BNWL-1333
UC-80

4-2-70

FFTF SHIELDING PROGRAM

W. L. Bunch

April 1970

AEC RESEARCH & DEVELOPMENT REPORT

REPORT NO.	PROJECT NO.	DATE	BY

BNWL-1333

BATTELLE

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UNITED STATES ATOMIC ENERGY COMMISSION UNDER CONTRACT AT(45-1)-1830

3 3679 00061 9165

BNWL-1333
UC-80, Reactor
Technology

FFTF SHIELDING PROGRAM

By

W. L. Bunch

Reactor and Plant Technology Department
FFTF Division

April 1970

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FFTF SHIELDING PROGRAM

I. INTRODUCTION

A shielding program was initiated in 1966 to support the design of the FFTF. This program subsequently has evolved and been coordinated with the LMFBR shielding program to the extent possible in order that information might be attained in a general form that has broad applicability. However, the timing of this program is dictated by the design schedule requirements of the FFTF project. Shielding of nuclear radiation sources is required to protect both operating personnel and equipment. Personnel protection requirements are established for AEC contractor activities by AEC Manual, Chapter 0524, Standards for Radiation Protection. The AEC standards are based on limiting the accumulated (lifetime) dose to operating personnel as well as limiting quarterly (13 consecutive weeks) exposure rates. For design application, these limits must be appropriately distributed on a time basis over all of the radiation work to be performed in order that accumulated doses of all operators remain within the established values. Battelle manuals BNWL-MA-3 and BNWL-MA-6 are being used as guides to provide a facility design that is operable within the AEC requirements. For materials, no simple set of standards exists, and consideration must be given to the effect of radiation on the material as well as other environmental effects, replaceability of the component, and the physical property of the material that is limiting.

Ideally, a shield should be designed to meet exactly the established criteria. Because of uncertainties associated with any calculated value, the design must make allowance for these uncertainties in order that it function adequately. The basic purpose of the shielding developmental program is to provide analytical techniques, nuclear data, and experimental results that are adequate for design application. Applicability for design use implies that the intensity and spectral distribution of the radiation field can be calculated at any point of interest with sufficient accuracy to assure that the design criteria are met. There is an attendant

desire to improve the accuracy of the calculation in order to make the shield as precise as possible and thereby eliminate the need for excessive costs.

No amount of development work or shield experimentation will make a good shield out of a poor design. The developmental work provides the analytical techniques and nuclear data required by the designer to achieve a satisfactory design. These tools can be used to evaluate various potential concepts or design approaches in order to select the most satisfactory shield arrangement for each radiation source. In general, it is much more practical and economical to evaluate arrangements analytically than to fabricate and evaluate experimentally a mockup that approximates the shield of interest.

The following sections discuss the shield program being carried out to support the design of the FFTF. As indicated, much of the information that is required is available or has been obtained already under this or earlier programs. In some cases, the need for development work is contingent upon the design; that is, in one conceptual arrangement a particular piece of equipment might be adequately shielded because of its location, whereas in another arrangement the inclusion of an adequate shield might be quite difficult. In many cases shielding considerations are not limiting and the shielding function is provided by materials whose primary function is something else. The shield is often integral with the rest of the design and may permit numerous alternate approaches. If adequate analytical techniques and nuclear data are available, these alternates can be evaluated by design calculations. If adequate confidence cannot be placed in the calculations because of complex geometries, special materials, inadequate cross section data, or other uncertainties, then experimental and analytical development work is required. Such experiments can be directed at simply evaluating the particular shield arrangement of interest; however, a more practical long range approach is to design experiments that provide a general solution to the class of problems in order that any future problem of a similar type can be solved satisfactorily without further experimentation.

In the paragraphs that follow, the FFTF shield program is discussed by project events that require specific shielding input information. Only those events are included where shielding considerations have a significant impact on the design. In some cases only potential problems are associated with the event at this time because the design is not completely defined. An indication is also given of available information that is germane to the problem. In many cases the available information will provide an adequate basis for the design unless some unusual problem is encountered.

II. SUMMARY

There are two basic experimental developmental areas in which additional effort is indicated to support the design of shields in the FFTF. These are: (1) penetrations, such as through the vessel cover and through the reactor cavity wall, and (2) special geometric arrangements of sodium and other shield materials in the lower axial region between the core and the support structure. On the basis of the available information and the impact of the development program on the project events discussed in this program plan, it is concluded that first priority should be given to the penetration experiments. Essentially no experimental data are available to provide a firm basis for the design of these penetrations or for evaluating the uncertainty associated with any particular design. Second priority is given to experiments in support of the design of the axial shield below the core. Some experimental information has been obtained for slab and pin-type shield arrangements that does provide a basis for the design of this region; however, additional experiments are indicated to refine and optimize the design.

In addition to the experimental developmental work, it is required that analytical techniques be refined and developed, and that nuclear data be upgraded and updated, for use in the design. The development of the analytical techniques is a parallel and companion effort to the experimental program. The analytical techniques are developed and validated on the basis of the experimental results to provide methods that are applicable for the designs of interest. Of primary importance are techniques for evaluating streaming effects through complex shield penetrations, and practical, economic methods for evaluating detailed two- and three-dimensional shield problems.

Lesser priority is given to obtaining basic nuclear data, such as the $^{18}\text{O}(\alpha, n)$ reaction cross section as a function of energy, because of timing and impact on the design.

III. DISCUSSION

A. PLACE ORDER FOR REACTOR VESSEL MATERIALS, MARCH 1970

1. Background

Prior to ordering the reactor vessel materials, the overall dimensions of the vessel must be established. At the present time, the layout and structural requirements of the vessel head dictate the basic diameter of the vessel. Within the diameter set by the head requirements, there is ample space between the core and the vessel to accommodate a reactor vessel shield that satisfies current radiation damage criteria for the core support structure, reactor vessel, and guard vessel. Storage of irradiated fuel in the annulus between the radial shield and the reactor vessel could influence the radial shield design but is not expected to influence vessel diameter. A number of alternate radial shield arrangements could be employed.⁽¹⁾ The length of the vessel is dictated primarily by the fuel element together with the pool depth requirements for in-vessel handling of core components. The thickness of the cover shield depends on the depth of the pool, together with the layout and arrangement of closed loop piping, control rod drives, refueling and instrument access requirements, and design of the work deck above the reactor. Personnel exposure rates above the work deck are to be less than 0.2 mrem per hour during reactor operation at full power. Access to the head region below the deck is to be practical following shutdown of the reactor and replacement of the inert atmosphere with air. Radiation damage to the electrical components associated with the control rod drives is a primary consideration both during operation and replacement of driver and test fuel. These latter considerations play relatively minor roles in meeting this event, but are important as detail design progresses.

In order to establish that adequate shields can be incorporated within the basic vessel dimensions, it is required that the attenuation characteristics of sodium, stainless steel, nickel, and other conceptual

shield materials of interest be known with sufficient accuracy that uncertainties can be accommodated. As the design progresses, accuracy requirements will become tighter.

Available Information

Attenuation characteristics of materials of interest is available primarily as a result of the series of attenuation measurements carried out in the Shield Test and Irradiation Reactor (STIR) facility operated for the AEC by Atomics International. These measurements include: (1) attenuation characteristics of sodium,⁽²⁻⁴⁾ which are required to determine the effects of varying the pool depth between the reactor core and the cover, (2) attenuation characteristics of an iron-sodium shield whose nuclear composition approximates that of the region between the Fast Test Reactor (FTR) core and the structural grid plate,⁽⁵⁾ and (3) attenuation characteristics of a stainless steel-sodium shield whose nuclear composition approximates that of the radial shield region between the FTR and the vessel.⁽⁶⁾ The overall nuclear composition of the slab shields employed in the experiments, and the distances employed, are typical of FTR conceptual shield compositions, though not of the physical material arrangements. These experiments provide preliminary information as to the effectiveness of the FFTF shield design, but more importantly establish the validity of analytical techniques and nuclear data employed in the design.

In addition to the above experimental information, other data reported in the literature can be employed to determine the validity of analytical techniques and nuclear data sets to be employed in the design of FTR shields. Available experimental data include attenuation of neutrons in sodium,⁽⁷⁻⁸⁾ attenuation measurements in fast reactor shields,⁽⁹⁾ and neutron spectral measurements through metal slabs.⁽¹⁰⁾

3. Development

Comparison of the measured and calculated spectral and spatial distribution of the neutron flux in sodium indicates that analytical techniques

and nuclear data employed by BNW in the conceptual design of the FTR shields are valid to depths of about five or six feet.⁽¹¹⁾ At greater depths the discrepancy between the calculations and measurements is attributed in part to limitations in the experiment. Similar good agreement was achieved in the case of the evaluation of the radial shield experiments at the vessel location.⁽¹²⁾ Because no spectrometer measurements were obtained in the axial shield experiment,⁽⁵⁾ it is not possible to determine the validity of the calculated spectrum at the grid plate.⁽¹³⁾ A measurement of the neutron spectrum at the simulated grid plate location would help to validate shield design calculations for placement of this key structural member.

It is concluded that adequate shielding information is available to satisfy this event and no additional development is required.

B. COMPLETE PRELIMINARY DESIGN OF THE CORE SUPPORT STRUCTURE, JULY 1970

1. Background

The core support structure is an important component located within the reactor vessel. In order to locate and design this item, it is necessary to define its radiation environment. In addition to the information discussed in Section III-A, it is necessary to be able to define the fission power distribution in stored fuel and the impact of the stored fuel on the flux distributions. Whereas the vessel is located at a relatively large distance from the core, the core barrel and support structure are located essentially at the minimum acceptable distance insofar as radiation damage is concerned. Therefore, a higher precision is required than for Section 111-A.

2. Available Information

Basic attenuation characteristics of the radial reflector and vessel shield region were measured as reported in Reference 6 and discussed in Section 111-A. In addition, an experimental evaluation of the radial

reflector-shield configuration was completed in the Zero Power Plutonium Reactor (ZPPR) loading FTR-2. These measurements included axial and radial neutron and gamma distributions in a mockup similar to FTR. Because of size limitations in the ZPPR facility, it was not possible to include the sodium region between the core barrel and the reactor vessel; therefore, the effect of fuel stored within the radial shield was measured. The experiments were completed by the end of 1969.

3. Development

Part of the developmental work required to meet this project event was carried out in the ZPPR/FTR-2 experimental program. Agreement between the ZPPR/FTR-2 experimental results and preliminary analyses completed by BNW⁽¹⁴⁻¹⁶⁾ indicate that the radiation environment in the vicinity of the core barrel can be predicted with sufficient accuracy to meet this project date, including the perturbations introduced by stored fuel. More refined analyses might indicate the need for subsequent development work in this area as the design evolves.

Additional development work is indicated for the region between the core and the lower core support structure. The sodium coolant to the driver fuel, open loop test assemblies, in-core safety and control rods, and peripheral control rods must pass through this region. It has not been established that the discrete geometric arrangement of sodium and shield materials will provide as effective a shield as the homogeneous approximations normally employed in design calculations. Westinghouse has submitted a preliminary test specification for experiments based on the current axial shield design. This development is required to complete the design of core components (driver fuel, control rods, closed loops) that will satisfactorily protect the support structure from core radiations. Existing information provides an adequate basis for preliminary design of the core support structure.

C. START FABRICATION OF REACTOR VESSEL, AUGUST 1970

1. Background

When fabrication of the vessel begins, all design features must be sufficiently evaluated to assure the necessary functions can be accommodated. As indicated in Section III-A, the vessel diameter is dictated by head space requirements. As the equipment arrangements within the vessel become firm, it will be necessary to calculate the radiation environment of each component within the reactor vessel and to determine whether or not the criteria are met. Items of interest from a shielding standpoint include radiation damage to the core barrel and support structure, to in-vessel handling equipment, instrument trees, and the reactor vessel; heat generation rates in stored fuel positions; in-vessel neutron sensor environment; and nuclear heat generation rates in components located near the core.

Another important factor in assessing the reactor vessel dimensions prior to fabrication is the effect of axial shielding requirements and penetration geometry in the vessel cover. The attenuation requirements of the vessel cover are integrally related to the depth of the pool above the core and to the arrangement of materials within the pool. Neutron streaming through the fission gas plenums in the fuel assemblies will influence the flux distribution in the pool, as will the physical arrangement of control rod extensions, instrument trees, in-vessel handling equipment, closed loops, and other in-vessel components. The analytical problem of realistically representing these components in attenuation calculations also must be resolved. The limitations of annular duct streaming formulas currently available and their impact on the cover design are considered in Section III-G.

2. Available Information

Measurements were made in the Zero Power Plutonium Reactor (ZPPR) assembly FTR-2 of the neutron and gamma intensity distributions in the core, radial reflector, and radial shield of a nuclear simulation of the FTR. Also included in the experiment was a measurement of the perturbation introduced by storing fuel in the shield region. These measurements were made in the radial shield since size limitations precluded inclusion of the sodium annulus between the core barrel and the reactor vessel.

Although these experiments provide a direct measure of the radiation environment in a simulation of the FTR, they also provide a firm basis for evaluating analytical techniques and nuclear data for use in the FTR. These measurements were completed near the end of 1969 and evaluation of the results is in progress. Preliminary experimental information obtained at ORNL employing pin-type axial shield knockups indicates the energy dependence of neutron streaming through the sodium-filled interstices.

3. Development

Agreement between the experimental results from ZPPR and preliminary analyses completed by BNW^(14,15) indicates that no additional development is required insofar as vessel diameter is concerned. Shielding that is associated with the fuel assembly requires further development to establish the radiation environment for the core support structure. This shielding is included in Section III-F, which discusses preliminary design of driver fuel assemblies. Development work associated with the design of penetrations through the cover is discussed in Section III-G. The vessel length is dictated basically by the 12-foot driver fuel length, together with the pool depth requirements for handling such an assembly. Given this vessel length, the shield development effort must be directed at achieving a satisfactory vertical arrangement of components within the vessel, and in finalizing the cover design.

D. INITIATE FABRICATION OF THE PROTOTYPE FUEL HANDLING EQUIPMENT, OCTOBER 1970

1. Background

Driver, test fuel, and closed loops are to be removed from the core and transported to interim storage employing a shielded handling machine. The irradiated fuel will be immersed in sodium to provide a heat sink and heat transfer medium to the gas coolant in the handling machine. It is necessary that adequate cooling be available to assure integrity of the fuel during the transfer process. In addition, adequate shielding must be utilized in the walls of the machine to protect operating personnel.

Because of the specific power of the fuel and the high exposure, a significant amount of shielding is required. From weight and space considerations, depleted uranium might be employed. The use of depleted uranium as part of the shielding of the fuel handling equipment will result in a transient increase in the neutron radiation level above the vessel cover and operating deck during fueling operations. This effect will occur because of fission events induced in the depleted uranium by neutrons from spontaneous fission and (α, n) reactions in the fuel.

The evaluation of the (α, n) neutron source strength in non-irradiated fuel together with the neutron spectrum from this reaction requires reexamination. Furthermore, the validity of homogenizing the fuel subassembly materials to calculate the gamma flux distribution external to the subassembly is in question.

2. Available Information

For the decay times of interest, a considerable amount of experimental data are available for fission products generated in uranium fuels irradiated in a thermal-neutron environment. Essentially no such data is available for plutonium fuels irradiated in a fast-neutron environment. By considering the different fission product yields for these two types of fuels, it is possible to calculate the radiation levels and the shielding requirements for plutonium.^(17,18) Such calculations are adequate for conceptual and preliminary design purposes.

Only gross yield data for the (α, n) reaction is available for thick oxygen targets and $^{238}\text{PuO}_2$ sources. In addition, all the associated spectral data for the emitted neutrons are not corrected for multiple scattering within the source. Furthermore, the data do not permit extrapolation to mixed UO_2/PuO_2 sources.

3. Development

The details associated with the design and operation of the fuel handling equipment for the FTR are not sufficiently well defined to establish the need for development work in the shielding area. The analytical

techniques and nuclear data that are currently available appear to be satisfactory. However, if subsequent evolution of the design indicates key areas in which shielding requirements will be critical, this conclusion will need to be reevaluated.

Fundamental measurements of the $^{18}\text{O}(\alpha, n)^{21}\text{Ne}$ neutron emission spectrum are desired. To evaluate the neutron yield strength in the fuel mixture, measurements of the (α, n) energy differential cross section are required.

E. FIRST PLACEMENT OF CONCRETE, DECEMBER 1970

1. Background

Prior to placement of concrete it is necessary to establish basic shield dimensions throughout the facility and to establish space requirements. The thickness of a given concrete shield wall depends on the strength and the spectral and spatial distribution of the source to be shielded together with the radiation zone requirements outside the shield wall. Those regions that require continuous occupancy are to be shielded to limit radiation levels to below 0.2 mrem/hr. Radiation zones in which either sources or personnel will be present only on an intermittent basis can be shielded to produce radiation rates in the range 0.2 to 1000 mrem per hour. The specific design guidelines given in manual BNWL-MA-3 (19) are being employed as an acceptable method of meeting the requirements established in AEC Manual Chapter 0524.

Primary sources of interest are associated with the sodium coolant used in both the reactor and the closed test loops. For sizing the reactor cavity wall it is necessary to know the intensity and spectral distribution of the radiation from the reactor vessel. Given these sources of radiation and the applicable criteria, shield thicknesses can be calculated.

2. Available Information

Attenuation characteristics of a large number of different types of concrete were measured as a function of temperature for reactor radiations. The results are available in a number of AEC reports.⁽²⁰⁻²⁶⁾ In addition.

attenuation calculations for gamma radiation through bulk shields are well established.

The activation of primary reactor and closed loop sodium can be predicted on the basis of the neutron flux distribution throughout the sodium. These calculations can be evaluated partially by comparison to the ZPPR/FTR-2 shield experiment results in which sodium reaction rates were measured. Because of the extremely high radioactivity level induced in the sodium, impurities in the sodium will not contribute to primary shield requirements during reactor operation.

3. Development

Although the detail design of the reactor and heat transport system components will influence the sodium activation levels, these changes are expected to be minor and should not influence the basic concrete wall dimensions in heat transport cells, closed loop cells, pipe spaces, and other building features. Likewise, corrosion products and fission products in the coolant will influence access requirements to cells, but will not influence wall dimensions. No shield development work is required to meet this project date.

F. COMPLETE PRELIMINARY DESIGN OF DRIVER FUEL ASSEMBLIES, DECEMBER 1970

1. Background

The driver fuel assembly length has been set at 12 feet. Given this length requirement, it is necessary to establish the position of the fuel zone with respect to the core support structure and the instrument plate. The core support structure is to have a design lifetime of 10 years; therefore, the attenuation characteristics of the region between the fuel and the structure must be known with sufficient accuracy to assure this lifetime. The design of the reflector-shield region below the fuel in each assembly must consider not only the attenuation characteristics of the material, but also its compatibility with the high temperature sodium environment. In addition, pressure drop of the coolant through the region should be minimized and the shield should not perturb the flow distribution

to the fuel in a deleterious manner. Although not directly affecting the design of the driver fuel assembly, it is also required that the attenuation characteristics of the region between the core support structure and the core region in closed loops, test assemblies, and control, safety, and shim rods be comparable to that provided by the driver fuel. The length of the driver subassembly could be affected if the shielding requirements of these other components cannot be met with the present clearance between the fuel and the core support plate.

2. Available Information

The basic attenuation characteristics of an iron-sodium slab shield, whose nuclear composition was similar to that of the conceptual design of the FTR, were measured in an STIR experiment.⁽⁵⁾ In the STIR experiments, the nuclear composition was achieved by the use of alternate slab layers of iron and sodium. In the conceptual arrangement the reflector and shield regions were to employ cylindrical pins that would be parallel to the sodium flow path. Thus, the shield between the core and the support structure would contain essentially pure sodium paths. In order to investigate possible anomalies that might be introduced by such an arrangement, a series of experiments was completed by ORNL at the Tower Shield Facility (TSF). In these experiments, steel pins were cast in sodium to form a square matrix containing the interstitial sodium region resembling that of the conceptual design. In addition, measurements were taken through pure steel and through pure sodium shields. A number of analyses were undertaken by ORNL personnel in order to interpret the experiments. Although not complete, the preliminary results indicate that bulk leakage of the total neutron flux through the pin-type assembly is enhanced by 10%-25% over that of a homogeneous assembly of the same composition. At high energies, the leakage is enhanced by a factor of 2-4; however, very little flux remains at these energies. Based on these preliminary results, it was decided to reduce the streaming component by using a helically fluted shield block between the bottom reflector and the core support plate.

3. Development

Evaluation of the pin-type shield arrangement measured by ORNL should be completed in order that the attenuation phenomena that are taking place can be appropriately accounted for in other similar arrangements. It will also be necessary to establish whether or not such phenomena will impact on the efficiency of the helical grooves currently being employed in the conceptual design. In addition to completing the preliminary design of the driver fuel assembly, it is also necessary by this event data to complete the design of the closed loops and the in-core and peripheral control rods to assure that the entire core support structure can be adequately protected at the established distance from the core. Additional developmental work is required to meet this project event. Westinghouse has submitted a preliminary test specification for further experiments to verify the attenuation characteristics of the latest axial neutron shield concept below the driver fuel.

G. INITIATE MACHINING OF THE REACTOR COVER, JANUARY 1971

1. Background

The design of the reactor cover is complicated by the need to provide openings to accommodate control rod drives, closed loops, instrumentation, refueling access, and mechanical restraint mechanism drive shafts. The control rod drive mechanisms need to be located in a sufficiently low-level radiation environment to eliminate the effects of radiation damage and to permit ready access during outages for maintenance work. The radioactive closed loop coolant is brought through the cover and provides a source of radiation exterior to the cover that must be shielded. In order to provide a flat working deck with unlimited personnel access during operation, the control rod drive mechanisms in the conceptual arrangement are located in an inert gas space between the reactor cover and the deck. Within this inerted space, the closed loop nozzles and piping are housed in a shielded structure. A multiple shield arrangement is thus employed. The cover must attenuate the neutron and gamma radiation from the core and sodium pool sufficiently to provide a suitable environment for the control rod

drive mechanisms during operation and to suppress activation of components external to the cover. The closed loop piping and nozzles are likewise shielded for this reason. Following shutdown and replacement of the inert atmosphere with air, the cover and closed loop shields must provide adequate attenuation of the gamma radiation to permit manned access to carry out the necessary maintenance operations. During discharge of the closed loops and of the driver fuel, adequate shielding must be provided to protect equipment and personnel from the gamma radiation. As in any shielding problem, it is necessary to be able to define the spectral and spatial intensity of all radiation sources at all receptor points of interest in order to design the shielding. Because of the numerous functions to be achieved, and the many alternatives that could be employed for each, the overall design of the vessel cover provides a complex challenge.

2. Available Information

Analytical techniques are available for calculating the radiation leakage through simple line-of-sight penetrations.⁽²⁶⁾ For more complex penetrations, in which steps or offsets are employed to eliminate the line-of-sight paths, the validity of analytical solutions has not been established. In the case of the control rods, it will probably be necessary to employ simple annular openings between fixed shield components and the movable rods required to translate poison into and out of the core. Leakage through these line-of-sight paths can be estimated realistically with present analytical techniques. Preliminary calculations indicate that the leakage rate from an individual control rod drive shaft can be tolerated because of the relatively small radii that are involved. Closed loops also appear to require line-of-sight gas spaces to provide insulation and separation of the inlet and outlet streams. These beams will be intercepted by structural material in the loop nozzle. In addition, the loops will be embedded within the required secondary gamma shield. The cover openings will employ steps for mechanical considerations. These steps also serve as radiation barriers to eliminate line-of-sight paths. However, because of the large number of openings that must be

provided, the steps must be minimized in order to retain structural integrity in the cover. **It** is necessary to integrate structural and shield requirements in establishing the number, size and location of steps and the clearances and tolerances that must be employed.

3. Development

It is necessary to have a parametric study made of the attenuation characteristics of a stepped, annular duct by varying the size of the offset, its position, and the clearances involved in the range of interest for the design of the cover. No firm development program in this area has been established at the present time. A preliminary test specification has been submitted by Westinghouse for experiments to determine the extent of radiation streaming through clearances in the FTR vessel head penetrations and to provide a firm basis for evaluation of this effect. **It** is proposed that continued consideration be given to initiating additional developmental programs to support design of the cover shield when specific problem areas are defined. One specific problem area may be neutron streaming through the fission gas plenums and its impact on the vessel cover design. In addition to experimental effort, **it** is necessary to carry out methods development for application of the results to the actual design.

H. COMPLETE FINAL DESIGN OF THE REACTOR PRIMARY HEAT TRANSPORT SYSTEM, FEBRUARY 1971

1. Background

Mixed plutonium- and uranium-oxide fuel will be used in the FTR, with about 90% of the fission events taking place in plutonium. In the event of a reactor shutdown prompted by malfunction of the heat transport system, the design should satisfactorily accommodate removal of the decay heat associated with these fission products to prevent overheating of the fuel. The decay heat as a function of time for the first few hundred seconds following a rapid shutdown is extremely important in establishing the required coastdown characteristics of the heat transport system and the eventual reliance on convection cooling. The requirement is to limit

cladding temperatures following shutdown to approximately normal operating temperatures.

2. Available Information

Almost all of the experimental data associated with fission products was obtained through the study of the thermal-neutron induced fission of ^{235}U . These studies have been of many types: fission yield of individual isotopes, cumulative yield of mass chains, charge distribution within mass chains, half life of individual isotopes, decay energy associated with beta particles and photons for individual isotopes, gross gamma associated with mixed fission products, gamma spectrum associated with mixed fission products, beta energy associated with mixed fission products, decay heat associated with mixed fission products, delayed neutrons associated with mixed fission products, etc. The experiments were carried out over a number of years, with the results exhibiting inconsistencies that are not resolved. Very little effort has been expended at investigating fission products associated with the fast-neutron induced fission of ^{239}Pu . Although chain yields are shown to be significantly different, very little difference is found between the measured decay heat rates associated with uranium and plutonium fission products at long decay times. Detailed calculations⁽²⁷⁻²⁹⁾ indicate that decay heat from plutonium fission products will be somewhat less than that from uranium, especially during the first few hundred seconds. These results are substantiated by the small amount of existing experimental data available.⁽³⁰⁻³²⁾ It is concluded that a design based on the use of ^{235}U data would be conservative.

3. Development

It would be desirable to undertake an experimental program to determine precisely the decay heat associated with fast reactor fuels; however, a review of the literature indicates there to be many shortcomings in the ^{235}U thermal reactor data, which has been obtained over the past quarter of a century. For this reason, it seems unlikely that an adequate development program can be defined and completed in time to contribute to meeting

this project date. **It** is strongly recommended that such development work be undertaken on behalf of the LMFBR. Although **it** will be possible to employ a conservative approach in the design of the FFTF heat transport system, **it** is undesirable and expensive to do so for any large number of reactors.

■. PLACE THE ORDER FOR NEUTRON DETECTORS FOR THE REACTOR INSTRUMENT SYSTEM, FEBRUARY 1971

1. Background

In order to operate the FTR safely and efficiently, **it** is required that a neutron-monitoring system be designed to provide essentially continuous and instantaneous operating information concerning the neutron flux level in the core. The nuclear environment at each sensor location depends on the deployment of shielding as well as the reactivity or power status of the core. Storage of fuel in the sodium annulus between the core barrel and the reactor vessel will also influence the spatial distribution of the neutron flux, as will the movement of peripheral and in-core control, safety and shim rods. The gamma environment during operation will largely be dependent on capture gamma radiation; whereas, following shutdown, the activation gamma radiation will predominate. Defining the neutron and gamma flux environment at every sensor location and for all operating conditions is a complex analytical problem.

2. Available Information

The same neutron and gamma attenuation data required to support the design of radiation shields are needed to assist in the design of the reactor instrument system. The experimental results discussed in References 1 through 13 are of primary interest. In addition, the neutron and gamma flux distributions obtained in the ZPPR/FTR-2, experiments provide a basis for estimating the effect of stored fuel on the neutron environment of the sensors. These same experiments also provide ^{56}Mn and ^{24}Na source distributions in the radial shield mockup, which assist in identifying the gamma background for the in-vessel sensors. The neutron and gamma

sensitivity of various detectors has been measured in order to select those whose characteristics are suitable for the environments of interest.

3. Development

Although the ZPPR/FTR-2 shield measurements were completed by December 1969, the results have not been fully assembled and analyzed. Therefore, **it** is not known at this time whether or not suitable in-vessel sensor locations are available for currently available detectors. Also, the impact of stored fuel on out-of-vessel sensors has not been evaluated. As a result of these analyses, further development work might be indicated in order to meet this project event. Analysis of the experimental results might require development of three-dimensional neutron and gamma transport methods to provide adequate design techniques.

J. COMPLETE FINAL DESIGN AND INITIATE FABRICATION OF IN-VESSEL SHIELD COMPONENTS, APRIL 1971

1. Background

In order to finalize the design of the in-vessel shield components, **it** is necessary to define the radiation environment at all those positions protected by the shield. Key protection points include the core support structure, the reactor vessel wall, and the stored fuel. In addition, **it** is necessary to define the radiation environment of the neutron sensors in the reactor instrument system. Finally, **it** is necessary to be able to predict the effects introduced by stored fuel and by movement of peripheral control rods, and to predict heat generation rates in stored fuel. The nuclear heat rate distribution in the radial shield is required to establish the minimum sodium volume fraction.

2. Available Information

The attenuation data in the STIR experiments⁽²⁻⁶⁾ and in the ZPPR/FTR-2 shield experiment were obtained to provide a basis for the design of the radial shield. These measurements provide attenuation characteristics of materials of interest, and, in the case of the ZPPR

measurements, in the geometry of interest. To the degree possible, the ZPPR measurements were designed to study the effect of stored fuel.

3. Development

Analysis of the ZPPR experiments is not complete. Preliminary results by BNW personnel indicate that analytical techniques and nuclear data currently employed might be expected to predict the spatial distribution of neutrons adequately as well as the effect of stored fuel. As the design becomes more firm, it might be necessary to employ the engineering mockup to investigate perturbations introduced by movement of peripheral control rods insofar as the in-vessel neutron sensor environment is concerned. Continuing consideration should be given to this area as the design progresses in order to meet this project date.

K. COMPLETE FINAL DESIGN OF EQUIPMENT LAYOUT WITHIN HEAT TRANSPORT CELLS, JUNE 1971

1. Background

The radioactivity level associated with the primary heat transport system sodium dictates the thickness of the concrete walls surrounding the heat transport equipment, except for the reactor cavity wall. The activity level of this sodium can be calculated with adequate accuracy based on the spatial neutron distribution information being obtained in support of other shield designs. However, the radiation levels within a heat transport cell during extended reactor outages will depend on the transport and deposition rates and patterns of radioactive corrosion and fission products. The deposition of radioactive isotopes within the components in the heat transport cells may establish the extent of contact maintenance and inspection within the cells and/or the need for remote methods.

2. Available Information

As a result of actual operating experience with sodium-cooled reactors, some experimental data exist concerning the transport and deposition of radioisotopes in various components and pipes. Contact maintenance of the primary heat transport system was carried out at SRE following major

cladding failures within the reactor. It has been found that the fission and corrosion products deposit within the walls of the components and are not removed by simple drainage of the sodium from the system. Sufficient experimental information does not exist to define the effects of temperature, purity levels, material differences, flow rates, and flux levels in order that accurate estimates be made of radiation levels in FTR cells.

3. Development

Experiments are in progress at PNL in high temperature sodium loops to establish the key parameters governing transport and deposition of corrosion and fission products of interest in the design of the FTR. Preliminary results of these measurements are becoming available.

L. COMPLETE FINAL DESIGN OF THE REACTOR CAVITY WALLS, JUNE 1971

1. Background

The inside diameter of the reactor cavity wall is established by the diameter of the reactor vessel together with the necessary surrounding pipe space. The thickness of the cavity wall is established either by the structural requirements, or by the nuclear attenuation requirements. Attenuation characteristics of concrete are relatively well known;⁽²⁰⁻²⁶⁾ therefore, the required wall thickness can be established based on the incident flux intensity that emanates from the reactor vessel and the leakage flux required to limit activation in the secondary sodium. The radiation levels from the reactor vessel will depend on the reflector-radial shield that is employed together with the effect of stored fuel. These effects are being studied in detail in order that a satisfactory radial reflector-shield be designed. Adequate information should soon be available to provide a basis for the preliminary design of the cavity. The final design of the cavity wall must include the penetrations for the primary heat transport piping. As the primary sodium changes temperature, the pipes and reactor vessel will expand or contract accordingly. The passageways through the cavity wall must provide space to accommodate the resultant movement. In addition, the pipes must be

insulated from the reactor cavity wall. Thus, the pipe spaces through the cavity wall provide leakage paths for neutrons. Design of these penetrations, together with placement of materials in either or both the cavity and heat transport cell, must provide adequate shielding to limit the activation of the secondary sodium, and to limit radiation levels if personnel access to the heat transport cells is required. The latter requirement is under review.

In the early conceptual layout arrangements, a pipe space was provided between the reactor cavity and the heat transport system cells. This pipe space made it possible to offset the penetrations through the cavity wall and those through the cell wall to the extent that neutron streaming effects could virtually be eliminated. As a result of the recent task force efforts to minimize the piping runs and the space within containment, the heat transport cells are now adjacent to the cavity wall, and the piping penetrations directly connect these two regions. The neutrons streaming through the annular space between the sodium-filled pipe and the steel-lined concrete provide a potential source for activation of secondary system sodium. This situation might be tolerable to the extent that activation of the secondary sodium is within design limits. If activation of secondary sodium exceeds tolerable limits, shielding can be placed on either side of the entrance to the cavity wall, around the piping within the cavity wall, around the secondary sodium in the heat exchanger and secondary loop pipes, or some combination of these could be employed. Placing any additional weight on the pipes is very undesirable.

2. Available Information

Measurements of the passage of neutrons through large rectangular ducts in concrete were made at ORNL as a part of an on-going study on the effects of nuclear weapons. Some measurements have also been made, employing neutron sources behind small slots and ducts through water.⁽³³⁾ The reported experiments do not provide an adequate basis for optimizing the design of neutron shielding of penetrations through concrete containing cylindrical, insulated piping filled with sodium.

3. Development

The analytical and experimental program required to provide a satisfactory solution will be complex. A developmental program needs to be initiated within the next few months to provide a basis for meeting this project date. The Appendix is a copy of a preliminary test specification, prepared by Westinghouse, that provides a basis for defining the experimental development effort. Parallel to these experiments, analytical techniques must be developed that successfully predict the measured streaming effect and form a basis for future design.

M. FINALIZE PHYSICS PARAMETERS OF INITIAL REACTOR CORE, JULY 1972

1. Background

The nuclear heat distribution in the radial reflector and the burnup rate of poison material in cocked control rods influence the physics parameters, hence the design, of the initial core. The sodium volume fraction in the radial reflector is related directly to the nuclear heating distribution. The outer zone fuel composition and the worth of peripheral control rods are influenced by the composition of the radial reflector. A minimum sodium volume fraction in the reflector is desirable both from the standpoint of shielding and core physics parameters.

Accurate evaluation of the burnup rate of control material in withdrawn control and safety rods is required to establish the design of these rods, including space allowance for helium gas. The nuclear heat distribution in control rods affects sodium volume requirements and, hence, reactivity worth. The effect of withdrawn control rods on the neutron distribution in the vicinity of the rods and throughout the sodium pool needs to be assessed to establish the influence on the shield design.

2. Available Information

A great deal of applicable shielding data was obtained in the ZPPR/FTR-2 experiments, which were included in the Phase B critical experiments program. These measurements were made on a rather idealized core whose

composition and volume approximated that of the FTR. Gamma dose rate measurements through the radial reflector and shield and in the axial reflector employed in the mockup provide supporting information in nuclear heating. However, the mockup contained neither peripheral control rods nor simulated withdrawn in-core control rods.

3. Development

An engineering mockup of the FTR core is planned in a zero power facility to finalize the physics parameters associated with the initial core. The mockup will employ a geometric and material arrangement that approximates the FTR design as closely as possible. Measurements made in this mockup will provide the basis for any required final adjustment in physics parameters. Measurements of the nuclear heat rate distribution through the radial reflector are required to establish the minimum sodium volume fraction that can be employed, consistent with thermal, mechanical, and neutronic design requirements. All relevant nuclear heating sources should be considered, as should the presence and location of peripheral control rods. Experiments are required to determine the neutron flux distribution in and around withdrawn in-core control rods, together with gamma dose rates and (n,α) heat generation distribution rates in the control rods. Although not required for evaluation of core physics parameters, an experiment to assess neutron streaming along fission gas plenums is required to establish more accurately, vessel cover shielding requirements, primary sodium activity, and radiation damage rates in the instrument assembly. Measurements of the neutron flux distribution in the vicinity of the in-vessel neutron sensor locations as a function of peripheral control rod movement would provide a basis for finalizing and optimizing the design of the low level startup instrument system.

IV. ACKNOWLEDGMENTS

The author acknowledges the comments to this plan contributed by P. S. Bland, S. A. W. Gerstl and J. Rathbun of Westinghouse; W. Neuendorf of Bechtel; and E. T. Boulette, C. R. Doman, C. A. Mansius and D. R. Marr of Battelle-Northwest.

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APPENDIX

FFTF PROJECT
WESTINGHOUSE DEVELOPMENT TEST REQUIREMENT SPECIFICATION
(Preliminary)

FFTF PROJECT
WESTINGHOUSE DEVELOPMENT TEST REQUIREMENT SPECIFICATION
(Preliminary)

1.0 Test Title: Duct Streaming Experiment

2.0 Test Number:

3.0 Test Objectives:

The purpose of this experiment is to measure the effect of neutron streaming through the sodium inlet and outlet piping system between the reactor cavity and the HTS equipment cells and to evaluate the validity and accuracy of available radiation shielding design methods.

4.0 Test Justification:

The sodium outlet pipes as well as the sodium inlet pipes penetrate the reactor cavity wall at a height of 19 ft **above** the top of the core (Ref. 1, 2, 7, 8). Hence, most neutrons (from the core) enter these ducts in a slant angle, which makes the geometric arrangement truly three-dimensional from a shielding calculation standpoint. Streaming of neutrons can take place along the sodium in the pipes as well as along the low density thermal insulation around the pipes. There are no adequate analytical techniques available to calculate the effect of neutron streaming through such three-dimensional arrangements with sufficient accuracy.

The prime reason for the need to accurately evaluate neutron streaming along the pipe chase are as follows:

(a) The suppression of secondary sodium activation, induced in the **HX** and secondary piping of the HTS equipment vault, to restrict the surface radiation level of secondary piping external to the equipment vault to less than 0.2 mrem/hr.

(b) Control of neutron activation of components in the HTS equipment vault to permit contact maintenance on decontaminated components. (This ~~may~~ be restricted to early years of operation.)

(c) Permit personnel access to the equipment vault for short periods with the loop isolated but the remaining two loops operational and the reactor operating at reduced power.

5.0 Test Description:

A series of measurements will be required to evaluate the effect of the several components of the piping system on the total neutron attenuation and neutron streaming. A 6 ft long sodium filled pipe through a concrete wall should be mocked up in a stepwise manner allowing spectral measurements of the neutron distribution after each step. The neutron source should be positioned at a distance of up to 8 ft from the entrance of the duct; its vertical position with respect to the horizontally orientated pipe must be variable to allow for slant incidence of neutrons from the source under various angles. The detector will be positioned in a variable location at the opposite end of the pipe. Neutron spectra should be measured after each of the following four consecutive steps:

Stage 1:

open duct in ordinary concrete
Length: 6 ft

Stage 2:

Insert a 1 in. thick stainless steel
pipe of 28 in. in OD concentrically,
leave a 10 in. air gap

Stage 3:

Fill the steel pipe
with sodium

Stage 4:

Fill 8 in. of the 10 in. gap with
thermal insulator material (KAYLO
10STS), Ref. 3, around the sodium
pipe and leave a 2 in. gap between
the insulator material and the concrete.

Depending on the final design, it might be necessary to increase the length of the test arrangement from 6 ft to about 20 ft, Ref. 4.

It is recommended that this series of measurements be repeated for an 18 in. OD sodium pipe with 8 in. insulation material and 4 in. air gap. The purpose of this additional experiment is to approximate the geometry of the sodium inlet pipes as well as the outlet pipes, and to perform parametric studies by comparing measured results from both experiments.

6.0 Test Results Expected:

A. Data Expected

It is required that measurements be made with neutron spectrometers over the energy range from thermal (0.025 eV) to fast neutrons (up to 15 MeV). Not only the scalar flux at the pipe exit is to be measured, but also the angular flux distribution at particular locations.

B. Accuracy Required

The accuracy of the measured data will be a function of the energy of the counted neutrons. The resolution of the counters should be such that a minimum accuracy of +25% can be achieved for the spectra of the scalar flux. The angular flux measurements are not required with this high accuracy for the absolute fluxes, but their relative accuracy should be good enough to allow the determination of the degree of anisotropy of the emerging neutron flux.

C. Format Desired

All unfolded measured data should be given in graphical form with error bars.

D. Dates Needed

The experiment should be completed prior to January 1971 (Ref. 5, 6).

7.0 Test Requirements:

A. The energy spectrum of the neutron source should approximate the spectra of the neutrons emerging from the reactor vessel wall. This spectral distribution will be a function of the height at which the neutrons penetrate the reactor vessel wall. Detailed information on the required source spectrum will be provided later.

B. The neutron detector must be movable in all three space directions. In addition to measurements along the axial traverse of the duct, measurements **across** the pipe exit are required horizontally as well as vertically. For some source positions, these traverse measurements will be required in three different planes with distances of 3, 5 and 10 ft from the exit end of the duct.

8.0 Test Component Description:

The concrete block should consist of ordinary concrete, the detailed composition of which will be provided by Bechtel Corporation.

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10.0 Cognizant WARD Engineer: S. A. W. Gerstl

11.0 Cognizant WARD Manager: Paul F. Fox

12.0 Signees:

/s/
S. A. W. Gerstl
Cognizant Engineer
WARD-FFTF Project

Approved: /s/
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