SODIUM REACTOR EXPERIMENT (SRE) SHIELDING
EVALUATION FOR THERMAL NEUTRON STREAMING
AT REACTOR VESSEL COOLANT
PIPE PENETRATIONS

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ABSTRACT

The experimental program performed in the SRE auxiliary and main primary galleries was part of a program to determine the adequacy of the shielding configuration for the SRE. The work discussed in this report is concerned with analysis of neutron streaming at coolant pipe penetrations of the reactor vessel, analysis of the shielding required, testing and evaluation of recommended shielding, and measurement and correlation of neutron streaming in labyrinths with theory. The activation analysis method using zinc sheets which was developed for the program of determining thermal neutron streaming in the SRE primary galleries was proven to be versatile, accurate, and reliable. A modified form of the theoretical method of Price, Horton, and Spinney, used to determine neutron scattering through labyrinths, was found to agree favorably with the experimental results obtained from the SRE primary galleries. The theoretical attenuation method used to determine the neutron shield configuration installed in the auxiliary primary gallery was found to give an overestimate of the actual attenuation properties of this shield. The neutron shield configuration installed in the auxiliary primary gallery proved to be adequate in reducing the thermal neutron streaming flux to an acceptable level. It is concluded that both SRE primary galleries are now adequately shielded to prevent excessive neutron-induced activation of the components and equipment located therein.

I. INTRODUCTION

The experiments which were initially performed to the SRE auxiliary and main primary galleries were part of a program to determine the adequacy of the shielding configuration for the SRE. Excessive neutron streaming in the SRE galleries was considered possible due to the cutouts required in the thermal shield in order to permit positioning of the coolant inlet and outlet pipes. The extent of neutron streaming in the SRE galleries was initially evaluated experimentally by the use of gold foils placed in the galleries. Analysis of this data indicated the possibility of excessive after shutdown radiation levels due to activation of components in the galleries. These levels were found to be in the range where they would limit after shutdown maintenance. Based on these results and data obtained during a radiation survey of the SRE, further experimentation was undertaken to determine the position, as well as the extent of the neutron streaming, in the SRE galleries. The problem that was to be investigated by the initial activation analysis experiments was the determination of the extent and position of neutron streaming (over the entire area of the bellow assemblies' entrance into each of the SRE primary galleries) using zinc sheets. Once these data were obtained, possible neutron shield configurations were determined and installed wherever excessive neutron streaming through the bellows assemblies existed.

As an extension of the experimental program initiated to determine the adequacy of SRE shielding, additional neutron streaming experiments were performed in the SRE auxiliary primary gallery after a proposed neutron shield configuration had been installed in the bellows region of this gallery. Experimental results which had been obtained from this region prior to the shield installation and experimental results which were necessary from this region after the shield installation were compared. In order to re-evaluate the neutron streaming problems in this region, the effectiveness of the neutron shield configuration was to be determined, and the gamma-ray dose rates due to activation of the piping in the auxiliary primary gallery were to be re-evaluated. Other activation analysis experiments, using zinc sheets, were performed to re-evaluate the effectiveness of right-angle bends for reducing the streaming neutron flux, and

to evaluate the possible neutron streaming paths through and along sodium piping in the auxiliary primary gallery, specifically in the exit region of the piping from the bellows. These results were compared with the initial experimental data, and with theoretical calculations, to determine the accuracy as well as the validity of an analytical method that could be used for evaluating such special shielding problems. Also, the method of detection was to be evaluated for its versatility and accuracy to determine if other experiments could be devised which would provide further useful information.

II. DETERMINATION OF EXPERIMENTAL METHODS

The method of activation analysis used in this experiment was developed so that the large areas to be investigated could be properly analyzed. Since the regions to be investigated would be inaccessible during operation (as well as for a few weeks after shutdown) due to the high radiation intensity of the radioactive sodium coolant, a method was devised so that personnel would not be exposed to this excessive radiation. The detection media had to be of sufficient sensitivity to provide the information desired. The only reliable, as well as practical, method would involve a material that could be activated by the neutron fluxes of the magnitude obtained by extrapolation of previous measurements and yet retain most of its activity for at least a few weeks after shutdown. Since the use of neutron detecting instruments was not feasible in the regions to be investigated, and neutron sensitive photographic film was undesirable from the standpoint of economics as well as reliability for the case at hand, only activation analysis was considered feasible for the experiment to be performed.

A number of materials were investigated to determine the best possible material that could be used to obtain the information required from these experiments. Materials were investigated for their nuclear properties, availability, and cost. The material, when activated, was required to have a relatively long half life, and had to emit relatively energetic gamma rays. The thermal neutron absorption cross section considering the expected thermal neutron flux levels and irradiation time had to be sufficient to cause activation of the material, so that the count rate from a scintillation detector would be at least twice the background count rate. The material had to be obtainable in sheets large enough to cover the areas indicated in Figures 1 and 2, and thin enough not to depress the thermal neutron flux in these areas. The material had to be able to withstand rather high temperatures ($\approx 100^{\circ}\text{C}$) without changing its physical properties. Since very little time was available for planning and installation of the experiment, the material had to be readily available. The cost of the material had to be reasonable because due to the type of experiment that would be performed, the material should only be used once.

Figure 1. Position and Areas of Zinc Sheets In Bellows Assembly of SRE Auxiliary Primary Gallery

Of the materials investigated, zinc was chosen for use in this experiment, since it met all of the necessary requirements that have been described above. Zinc has a thermal neutron absorption cross section of 1.1 barns and is an element which has few impurities when processed in sheets.⁷ The half life of Zn^{65} , the radioactive isotopes that would be counted after irradiation of the zinc sheets, is 245 days. This long half life proved to be very advantageous, since no decay correction of the data was necessary, even though the zinc sheets had different cooling periods. The decay scheme of Zn^{65} indicates two modes of decay, K-capture in 97.5% of the cases, and β decay in the remainder.⁸ The energy of the positron given off is 0.32 Mev and the gamma ray that is in cascade with the electron capture is 1.12 Mev. A soft X-ray (≈ 10 kev) is given off during the K-electron capture process, which also can be detected by the counter. Both of these gamma rays could easily be detected by the gamma scintillation detector used.

**Figure 2. Position and Areas of Zinc Sheets in Bellows
Assembly of SRE Main Primary Gallery**

Large zinc sheets (3 by 10 ft) with a thickness of 20 mils were found readily available commercially. This thickness of zinc would not depress the thermal neutron flux enough to give erroneous results. The cost of the zinc was reasonable (approximately 40 cents/ft²) and acceptable for the performance of this experiment. In addition, the probability of detecting activity due to any impurities in the zinc was further reduced, by adjusting the discriminator setting on the detector apparatus, so that only the 1.12-Mev gamma energy of Zn⁶⁵ would be counted by the recording apparatus. The 10-kev X-ray was not counted due to the excessive noise present at this discriminator setting.

The activation analysis performed on a sample of the zinc indicated that, after an SRE shutdown time of two weeks (time necessary for the primary sodium activity to decay sufficiently to allow entrance into the auxiliary primary gallery), the activity due to Zn⁶⁵ would account for all of the activity counted by the detector. These results are shown in Figure 3. At least a five-day decay period is necessary after irradiation of the zinc sheets to allow the activity due to the impurities and other zinc isotopes (Zn⁶⁹, Zn⁷¹, Zn⁷²) to decay to an activity level that will not contribute to the total activity of the zinc sheets.

The nuclear and physical properties for the commercial zinc (Zinalloy) used as the detection media for these experiments are summarized in Table I. Before an attempt is made to use this detection media for similar experiments in other reactor regions, the listed properties should be checked against the detector requirements for the particular experiment to be performed.

Figure 3. Gamma-Ray Decay Analysis Curve for Zinalloy

TABLE I

NUCLEAR AND PHYSICAL PROPERTIES FOR COMMERCIAL ZINC (ZINALLOY)

Thermal neutron absorption cross section*	$\sigma_{\text{abs}} = 1.1$ barns
Thermal neutron activation cross section (Zn^{64})*	$\sigma_{\text{act}} = 0.44$ barns
Natural abundance of Zn^{64} *	% = 48.89
Half life of Zn^{65} †	$T_{1/2} = 245$ days
Energy of Zn^{65} gamma ray†	$E_{\gamma} = 1.12$ Mev
Energy of Zn^{65} positron†	$E_{\beta^+} = 0.32$ Mev
Purity of commercial zinc (Zinalloy)§	Zinc content = 100% (activation analysis)
Absorption cross section relationship to neutron energy for Zinalloy*	Relationship up to 0.2 kev = $\frac{1}{V}$
Resonance energy peaks and total neutron cross section for Zn^{64} *=	a) at 2.75 kev - $\sigma_t = 59$ barns b) at 4.6 kev - $\sigma_t = 8.1$ barns
Size of available Zinalloy sheets (as used in experiment); other sizes available with thicknesses up to 1/2 in.	3 ft by 10 ft by 20 mils
Approximate cost of Zinalloy	40 cents/ft ² (20-mil thickness)
Availability of Zinalloy sheets	Delivery time = 2 weeks

*See reference 3

†See reference 4

§See reference 7

III. EXPERIMENTAL PROCEDURE

The experimental procedure for the installation, removal, counting, and analysis of the zinc sheets is relatively simple; depending, for the most part, upon the relative accessibility of the area, or areas to be investigated. For the experiments performed in the SRE primary sodium galleries, the main difficulty encountered was in the installation and positioning of the zinc sheets in the galleries. The relative inaccessibility of the areas under investigation was caused by the presence of sodium piping, tetralin piping, pipe expansion equipment, detection equipment, and heaters.

A. INITIAL EXPERIMENTS DURING NOVEMBER 1957 POWER RUN

In the experiments performed during the SRE November power run, zinc calibration sheets were placed in the same positions in the SRE galleries as were used for the preliminary experiments. The zinc calibration sheet for the main primary gallery was positioned below the west viewing device hole in shield block 1 so that zinc activity data would correspond to the gold foil data previously obtained in the same position (Figure 4). The calibration zinc sheets for the auxiliary primary gallery were positioned below the east viewing-device hole in shield blocks 12 and 13 (Figure 4) so that, again, the zinc-activity data would correspond to the gold-foil data obtained in the same position. The exact positions of the calibration zinc sheets in the SRE galleries are given in Figures 5 and 6.

After a total irradiation of approximately 59 Mwd, accumulated over a period from November 1957 to February 1958, at varying reactor thermal power levels, the SRE was shutdown for modification and inspection. As soon as the SRE galleries were accessible from the standpoint of the existing radiation levels and the operational schedule, the activated zinc sheets were removed and sent for activation analysis. The zinc sheets were cleaned and cut into strips so that they could be counted in a continuous scanner (Figure 7) which had a movable platform with a length of 3 ft and a width of 1.5 in. A continuous recorder, which was connected to the detector apparatus, plotted the count rate of zinc strip at a speed corresponding to the platform speed. In this way, the position of the recorded activity could be accurately determined in respect to the position of the

Figure 4. Floor Plan of the SRE Reactor Area

Figure 5. Position and Area of Calibration Zinc Sheets in SRE Auxiliary Primary Gallery

Figure 6. Position and Area of Calibration and Attenuation Determination Zinc Sheets in SRE Main Primary Gallery

Figure 7. Detection Equipment Used for Zinc Strip Counting

zinc sheets in the SRE galleries. The data obtained from the counting apparatus was in the form of a continuous graphic (chart) record, which indicated the activity detected for the zinc sheets.

The following information was noted on the data charts obtained from each zinc strip:

- 1) Position of the zinc strip counted in respect to the zinc sheet from which it was cut
- 2) Position of the zinc sheet in respect to the gallery from which it was taken
- 3) Time constant used for the count rate meter
- 4) Scale range on which the count rate meter was set
- 5) Background count rate (counts/min) for the above conditions
- 6) Peak activity count rate (counts/min)
- 7) Minimum activity count rate (counts/min)
- 8) Average activity count rate (counts/min)
- 9) General description of the count rate fluctuation over the length of the zinc strip.

Following the above procedure, the axial composite plot of the activity from the calibration sheets was compared with the corresponding thermal neutron density curves as obtained from the gold foils (Figures 8 and 9) in the preliminary experiments. A point by point comparison was made of the data curves from the respective positions of the zinc calibration sheets, the calibrated gold foils in the SRE auxiliary, and main primary galleries. The comparison was made in order to determine if the shape of the zinc data curve agreed with the shape of the gold foil data curve. Since the two sets of data curves were found to be in good agreement, i. e., the activity on the gold foils and the zinc sheets peaked at the same position and followed the same shape and pattern, it was felt that the data from the zinc calibration sheets could be used to calibrate the remaining zinc sheets. A calibration ratio was determined between the zinc sheets and the gold foils by obtaining the ratio between the net activity, in counts/min, and the

Figure 8. Neutron Data Obtained From Shield Blocks 1 and 2

Figure 9. Neutron Data Obtained From
Shield Blocks 12 and 13

gold foil activity extrapolated to thermal neutron density, in $\text{n/cm}^2\text{-sec}$. The net zinc activity was obtained by subtracting the background count rate from the total count rate for the zinc calibration sheet being calibrated.

In the main primary gallery, the calibration ratio was nearly constant over the entire height of the gallery. The variation in the net zinc activity over the length of the 48-in. calibration sheet was from 2400 to 3300 counts/min, corresponding to thermal neutron densities at a reactor power level of 20 Mw of 3.4×10^6 to 4.5×10^6 $\text{n/cm}^2\text{-sec}$ measured by the gold foils. Thus, the conversion factor determined by the ratio of the above values varied from 1410 to 1360, respectively. From a comparison of the data, an average conversion factor of 1380 $\text{n/cm}^2\text{-sec}$ at 20 Mw per count/min of the net activity recorded for the zinc sheets in the main primary gallery was used to determine the thermal neutron streaming in the main primary gallery at full power.

In the auxiliary primary gallery, the same method was used to determine the conversion factor. However, unlike the main primary gallery, where an average conversion factor could be used for the entire height of the gallery, the conversion factor in the auxiliary primary gallery varied from 5×10^3 $\text{n/cm}^2\text{-sec}$ per count/min to 3.75×10^4 $\text{n/cm}^2\text{-sec}$ per count/min of the net activity recorded for the zinc sheets extrapolated to full reactor power (20 Mw). Thus, the conversion factor used to determine the neutron density in the auxiliary primary gallery was dependent upon the specific location of the zinc sheets below reactor floor level. The difference in the conversion factors for the two primary galleries was probably due to a higher intermediate neutron flux present in the main primary gallery which was activating the bare zinc sheets and thus causing a higher conversion factor between neutron flux ($\text{n/cm}^2\text{-sec}$) and activated zinc atoms (counts/min).

B. FINAL EXPERIMENTS DURING MAY 1958 POWER RUN

The installation of the gold foils, used as the secondary standard to convert the zinc activity to thermal neutron density, was accomplished by taping gold foils every 6 in. on a steel rod which was positioned in the east viewing device hole in shield blocks 12 and 13. The individual gold foils were alternately placed in aluminum and cadmium boxes to determine the cadmium ratios given in curve C of Figure 12.

Zinc sheets were installed and positioned as shown in Figures 10 and 11. Zinc sheet C was positioned in the same location as the gold foils, which were located in the east viewing device hole in shield blocks 12 and 13. Zinc sheet V was positioned perpendicular to zinc sheet C and approximately 15-in. from the face of the neutron shield configuration. Zinc sheets were positioned along the south wall behind the neutron shield configuration. Zinc sheets A and B were all positioned after the first right angle bend in the auxiliary primary gallery. Zinc

Figure 10. Position of an Area Covered by Zinc Sheets of South Wall of SRE Auxiliary Primary Gallery

Figure 11. Position of Zinc Sheets Throughout the Labyrinth in the SRE Auxiliary Primary Gallery

sheet D was positioned in the far section of the auxiliary primary gallery. Zinc sheets A, B, C, and D, were positioned at the same distance from the north wall of the auxiliary primary gallery and were all the same length. Zinc sheet E was positioned after the second right angle bend in the far section of the auxiliary primary gallery.

After a total irradiation of approximately 950 Mwd, accumulated over the period of from May 5 to September 26, 1958, the zinc sheets and the gold foils were removed from their respective positions for activation analysis. The gold foils were counted, and by comparison of the activation of the bare and cadmium covered gold foils, the results given in Figure 12 were obtained. The procedure for obtaining the gold foil results was the same as used in all such analyses. Therefore, further discussion is unnecessary; the details of such procedures can be obtained from many references in counting techniques.

Figure 12. Comparison of Zinc Activity and Gold Foil Thermal Flux at East Viewing-Device Hole in Shield Block 12 and 13 (20 Mwt)

Upon receipt for counting, the activated zinc sheets were cleaned and cut into strips 3-ft long and 1-1/2-in. wide. By proper marking and recording, the zinc strips could be analyzed and later located in respect to the position coordinates of the zinc sheet in the SRE auxiliary primary gallery.

As previously implied, the counting efficiency for the detection equipment was not optimized. Calibration checks and background levels were run periodically so that the activities recorded on the graphs could be checked and the "true" zinc activity obtained.

The following information was necessary to analyze the zinc sheet data:

- 1) The position of the zinc strip in respect to the zinc sheet from which it was cut
- 2) The position of the zinc sheets in the SRE auxiliary primary gallery
- 3) The scale range of the count rate meter

- 4) The background activity level (counts/min) for the zinc strip being counted
- 5) The average zinc activity for the zinc strip being counted
- 6) A general description of the variation in the zinc activity over the length of the zinc strip.

Any unusual variations or peaks were noted on a composite map made for each zinc sheet in order to note the location of streaming paths or areas of increased shielding integrity. Then, a composite map was made of the data from each area in which zinc sheets were positioned. These maps are similar to that shown on curve B (Figure 12) which is for calibration zinc sheet C. As shown in Figure 12, the thermal neutron density distribution, as determined by the gold foils, was compared with the zinc activity determined by zinc sheet C. Both sets of data were obtained at the same time, and in the same position in the SRE auxiliary primary gallery, and during the same total irradiation. By obtaining the ratio between curve A and curve B, as shown in Figure 12, a relationship was determined between the thermal neutron density and the zinc activity of the various zinc sheets. This ratio $\left[3.4 \times 10^3 \text{ (n/cm}^2\text{-sec)/(counts/min)} \right]$ was used to convert the activity of all of the remaining zinc sheets in the SRE auxiliary primary gallery to a thermal neutron density at 20 Mwt.

From analysis of the count rate detected on the zinc sheets placed in the auxiliary gallery, a minimum exposure time of approximately 45 days in a thermal neutron flux of $5 \times 10^4 \text{ n/cm}^2\text{-sec}$ was found necessary to activate the zinc to a level that could be reliably counted by the detection equipment shown in Figure 7. Thus, the exposure time and thermal neutron flux to be used in any proposed experiment utilizing zinc as the detection media should be determined on the basis of the counting efficiency of the detection equipment and the data given in Table I above. The counting efficiency of any detection equipment is dependent upon many factors, e. g., distance between source and detector, crystal size, bias, voltage, discriminator settings, etc. Since only relative values (i. e., the zinc sheet activity was calibrated by use of gold foil data) were necessary for the analysis performed herein, the counting efficiency for the detection equipment shown in Figure 7 was not optimized. Consequently, by

proper adjustment of the many determining factors listed above, the counting efficiency for similar detection equipment could probably be increased substantially (probably by as much as a factor of 10) above that available for the zinc sheets analyzed herein.

IV. DESCRIPTION OF DETECTION EQUIPMENT

The detection equipment, as shown in Figure 7, consisted of the following components:

- 1) Reversible movable platform
- 2) NaI(Tl) scintillation counter
- 3) Continuous recorder (single-pen chart recorder)
- 4) Count rate meter
- 5) Linear amplifier (A-1)
- 6) Single channel differential analyzer.

The reversible movable platform was approximately 3-ft long and 1.5-in. wide. The width of the platform was determined by the shelf width of the shielded sample holder and the size of the NaI(Tl) crystal. The length of the platform was determined on the basis of convenience, i. e., ease of handling the zinc strips. The platform was made movable and reversible for the obvious reason of operational efficiency and speed for counting large areas of activated material. Micro-switches were attached at each end of the platform to act as limit switches for the motor drive.

The NaI(Tl) scintillation counter was installed in a shielded sample holder. The NaI(Tl) crystal was 1-1/2-in. in diameter and 1-1/2-in. thick. The NaI(Tl) scintillation counter was chosen for its high gamma-ray counting efficiency and its selectivity of radiation type.

The continuous graphic recorder was used to give a continuous record of the zinc activity on the zinc strips. By examination of these records, variations and peaks could easily be noted and plotted on the composite activity map for any area being analyzed. The continuous recorder was calibrated to record the percentage of full scale reading being detected on the scale of the count rate meter. The speed of the recorder was regulated to correspond to the speed of the platform moving under the NaI(Tl) scintillation counter.

The count rate meter was used to indicate any malfunction of the recorder or counter. Also, the scale of the count rate meter determined the extent of zinc activity recorded on the graphic recorder. The count rate meter would

indicate more accurately and quickly the fluctuations occurring in either background or zinc activity depending upon the time constant of the meter.

The linear amplifier is standard equipment to be used with a differential analyzer following a scintillation counter.

The single channel differential analyzer was used to determine the energy level (pulse size) of the gamma-rays to be detected by the count rate meter. By varying the energy differential (bias) setting, the gamma energy peak of Zn^{65} ($E_\gamma = 1.12 \text{ Mev}$) could be isolated for counting, thus excluding other gamma energies and background noises.

V. RESULTS

A. ANALYSIS OF EXTENT OF THERMAL NEUTRON STREAMING

1. Preliminary Experimental Data

In the preliminary experiments, measurements were made at several positions in the main and auxiliary primary galleries as well as in portions of the secondary sodium galleries. Data obtained in locations of thermal neutron streaming in the vicinity of the primary sodium pipe penetrations of the reactor vessel are given in Figures 8 and 9. These preliminary measurements substantiated the prior area survey results. Therefore, it was believed that this source of neutrons in the galleries would present an after shutdown access problem in the areas directly adjacent to the reactor but would not appreciably limit access to other portions of the galleries.

Perhaps of even greater consequence was the fact that significant thermal neutron fluxes were present around both intermediate heat exchangers and that these fluxes were great enough to cause activation of the secondary sodium. Thus, the activated secondary sodium would be present in reactor areas that were considered unlimited access areas and would contaminate supposedly non-radioactive components and areas.

2. Initial Data from Zinc Detectors

In the experiments performed during the November, 1957 power run, zinc sheets were placed over most of the accessible area of the bellows assemblies in each of the SRE primary galleries (see Figures 1 and 2).

At the time that the zinc calibration sheets were removed from their respective galleries, after the completion of the November, 1957 power run, the zinc sheets in the bellows regions were also removed and submitted for counting analysis. The activity data obtained for the zinc sheets was then analyzed in the same manner as previously described. Using the appropriate conversion factors as determined in Section III-A., above, the net activity of the individual zinc sheets were converted to the corresponding thermal neutron density at full power (20 Mw). The thermal neutron density distribution as was determined for each zinc sheet positioned in the auxiliary primary gallery was placed on a composite map of the entire bellows region to determine the axial

and radial variance of the thermal neutron density in this region. When these values were plotted on a scaled map of the bellows region, such as in Figure 13, the thermal neutron density levels could be separated into distinctly different intensity regions. Once these regions had been isolated, the extent and the location of the streaming in the bellows region were evident. The same method was used to determine the extent and location of the streaming in the bellows region of the main primary gallery (Figure 14).

Figure 13. Thermal Neutron (2480 m/sec) Density Distribution Through Bellows Assembly in the SRE Auxiliary Primary Gallery (n/cm²-sec)

The thermal neutron density distribution through the bellows assembly in the auxiliary primary gallery, which was obtained from zinc sheets activated in the auxiliary primary gallery during the November 1957 SRE power run, is given in Figure 13. The area below the sodium pipes at the bellows assembly entrance has a thermal neutron density of approximately 10^9 n/cm²-sec, when extrapolated to full reactor power (20 Mw).

Figure 14. Thermal Neutron (2480 m/sec) Density Distribution Through Bellows Assembly in the SRE Main Primary Gallery ($\text{n/cm}^2\text{-sec}$)

From the composite map of the auxiliary primary gallery (Figure 13), the area of greatest thermal neutron streaming is located below the inlet and outlet sodium pipes. The highest values measured, i.e., approximately $2 \times 10^9 \text{ n/cm}^2\text{-sec}$, were located between the two sodium pipes. Immediately above and to the right of the outlet sodium pipe, the thermal neutron streaming is reduced by a factor of 2. Although these two areas did not show a wide variance in thermal neutron streaming, the data obtained from the zinc sheets distinctly indicated the two areas of streaming intensity noted in Figure 13. Thus, the method used for the detection of the "holes" in the auxiliary and main primary gallery thermal shields was adequate enough to distinguish between areas which differed by a factor of as low as two in the thermal neutron streaming intensities.

The thermal neutron distribution through the bellows assembly in the main primary gallery, which was obtained in the same manner as the above, is given in Figure 14. The area surrounding the sodium pipes at the bellows assembly entrance has a thermal neutron density of approximately 10^7 n/cm²-sec when extrapolated to full reactor power (20 Mw).

From the composite map of the main primary gallery (see Figure 14), the highest thermal neutron streaming is located below the pipe area and above the outlet sodium pipe. The highest values measured, i. e., approximately 2×10^7 n/cm²-sec, were located immediately below the pipe area. Immediately above the outlet sodium pipe, a distinct peaking of the thermal neutron density was found, i. e., approximately 1.2×10^7 n/cm²-sec. The remaining area of Figure 14 is relatively uniform in both intensity of thermal neutron density and symmetry.

3. Analytical Methods and Procedures

The thermal neutron density data obtained from the zinc activation experiments agreed favorably with similar data obtained in previous measurements from several well separated regions in both the auxiliary and main primary galleries. Using the thermal neutron data obtained from this experiment, a complete analysis was performed to determine the after shutdown gamma radiation levels that would exist in the SRE galleries.

For all of the calculations performed in order to evaluate the radiation levels in the SRE galleries, the assumption was made that the neutron densities that were measured in the SRE galleries are approximately the same numerical value as the neutron fluxes in these regions. Thus, all of the neutron flux values that are used in the following calculations are actually the neutron densities that have been measured in the SRE galleries and denoted as neutron fluxes.

The following assumptions and data were used in the gamma-ray dose rate calculations performed for the auxiliary primary gallery to determine the accessibility of the gallery after reactor shutdown:

$$\text{Average thermal flux in the bellows assembly region } (\bar{\phi}_{th}) = 4 \times 10^8 \text{ n/cm}^2\text{-sec}^*$$

$$\text{Reactor power level} = 20 \text{ Mwt}$$

*See reference 4

irradiation time = 5 years

Cooling time sufficient to reduce sodium activity in pipes to safe level (assuming no fuel element ruptures or buildup of long-lived impurities in the sodium) = 2 weeks.

The basic equation which can be used to calculate the dose rate at the surface of the iron plate in the bellows assembly of the auxiliary primary gallery is as follows:

For a uniform slab source:

$$D = \bar{\phi}_{th} \frac{\sum_{act} E e^{-\mu_t t}}{2k\mu_s} \left\{ \frac{1}{\mu_t t + 1 + f_1(\mu_t t)} + a + b(2 + \mu_t t) - e^{-\mu_s s} \left[\frac{1}{P + 1 + f_1(P)} + a + b(2 + P) \right] \right\} \quad \dots (1)$$

where

D = gamma dose rate (r/hr)

$\bar{\phi}_{th}$ = average thermal neutron flux (n/cm²-sec)

$\sum_{act}$ = activation cross section for the slab (cm⁻¹)

E = gamma-ray energy released per thermal neutron capture (Mev/capture)

k = conversion factor from energy flux to dose rate (Mev/cm²-sec per r/hr)

μ_s = gamma-ray linear absorption coefficient for the source region (cm⁻¹)

μ_t = gamma-ray linear absorption coefficient for the shield region (cm⁻¹)

s = thickness of the source region (cm)

t = thickness of the shield region (cm)

a and b = constants associated with a quadratic approximation to the dose buildup factor, B_r , in which $B_r = 1 + a(\mu x) + b(\mu x)^2$

$$f_1(x) = \frac{e^{-x}}{E_2(x)} - (x + 1)$$

$$E_2(x) = \int_1^{\infty} \frac{e^{-xt}}{t^2} dt$$

$$P = \mu_s s + \mu_t t$$

Since there is no shield region for the iron plate, Equation (1) can be reduced to the following equation:

$$D = \frac{\bar{\phi}_{th} \Sigma_{act} E}{2k \mu_s} \left\{ 1 + a + 2b - e^{-\mu_s s} \left[\frac{1}{\mu_s s + 1 + f_1(\mu_s s)} + a + b(2 + \mu_s s) \right] \right\} \dots (2)$$

The following values were used to determine the dose rate at the surface of the iron plate after the plate has reached saturated activity:

$$\bar{\phi}_{th} = 4 \times 10^8 \text{ n/cm}^2\text{-sec}$$

$$\Sigma_{act} = 2.34 \times 10^{-4} \text{ cm}^{-1}*$$

$$E = 1.2 \text{ Mev/capture} \left[(1.1 \text{ Mev (50\%) and } 1.3 \text{ Mev (50\%)}) \right]^\dagger$$

$$a = 0.81^* \text{ and } b = 0.45^*$$

$$k = 5.4 \times 10^5 \text{ Mev/cm}^2\text{-sec per r/hr}^*$$

$$\mu_s = 0.41 \text{ cm}^{-1} \text{ for } 1.2 \text{ Mev photons}^*$$

$$s = 1.27 \text{ cm}$$

Since the half life of Fe^{59} is 45 days, the iron plate would have reached saturated activity after 5 years of irradiation. The other elements present in the iron plate will not contribute to the dose rate because of their abundance, cross section or half life (for a 14-day cooling period).

*See reference 11

†See reference 2

Substituting the above values into Equation (2) the dose rate at the surface of the iron plate was found to be 318 mr/hr.

The following values were used to determine the dose rate from the steel piping in the auxiliary primary gallery.

$$\bar{\phi}_{th} = 4. \times 10^8 \text{ n/cm}^2\text{-sec}$$

$$\Sigma_{act} = 3.36 \times 10^{-3} \text{ cm}^{-1}*$$

$$E = 1.9 \text{ Mev/capture}$$

$$k = 5.4 \times 10^5 \text{ Mev/cm}^2\text{-sec per r/hr}^*$$

$$\mu_s = 0.41 \text{ cm}^{-1} \text{ for 1.2 Mev photons}^*$$

$$s = 0.635 \text{ cm}$$

$$a = 0.81^* \text{ and } b = 0.45^*$$

It should be noted that the value of the thermal neutron flux ($\bar{\phi}_{th}$) used is the same as that in the bellows assembly region. Thus, the dose rate calculated for the steel piping is the maximum value, since the calculations assume no change in the neutron flux with position.

The macroscopic activation cross section for the steel piping was determined assuming the following elemental composition and wt %: *

<u>Composition</u>	<u>%</u>
Iron	67
Cobalt	0.1
Chromium	19
Nickel	8.9
Manganese	2
Silicon	2
Phosphorus	1

Analysis of the activation products of these elements indicated that, following a two-week decay time, only the iron and cobalt would be important. Since chromium emits only a weak gamma ray in 3% of its decays, it was not

*See reference 11

considered significant for the gamma dose rate calculations. Nickel and manganese activities have short half lives in respect to the assumed cooling time, and therefore were not considered significant. Silicon and phosphorus are not important contributors to the total gamma-ray dose rate, due to their abundance, cross section, half life, and/or stable activation products.

The average energy of 1.2 Mev for the emitted photons was obtained by averaging the iron photons of 1.1 Mev and 1.3 Mev and cobalt photons of 1.17 and 1.33 Mev. The average energy released per capture (1.9 Mev) was determined from the total energy released per disintegration for iron (1.2 Mev) and cobalt (2.5 Mev) and from the iron and cobalt macroscopic activation cross sections.

Substituting the above calculated values in equation (2) the gamma-ray dose rate at the surface of the piping after an irradiation time of 30 to 40 years was determined to be 4.5 r/hr. It should be noted that the dose rate calculation was based on an infinite slab source. However, since the primary galleries of the SRE are rather completely filled with piping, tubing, and plates, use of this geometry to represent this situation is not out of line. The personnel working in the gallery would see essentially two such slab sources from the surrounding piping, etc. Thus, the dose rate from the piping would be twice the calculated value or 9 r/hr. After 5 years of reactor operation at full power (20 Mw), the dose rate from the piping would be 4.63 r/hr. This value is determined from the fact that the cobalt component contributed 97% of the dose rate after 30 to 40 years operation and will have only reached one-half of its saturated activity at the end of 5 years (the half-life of Co^{60} being 5.27 years). After 5 years of reactor operation at full power (20 Mw), cobalt would contribute 4.36 r/hr while iron would contribute 0.27 r/hr to the dose rate. Because of the nature of the SRE, i. e., the reactor is an experimental unit, the plant factor will probably be quite low. Due to the above, the calculations were based on the reasonable design value of 5 years at full power (20 Mw) or equivalent total integrated power.

Since the thermal neutron flux in the bellows assembly of the main primary gallery is a factor of 50 lower than that found in the bellows assembly of the auxiliary primary gallery, the gamma-ray dose rate would be $318/50$, or 6.4 mr/hr after 5 years of reactor operation at 20 Mwt.

Also, the average thermal neutron flux outside the bellows assembly in the main primary gallery is a factor of 5×10^2 less than the thermal neutron flux outside the bellows assembly in the auxiliary primary gallery. Thus, the neutron-induced activation of the 6-in. steel pipes in the main primary gallery will result in a gamma dose rate which is reduced by a factor of 5×10^2 from that found for the steel pipes in the auxiliary primary gallery. Therefore, the gamma-ray dose rate would be 10 mr/hr after 5 years of reactor operation at 20 Mw thermal.

From the above calculations, it was found that the maximum neutron-induced gamma-ray dose rate in the main primary gallery would be 16 mr/hr, again occurring near the bellows assembly.

4. Shielding Determination and Recommendations

From the results obtained in section (V-A-3) above, the thermal neutron flux in the auxiliary primary gallery must be reduced by at least a factor of 10^3 in order to reduce the gamma-ray dose rate to the maximum occupational level (7.5 mr/hr). Even though the streaming fast neutron flux is most probably less than the streaming thermal neutron flux, the fast neutron flux was assumed to be equal in magnitude to the thermal neutron flux in the galleries. Previous measurements indicated that the total activity of the bare gold foils was equally divided between the activation due to thermal neutrons and that due to all other neutrons. Since it was expected that the major fraction of the neutrons detected in the galleries are the result of fission neutrons scattered in the sodium pool, the streaming neutron density would be comprised mostly of low energy neutrons. The radius of the sodium pool plus the large reflector thickness between the core and the line-of-sight streaming would seem to indicate that the streaming fast neutron flux would be much lower than the streaming thermal and intermediate neutron fluxes, probably by as much as a factor of 10 to 100. Thus, in the shielding configuration, it was necessary to place a thermal neutron shield first and follow this by a shield for attenuating both fast and thermal neutrons and gamma radiation as well. Based on the above requirements, a neutron shield configuration was determined. The calculations for the neutron shield configuration installed in the auxiliary primary gallery are given below.

For the neutron shield configuration which was installed, a 0.25 boral plate was used in the thermal neutron shield. This shielding was followed by 6 in. of steel, 12 in. of ordinary concrete ($\rho = 2.3 \text{ gm/cm}^3$), and a 20-mil layer of cadmium. The following equation was used to determine the emerging neutron flux.

$$\phi = \bar{\phi}_0 e^{-\sum_i \mu_{a_i} x_i} \quad \dots (3)$$

where

$\bar{\phi}_0$ = average neutron flux incident on the shield ($\text{n/cm}^2\text{-sec}$)

ϕ = emerging neutron flux ($\text{n/cm}^2\text{-sec}$)

μ_{a_i} = macroscopic absorption or removal cross section for the i^{th} shield region (cm^{-1})

x_i = thickness of i^{th} shield region (cm).

The following values were used to determine the emergent thermal neutron flux:

$$\bar{\phi}_0 = 4 \times 10^8 \text{ n/cm}^2\text{-sec}$$

$$\mu_{a_1} = 7.98 \text{ cm}^{-1} \text{ for boral plate region}^9$$

$$x_1 = 0.635 \text{ cm for boral plate region}$$

$$\mu_{a_2} = 0.197 \text{ cm}^{-1} \text{ for steel region}^{11}$$

$$x_2 = 15.25 \text{ cm for steel region}$$

$$\mu_{a_3} = 0.078 \text{ cm}^{-1} \text{ for concrete region}^{11}$$

$$x_3 = 30.5 \text{ for concrete region.}$$

The boral plate macroscopic absorption cross section was obtained from experimental results given by the manufacturer and confirmed in Reference 1.

Substituting the above values into Equation (3), the emerging thermal neutron flux would be $1.2 \times 10^4 \text{ n/cm}^2\text{-sec}$, or the shield would provide an attenuation factor of 3.5×10^4 .

The following values were used to determine the emergent fast neutron flux:

$$\begin{aligned}\bar{\phi}_0 &= 4 \times 10^8 \text{ n/cm}^2\text{-sec} \\ \mu_{r1} &= 0.126 \text{ cm}^{-1} \text{ for boral plate region}^{10} \\ \mu_{r2} &= 0.171 \text{ cm}^{-1} \text{ for steel region}^{11} \\ \mu_{r3} &= 0.078 \text{ cm}^{-1} \text{ for concrete region}^{11} .\end{aligned}$$

All other values for the regions remain the same. Substituting these values into Equation (3), where μ_a becomes μ_r , the macroscopic removal cross section, the emerging fast neutron flux would be $2.5 \times 10^6 \text{ n/cm}^2\text{-sec}$, or the shield would provide an attenuation factor of 1.6×10^2 . The assumption has been made that the addition of the cadmium liner after the concrete will serve to capture the thermal neutrons produced by moderation of the fast neutrons as they penetrate the shield configuration.

Once the shield configuration was determined, the area of the shield region was considered. For the auxiliary primary gallery, the area of an adequate shield region was based on the initial neutron streaming data and the lack of space near the source of neutron streaming, i. e., in the region of the bellows assembly of the auxiliary primary gallery. Therefore, both the thermal neutron and steel shields were placed inside the 3 ft diameter region surrounding the sodium pipes (i. e., within the concrete recess) and as close to the individual sodium pipes as possible. The concrete shielding was extended above this recess region by at least 1 ft in order to present equivalent shield paths to all scattered streaming neutrons and gamma radiations. The 20-mil thickness of cadmium enclosed the entire area of the concrete shielding in order to capture the thermal neutrons produced by moderation of the fast neutrons as they penetrate the shield configuration.

From the results obtained in section V-A-3 above, the gamma-ray dose rate in the main primary gallery would not limit the accessibility of the area after a reactor shutdown period of 2 weeks. Therefore, the recommendation was made that no additional shielding would be necessary in the main primary gallery.

B. ANALYSIS OF THERMAL NEUTRON STREAMING AFTER NEUTRON SHIELD INSTALLATION

1. Final Data From Zinc Detectors

The thermal neutron density distribution through the bellows assembly in the auxiliary primary gallery, which was obtained from zinc sheets activated during the May-September 1958 SRE power runs, is given in Figure 15. To compare these results with those obtained during the November, 1957 SRE power run, see Figure 13 which gives the thermal neutron density distribution through the bellows assembly in the SRE auxiliary gallery before installation of the proposed neutron shielding. The range of values for the thermal neutron density in Figure 13 is 5×10^7 to 2×10^9 n/cm²-sec while the range of values in Figure 15 is 2×10^6 to 1×10^7 n/cm²-sec. The average thermal neutron density for Figure 13 was determined to be approximately 100 greater than that determined for Figure 15. All values have been extrapolated to full reactor thermal power, i. e., 20 Mw.

From the composite flux map of the bellows region in the auxiliary primary gallery after the neutron shield installation (Figure 15), the highest thermal neutron streaming area was found to be located below the outlet sodium pipe, the flux being approximately 10^7 n/cm²-sec. The entire area investigated, as shown in Figure 15, was relatively uniform in both intensity of thermal neutron density and symmetry.

2. Comparison of Experimental and Analytical Results

In order to analyze the attenuation properties of the neutron shielding that was installed in the SRE auxiliary gallery prior to the May-September, 1958 power run, a comparison of the thermal neutron density distribution in this area was made before and after installation of the neutron shields. The thermal neutron density distribution obtained during the November, 1957 power run, which was prior to the installation of the neutron shield, is shown in Figure 13. The magnitude varies by a factor of 40 over the entire region. Also, the conversion factor necessary to change the zinc activity to thermal neutron density varies from 5×10^3 to 3.75×10^4 n/cm²-sec per count/min, which indicated that the energy distribution of the streaming neutron flux was not constant throughout the bellows assembly region. The average thermal neutron density for the bellows

Figure 15. Thermal Neutron Density Distribution Through Bellows Assembly in the SRE Auxiliary Primary Gallery After Installation of Proposed Neutron Shielding At 20 Mwt (n/cm^2 -sec)

assembly was determined to be approximately $4 \times 10^8 n/cm^2$ -sec, which would cause activation of the steel piping to such an extent that the gamma-ray dose rate in the auxiliary gallery after 5 years operation at 20 Mwt and two weeks shutdown would be 4.6 r/hr. For comparison purposes, Figure 15 shows the thermal neutron distribution obtained in the same region as the data in Figure 13, only after installation of the proposed neutron shield. As can be noted from this figure, no pronounced flux level distinction is apparent except for a gradual change by a factor of 5 over the entire region. In comparison, the factor for

converting the zinc activity to thermal neutron density is nearly constant, i. e., a ratio of 3.4×10^3 n/cm²-sec per count/min was obtained (Figure 12). The apparent uniformity of this ratio indicates that the neutron flux leaving the neutron shield has a relatively uniform energy distribution over the entire surface. Thus, the neutron shield has made the energy distribution of the neutron flux in the SRE auxiliary primary gallery as uniform as that previously detected in the SRE main primary gallery. The average thermal neutron density for the area investigated was determined to be nearly a factor of 10^2 less than the thermal neutron density previously detected. Thus, the expected gamma-ray dose rate from the steel piping in the SRE auxiliary primary gallery after 5 years³ operation at 20 Mwt and two weeks shutdown would be reduced from the previously calculated value of 4.6 r/hr to 46 mr/hr. This radiation level, if reached in the gallery, would be prohibitive to extended operational maintenance. However, since the SRE is an experimental plant, the expected plant factor will be low, which is expected to reduce the calculated gamma-ray dose rate (46 mr/hr) to a reasonable operational radiation level.

As noted in section V-A-4., a thermal neutron density reduction factor of 3.5×10^4 had been calculated for the proposed neutron shielding prior to installation, whereas the actual reduction factor is approximately 10^2 . The large difference between theory and experiment can be accounted for by three phenomena:

- a) Channeling through the boral shielding¹
- b) Cracks between the concrete blocks in the proposed neutron shielding configuration as installed
- c) Reduction of the water content in the bulk shield concrete wall between the reactor core tank and auxiliary primary gallery after the 1957 power run.

Although the neutron shield attenuation was not as great as expected (i. e., 3.5×10^4), the recommendation was made that no additional neutron shielding be incorporated in the auxiliary primary gallery at the present time. As can be seen in Figure 15, no excessive neutron streaming was detected either along or through the primary sodium piping that penetrated both the bellows assembly and the installed neutron shielding. The installed shielding, consisting of 0.25-in.

boral plate, 6-in. of steel plate, 12-in. of ordinary concrete blocks and a 20-mil layer of cadmium, not only flattened the axial and radial thermal neutron density streaming through the bellows assembly but also reduced the average thermal neutron density by a factor of 100. The ratio of the maximum to minimum flux density before installation of the neutron shield was 40, whereas this ratio was only 5 after installation.

As previously stated, no specific areas of neutron streaming were detected by the zinc sheets located behind the bellows assembly region and only a factor of 5 was detected between the maximum and minimum thermal neutron densities recorded. Such a factor is certainly possible from the relative vertical distance from the reactor core to the detector. Variance in the thermal neutron density for the SRE auxiliary primary gallery was detected mostly in the vertical direction directly behind the bellows assembly region. Therefore, no excessive increase or decrease in the thermal neutron density was observed along or through the sodium piping penetrating the neutron shielding.

C. ANALYSIS OF THERMAL NEUTRON SCATTERING THROUGH GALLERY LABYRINTHS*

1. Data from Zinc Detectors

a. Initial Experimental Data

For the auxiliary primary gallery, the data from the zinc sheets in the bellows region were used to determine the thermal neutron flux before the right-angle bend and the data from zinc calibration sheets A and B (Figure 5) were used to determine the thermal neutron flux after the right-angle bend. However, the thermal neutron flux is not uniform in the bellows region of the auxiliary primary gallery and was only determined by both sets of zinc sheets over a limited region of the total area of the auxiliary primary gallery. Therefore, the assumption was made that the area of the gallery which was covered by both the calibration and bellows region zinc sheets, i.e., 6.7 ft^2 , would determine the effective attenuation of the right-angle bend. The average thermal neutron flux determined for the 6.7 ft^2 area before the right-angle bend was $2.8 \times 10^8 \text{ n/cm}^2\text{-sec}$ and the average thermal neutron flux determined for the 6.7-ft^2 area after the right-angle bend was $4 \times 10^7 \text{ n/cm}^2\text{-sec}$. Therefore, the attenuation factor for the right-angle bend in the auxiliary primary gallery was:

*In this section of the report, the assumption will be made that the neutron flux values will be numerically equivalent to the neutron density values obtained for thermal streaming.^{5,6}

$$\text{Attenuation} = \frac{2.8 \times 10^8 \text{ n/cm}^2\text{-sec}}{4 \times 10^7 \text{ n/cm}^2\text{-sec}} = 7.0$$

The geometry in the auxiliary primary gallery was not considered good for the experimental results obtained above since no distinct offset or right-angle was present between the bellows region and the region of zinc sheet A and B.

For the main primary gallery, the data from calibration zinc sheet C were used to determine the thermal neutron flux before the right-angle bend and the data from calibration zinc sheet D, which had been hung at a right angle to the calibration zinc sheet, were used to determine the thermal neutron flux after the right-angle bend (Figure 6). Since the experimental arrangement in the main primary gallery was considered good geometry for determining this attenuation, and the thermal neutron flux distribution was uniform in the bellows region of the main primary gallery, the overall thermal neutron flux was used for the determination of the effective attenuation of a right-angle bend. The average thermal neutron flux determined before the right-angle bend was $4 \times 10^6 \text{ n/cm}^2\text{-sec}$ and the average thermal neutron flux determined after the right-angle bend was $9.1 \times 10^5 \text{ n/cm}^2\text{-sec}$. Therefore, the attenuation factor for the right-angle bend in the main primary gallery, as determined by the zinc sheets, was:

$$\text{Attenuation} = \frac{4 \times 10^6 \text{ n/cm}^2\text{-sec}}{9.1 \times 10^5 \text{ n/cm}^2\text{-sec}} = 4.4$$

b. Final Experimental Data

Using the method described in Section III-B, the following results were obtained from the zinc sheets positioned in the auxiliary primary gallery. These results were used to evaluate the thermal neutron flux attenuation in the auxiliary primary gallery. As noted in Table II, the thermal neutron fluxes (densities) are given throughout the gallery for each of the different positions of the zinc sheets. Using Figure 11 as a reference for the position of the zinc sheets and, therefore, the thermal neutron fluxes, an attenuation factor can be determined between the different positions of the zinc sheets and for the right-angle bends in the gallery.

TABLE II

**THERMAL NEUTRON DENSITY DISTRIBUTIONS THROUGH
SRE AUXILIARY PRIMARY GALLERY AT 20 Mwt**

Zinc Sheet	Thermal Neutron Density at Given Distance From Reactor Floor Level (n/cm ² -sec)		
	8.0 ft	10 ft	13 ft
V	2.3×10^6	4.1×10^6	9.2×10^6
A	1.2×10^6	1.4×10^6	2.0×10^6
B	7.1×10^5	7.1×10^5	9.2×10^5
D	1.0×10^5	1.2×10^5	1.7×10^5
E	1.0×10^5	1.2×10^5	1.0×10^5
C	1.3×10^6	1.7×10^6	4.5×10^6

Case 1. Right-Angle Bend Between Zinc Sheet V and C

The following typical equation will be used to determine the attenuation between different zinc sheet positions and at varying distances from the reactor floor level. The average attenuation for the positions will be obtained by averaging the results for each of the three distances from the reactor floor level.

$$\text{Attenuation (Distance from reactor floor level)} = \frac{\phi_{th}(\text{at zinc sheet V})}{\phi_{th}(\text{at zinc sheet C})} \dots (4)$$

$$\text{Attenuation (8 ft)} = \frac{2.3 \times 10^6 \text{ n/cm}^2\text{-sec}}{1.3 \times 10^6 \text{ n/cm}^2\text{-sec}} = 1.8$$

$$\text{Attenuation (10 ft)} = \frac{4.1 \times 10^6 \text{ n/cm}^2\text{-sec}}{1.7 \times 10^6 \text{ n/cm}^2\text{-sec}} = 2.4$$

$$\text{Attenuation (13 ft)} = \frac{9.2 \times 10^6}{4.5 \times 10^6} = 2.0$$

$$\text{Average} = \frac{1.8 + 2.4 + 2.0}{3} = 2.1$$

Thus, the experimentally determined reflection albedo (β), which is the inverse of the attenuation, would be 0.48 for the ordinary concrete rectangular duct. This is to be compared with the experimental value quoted in reference 2 of 0.51 for a steel-lined barytes concrete annulus.

Case 2. Experimental Attenuation of Streaming Thermal Neutrons Through SRE Auxiliary Primary Gallery at 20 Mwt

Using the same method as shown in Case 1 for determining the reflection albedo (β), the attenuation results given in Table III were determined for the different zinc sheet positions in the SRE auxiliary primary gallery.

TABLE III
EXPERIMENTAL ATTENUATION RESULTS IN THE SRE AUXILIARY
PRIMARY GALLERY FOR STEAMING THERMAL NEUTRONS

	Attenuation at Various Heights			
	8 ft	10 ft	13 ft	Ave
Bellows to Zinc Sheet A	3.6	3.6	3.5	3.6
Zinc Sheet A to B	1.7	2.0	2.2	2.0
Zinc Sheet B to D	(data not applicable)		5.4	5.4
Zinc Sheet D to E	(data not applicable)		1.7	1.7

In the attenuation calculation for zinc sheet B to D and zinc sheet D to E, the only reliable data is at the 13 ft level of the gallery. This fact is apparent if one considers that an opening only large enough to allow the sodium pipes to pass through is available for neutron streaming at the right angle bend between zinc sheet B and D.

2. Analytical Method and Results

In order to determine if there was a reliable theoretical method which would predict the above experimentally determined attenuation factors, and could

therefore be used in analysis of similar situations in other plant designs, the following equation² was modified to fit the geometry that was present in the SRE galleries:

$$\phi(f) = \frac{\phi(i)Ka^2\beta_i}{2Z_f^2} \csc \theta, \quad \dots (5)$$

where

$\phi(f)$ = thermal neutron flux at distance f around the bend (n/cm^2 -sec)

$\phi(i)$ = thermal neutron flux incident on the scattering media (n/cm^2 -sec)

K = an empirical constant chosen to yield agreement with experimental results²

β_i = effective albedo for the scattering media²

Z_f = distance from the nearest wall of the emitting duct to the detector (ft)

θ = angle of the duct bend (degrees)

a = radius of the duct from which the thermal neutrons are emitted (ft)

The equation was modified so that it could be used with the geometry present in the SRE sodium galleries. Equation (5) was derived for a cylindrical duct with radius "a," but the duct present in the SRE sodium galleries was a rectangle. In order to analyze the rectangular duct, an effective radius was determined from the cross-sectional area of the rectangular duct by assuming a cylindrical duct with the same cross-sectional area.

According to Reference 2, Horton and Halliday have found empirically that the product of $K\beta_i$ in the above equation is approximately equal to 1/3 for both steel-lined barytes concrete and steel enclosed water ducts. Thus, in the following calculations, a value of 1/3 was assumed for the ordinary and/or magnetite concrete ducts of the SRE galleries.

a. Initial Theoretical Calculations

Using the above theoretical method, calculations were performed for the conditions that existed during the initial experiments. These results were to be compared with the initial experimental results.

The following assumptions and dimensions were determined for the auxiliary primary gallery:

Effective scattering area seen by both zinc sheets = 6.7 ft^2

Effective radius of above area (a_{eff}) = 1.45 ft

Distance from nearest wall of the emitting duct to the calibration zinc sheet (Z_f) = 1.5 ft

Product of $K\beta_i = 0.33^*$

Angle of the duct bend (θ) = 90°

As given below, the theoretical attenuation factor was found by using a modified form of Equation (5).

$$\text{Attenuation} = \frac{\phi(i)}{\phi(x)} = \frac{2Z_f^2}{K\beta_i(a_{\text{eff}})^2 \csc \theta} \quad \dots (6)$$

Substituting the above data for the auxiliary primary gallery into Equation (6), a theoretical attenuation factor of 6.4 was determined for those conditions.

The following assumptions and dimensions were determined for the main primary gallery:

Areas of the wall from which the thermal neutrons are scattered = 72 ft^2

Effective radius of the above area (a_{eff}) = 4.8 ft

Distance from nearest wall of the emitting duct to the zinc sheet detector (Z_f) = 4 ft

Product of $K\beta_i = 0.33^*$

*See reference 2.

Angle of the duct bend (θ) = 90°

Using the above data, a theoretical attenuation factor of 4.2 was determined.

b. Final Theoretical Calculations

Using the theoretical method previously described and used in Section V-C-2-a., calculations were performed for the conditions existing during the final experiments. These results were to be compared with the final experimental data.

The following assumptions and dimensions were determined for the auxiliary primary gallery:

Scattering area of south wall seen by zinc sheets = 32 ft^2

Effective radius of above area (a_{eff}) = 3.2 ft

Product of $K\beta_1 = 0.33^*$

Angle of the duct bend (θ) = 90°

Using the above data, the theoretical attenuation factor was calculated by using Equation (6).

Case 1. Attenuation Between Zinc Sheets of Figure 15 and Zinc Sheet A

$$\text{Attenuation} = \frac{2(2.4)^2}{(0.33)(10)} = 3.8 \text{ at a distance of 2.4 ft from right-angle bend.}$$

Case 2. Theoretical Attenuation of Streaming Thermal Neutrons Through SRE Auxiliary Primary Gallery at 20 Mwt

Using the same method as shown in Case 1, the attenuation results given in Table IV were determined for different positions in the SRE auxiliary primary gallery. Since a right angle exists between zinc sheet D and E, the following values were used in the determination of that attenuation value.

Effective scattering area seen by both zinc sheets = 6 ft^2

Effective radius of above area (a_{eff}) = 1.4 ft

Distance from nearest wall of emitting duct to zinc sheet = 9 in.

*See reference 2.

TABLE IV
THEORETICAL ATTENUATION RESULTS IN THE SRE AUXILIARY
PRIMARY GALLERY FOR STREAMING THERMAL NEUTRONS

Gallery Position	Attenuation
Bellows to Zinc Sheet B	17.4
Zinc Sheet A to B	4.6
Bellows to Zinc Sheet D	92.0
Zinc Sheet A to D	24.0
Zinc Sheet B to D	5.3
Zinc Sheet D to E	1.8

Whether or not this type of approach can be used in computing the attenuation between sheets D and E, i. e., for the case of a small opening at the bottom of the gallery and a relatively short distance from the nearest wall to the zinc sheet (Z_f), cannot be determined with the knowledge and data presently available. Although, the calculated and experimental results for this region will be compared in the next section, no conclusions should be based on these results.

3. Comparison of Experimental and Analytical Results

a. Initial Experimental and Analytical Comparison

If the calculated values are compared with the experimental values, one finds that the results compare favorably, even in the auxiliary primary gallery where the geometry as well as the experimental data were not too applicable to the problem of determining the attenuation factor for a right-angle bend. The two values for the main primary gallery are in excellent agreement, verifying the fact that the geometry as well as the experimental data were applicable to the problem of determining the attenuation factor for a right-angle bend.

b. Final Experimental and Analytical Comparison

Comparison of the final experimental results and the analytical results for the auxiliary primary gallery indicate good agreement. Since the experimental arrangements for the final tests were specifically set up for the determination of labyrinth attenuation properties, the good agreement indicated in Table V below reaffirmed the comparable results obtained in the initial experiments.

The experimental and theoretical attenuation results are given for comparison and comment in Table V.

TABLE V

**EXPERIMENTAL AND THEORETICAL ATTENUATION RESULTS
FOR LABYRINTH IN SRE AUXILIARY PRIMARY GALLERY**

Gallery Position	Attenuation Factor		Comments
	Experimental	Theoretical	
At Zinc Sheet A	3.6	3.8	Relatively good agreement
At Zinc Sheet B	7.2	17.4	Indicates a thermal neutron buildup
At Zinc Sheet D	39.0	92.0	Less of an overall thermal neutron buildup
Zinc Sheet A to B	2.0	4.6	Indicates a thermal neutron buildup
Zinc Sheet A to D	10.8	24.0	Less of an overall thermal neutron buildup
Zinc Sheet B to D	5.4	5.3	Relatively good agreement
Zinc Sheet D to E	1.7	1.8	Relatively good agreement
At Zinc Sheet E	66.0	165.0	Composite results of above
At Zinc Sheet C	2.1	2.0 ⁴	Good agreement for albedo value

The primary reason for the thermal neutron buildup indicated in Table V is that the neutron flux streaming through the bellows assembly is made up of both intermediate and higher energy neutrons as well as thermal neutrons (curve C, Figure 12). Thus, the neutrons above thermal energies scatter in the galleries and are eventually reduced to thermal energies (in distances not less than 3 ft from the right-angle bend, as estimated from zinc sheet results). From the cadmium ratio (curve C, Figure 12), nearly one-half of the gold foil activity, as recorded in the auxiliary primary gallery at zinc sheet C, is due to neutrons with energies greater than the cadmium cut-off, which means that the flux of neutrons above the cadmium cut-off is greater than the thermal neutron flux. Therefore, a buildup of the thermal neutron flux can be expected which will partially offset the reduction due to the increase in distance from the scattering wall. At distances greater than 10 ft from the right angle bend, the theoretical and experimental attenuation results were found to be in better agreement provided that the thermal neutron flux experimentally determined at the 5.4-ft distance was used as the basis for the theoretical attenuation calculations.

An average thermal neutron flux of approximately 1×10^5 n/cm²-sec was determined in the far section of the auxiliary primary gallery near the primary sodium pumps. Since the zinc sheets activated in this region had a very low counting rate (approximately twice the normal background for the counting instrumentation being used), these results would contain large uncertainties due to poor counting statistics.

Whether or not this thermal neutron buildup effect would be great enough to explain the discrepancy between the theoretical and experimental results obtained between zinc sheet A and B is not known, even though such a condition does exist. Otherwise, the theoretical method used in these comparisons seems to give relatively good agreement for all cases where such buildup conditions as stated above do not exist. Therefore, the recommendation made at the present time is to use this method to analyze all similar problems until more experimental data becomes available that might prove the method in error or give more information as to its limitations.

VI. EVALUATION OF EXPERIMENTAL METHOD

Although the values obtained from these experiments were important and necessary for the problem discussed herein, the main conclusion that can be drawn from this report does not concern itself with the results obtained but rather with the versatility of the method which was devised for these experiments. The accuracy with which the thermal neutron streaming intensities, and consequently the areas of these intensities, can be determined enables the experimentalist to obtain irregular thermal neutron densities in critical but often inaccessible areas. Also, the nuclear properties of the detector will enable a more extensive evaluation of neutron shielding and streaming in areas which are not accessible by other detection method. The conditions stated in Sections II and III-A give the limitations of the detector as well as of the methods used in this experiment. Thus, if this method is considered for use in other experiments, the limitations that have been discussed herein should be closely examined before such an experiment is undertaken.

The versatility and accuracy of this method was determined from a number of experiments performed in the SRE primary galleries. The results obtained from this method have been given in the analysis which is contained herein. The experiments that were performed utilizing the zinc sheets as the detection media are listed in Table VI.

TABLE VI
EXPERIMENTS PERFORMED IN SRE EMPLOYING ZINC SHEETS
AS THE DETECTION MEDIA

	Experiment
Thermal Neutron Streaming at Reactor Vessel Pipe Penetrations	1
Attenuation Properties of Proposed Neutron Shield	2
Neutron Attenuation Properties for Right-Angle Bends in SRE Galleries	3
Neutron Attenuation Properties for Labyrinths in SRE Galleries	4
Neutron Streaming Along and Through Sodium-Filled Pipes	5
Neutron Attenuation Properties of Rectangular Cells Containing a Maze of Piping, Equipment, etc.	6

VII. CONCLUSIONS

The following conclusions can be made about the results obtained from the experiments conducted in the SRE sodium galleries.

The experimental method of determining thermal neutron streaming in the SRE primary galleries has proven to be versatile, accurate and reliable. Using the method of detection and analysis described herein, many experiments that could not be performed previously, due to lack of accessibility and the limitations of the available detectors, can now be undertaken. However, if this method is considered for use in experiments other than those described in this study, the limitations stated in Sections II and III, should be closely examined before any such experiment is undertaken.

The streaming thermal neutron flux in the auxiliary primary gallery of the SRE has been reduced by the addition of neutron shielding in the auxiliary primary gallery. However, measurements of the gamma-ray dose rate should continue to be performed before personnel attempt access to these areas.

The theoretical calculational method outlined herein should be used to determine the thermal neutron flux levels in similar labyrinths. Although the experimental data obtained from the SRE experiments has not determined the limitations of this theoretical method, the results indicate good agreement for distances up to 15 ft from the source and after two right-angle bends.

The zinc activity did not show an increase in the vicinity of the sodium pipes. Thus, no excessive thermal neutron streaming was apparent due to sodium pipe penetrations through the neutron shield installed in the bellows assembly of the SRE auxiliary primary gallery.

Apparently the presence of piping, equipment, etc., in the SRE galleries does not reduce the streaming thermal neutron flux. In calculations for such cases, it appears warranted to assume an empty duct.

The theoretical attenuation calculation for the installed neutron shield predicted that the shield would be considerably more effective than was measured. The effect of boral channeling and cracks in the concrete blocks are probably the major errors in the theoretical approach. Thus, caution should be taken if other calculations are made for similar neutron shields.

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