core -
Control Drums Worth

Drums Out	1.046988 ± 0.0023	
Drums In	1.024777 ± 0.0027	
Drums Worth (Delta k %)		2.22

Table VII. 40 kWe Zirconium Hydride Core -
Moderator Temperature Coefficient
(MCNP Analysis/Core Model/No Drums or Vessel/With 10 w% H₂ Loss)

Case 1: ZrH at 300 K	k = 1.039645 ± .0011
Case 2: ZrH at 600 K	k = 1.046198 ± .0009
	($\Delta k/k$)/°K = 2.101E-5 (from 300 to 600 K) ^a
Case 3: ZrH at 800 K	k = 1.050015 ± .0010
	($\Delta k/k$)/°K = 1.824E-5 (from 600 to 800 K) ^a
Case 4: ZrH at 1200 K	k = 1.064441 ± .0010
	($\Delta k/k$)/°K = 3.435E-5 (from 800 to 1200 K)

^amean value

Table VIII. 40 kWe Zirconium Hydride Core -
Temperature Reactivity Coefficients
[($\Delta k/k$)/°K]

Doppler:	
Fuel	-3.94E-6
Emitter:	
Mo	-2.49E-6
W	-1.06E-6
(Mo + W)	-3.54E-6

Table IX. 40 kWe Zirconium Hydride Core - Neutron Kinetics Parameters

Delayed Neutron Group i	Beta	Lambda 1/SEC
1	2.7532E-4	1.2720E-2
2	1.5325E-3	3.1740E-2
3	1.3523E-3	1.1600E-1
4	2.9283E-3	3.1100E-1
5	9.2095E-4	1.4000E+0
6	1.8707E-4	3.8700E+00
TOTAL	7.1965E-3	

- Neutron Generation Time: 2.640E-06 sec
- Prompt Lifetime: 2.752E-06 sec
- Effective Delayed Neutron Lifetime: 1.274E+01 sec
- The effect of photoneutrons from Be is not accounted for in these calculations.

Table X. 40 kWe Rod Moderated Core - TFE Cross Section

Region	OR (cm)	Material	Density (g/cm)
Void	0.5375	--	-
Fuel	0.85	UO ₂ (90% enr)	10.95
Emitter	0.995	Mo	10.2
Emitter	1.00	Nat Tungsten	19.1
Gap	1.03	-	-
Sheath	1.245	SS316L	7.9

Table X1. 40 kWe Rod Moderated Core - Moderated Rod Cross Section

Region	OR (cm)	Material	Density (g/cm)
Hydride	0.968	ZrH	5.5
Beryllium	1.1675	Be	1.85
Gap	1.20	-	-
Clad	1.25	Hastelloy w/o Tungsten	9.0

Table XII. 40 kWe Rod Moderated Core - Core Characteristics

Number of TFEs	151
Number of Moderator Rods	120
Pitch (cm)	2.525
Equivalent Core Diameter (cm)	44.2
Core Active Length (cm)	35.0
U ²³⁵ Fuel Loading (kg)	78.2
U ²³⁵ Enrichment (%)	90
Fuel Smear Density (%)	60
Moderator	ZrH + Be
Radial Reflector	
Material	Be
Thickness (cm)	10
Axial Reflector	
Material	Homogeneous Mixture of: Mo, Be, SS316L, and NaK
Thickness (cm)	10
Control Drums	
Number	12
Diameter (cm)	10.4
Material	Be with 1 cm of 90% enriched B ₄ C (120° arc)
Interstitial Material	Be + SS316L + NaK

Table XIII. 40 kWe Rod Moderated Core - Control Drums Worth

Drums Out	$k = 1.018531 \pm 0.0011$	
Drums In	$k = 0.951172 \pm 0.0023$	
Drums Worth ($\Delta k\%$)		6.74

TABLE XIV. 40 kWe Rod Moderated Core - Submersion Analysis

- Water Replaces NaK in Core and Axial Reflector
- Water Surrounds Reactor
- Base Case: $k = 1.018531 \pm 0.0011$

Configuration	k-Effective
Normal Configuration/No Water	1.018531 ± 0.0011
Drums In/Water Replaces NaK/Water Around Axial and Radial Reflector	1.025277 ± 0.0021
Drums In/Safety Rod In/Water Replaces NaK/Water Around Axial and Radial Reflector ^a	0.997932 ± 0.0021
• Submersion Problem	

^aCentral Safety Rod added to design

Table XV. 40 kWe Rod Moderated Core - Sensitivity Study

Base Case: Enrichment = 90% U^{235} Radial Reflector = 10 cm Be (between peaks)	$k = 1.018531 \pm 0.0011$
Case 1: Enrichment = 97%	$k = 1.038257 \pm 0.0023$ $\Delta k = 0.019726 \pm 0.0026$
Case 2: Enrichment = 97% Radial reflector thickness increased to 10 cm between flats	$k = 1.045312 \pm 0.0018$ $\Delta k = 0.026781 \pm 0.0021$

Table XVI. 40 kWe Rod Moderated Core Moderator/Reflector Temperature Reactivity Coefficients $[(\Delta k/k)/^{\circ}\text{K}]$

Base Case (all at 300 K)	$k = 1.018531 \pm 0.0011$
Moderator (ZrH and Be) at 800 K	$k = 1.024548 \pm 0.0011$ $\Delta k = 0.006017 \pm 0.0016$
Radial Reflector (Be) at 800 K	$k = 1.020466 \pm 0.0011$ $\Delta k = 0.001935 \pm 0.0016$
*Uncertainty still too large compared with Δk values; calculation to decrease uncertainty not performed because concept was dropped from consideration.	

Table XVII. 40 kWe Yttrium Hydride Core - Core Characteristics

Number of TFEs	150
Pitch (cm)	3.4125
Equivalent Core Diameter (cm)	45.4
Core Active Length (cm)	35.0
U^{235} Fuel Loading (kg)	69.5
U^{235} Enrichment (%)	90
Fuel Smear Density (%)	60
Moderator	YH + Be
Radial Reflector	
Material	Be
Thickness (cm)	10
Axial Reflector	
Material	Homogeneous Thickness (cm) Mixture of: Mo, Be, SS316L, and NaK
Control Drums	
Number	12
Diameter (cm)	10.4
Material	Be with 1 cm lens of 90% enriched B_4C
Safety Rod	
Follower Material	Be
Poison	90% enriched B_4C

Table XVIII. 40 kWe Yttrium Hydride Core - TFE Cross Section

Region	OR (cm)	Material	Density (g/cm)
Void	0.5692	--	-
Fuel	0.90	UO ₂ (90% enr)	10.95
Emitter	1.0375	Mo	10.2
Emitter	1.05	W (2 pass) ¹	19.1
Gap	1.08	-	-
Collector	1.18	Mo	10.2
Gap	1.22	-	-
Sheath	1.30	SS316L	7.9
Coolant	1.375	NaK	0.865
Clad for Be	1.40	SS316L	7.9
Be Moderator	1.55	Be	1.85

Table XIX. 40 kWe Yttrium Hydride Core - Central Control Rod

Region	OR (cm)	Material	Density (g/cm)
Follower	1.3	Be	1.85
Poison	1.3	B ₄ C (90% enr)	2.5
Coolant	1.375	NaK	0.865
Clad	1.4	SS316L	7.9
Be	1.55	Be	1.85

Table XX. 40 kWe Yttrium Hydride Core - Control Drums/Safety Rod Worth

Drums Out/Safety Rod Out	1.014147 ± 0.0007	
Drums In/Safety Rods Out	0.937389 ± 0.0017	
Drums Worth (Δ %)		7.68
Drums Out/Safety Rod In	0.986213 ± 0.0016	
Safety Rod Worth (K %)		2.79
Drums In/Safety Rod In	0.902900 ± 0.0016	
Drums Plus Safety Rod Worth (%)		11.12

¹ W-182/W-183/W-184/W-186 = 1 / 18 / 80 / 1 (weight %)

Table XXI. 40 kWe Yttrium Hydride Core - Cross Section Validation

I.	Core Analysis	
	MCNP	1.014147 ± 0.0005
	RZ Diffusion/15 Group XS	1.017136
II.	Unit Cell Analysis	
	MCNP	1.550869 ± 0.0007
	Transport (P3S8)/97 Groups	1.535071

Table XXII. 40 kWe Yttrium Hydride Core -
Temperature Reactivity Coefficients
[($\Delta k/k$)/°K]

Moderator (YH + Be) ¹	+3.1E-6
Radial Reflector Be) ²	+3.9E-6
Doppler: ³	
Fuel	-4.1E-6
Emitter:	
Mo	-2.4E-6
W	-6.8E-7
(Mo + W)	-3.1E-6
(Collector + Sheath) Mo	-2.4E-6

¹MCNP Analysis/Reactor Core Model/Temperature Range: 300 to 800K/Mean Value Reported

²Same as above

³1-D Transport (P3S8) Unit Cell/97-Group XS generated for the 40 kWe ZrH Moderated Core/Temperature Range: 300 to 1000 K

Table XXIII. 40 kWe Yttrium Hydride Core -
Summary of Reactivity Requirements

Burnup Reactivity Swing	+3.62	
TFE Doppler:		
Fuel (300 to 1908 K)	+0.66	
Emitter (300 to 1850 K)	+0.48	
(Collector + Sheath) Mo		
(300 to 1500 K)	+0.29	
Moderator (300 to 800 K)	-0.15	
Radial Reflector (300 to 800 K)	-0.20	
One Stuck Drum	+0.64	
Uncertainties (minimum)	+1.00	
Subtotal		+6.34
Radial Expansion	TBD	
Axial Expansion	TBD	
Axial Reflector	TBD	
Coolant	TBD	
Power Shaping Reactivity	TBD	
Total		TBD

Table XXIV. 40 kWe Yttrium Hydride Core - Sensitivity Study

Base Case:	
Enrichment = 90% U^{235}	$k = 1.014147 \pm 0.0007$
Radial Reflector = 10 cm Be	
Collector = Mo	
Case 1: Enrichment = 97%	$k = 1.036640 \pm 0.0011$
	$\Delta k = 0.02493 \pm 0.0017$
Case 2: Enrichment = 97%	
Weight Fraction in Homogeneous	
Axial Reflector:	
Case 2a: Increase Be by	$k = 1.052253 \pm 0.0020$
10% (decrease Mo by 10%)	$\Delta k \text{ (Case 2a-Case 1)} = 0.15610 \pm$
Case 2b: Increase Be by	0.0025
20% (decrease Mo by 20%)	$k = 1.062592 \pm 0.0018$
	$\Delta k \text{ (Case 2b-Case 1)} = 0.025950 \pm$
	0.0024
Case 3: Increase Radial Reflector	$k = 1.027589 \pm 0.0018$
Thickness to 12 cm	$\Delta k = 0.013451 \pm 0.0019$
Case 4: Replace Mo by Nb in	$k = 1.005753 \pm 0.0015$
Collector	$\Delta k = -0.008354 \pm 0.0016$
Case 5: Replace Homogeneous Axial	$k = 1.012573 \pm 0.0020$
Reflector by Detailed Axial	$\Delta k = -0.001574 \pm 0.0021^1$
Reflector	

¹Uncertainty too large when compared with Δk

Table XXV. 40 kWe Yttrium Hydride Core - Submersion Analysis

- Water Replaces NaK in Core and Axial Reflector
- Water Surrounds Reactor
- Base Case: $k = 1.014147 \pm 0.0007$

Configuration	k-Effective
Normal Configuration/No Water	1.014147 ± 0.0007
Drums In/Safety Rod Out/Water Replaces NaK/Water Around Axial and Radial Reflector	0.999050 ± 0.0021
Drums In/Safety Rod In/Water Replaces NaK/Water Around Axial and Radial Reflector	0.967408 ± 0.0010
Safety Rod In/No Radial Reflector/Water Replaces NaK/Water Around Vessel and Axial Reflector	0.984416 ± 0.0019

Table XXVI. 40 kWe Yttrium Hydride Core - Neutron Kinetics Parameters¹

Delayed Neutron Group i	Beta	Lambda 1/Sec
1	$2.6448E-4$	$1.2720E-2$
2	$1.4745E-3$	$3.1740E-2$
3	$1.3030E-3$	$1.1600E-1$
4	$2.8182E-3$	$3.1100E-1$
5	$8.8632E-4$	$1.4000E+0$
6	$1.8003E-4$	$3.8700E+00$
TOTAL	$6.9265E-3$	

- Neutron Generation Time: $1.267E-05$ sec
- Prompt Lifetime: $1.289E-05$ sec
- Effective Delayed Neutron Lifetime: $1.274E+01$ sec

¹The effect of photoneutrons from Be is not included in these results.

Table XXVII. 40 kWe Yttrium Hydride Core - Hydrogen Loss

Base Case	$k = 1.014147 \pm 0.0007$
5 w% Loss	$k = 1.005871 \pm 0.0014$ $\Delta k (5\%) = -0.008276 \pm 0.0016$
10 w% Loss	$k = 0.997507 \pm 0.0012$ $\Delta (10\%) = -0.01664 \pm 0.0014$
*Basically: $-1.66E-03 (\Delta k/\% H_2 \text{ loss})$	

Table XXVIII. 40 kWe Yttrium Hydride Core -
Modification 1 - Control Drums/Safety Rod Worth

Drums Out/Safety Rod Out	1.056448 ± 0.0021	
Drums In/Safety Rods Out Drums Worth (Δk %)	0.992458 ± 0.0020	6.40
Drums In/Safety Rod In Drums Plus Safety Rod Worth (%)	0.957652 ± 0.0022	9.88
*Not enough control worth in drums		

Table XXIX. 40 kWe Yttrium Hydride Core -
Modification 1 - Submersion Analysis

- Water Replaces NaK in Core and Axial Reflector
- Water Surrounds Reactor
- Base Case: $k = 1.056448 \pm 0.0021$

Configuration	k-Effective
Normal Configuration/No Water	1.056448 ± 0.0021
Drums In/Safety Rod In/Water Replaces NaK/Water Around Axial and Radial Reflector	1.024317 ± 0.0019
*Problems for submersion	

Table XXX. 40 kWe Yttrium Hydride Core - Modification 2

A. No Gadolinia

- A.1 Drums Out/Safety Rod Follower In
 $k = 1.079526 \pm 0.0006$
- A.2 Drums In/Safety Rod Follower In
 $k = 1.009061 \pm 0.0017$
Drums Worth = 0.070465 ± 0.0018
- A.3 Same as A.1 with Fuel Redistribution Axially
(shape provided by SPI)
 $k = 1.075501 \pm 0.0025$
- A.4 Same as A.1 with Fuel Redistribution Radially
(no central void in fuel region)
 $k = 1.083954 \pm 0.0021$
- A.5 Submersion/Safety Rod In/No Drums or Radial Reflector
 $k = 1.036091 \pm 0.0023$
- A.7 Same as A.1 but core dimensions and densities for hot isothermal
 $k = 1.079422 \pm 0.0009$
 $\Delta k = -0.000104 \pm 0.0011$
- A.8 Same as A.1 but NaK density at Hot Isothermal
 $k = 1.078837 \pm 0.0010$
 $\Delta k = -0.000689 \pm 0.0012$

Table XXXI. 40 kWe Yttrium Hydride Core - Modification 2

B. With Gadolinia

B.1 Drums Out/Safety Rod Follower In
 $k = 1.043814 \pm 0.0020$

B.2 Drums In/Safety Rod Follower In
 $k = 0.962836 \pm 0.0012$
 Drums Worth = 0.080978 ± 0.0023

B.3 Submersion/Safety Rod In/No Drums or Radial Reflector
 $k = 0.996532 \pm 0.0017$

* Problem with submersion

Table XXXII. 10 kWe Yttrium Hydride Core - Core Characteristics

Number of TFEs	48	
Number of Drivers	6	
Pitch (cm)		4.5
Equivalent Core Diameter (cm)	36.4	
Core Active Length (cm)	35.0	
U ²³⁵ Fuel Loading (kg)		
TFEs	27.1	
Drivers	10.2	
U ²³⁵ Enrichment (%)	90	
Fuel Smear Density (%)		
TFEs	73	
Drivers	85	
Moderator	YH + Be	
Radial Reflector		
Material	Be	
Thickness (cm)	10	
Axial Reflector		
Material	Heterogeneous (detailed)	
Thickness (cm)	10	
Control Drums		
Number	12	
Diameter (cm)	10.4	
Material	Be with 1 cm lens of 90% enriched B ₄ C	

Table XXXIII. 10 kWe Yttrium Hydride Core -
TFE Cross Section

Region	OR (cm)	Material	Density (g/cm)
Void	0.467	--	-
Fuel	0.90	UO ₂ (90% enr)	10.95
Emitter	1.0375	Mo	10.2
Emitter	1.05	W (2 pass) ¹	19.1
Gap	1.08	-	-
Collector	1.18	Mo	10.2
Gap	1.22	-	-
Sheath	1.30	SS316L	7.9
Coolant	1.375	NaK	0.865
Clad for Be	1.40	SS316L	7.9
Be Moderator	1.94	Be	1.85
¹ W-182/W-183/W-184/W-186 = 1 / 18 / 80 / 1 (weight %)			

Table XXXIV. 10 kWe Yttrium Hydride Core -
Driver Cross Section

Region	OR (cm)	Material	Density (g/cm)
Void	0.562	-	-
Fuel	1.45	UO ₂ (90% enr)	10.95
Clad	1.525	SS316L	7.9
Coolant	1.600	NaK	0.865
Clad	1.625	SS316L	7.9
Be	1.94	Be	1.85

Table XXXV. 10 kWe Yttrium Hydride Core -
Control Drums/Safety Rod Worths

Drums Out/Safety Rod Out	1.039175 ± 0.0021	
Drums In/Safety Rod Out	0.965775 ± 0.0020	
Drums Worth (Δk %)		7.34
Drums In/Safety Rod In	0.903630 ± 0.0015	
Drums Plus Safety Rod Worth (Δk %)		13.61

Table XXXVI. 10 kWe Yttrium Hydride Core -
Temperature Reactivity Coefficients
[$(\Delta k/k)/^{\circ}\text{K}$]

Moderator:	Core (YH + Be) ¹	+2.4E-5
Doppler: ²		
A)	TFE	
	Fuel	-4.5E-6
	Emitter	-2.3E-6
	(Collector + Sheath) Mo	-1.6E-6
B)	DRIVER	
	Fuel	-4.5E-6
	Mo	-2.0E-8

Table XXXVII. 10 kWe Yttrium Hydride Core -
Submersion Analysis

- Water Replaces NaK in Core and Axial Reflector
- Water Surrounds Reactor
- Base Case: $k = 1.039175 \pm 0.0021$

CONFIGURATION	k-EFFECTIVE
Normal Configuration/No Water	1.039175 ± 0.0021
Drums In/Safety Rod In/Water Replaces NaK/Water Around Axial and Radial Reflector	0.941820 ± 0.0020
Safety Rod In/No Radial Reflector/Water Replaces NaK/Water Around Vessel and Axial Reflector	0.951466 ± 0.0025

¹MCNP Analysis/Reactor Core Model/Temperature Range: 300 to 800K/Mean Value Reported

²1-D Transport (P3S8) Unit Cell/97-Group XS generated for the 40 kWe ZrH Moderated Core/Temperature Range: 300 to 1000K

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