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LMFBR IN-CORE INSTRUMENTATION AND IN-VESSEL SURVEILLANCE SYSTEMS

SECOND QUARTERLY REPORT
APRIL-JUNE 1972

PREPARED FOR THE
U.S. ATOMIC ENERGY COMMISSION
CONTRACT AT(04-3)-189 PROJECT AGREEMENT 65

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**LMFBR IN-CORE INSTRUMENTATION AND IN-VESSEL
SURVEILLANCE SYSTEMS**

**Second Quarterly Report
April-June 1972**

Approved:

G. R. Pfisterer
G. R. Pfisterer
Technical Project Engineer

Approved:

C. L. Larson
C. L. Larson
Program Manager

Prepared for the
United States Atomic Energy Commission
Under Contract AT(04-3)-189
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ABSTRACT

This report summarizes the work performed on the research and development program for LMFBF In-Core Instrumentation and In-Vessel Surveillance Systems during the period from April 1, 1972 through June 30, 1972. The work was performed under Contract AT(04-3)-189, Project Agreement 65.

1. INTRODUCTION

This program was initiated in January 1972, with the objective of developing in-core and in-vessel instrumentation for application to LMFBF's. The program is presently divided into three tasks. Task A provides development of in-vessel high temperature neutron detector and cable assemblies. Task B provides development of acoustic monitoring systems for application to the reactor core and vessel. Task C provides development of other critical areas of instrumentation for reactor application. Thus far, Task C has concentrated on the development of (1) a subcriticality monitoring system, and (2) an on-line reactivity monitoring system.

Primary emphasis is on overall system development, making maximum use of base technology information available from the National Laboratories.

The following reports have been written in partial fulfillment of USAEC Contract AT(04-3)-189, Project Agreement 65:

1. *LMFBF In-Core Instrumentation and In-Vessel Surveillance Systems, First Quarterly Report, January-March 1972, July 1972 (GEAP-13825-1).*
2. Wimpee, L. C., *State-of-the-Art Review for LMFBF Source Range Fission Counter, July 1972 (GEAP-10611).*

2. SUMMARY

2.1 TASK A—NEUTRON MONITORING

The objective of this task is to develop in-vessel neutron detector and cable assemblies that do not require auxiliary cooling or special shielding. Both startup (source range) and intermediate or wide-range detector assemblies are planned with first priority given to startup detectors.

Activity for the quarter was predominantly in the following areas:

- a. reactor nuclear analysis
- b. detector orientation
- c. state-of-the-art review
- d. RDT standards review
- e. detector physics study
- f. detector conceptual design
- g. detector development modeling
- h. procurement and testing of cables.

The reactor nuclear analysis provides the calculations and data of the nuclear environment in which the neutron detectors must function. The analysis effort is being concentrated in the vessel shield just outside the lateral core support, with provision for modification of the neutron flux density in this region by changing the shielding materials inside the core support cylinder. Detector in-vessel orientation cases were investigated, and location and arrangement for the source range detector were made. Adjustment of the in-vessel shielding can be made to obtain the U-235 fission rate necessary to meet system requirements.

The initial phase for the source range detector development, consisting of reviews of the state of the art and RDT standards and detector physics studies, has been completed. The second phase, more hardware-oriented, is underway with completion of the first development model.

2.2 TASK B—ACOUSTIC MONITORING

The principle objective of the PA-65 acoustic monitoring task is to develop a system for the continuous in-service monitoring of the operating conditions of an LMFBR core and in-vessel components.

The important areas of activity during the quarter were as follows:

1. definition of functional objectives;
2. preparation of the development plan; and
3. planning of signal characterization tests.

Several activities have been performed to permit selection of the functional objectives for this task. The state of the art was reviewed and a report issued. General Electric participated, with other reactor manufacturers, in an AEC-sponsored meeting to select functional objectives. Design requirements were solicited from the various engineering groups within GE-BRD.

On the basis of information gathered, the following candidate functional objectives were identified:

- a. core boiling detection
- b. component vibration
- c. component failure detection by stress emission
- d. control element signature analysis
- e. core clamping effectiveness
- f. channel clearance monitoring
- g. reactor inlet gas or particulate entrainment
- h. channel flow monitoring
- i. channel temperature monitoring
- j. acoustic ranging/under sodium viewing.

The above objectives were evaluated for application to the vessel and in-vessel components of LMFBF's, in general, as well as to the demonstration plant. The following criteria were used:

- 1. value of the data for protection, operation, and maintenance of the plant;
- 2. probability of successful implementation based on current state of the art and signal quality; and
- 3. relative merits of obtaining the same information by alternate methods.

The following functional objectives were selected:

- 1. detection of anomalous in-vessel component impacting;
- 2. mechanical signature analysis of control drives;
- 3. monitoring of in-vessel component vibration;
- 4. monitoring of acoustic signals in upper plenum sodium for core anomalies and boiling; and
- 5. detection of entrained gas or particulate matter in the sodium stream at the vessel inlets.

An internal review of these objectives will be held before the final recommendations are reported.

2.3 TASK C—OTHER CRITICAL INSTRUMENTATION

2.3.1 Subcriticality Monitoring

A preliminary Subcriticality Monitoring System (SMS) design was established to meet the functional requirements which were identified in the previous quarter. The characteristics of this system are summarized as follows:

Preliminary SMS Design

- a. **Technique** — Neutron source multiplication supplemented by calculated data analysis calibration near critical with control rod drop analyzed by inverse kinetics.
- b. **Detector** — U-235 fission chamber (1 to 2 gm of fissionable material).
- c. **Detector Location** — Outside core lateral support in vessel shield.

The results indicated that the SMS detector could also serve the source range monitoring (SRM) function.

2.3.2 On-Line Reactivity Computer

A task force to work on the conceptual design of the On-Line Reactivity Computer (ORC) was established, which included representation from the areas of core design, nuclear engineering, safety and licensing, and control and instrumentation. The work was of a scoping nature which accomplished the following:

1. established mathematical model for ORC;
2. estimated magnitude of reactivity feedback coefficients;
3. estimated uncertainty in the reactivity balance calculation;
4. listed core malfunctions which ORC could detect; and
5. evaluated practical problems associated with incorporating ORC in the Plant Protection System.

These preliminary results indicated that the ORC could detect several reactor malfunctions, but further study is needed to determine the incremental benefit of the ORC in relation to the other detection systems in the safety system.

3. TECHNICAL ACCOMPLISHMENTS

3.1 TASK A—NEUTRON MONITORING

3.1.1 Reactor Nuclear Analysis

The purpose of this effort is to determine the nuclear environment in which the out-of-core source range neutron monitor (SRM) will be used. It is expected that the SRM will be positioned at core midplane elevation exterior to the radial blankets and interior to the reactor vessel. The near-core radial neutron shield locations, selected as acceptable locations for the detector, are shown in Figure 3-1.

Nuclear analysis effort is being concentrated on detector locations in the vessel shield just outside the lateral core support structure with secondary effort being applied in the radial shield area (closer to the active core) as a backup in case of insufficient count rate at the other location. In order to include a wide range of possible fast reactor radial neutron shield designs within the scope of the analysis, two shield compositions are being considered in the radial shield (adjacent to the radial blankets), one of steel and one of steel plus B₄C. The large neutron attenuation associated with the amount of B₄C assumed present in the radial shield (assumed for most of the calculations) unduly restricts the detector design for placement in the vessel shield. Some effort was made to determine the effects of removing the B₄C and replacing it with stainless steel. Thus, effects with both shields will be presented whenever data are available.

If a particular shield design does include some amount of B₄C in order to limit neutron fluence on the surrounding structure, many options are available to retain or even amplify the no-B₄C flux and resulting count rate at the detector location. Among these options are the following:

- 1 allow a larger fluence on the core support and structure at the core midplane level taking credit for a possibly lower value at stress points (i.e., at the clamping plane, ~top of axial blanket level),
- 2 retain the B₄C in the radial shield while providing a no-B₄C neutron "window" at the detector location, or
- 3 replace the B₄C with a neutron moderating material (graphite, BeO, nickel, etc.), thus decreasing high energy neutron fluence and, additionally, increasing the U-235 fission rate in the detector due to spectrum softening.

It appears that for the detector design purposes there are no insurmountable problems associated with achieving the count rate associated with a steel and sodium radial shield.

3.1.1.1 U-235 Fission Rate

The U-235 fission rate profile was calculated in one-dimensional radial geometry utilizing transport theory. In order to provide an upper estimate of the gamma flux in the vessel shield, a core model descriptive of the end of an equilibrium burnup cycle was chosen. Although the fission rate (and, hence, fission gamma source) in the blanket is highest at this time (due to production of fissile plutonium), the neutron leakage out of the blanket is not appreciably changed and the U-235 fission rate results should be applicable to any time within the operating cycle.

The radial U-235 fission rate profile from core center into the vessel shield with and without B₄C in the radial shield is compared in Figure 3-2. The slight rise in fission rate in the radial shield for the no-B₄C case is due to thermalization of neutrons into the high U-235 fission energy range at low energies.

Figure 3-1. Near-Core Upper Axial and Radial Neutron Shields

Figure 3-2. U-235 Fission Rate Distribution versus Radial Distance from Core Center (Core Power = 935 MWt)

The U-235 fission rate, at other than full power (as presented in Figure 3-2) is directly proportional to the fission power level. The power level at the fully shutdown condition has been estimated using the neutron source due to spontaneous fission and the (α, n) reaction with oxygen. To estimate the power level at shutdown conditions, the shutdown source level, S_s , together with the shutdown multiplication,

$$m = \frac{1}{1 - k_s} ,$$

is compared to the full power neutron source, S_p ,

$$\frac{\text{Shutdown Power}}{\text{Full Power}} = \frac{m \times S_s}{S_p}$$

Using the expected shutdown reactivity of $k_s \approx 0.85$, the power level at fully shutdown conditions is estimated to be $\sim 5.7 \times 10^{-11}$ times rated power.

3.1.1.2 Gamma Field

In order to evaluate the gamma flux profile at various power levels and times after shutdown, it is necessary to consider both high and low level gamma sources. The gamma sources considered for this study include the following:

- a prompt fission gammas and prompt capture gammas in the core and structure outside the core;
- b fission product decay gammas at various times after shutdown, and
- c sodium and structural decay gammas from material made radioactive during neutron exposure.

FISSION AND PROMPT CAPTURE GAMMA

The prompt gamma release from neutron capture and the fission event are calculated by the radial geometry transport theory code utilized to analyze the U-235 fission rate profile. Since the fission gamma source constants used by the code include both the prompt and delayed gammas, some method was needed to separate the two. To accomplish this, the gamma flux profile resulting from only delayed fission gammas was generated separately and subtracted from the gamma flux profile for both total fission gammas and prompt capture gammas.

The resulting gamma flux profile is, then, directly proportional to the power level and is given in Figure 3-3 for full power operation for the case of B+C in the radial shield.

The gamma flux profile, including delayed fission product, gammas, was calculated for the no-B+C case and is compared to the B+C case in Figure 3-4.

FISSION PRODUCT DECAY GAMMAS

A conservative (maximum) fission product decay gamma source rate was established for the core and axial blankets by assuming infinite reactor operation.

The gamma source strength was established for 1 and 100 min after reactor shutdown by relating the calculated fission rate at full power for each core and blanket region to the total and spectral fission product decay gamma activity information given in Reference 1.

Using the multigroup decay gamma source rate generated above, the gamma flux profile was calculated for the B+C in radial shield case, for both 1 and 100 min after shutdown, using the before-mentioned transport code. The results are given in Figure 3-5. (The "bouncy" shapes across the core and blankets are due to assuming a flat source distribution across each core and blanket region.)

Figure 3-3. Prompt Fission and Capture Gamma Dose Rate versus Radius

Figure 3-4. Total Fission and Capture Gamma Dose Rate versus Radius

Figure 3-5. Fission Product Decay Gamma Dose Rate versus Radius

SODIUM AND STRUCTURE ACTIVATION GAMMAS

Na

The primary sodium coolant will become highly radioactive because of the Na-23, (n, γ) Na-24 reaction. For short-term shutdown, the other primary sodium reaction, Na-23 (n,2n) Na-22 with its long half-life, is insignificant due to the much smaller production in Na-22. The resultant gamma flux levels were calculated with a one-dimensional transport calculation.

Structure (SS-304)

Neutron-induced structure activation and resulting gamma source strength per unit volume of SS-304 was calculated at various points outside the active core and blankets. The activation gamma code ACT-II² was used for this purpose with an assumed irradiation time of 30 years.

The actual gamma activity radial profile was estimated by interpolation between the calculated points. The resulting gamma source profile and the before-mentioned one-dimensional transport code were used to produce the gamma flux profile for structure activation presented in Figure 3-6.

GAMMA FLUX AS A FUNCTION OF REACTOR POWER LEVEL AND TIME SINCE SHUTDOWN

The assumption is made for the presentation of the following information that the reactor has been shut down for either 1 or 100 min and is being promptly brought back up to either full or fractional power.

The gamma flux resulting from sodium and structure activation plus the decay fission product gammas is essentially independent of the returning power level for the short time scale being considered, while the prompt flux (prompt fission and capture) is directly proportional to the instantaneous power level.

Thus, the "non-prompt" gamma flux at either 1 or 100 min after shutdown can be considered "background gammas" and the "prompt" gamma flux, the instantaneous power level contribution.

The gamma flux at various locations exterior to the core as a function of power level is shown in Table 3-1 for the case of B₄C in the radial shield. Estimates of the power level dependent flux for B₄C removed from the radial shield have been made and are presented in Table 3-2.

The following simplifying assumptions were made to estimate the gamma flux for the no-B₄C case:

1. *Fission Product Decay Gammas* — Due to the "improved" reflector with removal of B₄C, the radial blanket edge fission rate could increase by as much as 30%. The replacement of SS for B₄C, however, will more than compensate for the increased gamma because of the greater attenuation. Thus, to be conservative, the blanket and radial shield fission product decay gamma flux is increased 30%, while the flux outside the shield is kept the same as the B₄C-in case.
2. *Sodium Activity Gammas* — The increased attenuation (absorption) of gammas in the radial shield is ignored. Thus, the gamma flux profile is assumed equal to the B₄C in radial shield case.
3. *Structure Activation Gammas* — It is assumed that the structure activation gamma source (and, hence, gamma flux) increases directly proportional to the U-235 fission rate increase (between B₄C-in and no-B₄C case).

FISSION CHAMBER ACTIVATION

The purpose of this work was to demonstrate that the gamma flux within the detector due to detector activation is insignificant compared to the background gamma flux.

For this purpose, the unrealistic but maximized activation time of 30 years at full power flux conditions and 1 min after shutdown was used. The reference chamber material, SS-304, activated in the vessel shield and the corresponding U-235 fission rate at that location were used to compensate the in-chamber gamma source.

Table 3-1
GAMMA FLUX DOSE (mR/hr) AS A FUNCTION OF OPERATING POWER LEVEL
AT ONE MINUTE AFTER SHUTDOWN AT SELECTED LOCATIONS
(With B₄C in Radial Shield)

Location→	Blanket	Radial Shield	Vessel Shield		
			Inner	Mid	Outer
Background Gamma					
Na	6.907×10^6	1.578×10^7	1.118×10^7	1.245×10^7	1.244×10^7
Structure	2.670×10^8	1.882×10^8	4.471×10^7	3.191×10^7	1.724×10^7
Decay Fission Product	1.034×10^{10}	2.578×10^8	2.805×10^6	2.972×10^4	8.082×10^2
Total Background	1.061×10^{10}	4.618×10^8	5.870×10^7	4.439×10^7	2.968×10^7
For Initial Prompt Gamma	1.804×10^{11}	5.642×10^9	3.574×10^8	1.388×10^8	5.785×10^7
Total Gamma (Prompt + Background)					
For Various Initial Power Levels					
1.0	1.910×10^{11}	6.104×10^9	4.161×10^8	1.832×10^8	8.753×10^7
0.1	2.865×10^{10}	1.026×10^9	9.444×10^7	5.827×10^7	3.546×10^7
0.01	1.241×10^{10}	5.182×10^8	6.227×10^7	4.578×10^7	3.026×10^7
0.001	1.079×10^{10}	4.674×10^8	5.906×10^7	4.453×10^7	2.974×10^7
Zero	1.061×10^{10}	4.618×10^8	5.870×10^7	4.439×10^7	2.968×10^7

Table 3-2
GAMMA FLUX DOSE (mR/hr) AS A FUNCTION OF OPERATING POWER LEVEL
AT ONE MINUTE AFTER SHUTDOWN
(Without B₄C in Radial Shield)

Location→	Blanket	Radial Shield	Vessel Shield		
			Inner	Mid	Outer
Background Gamma					
Na	6.907×10^6	1.578×10^7	1.118×10^7	1.245×10^7	1.244×10^7
Structure	3.418×10^8	8.234×10^9	1.463×10^9	5.220×10^8	1.915×10^8
Decay Fission Product	1.344×10^{10}	3.351×10^8	2.805×10^6	2.972×10^4	8.082×10^2
Total Background	1.379×10^{10}	8.585×10^9	1.477×10^9	5.345×10^8	2.039×10^8
For Initial Prompt Gamma	1.865×10^{11}	3.446×10^{10}	8.849×10^9	2.175×10^9	6.899×10^8
Total Gamma (Prompt + Background)					
For Various Initial Power Levels					
1.0	2.003×10^{11}	4.304×10^{10}	1.033×10^{10}	2.710×10^9	8.938×10^8
0.1	3.244×10^{10}	1.203×10^{10}	2.362×10^9	7.520×10^8	2.729×10^8
0.01	1.566×10^{10}	8.930×10^9	1.565×10^9	5.562×10^8	2.108×10^8
0.001	1.398×10^{10}	8.619×10^9	1.486×10^9	5.367×10^8	2.046×10^8
Zero	1.379×10^{10}	8.585×10^9	1.477×10^9	5.345×10^8	2.039×10^8

Figure 3-6. Structure Activation Gamma Dose Rate versus Radius

With four 0.050-in. walls and a steel activity of 4.2×10^7 mR/hr-cc, the surface dose source is 2.13×10^7 mR/hr-cm² for the steel. Using data from Reference 2 giving 3.93 MeV/fission and an at-power U-235 fission rate of 2.23×10^{10} fission/gm-sec at 174 cm, the four surfaces of U-235 coated with 1 mg/cm² produce a net surface dose source of 6.06×10^5 mR/hr-cm².

To convert the surface gamma source to flux at the center of the cylindrical fission chamber, the familiar equation for the flux at a point of radius R from a line of length h and at the midpoint of the line;

$$\phi = \frac{S_L}{4\pi R} \tan^{-1} \frac{h}{R}$$

where S_L = photons/cm-sec (or mR/hr-cm)

is integrated over γ the angle of sweep generating the right circular cylinder giving:

$$\phi = S_V \tan^{-1} \frac{h}{R}$$

where S_V = mR/hr-cm².

Being conservative and assuming an infinite height cylinder gives

$$\phi = \pi S_V / 2$$

Using the SS and U-235 surface sources mentioned above produces a gamma flux at the center of the cylinder of 3.4×10^7 mR/hr; however, since the chamber will be withdrawn* such that the count rate at full power is at least 3 decades less than it would be at the nominal location, the chamber activation and, hence, flux will be no more than $\sim 3 \times 10^4$ mR/hr compared to a background gamma flux of $\sim 5.9 \times 10^7$ mR/hr. Thus, it is seen that chamber activation does not significantly increase the net gamma flux in the detector.

3.1.2 Detector Location

3.1.2.1 Introduction

The location of the SRM detector in the LMFBR vessel is an essential factor in determining the basic required specifications for the detector. In this work, the technical efforts of the reactor vessel internals designers, the core physics and shielding engineers, the detector-cable designers, and the neutron monitoring systems design engineer were coordinated.

It is not within the scope of this program to determine the exact location for the high temperature rated neutron detector for all LMFBR designs; however, this report shows that the detector sensitivity selected can be applied within the vessel of a typical LMFBR. While the typical LMFBR design used as the basis for this study utilizes open-head refueling, the basic detector system results are equally applicable to under-head refueling designs.

The reference detector (Figure 3-7) is a fast fission counter, approximately 100 ns collection time, operated with a wide band, low noise preamplifier in the current mode.

* The present objective of the detector study is to have the fission chamber reading operative at all levels of reactor operation. In order to assure that the detector does not saturate, it is expected that it will have to be removed to an axial location where the U-235 fission rate is at least 3 orders of magnitude less than it would be at the core midplane level at full power.

Figure 3-7. Conceptual Fission Counter Design

The reference design is an integral cable assembly with two coaxial cables and hermetic seals between the detector and cable. The detector is 1.88-in o.d., 6.0 in long with a 4.5-in sensitive length, and the cables are 30 ft long. The detector housing and electrodes are Inconel 600, the electrode spacing is 0.040 to 0.080-in, and the gas fill is planned to be 90% argon and 10% nitrogen. The 500 cm² of electrode surface area is coated with 2 mg/cm² of U-235. The cables have copper conductors with a composition sheath of copper and protective stainless steel cladding.

3.1.2.2 System Design Considerations

The following major system design features were considered in the orientation of the source range detector within the reactor vessel:

- 1 The detector shall be in a guide tube and be retractable for range extension during reactor startup, so that it remains operational at full power.
- 2 The detector and monitoring equipment shall cover approximately five decades of operation from full shutdown in the inserted (most sensitive) position, to provide approximately two decades of "overlap" with the wide range monitor system.
- 3 The detector and monitoring equipment shall provide a nominal 10 counts/sec from neutron flux at full shutdown.
- 4 The detector-cable assembly should be replaceable during reactor operation.
- 5 The design life should be two to three years.
- 6 The source range detector should remain in-vessel and operational during all phases of operation including reactor vessel closure removal and refueling.
- 7 The environment temperature is 850°F in the startup position and 1050°F in the retracted position.

The following factors were considered but are not requirements:

- 1 The source range detector may be useable for the subcriticality monitoring function if located near enough to the core.
- 2 It is desirable to keep the guide tube temperature low by use of inlet sodium as a coolant. For this reason and also to provide adequate space for guide tube support, the guide tube should be no larger than a nominal 2 in (i.d.).
- 3 The connection between the integral cable and the flexible extension cable should be well out of the high temperature, high neutron and gamma flux regions and where access can be provided during reactor operation.

3.1.2.3 Reactor Vessel Internal Mechanical Considerations

Some important mechanical considerations in getting a guide tube into the vessel internal structure are:

- 1 The guide tube must pass through a nozzle on the side of the vessel in order to bypass the vessel closure and remain in place during closure removal.
- 2 The drywell guide tube must have generous (greater than 60 in.) radii of curvature to minimize bending of the detector cables. The total included angle of bend should be minimized.

3. At least three independent guide tubes must be provided and oriented in a uniform dispersal pattern for the least common mode effects. Three additional guide tubes may be required for power range detectors.
4. The neutron fluence should be kept to a level such that reactor internal parts that are difficult to remove do not have to be replaced during the reactor design life. This includes the detector guide tubes.
5. During reactor startup, the detector will be withdrawn from the location opposite the core centerline to a region where the neutron flux is approximately five decades less. It is desirable to have the detector cooler than the bulk mean core outlet temperature because the detector will operate at this location the greater portion of its lifetime.

3.1.2.4 Nucleonics and Shielding Considerations

The following are the most important nuclear design and in-vessel shielding considerations:

1. The fertile material of the detector will be U-235.
2. The neutron fluence on the reactor vessel wall should be kept well below design limits (less than 10^{22} nvt for neutrons above 100 kev).
3. Shielding changes should be kept within the mechanical and physical design limits now existing on any reference designs.
4. Nonlinearity effects on the detector from such things as sodium temperature or physical movement components should be minimized over the power range of reactor operation.
5. Gamma level at plant shutdown is approximately 10^6 R/hr.

3.1.2.5 Discussion

With B₄C in the shield between the blanket assemblies and the core support cylinder, the flux level just outside the core support was calculated to be approximately a factor of 40 lower than required for a detector which was selected as the reference design (see Figure 3-8 and Table 3-3). To increase the detector sensitivity by a factor of 40 was not given serious consideration because the size of such a detector would be beyond reasonable dimensions for insertion into the vessel, or would be extremely complicated by the number of concentric cylinders required to provide the necessary sensitive area.

The approach that seemed the most logical was to locate the detector in a higher flux zone or to increase the flux at the location just outside the core support cylinder. Two schemes for getting the detector in a higher flux situation were studied. One case provided for the detector guide location to be in the blanket region. While this location provides more than ample flux, the mechanical problems are considerable and some key system design considerations would be compromised. Some of the mechanical and physical problems are:

1. The guide tube interferes with the closure such that deep slots would have to be cut into the closure. The size and nature of these slots would compromise the closure shielding and strength.
2. The guide tube interferes with removal of key in-vessel components such as the core support, and core restraint mechanisms.
3. Guide tube support to account for core expansion during power ascension may not be practical due to the extreme differential movement.

Figure 3-8. U-235 Fission Rate Distribution versus Radial Distance from Core Center (Core Power = 935 MWt)

Table 3-3
COMPARISONS OF SHIELDING, DETECTOR LOCATION, AND DETECTOR SIZE VARIATIONS

	Detector Location	Shutdown Fission Rate (Fission/sec/gm U-235)	4.5-inch Sensitive Length 500 cm ² of 1 mg/cm ² U-235		6.75-inch Sensitive Length 767 cm ² of 1 mg/cm ² U-235	
			Absolute Count Rate (cps)	Effective Count Rate ^(a) (cps)	Absolute Count Rate (cps)	Effective Count Rate ^(a) (cps)
With B ₄ C in the Radial Shield	Vessel Shield	0.57	0.29	0.20	0.44	0.31
	Structure	2.8	1.42	0.98	2.16	1.52
	Radial Shield	3.5	1.75	1.2	2.68	1.88
	Blanket	290	145	102	223	156
With Steel in the Radial Shield	Vessel Shield	8.25	4.13	2.9	6.35	4.45
	Structure	28.5	14.3	10.0 ^(b)	21.9	15.4
	Radial Shield	140	70	49	108	75.3
	Blanket	315	158	111	242	170

(a) Estimate due to loss of neutron counts from gamma discrimination

(b) Optimum location and situation.

- 4 The guide tube must be immersed in reactor core exit temperature sodium in the area where the detector will be located during power operation
- 5 Refueling operation movements will be restricted by the guide tubes projection over the lateral core support into the blanket area

Another consideration to be taken into account for the blanket region location is the possible need for a shroud that separates the core-blanket exit flow from the higher temperature sodium. The guide tube must penetrate this shroud to reach the blanket region. To allow for both lateral and axial expansion and low leakage at the point where the guide tube penetrates the shroud would compromise the reliability of both the shroud and the guide tube.

The other case provided for the guide tube to be located outside of the core support cylinder (and outside the shroud) and to increase the U-235 fission rate at the detector by modification of the shield between the blanket and the core support. By changing the B₄C to stainless steel an increase in the U-235 fission rate of greater than a factor of 10 was calculated. Figure 3-8 and Table 3-3 compare the two shield material configurations. To gain the additional factor of 4 requires replacing of the steel shielding with sodium just in front of the detector. The neutron fluence on the core support cylinder is approximately 10²² nvt for 30-year life (see Table 3-4). The metal at the points where the flux is the maximum could be omitted without affecting structural integrity by making holes in the metal just in front of the detector location.

With the detector located outside of the core support cylinder all of the concerns raised for the blanket location are resolved. In addition, preliminary design calculations (subsection 3.3) indicate that subcriticality monitoring is possible with the reference detector in the outermost location.

Table 3-4
NEUTRON FLUENCE AT THE STRUCTURE AND REACTOR VESSEL WALL
(@ 30-year Operation With 0.8 Load Factor)

Fluence nvt	Structure	Reactor Vessel
With B ₄ C:		
E > 821 keV	1.68×10^{20}	7.88×10^{16}
E > 100 keV	9.83×10^{20}	6.17×10^{18}
Total	2.02×10^{22}	2.37×10^{19}
With steel:		
E > 821 keV	3.84×10^{20}	1.67×10^{17}
E > 100 keV	1.18×10^{22}	3.69×10^{19}
Total	4.60×10^{22}	1.74×10^{20}

3.1.3 Detector and Cable Development

3.1.3.1 State-of-the-Art Review

A report on the state-of-the-art review of high temperature neutron detectors was written and will be published as GEAP-10611 (see Section 1).

3.1.3.2 RDT Standards Review

The issued RDT standards were reviewed for applicability to the SRM design. Of 38 potentially applicable standards which were reviewed, it was determined that 20 might be applicable; however, changes would be required on many of these. Six standards are indirectly applicable in that they describe systems or electronic components with which the source range detector must be integrated. Ten of the standards reviewed were not applicable.

It appears that many of the electrical component RDT standards reviewed have been written as specifications for a specific application; therefore, it would be difficult to comply with them without modification by revisions. It is not feasible at this time to write detailed modifications or revisions to each standard because the source range detector design has not progressed to the point of application of standards and specifications to engineering drawings, and a detailed discussion of each RDT standard now would detract from the primary effort. A few examples of the types of areas where changes would probably be necessary are presented below.

1. RDT C2-1T Insulation Compaction

The permitted use of a liquid absorption test to determine compaction leaves an element of doubt; complete knowledge of the presence of voids is not possible from this test. A better determination of compaction may be required.

2. RDT C7-14T Metal Sheath Ceramic Insulated Cable

- a. Refers to C18-1T for which exceptions are taken (see item 4 below).
- b. The straightness requirement of ± 2 in. from a straight line is impractical for cable handling.
- c. This standard does not cover silica insulated cable; therefore, it would not be applicable if that cable were selected.

3. RDT C17-1T High Temperature Electrical Connectors and Hermetic Seals

- a. This standard is not applicable to the type of seal planned for the source range detector.
- b. The insulation resistance specified is lower than can practically be achieved.

4. RDT C18-1T Ceramic Electric Insulators

- a. Requirements on webs between holes, wall thickness and straightness are not applicable.
- b. May be difficult to meet requirement for 99.5% alumina because of metallizing difficulties for high purity alumina. A lower purity may be acceptable.
- c. In Appendix B the 1150°F air fire by the vendor is unnecessary, vendor cleaning is not acceptable as a final cleaning step, and 1150°F will be exceeded in internal air firing.

5. RDT F5-1T Cleaning and Cleanliness Requirements

- a. In general, the requirements for cleaning are too loose for neutron detector application.
- b. Air jets should not be used for drying because of contamination problems.
- c. The intent can be met with existing facilities with some modifications in both facilities and standard.

6. RDT F6-5T Welding Qualification (Supplement to ASME Boiler and Pressure Vessel Code Section IX)

This is not applicable since the source range detector wall is not a pressure boundary.

7. RDT F6-14T Brazing Fabrication Requirements

Does not permit use of copper which would be preferable to copper gold in neutron detectors.

8. RDT M Series

The technical requirements of these standards can be met; however, since this work involves only small quantities of materials, modification may be necessary to allow the material to be qualified by organizations other than the supplier.

Table 3-5 summarizes the review of the RDT Standards and the decisions made concerning each one.

3.1.3.3 Detector Physics Study

The detector physics study was completed. Six computer programs were written for analysis of sensitivity, gamma discrimination capabilities, and pulse height distribution of possible detector designs. These programs are being used to evaluate detector parameters.

3.1.3.4 Detector Conceptual Design

The detector and cable assembly conceptual design was completed and a design review held. The electrode spacing, gas type, and gas pressure will be selected by evaluation of development models and by use of the computer programs developed in the detector physics study. The selection of electrode and housing material and pretreatment of the material will be dependent on the degree of combination of nitrogen from the fill gas with the material selected, or the acceptability of a completely inert gas fill (argon-helium).

Table 3-5
RDT STANDARDS REVIEWED FOR
HIGH TEMPERATURE NEUTRON DETECTOR APPLICATION

Standard	Potentially Applicable (With Some Changes)	Not Applicable	Indirectly Applicable	Applicable
C2-1T	X			
C7-14T	X			
C15-2T			X	
C15-3T			X	
C15-10T			X	
C16-11			X	
C17-11	X			
C18-1T	X			
E15-2T		X		
F2-2T				X
F2-4T				X
F3-2T	X			
F3-6T	X			
F3-8T	X			
F3-37T	X			
F3-39T	X			
F4-20T			X	
F5-1T	X			
F6-5T	X			
F6-7T		X		
F6-14T	X			
F7-2T			X	
F7-3T	X			
M1-2T	X			
M1-9T		X		
M1-11T		X		
M3-2T	X			
M3-3T	X			
M3-5T	X			
M3-6T	X			
M7-3T		X		
M7-8T	X			
M7-10T		X		
M8-1T	X			
M9-1T		X		
M9-2T		X		
M11-1T		X		
M7-12T		X		

3.1.3.5 Detector Development Model

The first development model was completed and tested and incorporated a 0.020-in. electrode spacing and one atmosphere of 95% argon/5% helium fill gas. The neutron pulse height of the detector was unacceptably low, so the detector was rebuilt with a 0.050-in. electrode spacing and three atmospheres of 95% argon/5% helium fill gas. Test performed on the rebuilt detector demonstrated that the pulse height was acceptable and that the collection time was as predicted, although it was too slow for satisfactory gamma discrimination. A wide band low noise preamplifier for development model testing and analysis was assembled and tested using information supplied by N. C. Hoitink of HEDL. The preamplifier design is based on a design by ORNL.

3.1.3.6 Procurement and Testing of Cables

Cable samples from four suppliers have been received and breakdown pulse noise tests started. A sample of Kaman Sciences silica insulated cable with temporary glass end-seals performed very well with both cable and seal heated to 1200°F. Noise appeared briefly at 600 and 1000 V; however, it did not recur at voltages up to 1100 V. After repeated attempts to reproduce the noise, it was decided that the noise was from external sources.

Three quotations for a high temperature seal design have been received. One order for 20 seals will be placed with Basic Ceramics.

3.2 TASK B—ACOUSTIC MONITORING

3.2.1 Overall Task Planning

Work plans were further detailed by revising activities to agree with current funding and by coordinating and integrating testing plans with the National Laboratories and other AEC contractors.

A development plan is being prepared which will define the responsibility for the various technical activities required to perform the acoustic monitoring task. Sites and facilities for source characterization are currently being identified by personnel of the Research and Development Center of General Electric. The assignments of responsibility within BRD are also being defined. The development plan will be issued in the next quarter.

3.2.2 Functional Objectives

Each of the functional objectives will require essentially separate instrumentation channels, although, in some instances, certain transducers might supply inputs to more than one channel and, eventually, the data display will be integrated. Generally, each functional objective implies its own sensors, special signal conditioning, and data handling. At this point, it is practical to consider the respective instrumentation for each channel as a separate entity.

Preliminary data sheets have been written for each application. Source characterization and sensor functional requirements were listed. Detailed sensor specifications can be written only after experimental source characterization data have been gathered and analyzed and after the siting environment has been ascertained.

The preliminary data sheets are as follows:

DATA SHEET FOR IMPACT MONITORING**Source Characterization**

Description	Metal impact
Location	In reactor vessel
Frequency (Hz)	1 to 50,000

Sensor Requirements

Location	Mounted outside reactor vessel
Type	Accelerometer
Major Function	Detect in-vessel components impacting
Response Time (sec)	1
Sensitivity	Later
Operating Temperature (°F)	1000
Sensor Atmosphere	Nitrogen
Total Flux (nv)	$< 10^{13}$
Life (hr)	1000

DATA SHEET FOR CONTROL DRIVES MECHANICAL SIGNATURE ANALYSIS

Source Characterization

Description	Noise produced by rubbing of control element against guide tube and by operation of the drive mechanism
Location	Control element and drive mechanism
Frequency (Hz)	10 to 50,000, both continuous and impulse

Sensor Requirements

Location	Control drive housing and vessel periphery
Type	Accelerometer
Major Functions	Detect incipient binding of control element and abnormal operation of control drives
Response Time (sec)	1 to 10
Sensitivity	Later
Operating Temperature (°F)	~ 300
Atmosphere	Nitrogen
Life (hr)	1000
Total Neutron Flux	Zero

DATA SHEET FOR VIBRATION MONITORING**Source Characterization**

Description	Component vibrations
Location	Selected positions inside reactor vessel
Frequency	Hz to KHz

Sensor Requirements

Location	Mechanically linked to component of interest (in dry guide tubes)
Type	Accelerometer
Response Time (sec)	~1
Accuracy	Later
Operating Temperature (°F)	720 to 1150
Environment	Sodium/Argon
Total Flux (nv)	$< 10^{13}$
Operating Life (hr)	10,000

DATA SHEET FOR CORE MONITORING

Source Characterization	Immersible Detector	Wave Guide and External Detector
Description	Mechanical impact, pressure fluctuations	Subcooled boiling
Location	In-vessel	In-core
Frequency (Hz)	1 to 100	10 to 400
Sensor Requirements		
Location	Upper plenum	Above head
Type	Dynamic pressure	Accelerometer
Major Functions	Detect mechanical failure and vapor bubble collapse plenum	Detect subcooled boiling
Response Time (sec)	0.05	1.0
Accuracy	Later	Later
Coolant Velocity (ft/sec)	10 to 20	—
Coolant Temperature (°F)	1200 maximum	250
Flux, total (nv)	$< 10^{13}$	Negligible
Life (hr)	10,800	1000

DATA SHEET FOR ENTRAINMENT MONITORING**Source Characterization**

Description	Active systems
Location	Reactor inlet (cold leg) piping
Frequency (MHz)	~ 2 to 4

Sensor Requirements

Location	Cold leg piping external surface
Type	Active, 2 to 4 MHz, pulse
Major Function	Detect attenuation produced by dangerous levels of gas or particulates
Response Time (sec)	~ 0.1
Accuracy	Later
Coolant Velocity (ft/sec)	—
Operating Temperature (°F)	720
Flux, total	~ 0
Life (hr)	1 to 3000

3.2.3 Source Characterization

Source characterization work is being performed both within GE-BRD and at the General Electric R&D Center (GE-R&DC). Internally, in-vessel components subject to destructive vibration are being identified. Their vibrational modes and their permissible vibrational amplitudes will be calculated. Care will be taken to make the results generally applicable. At the GE-R&DC tape recordings of acoustic signals previously recorded as well as opportunities for further work are being reviewed. In general, acoustic monitoring tests for this task must utilize tests which are part of component development tasks, since providing facilities solely for acoustic source characterization is generally not economical.

3.3 TASK C—SUBCRITICALITY MONITORING

3.3.1 Summary

Subcriticality Monitoring System (SMS) functional requirements for the measurement of reactor shutdown margin were developed from considerations for safe reactor refueling and startup (to ensure against accidental power excursions from a near-critical core while the reactor control systems are partially disabled). These functional requirements were translated into design requirements based on a typical LMFBR Demonstration Plant reactor description. A review and evaluation of present state-of-the-art technology in subcritical measurement techniques has identified a practical preliminary design. This uses low level neutron flux monitoring outside the reactor which can reliably be interpreted as a measure of reactor shutdown margin. Measurement at near-critical by inverse kinetics will normalize far-subcritical measurements by neutron source multiplication techniques. These measurements are designed not to interfere with normal refueling, maintenance, and startup operations on a subcritical core. Alternative measurements of reactor shutdown margin are discussed to summarize the development of the present design. Further design development problems were identified in terms of their importance.

3.3.2 Background

LMFBR's will be designed and built to assure safe operation. Reactor power surveillance and control systems are designed to recognize and limit any opportunity of reactor overpower conditions. The effects of an unlikely power excursion are further minimized by the inherent shutdown capability of the Doppler effect in the nuclear fuel, reactor vessel, and plant containment. During power operation, these safeguards are in service to assure a safe and reliable power plant.

Routine refueling and maintenance operations require many of these safeguard systems to be out of service. Although the reactor core is fully shut down during these periods, most of the design safeguards are inoperable. It is necessary, then, to assure the core will remain subcritical throughout these intervals of plant outage.

With no measure of the actual reactor shutdown margin of reactivity, procedures for these routine refueling, maintenance, and startup operations become quite conservative for safety purposes. There are safety considerations that certainly can be outlined to protect against an unexpected critical or supercritical core occurring during these periods. However, a reliable, accurate monitoring of the subcriticality would provide the assurance of safe operations without interfering with the normal procedures.

This measurement would be used to evaluate the potential of a critical or supercritical core prior to scheduled fuel or control movements while the reactor is shut down. The evaluation is based on knowledge of the maximum reactivity changes that could possibly occur and reactor physics interpretations of the measurement of the low level neutron flux from the core. Maximum reactivity additions are based on calculated or measured core parameters. Measurement and evaluation techniques of subcriticality have been developed by previous work in the USAEC LMFBR program. With a typical LMFBR reactor core design and this state-of-the-art measurement capability, a subcriticality monitoring system (SMS) has been defined. Functional requirements, along with additional desirable objectives, for the design have been identified. These were translated into design requirements which were used to select a reference conceptual SMS design. Remaining design development problems that further establish the practicality, accuracy, and reliability of the SMS can be determined.

3.3.3 Functional Requirements

The SMS will provide a measurement of reactor shutdown margin which will be used to evaluate the safety of scheduled fuel and control movements in a shutdown reactor. To clarify this evaluation of the measurement, a conservative safety philosophy has been assumed for the use of the SMS. After each change in the composition of the shutdown core, the reactor will remain subcritical. The primary function of the SMS is to demonstrate this inherent safety of the anticipated refueling sequence or a shutdown control rod withdrawal. The reactor shutdown margin should be measured to be greater than the maximum reactivity insertion possible before safety can be assured.

During refueling, all reactor control is inserted into the core. In this full shutdown condition, spent fuel assemblies are replaced with fresh assemblies to provide excess reactivity for burnup during the next operating cycle. As the end-of-cycle (EOC) core is gradually changed by fresh fuel loading to the beginning-of-cycle (BOC) core, the reactor shutdown margin is reduced. If no error in fuel replacement occurs, the operation remains safely routine. Acknowledging the possibility of oversight, error, or mistake in the refueling operation introduces the need to monitor the reactor shutdown margin. The evaluation of this measurement should determine the feasibility of safely continuing with the scheduled fuel replacement.

The SMS function during reactor startup is to prevent unexpected or premature criticality from control withdrawal by providing a measure of shutdown margin. The control reactivity worth will be well known from reactor calculations and experiments. A measure of subcriticality then determines the amount of control that can be withdrawn prior to reaching criticality. This becomes important during the startup operation with the use of single-purpose control systems. To further identify the particular SMS function during startup, a summary procedure for withdrawing the single-purpose primary or alternate shutdown scram control systems is used. The control system is assumed withdrawn from a subcritical core. The core remains subcritical as it is placed in readiness to offer scram capability. Criticality is assumed to be reached with the shim control system.

A measurement of the reactor shutdown margin prior to startup would demonstrate, in normal circumstances, that the scram system can be removed safely from the core. Anomalous excess reactivity that prevented safe startup would then be easily detected without perturbing the fully shutdown core. An evaluation of the measured subcriticality would determine the potential of safe startup operations prior to their schedule. Any required changes in the core could be introduced without delay.

3.3.4 Functional Objectives

Measurements of subcriticality can be used to determine excess reactivity being introduced during refueling or unscheduled changes in the shutdown margin. These measurements are classified as functional objectives for the SMS. Primarily, safety considerations that define system functions will be satisfied by the design. In addition, reactor operations can be assisted by particular measurements during the refueling operations. The SMS should be used to its maximum capability of providing convenient monitoring of the reactor during the shutdown condition.

Typical LMFBR designs have refueling access to the core assemblies by either rotating or removing the head closure. For refueling with the closure removed, all control rods remain fully inserted into the core.

For head removal, mechanical and electrical components will disengage from the control rods resting in the core. As the closure is removed into the refueling cell, the control should then remain behind. By monitoring the subcriticality during this operation, it is possible to verify that the control is not accidentally withdrawn. Other procedures could be devised that would exclude the possibility of inadvertent withdrawal, but a SMS measurement easily determines any actual changes in the shutdown margin. The monitoring does not interfere with normal events required just to remove the head closure.

During the refueling interval, the SMS is used to verify the safety of fuel replacements. The subcriticality measurement is also used to verify the core excess reactivity. New fuel replaces spent fuel to provide burnup potential to operate until the next scheduled refueling. A direct measurement of the change in reactor shutdown margin during refueling would determine the amount of excess reactivity for burnup being loaded. Under normal conditions, this monitoring would verify that the refueling is successful. Alternatively, an incident of not providing enough burnup reactivity would be detected early and corrected during the normal refueling operations. Without the actual measurement, a lack of fuel would be apparent only after the reactor has again reached power operation. To immediately refuel or shorten the power operation cycle are both time consuming, unpleasant corrective measures. A SMS measurement contributing to the description of the BOC core removes the doubt whether the refueling objectives were met.

These two additional functions, associated with the functional safety requirements, extend the use of the SMS to benefit reactor operations. In a general sense, the SMS will provide surveillance for the fully shutdown reactor. Normal control and surveillance systems are primarily designed to function with an operating critical reactor core. A system designed primarily for use during subcritical operation will increase positive reactor monitoring capability. Decisions for increased safety operation can be made based on this additional measurement information.

3.3.5 Design Basis

The measurements for a SMS to fulfill the functional requirements and objectives are based on a typical LMFBR demonstration plant. The subcriticality measurements are obtained from a description of the design and the calculated reactor parameters. To translate the SMS functions into the selection of a measurement technique, this anticipated subcriticality, combined with reactor environment around the detector, can identify a practical and reliable design.

To demonstrate the safety of continuing the refueling sequence, the means of maximum reactivity insertion by fuel and control replacement is identified. This event would be a gross error in the refueling operation and unlikely to occur. It is, however, the limit that can be reached for a single assembly replacement, control assembly from zone 1 is removed from the core and replaced by a fresh zone 2 (higher enriched) fuel assembly. The SMS measurement must accurately demonstrate prior to a scheduled replacement that, in any such event, the reactor shutdown margin will remain at least 1\$. If the mistake occurs, it will be recognized by the next measurement and an immediate correction can be made.

After the core has been refueled, a SMS measurement would verify the subcriticality is enough to allow scram control removal without reactor criticality. This assumes the reactor will be 1\$ subcritical when either a primary or alternate scram shutdown control system is withdrawn from the core and in operating position. The reactor can reach criticality and power operation from movement of the remaining scram control plus some shim control. The single-purpose scram control then will be available before the reactor is operated at appreciable power.

The reactivity worth of both primary and secondary shutdown control will be well known from independent measurements during the design and construction testing of the reactor. For the purposes of the preliminary SMS design, the calculated worth of each from the LMFBR core description will be used as a representative value.

The total subcriticality of a fully shutdown reactor changes from the EOC core as it is refueled to a BOC core. The magnitude of this shutdown margin in each condition is the nominal value the SMS will measure. To fulfill the safety functions, a measurement need not accurately infer the actual shutdown margin. Rather, it must demonstrate with confidence that the margin is greater than the prescribed safety limit prior to the scheduled event. As the core becomes more subcritical, several measurement techniques lose sensitivity. The actual measure of subcriticality, while not determined for functional requirements, does affect the capability of the SMS.

To meet functional objectives, the SMS will verify the shutdown margin for operational rather than safety purposes. These objectives require measurements on the EOC and BOC shutdown cores. For the EOC core, the removal of the head closure should not reduce subcriticality to less than a minimum shutdown margin expected for fully inserted control. The SMS measurement will demonstrate that this minimum is present. A breach of this minimum is assumed to identify the withdrawal of control along with the head removal. For the BOC core, a maximum level of reactor shutdown margin can be expected if ample excess reactivity for burnup is present. A measurement that verifies that the actual subcriticality is less than this maximum would reasonably assure that the core is properly fueled. To find the refueled core having a greater degree of shutdown would identify a possible oversight.

These descriptions of SMS measurements are further defined in terms of typical reactor description data. For the preliminary SMS design these data become the design requirements. The SMS functions and basis for design will remain unchanged if additional effort refines the LMFBR reactor description. The data only provide representative design requirements that can be used to specify a particular SMS design. The general design will not change with additional data for the core parameters.

3.3.6 Design Requirements

Core parameters applicable to reactor subcriticality are presented in Table 3-6. These are only representative values for a typical demonstration plant design. By using these data, a SMS design is selected that would be compatible with a future final LMFBR plant design.

Table 3-6
CORE PARAMETERS FOR SMS DESIGN REQUIREMENTS

Reactor Shutdown Margin

End of Operating Cycle, Refueling Conditions, All Control Inserted	-40\$ → -34\$
Beginning of Operating Cycle, Refueling Conditions, All Control Inserted	-15\$ → - 9\$

Component Worths

Average Worth of Replacing One Spent Fuel Assembly with One Fresh Fuel Assembly	0.3\$
Maximum Worth of Zone 2 Fuel Assembly Placed in Zone 1	2.1\$
Maximum Control Element Worth	2.5\$

Nuclear Control System Worths (Nominal)

Total Control	46\$
Primary Shutdown System	7.5\$
Alternate Shutdown System	7.5\$

Reactor Operation Data

Fuel Burnup for One-Year Operation	24\$
Time Estimate for Individual Refueling Events	8 to 30 minutes

From the description of measurements that meet functional requirements, these core data are used to formulate design requirements. The approximate maximum reactivity change for an error in refueling is the sum of maximum control assembly worth and maximum fuel assembly worth. This necessary level of shutdown margin required to safely proceed with refueling must be determined while the actual subcriticality is much greater. The nominal full shutdown margin is estimated by the expected range of the reactivity worth for the fully inserted control system. To verify safe withdrawal of the shutdown control, reactor subcriticality is measured to be greater than the reactivity increase by its removal from the core. Further, the reactor will remain greater than -1% subcritical after this control removal. The SMS measurement will demonstrate that the actual shutdown margin is greater than this limit established for safety requirements.

The functional objectives can be met by similar measurements. For a LMFBR design that uses hot cell refueling, the removal of the reactor closure should not change the measure of subcriticality. If the measurement shows the core remains shut down beyond the minimum expected level, then no control rod removal with head closure has occurred. Likewise, to determine if refueling has properly restored sufficient fuel reactivity for burnup, the measurement should show subcriticality no larger than the BOC maximum.

These measurements must occur with accuracy in a practical time limit. The estimate for the time required to efficiently refuel or accomplish a remote handling event is 15 min. The SMS measurement will occur during this interval, providing a verification of the safety for the next operation.

These design requirements are given in Table 3-7. Based on the compilation of representative reactor data and the design basis, these measurements respond to the requirements and objectives of the SMS.

Table 3-7
PRELIMINARY SMS DESIGN REQUIREMENTS

Design Requirement	Functional Requirement or Objective Responsive to
1 At -9 to -40% demonstrate that reactor is at least 6% subcritical in ~ 10 to 30 min	Verification that refueling may safely proceed
2 At $\sim -15\%$ demonstrate that the reactor is at least 9% subcritical in 30 to 60 min	Verification that scram shutdown system can be safely withdrawn
3 At $\sim -15\%$ demonstrate that the reactor is no more than 17% subcritical in 30 to 60 min	Assurance that sufficient reactivity is available for planned burnup reactivity loss
4 At -15% to -40% demonstrate that the reactor is at least 10% subcritical in ~ 30 min	Verification that control elements are not moving with reactor closure

3.3.7 Review of Subcriticality Measurement Techniques

Research and development of measurement techniques provides a summary of several candidates for consideration in a SMS design (References 3 through 9). The feasibility and capability of an accurate measure of subcriticality are more fully discussed in these references. This review summarizes these methods only in enough depth to select a system applicable for the Demonstration Plant.

Each of these methods is based on a reactor physics interpretation of a detector response to the low level neutron flux. Three detectors were considered for their detection capabilities: (1) He³-filled, (2) boron (B¹⁰), and (3) U-235 fission. The result demonstrated that the neutron detection efficiencies of both helium-filled and boron detectors were superior to a similar U-235 detector. Anticipating the high gamma flux present in an operating reactor core, degradation of the detection efficiencies was also examined. Measurement by U-235 fissions is much less sensitive to high gamma background than the other detectors. By considering both aspects of detector response, the advantages of the U-235 fission detector were determined. In gamma background above 10⁵ r/hr, this detector clearly gave better performance.

The measurement techniques which present state-of-the-art technology offers for a SMS design are interpretations of the core response to either an inherent or applied source. An inherent source must be large enough to give meaningful results. The applied source must not only be large enough but also conveniently introduced.

Two methods utilize an additional neutron source: (1) pulsed neutrons and (2) asymmetric sources. The pulsed-neutron technique is based on the core transient response after the introduction of a large burst of neutrons. The method requires a pulsed neutron generator to operate in or near the core.

The interpretation of this measurement has proven difficult for far-subcritical reactors in its current development. Another technique utilizes a source placed in an asymmetric position in the core. The interpretation of changes in the neutron flux distribution provides the measure of subcriticality. To produce the observable flux distortions, however, a source greater than inherent core sources must be used. As the reactor becomes far subcritical, this perturbation in flux also becomes difficult to interpret for measurement.

The two above methods not only require in-core or near-core detection equipment, but also the additional means of providing a flux perturbation. By using the inherent source of neutrons within the core, this requirement is eliminated. Several methods have been devised to monitor the flux to provide a measure of core subcriticality: (a) neutron noise measurements, (b) inverse kinetics, and (c) neutron source multiplication.

Reactor subcriticality can be inferred from analysis of random fluctuations (noise) occurring in the measured neutron flux density. This technique relies on transient information contained in stochastic fluctuations to provide a measurement. Accurate determination, however, requires long measurement time or high detection efficiencies. This efficiency is the rate of detection events normalized to the fission rate in the core. Present technology in noise measurements limits accurate determinations of subcriticality to detection efficiencies above 10⁻⁶ detector events per fissions in the core. An additional limitation from physical considerations limits practical measurements to above -5\$ subcritical. Below this level of shutdown the information in the noise becomes quite small. The detection of neutron flux becomes primarily a random event unusable for subcriticality measurement.

Subcriticality can be measured from the detector time response following a change in reactivity (rod drop) through the inverse solution of the reactor kinetics equations. This inverse-kinetics method can be easily implemented using reactor control rods and normal neutron detection equipment. The technique is more successful near critical when the transient flux is large. As the reactor becomes far shut down, the changes in the flux become indistinct when a control rod is moved. The count rate in a detector is also limited in accuracy by statistical uncertainties. For far subcritical, the core fission rate is naturally less. Related by the detection efficiency, this reduces the count rate and increases inaccuracy in the measurement. The reduced transient data with large uncertainties generally make this method imprecise for reactors over -5\$ subcritical. It is a convenient, accurate technique only near critical.

The neutron source multiplication technique for subcriticality measurement is advantageous for its simplicity and ease of application. This basic concept in subcriticality determination uses a steady-state detector count rate related to subcriticality by knowledge of the inherent neutron source in the core. While the source level can be obtained from calculation, the technique is usually calibrated near critical by another measurement.

As the reactor becomes far subcritical, changes in the stable neutron flux level can be attributed to either changes in reactivity or sources. By using the calibration measurement and calculated changes in the source and detector response, the reactivity can be inferred quite accurately. Reduction in the flux level lowers the statistical accuracy of the detector count rate as with the other techniques. With reasonably longer measurement times, this difficulty is overcome.

A general difficulty is associated with each of these methods of measurement. The level of subcriticality is known only after accurate reactor physics interpretation of the detector response. The individual analytical techniques have been established that demonstrate the feasibility of their use. In practice, specific experimental verification of this has only begun to be investigated. Each method is therefore limited by the ability to properly and accurately use the measured data.

3.3.8 Design Selection

A measurement technique using equipment and detectors outside the core will have a distinct advantage. By not interfering with head closure removal and refueling activities over the core, the SMS will conveniently provide measurement of subcriticality. Techniques requiring external sources introduced to the reactor are quickly eliminated from design consideration.

Subcriticality measurements to -30% and -40% are part of the preliminary design requirements. The only practical means of measurement at far subcritical is by the neutron source multiplication technique. Unfortunately, this method requires a calibration measurement near critical and supporting calculated corrections as supplements for an accurate, reliable determination of subcriticality. The use of corrections has been shown practical from development results. The initial calibration measurement is an independent measure of reactor subcriticality. The use of noise or inverse-kinetic techniques is convenient to provide this supplemental measure at near critical.

Noise measurements require detector sensitivities that are generally not available for locations beyond the reactor core boundary. Figure 3-9 shows the radial profile of expected detector efficiency at the core midplane. The curve is based on a 2 gram U-235 fission detector with counting losses estimated for a 10^7 to 10^8 r/hr gamma background. Detector efficiencies shown for outside the core are much less than the 10^{-6} minimum accepted for practical subcritical measurements by noise techniques.

By the rod drop method of inverse kinetics, calibration can be conveniently obtained for the measurement by the neutron source multiplication technique. Aside from proper physics interpretation of the measurement, the statistical uncertainty from the detector count rate is the primary operational inaccuracy. Table 3-8 gives a summary calculation showing these statistical uncertainties from a detector location outside the core. Satisfactory accuracy can be obtained from these two techniques from a detector location beyond the lateral core support. Here, the detector is not interfering with any maintenance or refueling operation in the core.

By having no special requirements other than low level neutron monitoring detection, the SMS equipment can be integrated into other reactor systems. It is feasible to have a U-235 fission detector used for subcriticality measurements and also serve other source range neutron monitoring functions. To fully utilize the detection system supporting the SMS, it will also be incorporated into other neutron monitoring systems. The requirements for this detector can be met with an efficiently design device with 1 to 2 gm of U-235 fissionable material. Such a detector was devised during the ORNL measurement development program described in References 3 through 9. A summary of the preliminary SMS design is presented in Table 3-9.

3.3.9 Additional Design Development Required

The SMS has been demonstrated as a practical, reliable procedure of determining reactor subcriticality over the full shutdown range. The uses of calculated reactor physics interpretations have been identified. From these results, a selection is made of measurement techniques suitable to demonstration plant design. The additional design development needs to be directed toward confirming the reactor physics interpretation of detector measurements. Large irregular changes in the core composition occur during refueling operations.

Table 3-8
STATISTICAL UNCERTAINTY OF SMS MEASUREMENT

Reactor Environment

Inherent Neutron Source	10 ⁸ f/sec
Multiplication at -2\$	150 X
Multiplication at -45\$	7 X
Detector Efficiency in Vessel Shield	3 X 10 ⁻⁸

Uncertainty in Measured Data

Transient Measurement (Inverse Kinetics)

Detector Countrate at -2\$	450 cps
Uncertainty	± 5%

Static Measurement (Source Multiplication)

Detector Countrate at -45\$	30 cps
Measurement Time	15 min
Uncertainty	± 1%

Table 3-9
PRELIMINARY SUBCRITICALITY MONITORING SYSTEM DESIGN

Technique	Neutron source multiplication supplemented by calculated data analysis calibration near critical with control rod drop analyzed by inverse kinetics
Detector	U-235 fission chamber, 1 to 2 grams of fissionable material
Detector Location	Outside core lateral support in vessel shield

The adequate prediction of these effects on SMS data remains uncertain without continued efforts defining specific problem areas and developing acceptable solutions. The calculated interpretations of measurements require more design effort to develop procedural standards assuring accuracy. These calculations are feasible, effectively utilizing their contribution to the measurement requires more specific definition of the actual measurements during refueling and startup. The systematic uncertainties associated with these calculations must be experimentally determined. Testing should be directed toward confirming their proper and reliable use in an actual LMFBR environment.

In another area, the SMS measurement procedures and detection equipment need integrating with the total reactor system. Procedural requirements are derived by specification of actual refueling, maintenance, and startup activities. Proper SMS integration will assure minimum interference but complete contribution. Similarly, by incorporating SMS detector requirements with other reactor low level neutron flux monitoring systems, the most efficient and reliable use of a subcriticality measurement is attained.

Figure 3-9. Detector Efficiency at the Core Midplane

3.4 TASK C—ON-LINE REACTIVITY COMPUTER

3.4.1 Summary

A conceptual On-Line Reactivity Computer (ORC) was developed and a preliminary evaluation was made of: (1) its recommended use in the LMFBR program; (2) practical problems associated with including it in the Plant Protection System (PPS), and (3) recommendation of additional work.

3.4.2 Conceptual On-Line Reactivity Computer

The ORC is a computer (analog, digital, or hybrid) used to solve the equations representing the reactivity state of the reactor. "On-line" means that the calculations are performed continuously or at a predetermined frequency as opposed to on-demand or at a later time. The successful design of the ORC depends on identifying and mathematically representing all significant reactivity effects. These reactivity effects are related to plant operating conditions such as power, temperature, and time. The ORC is programmed with the reactivity equations and has as signal inputs the plant operating conditions. In operation, the ORC essentially sums the reactivity terms which should equal zero. The zero sum indicates that the reactor is performing as predicted and there are no anomalous effects. However, in initial operation this sum will probably not be zero because not all reactivity coefficients and effects will be known precisely. The presence of anomalous reactivity will then have to be determined from other indications. Initial reactor testing will permit refining the reactivity equations and will bring the ORC into balance. The ORC will require this type of adjustment each time the reactor core is reloaded.

The purpose of the ORC is to detect discrepancies between the expected and actual reactivity state of the reactor and thus detect abnormal reactor conditions. The reactivity model accounts for the following reactivity effects:

1. Doppler
2. fuel and cladding expansion
3. coolant expansion
4. structure radial expansion
5. fuel bowing
6. fuel burnup
7. Pu-241 decay
8. control system reactivity.

The sum of the above core reactivity effects is compared to the measured reactivity of the reactor. A block diagram of the ORC conceptual design is given in Figure 3-10.

As the design of LMFBR plants progresses it may be possible to simplify ORC by eliminating factors which have negligible reactivity effects. Likewise, the kinetic reactivity calculation may be eliminated if analysis indicates that conceivable core malfunctions will be detected without this term. Subsection 3.4.6 describes the mathematical model of an ORC design and gives a discussion of some of its uncertainties. In principle, the ORC for the LMFBR plant may cover the range from zero power to full power. However, practical considerations (tradeoff of equipment complexity against additional protection) may restrict the operation to the power range. In any case, a reference point would be chosen and any change in core reactivity with respect to the reference point will be recorded as the reactor rises in power. A careful record of all reactivity changes from the initial reference point will be maintained in order to check the reactivity model from time to time. Periodic rezeroing of the computer will probably be made to update the reference point. This rezeroing will permit greater sensitivity for detection of anomalous reactivity.

Figure 3-10. On-Line Reactivity Computer Conceptual Design

All of the reactivity effects can be related to reactor vessel inlet coolant temperature, reactor vessel outlet coolant temperature, control rod position, neutron flux, and operating time. The requirements of the ORC are not anticipated to place any additional measurement requirements on the LMFBR plant.

The following malfunctions are examples of the type which ORC will be designed to detect:

- 1 flow blockage in one or more channels,
- 2 shift in flow distribution between the core and blanket region,
- 3 reposition of core mechanical components,
- 4 partial core voiding due to gas entrainment,
- 5 control rod disengagement, and
- 6 fuel redistribution, as a result of overpower, without cladding failure.

An estimate of the magnitude of the anomalous reactivity resulting from the first five of the above malfunctions is made in the following paragraphs. In detecting these core malfunctions the ORC will work together with such other systems as the FEDAL, the acoustic monitoring, and the channel instrumentation. An estimate of the effectiveness of the ORC in relation to these other detection systems will be made as the LMFBR plant design progresses.

As an example of the first malfunction, it is assumed that there is flow blockage of one channel in the innermost ring. This causes local coolant temperature increase and a fuel temperature increase. The increase in coolant temperature could lead to local boiling (20 cm below the midplane to 20 cm above the midplane) with the reactivity effect due to void calculated to be approximately 6¢. Following the voiding, the fuel temperature would further increase and could finally reach the melting point. The reactivity effect resulting from the temperature increase from full power to the melting point in the assembly is about -2¢. If the fuel reaches the melting point, the molten fuel could leak out of the cladding. The maximum possible reactivity effect due to fuel compaction toward the midplane is approximately 41¢ for one assembly. Eventually, the flow in a plugged channel could resume and the molten fuel could be swept out. The net reactivity effect due to complete removal of the fuel assembly is -2.05¢.

In the second malfunction it is assumed that a 10% decrease in core flow occurs, causing a 30°F increase in core outlet temperature with the core inlet temperature remaining constant. The reactivity change resulting from 10% core flow decrease is calculated to be -12¢.

In the third malfunction it is assumed that the clamping system failure causes a change in core diameter. The reactivity effect of the core diameter increase is calculated to be -2.1¢/in.

In the fourth malfunction, two arbitrarily chosen large gas entrainment cases are considered:

- a entrance of 1 ft³ of inert gas in six fuel channels near the center of the core, and
- b dispersal of 1 ft³ of inert gas in the full core.

The reactivity resulting from case a has been calculated to be a positive 44¢. Case b leads to reactivity reduction of 62¢ because the increased leakage effect dominates the effect of a reduction in inelastic scattering.

In the fifth malfunction, since LMFBR control rods have reactivity worths from one to several dollars, a disconnected rod would give a relatively large ORC signal as the drive moved without its poison.

A detection sensitivity of 10 ¢ appears to be a reasonable design goal for ORC (see subsection 3.4.6). The ORC used in the Enrico Fermi FBR plant (EFFBR) has a sensitivity of approximately 3 ¢. Based on what EFFBR has accomplished and the knowledge that the reactivity effects in the LMFBR plant are considerably more complex, the 10 ¢ goal is considered reasonable. A sensitivity of 10 ¢ leads to the following conclusions relative to the five malfunctions investigated:

Malfunction Number	Conclusion
1	Detected during fuel compaction
2	Could detect approximately 8% core flow change
3	Could detect 0.05-in. core diameter change
4a	Could detect 1/4 ft ³ gas entrainment
4b	Could detect 1/6 ft ³ gas entrainment
5	Could detect relatively small control rod movement

3.4.3 Practical Problems Relative to Incorporating ORC in the PPS

RDT Standard C16-1T has been studied to determine practical problems relative to the use of ORC in the Plant Protection System (PPS). The first problem appears to be justification for its incorporation in the PPS. If a postulated accident can be identified where the ORC is the only defense for protection of plant or public, then its incorporation would be obvious. However, the ORC falls into the category of an anticipatory signal, providing indication of impending accidents. Section 3.5 of RDT Standard C16-1T states that anticipatory trip variables should be included in the PPS only when absolutely essential as justified by analysis. If the ORC is included in the PPS, the following are considered to be additional practical problems:

1. Three separate computers would be required to meet the single-failure and redundancy criteria.
2. Sensors such as those for control rod position would have to be made safety grade and provided in triple redundancy.
3. Increased equipment cost would be small compared to expanded engineering, installation, and quality assurance effort required to meet redundancy, separation, isolation, and all other criteria pertinent to the PPS.

3.4.4 Recommended Use of the On-Line Reactivity Computer

Based on these preliminary results, it is recommended that the ORC be developed for incorporation in the LMFBR plant. Analysis of five possible core malfunctions indicated that they would be detected with an achievable ORC sensitivity. Early detection provides the chance to prevent or minimize plant damage due to a malfunction. Since ORC is a developmental system, its use should be limited to indication and warning.

3.4.5 Suggested Additional Work

Additional work on ORC should include generation of a program plan and schedule based on the results of this study. This program plan should consider such items as:

- 1 reactivity effect and speed of response required to detect different hypothetical malfunctions,
- 2 significance of ORC in relationship to other measurement systems,
- 3 calculation of reactivity feedback coefficients,
- 4 calculation of control rod reactivity worth for different patterns,
- 5 sensitivity requirements for ORC,
- 6 computational techniques for ORC,
- 7 experimental verification of ORC during plant startup,
- 8 specification for ORC (including type of computer, software requirements, etc),
- 9 frequency of calculation,
- 10 operational range of ORC, and
- 11 frequency of reactivity trim adjustment

3.4.6 Reactivity Model

3.4.6.1 Mathematical Expression

In a typical LMFBR, the static core reactivity balance with respect to an initial condition (zero power) is given by

$$\Delta K_s = A \ln \left(1 + \frac{\Delta T_c + R_f P}{T_{co}} \right) + C_p P + C_i \Delta T_i + C_o \Delta T_o + C_q \Delta T_c + \Delta K_{bu} + \Delta K_{pu} + \Delta K_{bo} + \Delta K_{control} \quad (1)$$

where

- A = Doppler coefficient
- P = Reactor power level
- T_{co} = Initial average coolant temperature
- ΔT_c = Average coolant temperature change
- R_f = Change in effective fuel temperature with respect to power
- C_p = Fuel axial expansion and cladding axial expansion reactivity coefficient due to power change
- C_q = Coolant reactivity coefficient (fuel axial expansion, cladding axial expansion and coolant expansion due to coolant temperature change)
- C_i = Structure reactivity coefficient due to inlet temperature change

- C_o = Structure reactivity coefficient due to outlet temperature change
- ΔT_i = Inlet temperature change
- ΔT_o = Outlet temperature change
- ΔK_{bu} = Reactivity change due to fuel burnup in the core and reactivity change due to fuel generation in the blanket
- ΔK_{pu} = Reactivity change due to Pu-241 decay
- ΔK_{bo} = Reactivity change due to bowing effect
- $\Delta K_{control}$ = Reactivity change due to control rod position changes.

It should be noted that

$$\Delta T_c \cong \frac{1}{2} (\Delta T_i + \Delta T_o) \quad (2)$$

With regard to Equation 1:

- The temperature exponent m in the Doppler effect is assumed to be 1. The actual m for the demonstration plant may lie between 0.8 and 1.0.
- The Doppler coefficient A is a function of the integrated power, i.e.,

$$A \cong A_o + \alpha \sum_i P_i t_i \quad (3)$$

where

A_o = Initial Doppler coefficient

α = Doppler coefficient change per unit integrated power

$\sum_i P_i t_i$ = Integrated power

- The burnup reactivity change may be approximated by the following expression:

$$\Delta K_{bu} \cong b_1 \sum_i P_i t_i \quad (4)$$

where

b_1 = Reactivity change per unit integrated power due to fuel burnup and fuel generation in the blanket

- d The reactivity change due to bowing effect may be approximated by

$$\Delta K_{bo} \cong c_1 \Delta T + c_2 \left(\frac{\sum_i P_i t_i}{P_o t_o} \right)^{1.57} \quad (5)$$

where

c_1 = Short-term reactivity coefficient due to bowing effect

ΔT = Core temperature rise

c_2 = Long-term reactivity coefficient due to bowing effect

$\sum_i P_i t_i$ = Integrated power

$P_o t_o$ = Reference integrated power

- e The reactivity change due to Pu-241 decay can be expressed by

$$\Delta K_{pu} = d_1 t \quad (6)$$

where

d_1 = Reactivity change per unit time due to Pu-241 decay

t = Time

- f The control rod position reactivity effect, $\Delta K_{control}$, will be determined from the changes in rod position from the datum condition and the insertion pattern of the remaining rods. The form of this model has not yet been determined.

- g Substituting Equations 3 through 6 into Equation 1 yields the following approximation

$$\begin{aligned} \Delta K_s \cong & \left(A_o + \alpha \sum_i P_i t_i \right) \ln \left(1 + \frac{R_f P}{T_{co} + \Delta T_c + 460} \right) + C_p P \\ & + e_1 \Delta T_c + e_2 (\Delta T_c)^2 + C_1 \Delta T_1 + C_o \Delta T_o + b_1 \sum_i P_i t_i + d_1 t \\ & + c_1 \Delta T + c_2 \left(\frac{\sum_i P_i t_i}{P_o t_o} \right)^{1.57} + \Delta K_{control} \end{aligned} \quad (7)$$

where

$$e_1 = c_q + \frac{A}{T_{co} + 460}$$

$$e_2 = \frac{-A}{2(T_{co} + 460)^2}$$

A series of static tests during initial rise to power can be developed to determine the coefficients A , C_p , e_1 , e_2 , C_1 , C_o , and c_1 experimentally and to evaluate the necessary control rod worths and interaction effects.

3.4.6.2 Preliminary Results and Uncertainties

Referring to Equations 1 and 7, the following values are based on preliminary calculations of a typical LMFBR:

$$\begin{aligned}
 A_0 &= -142\phi \\
 \alpha &= -0.104\phi/\text{full power day} \\
 b_1 &= -9.16\phi/\text{full power day} \\
 c_1 &= -0.203\phi/^{\circ}\text{F} \\
 c_2 &\cong 0 \\
 d_1 &= -3\phi/\text{month} \\
 R_f &= 1.5^{\circ}\text{F/mW}
 \end{aligned}$$

The calculated reactivity coefficients at the rated power are

$$\begin{aligned}
 C_p &= -0.058\phi/\text{MWt} \\
 C_i &= -0.032\phi/^{\circ}\text{F} \\
 C_o &= -0.180\phi/^{\circ}\text{F} \\
 C_q &= -0.004\phi/^{\circ}\text{F}
 \end{aligned}$$

The uncertainty associated with control rod reactivity is assumed to be 4% and with every other reactivity term in the reactivity model is estimated to be 20%. Therefore, the uncertainty in the ΔK_s calculation is approximately

$$\delta\Delta K_s \cong \left[\sqrt{(\Delta K_D)^2 + (C_p P)^2 + (C_i \Delta T_i)^2 + (C_o \Delta T_o)^2 + (C_q \Delta T_c)^2} \right. \\
 \left. + \frac{(\Delta K_{bu})^2 + (\Delta K_{pu})^2 + (\Delta K_{bo})^2 + (\Delta K_{\text{control}})^2}{4\%} \right] 20\% \quad (8)$$

where

$$\Delta K_D = A \ln \left(1 + \frac{\Delta T_c + R_f P}{T_{co}} \right) \quad (9)$$

For a typical case in which the power is increased approximately 100 mW, and after more than two days operation the following are estimated values:

$$\Delta K_{\text{control}} = 50\phi$$

$$\Delta K_{bo} = 0$$

$$\Delta K_{pu} = 0$$

$$\Delta K_{bu} = -20\phi$$

$$C_q \Delta T_c = 0$$

$$C_o \Delta T_o = 0$$

$$C_l \Delta T_l = 0$$

$$C_p P = -5\phi$$

$$\Delta K_D = -25\phi$$

the calculated uncertainty associated with the reactivity balance is approximately 6.8ϕ , which constitutes about 13% of the core reactivity change. However, as time goes on, the uncertainty will be much reduced because of the updating of the reactivity model and thus more accurate prediction of the core reactivity change can be expected. For example, the uncertainty in reactivity balance, $\delta \Delta K_s$, would be reduced to 0.6ϕ for the same case, if the uncertainty associated with each reactivity term in the reactivity model is changed to 1%.

3.4.6.3 Kinetics Inversion

The kinetic reactivity is obtained from solving the kinetics equations using kinetics inversion technique. The delayed neutron and neutron lifetime information is given in Table 3-10 and the kinetic reactivity is given by the following equation:

$$\left(\frac{1 - \beta_{eff}}{\beta_{eff}} \right) K_{ex}(t) = 1 + \left(\frac{\ell}{\beta_{eff}} \right) \frac{1}{\phi(t)} \frac{d\phi(t)}{dt} - \frac{\phi(0)}{\phi(t)} \sum_{i=1}^6 \alpha_i e^{-\lambda_i t} - \frac{1}{\phi(t)} \sum_{i=1}^6 \alpha_i \lambda_i e^{-\lambda_i t} \int_0^t \left[1 + K_{ex}(t') \right] \phi(t') e^{\lambda_i t'} dt' \quad (10)$$

where

$\phi(t)$ = neutron flux at time t

$K_{ex}(t)$ = time dependent excess reactivity

α_i = fraction of the total effective delayed neutron fraction in the i^{th} group = β_i/β_{eff}

Table 3-10
DELAYED NEUTRON CONSTANTS

Delayed Neutron Precursor Group i	Effective Delayed Neutron Fraction β_i	Delayed Constant for Group i $\lambda_i, \text{sec}^{-1}$
1	8.19698×10^{-5}	1.28×10^{-2}
2	7.22549×10^{-4}	3.14×10^{-2}
3	6.15784×10^{-4}	1.352×10^{-1}
4	1.18735×10^{-3}	3.434×10^{-1}
5	5.24118×10^{-4}	1.366
6	1.68235×10^{-4}	3.804

Total effect Delayed Neutron Fraction: $\beta_{\text{eff}} = 3.3 \times 10^{-3}$

Prompt Neutron Lifetime: $\ell = 5.76 \times 10^{-7} \text{sec}$

4. ADMINISTRATIVE

4.1 TRIPS

April 4, 1972 — L. C. Wimpee to HEDL & WHCO (N. C. Hoitink, L. Phillipp, J. Stringer, M. Wood) to discuss subjects related to neutron detectors and cables in high gamma and high temperature environments.

April 27, 1972 — R. Pflasterer, C. Larson, D. Hines and L. McCarty, AEC-RDT to discuss preliminary 189 proposals for FY-1973.

April 27, 28, and 29, 1972 — L. C. Wimpee to Argonne National Laboratory for discussions with G. F. Popper and A. E. Hirsch, and to Oak Ridge National Laboratory for discussions with D. P. Roux and J. DeLorenzo.

May 26—June 11, 1972 — G. R. Pflasterer participated in an AEC-sponsored trip of U.S. instrument and control specialists to Germany, France, and the United Kingdom.

4.2 VISITS

April 6, 1972 — Dr. T. P. Mulcahey of ANL and Dr. J. Wallach of GE-R&D Center visited Sunnyvale to discuss the acoustic monitoring program.

May 19, 1972 — Past accomplishments and future plans were presented to representatives of RDT, Atomics International (AI) and Westinghouse (W) in Sunnyvale. Dr. E. A. Womack of RDT, D. W. Staub and E. B. Ash of AI and G. Macrae of W were the visitors. After the GE presentations, discussions concerning each task and other instrumentation and control areas were held and comments by each reactor manufacturer's representative were made.

4.3 LIST OF CONTRIBUTORS

Breeder Reactor Department

H. S. Bailey
G. Billuris
L. A. Carmichael
E. R. Craig
S. L. Derby
J. N. Fox
T. E. Gleason
D. Glover
P. Greebler
P. J. Hendriksz
D. P. Hines
C. L. Larson

Y. S. Lu
L. McCarty
W. G. Meinhardt
E. E. Olich
G. R. Pflasterer
J. J. Regimbal
C. E. Russell
G. I. Samstad
B. F. Shoopak
P. W. Swartz
P. C. Vaughan
M. L. Weiss

Atomic Power Equipment Department (Nuclear Instrumentation Group)

Y. Dayal
V. W. Perry
G. W. Titus
L. C. Wimpee

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Division of Reactor Licensing U. S. Atomic Energy Commission Washington, D. C. 20545 Attn: Assistant Director Reactor Technology	2	Manager FFTF Project Hanford Engineering Development Laboratory P. O. Box 1970 Richland, Washington 99352	5
Division of Reactor Standards U. S. Atomic Energy Commission Washington, D. C. 20545 Attn: Director	2	Director LMFBR Program Office Argonne National Laboratory 9700 South Cass Avenue Argonne, Illinois 60439	2
U. S. Atomic Energy Commission Washington, D. C. 20545 Attn: Librarian	2	Associate Director Engineering Research & Development Argonne National Laboratory 9700 South Cass Avenue Argonne, Illinois 60439	3
RDT Site Office U. S. Atomic Energy Commission Argonne National Laboratory P. O. Box 2108 Idaho Falls, Idaho 83401	1	Associate Director Reactor and Engineering Sciences Oak Ridge National Laboratory P. O. Box X Oak Ridge, Tennessee 37830	1
RDT Site Office U. S. Atomic Energy Commission Argonne National Laboratory Building 2 Argonne, Illinois 60439	1	Director Liquid Metal Engineering Center Atoms International P. O. Box 1449 Canoga Park, California 91304	5
RDT Site Office U. S. Atomic Energy Commission Atoms International P. O. Box 1446 Canoga Park, California 91304	1	Manager Idaho Nuclear Corporation P. O. Box 1845 Idaho Falls, Idaho 83401	1
RDT Site Office U. S. Atomic Energy Commission General Electric Company 310 DeGuigne Drive Sunnyvale, California 94086	1	Office of the President Atoms International A Division of North American Rockwell Corporation P. O. Box 309 Canoga Park, California 91304	1
RDT Site Office U. S. Atomic Energy Commission Oak Ridge National Laboratory P. O. Box X Oak Ridge, Tennessee 37830	1	Manager Advanced Reactor Division Westinghouse Electric Corporation P. O. Box 158 Madison, Pennsylvania 15663	1
RDT Site Office U. S. Atomic Energy Commission Westinghouse Electric Corporation P. O. Box 154 Madison, Pennsylvania 15663	1	General Manager Breeder Reactor Development Operation General Electric Company Sunnyvale, California 94086	1
President Hanford Engineering Development Laboratory P. O. Box 1970 Richland, Washington 99352	1	Contracting Officer U. S. Atomic Energy Commission 2111 Bancroft Way Berkeley, California 94704	3