CHAPTER II
LITERATURE SURVEY

Radiation techniques based on neutron scattering and transmission have received considerable attention in previous studies. Before discussing those studies, the concepts of both techniques are reviewed. The three essential components involved in the neutron technique are subsequently identified. Previous work on investigating these components and some other factors, which may affect the technique, are then discussed and assessed. At the end of this chapter the fundamentals of the Monte Carlo method are reviewed.

2.1 Steam Quality Measurement Techniques

As mentioned before, steam quality can be estimated from void fraction. Therefore, there are a large number of techniques for steam quality measurement that have been used with varying degrees of success and have been reviewed by early workers (Hewitt, 1982). The void fraction measurement techniques can be classified into two general types, intrusive and non-intrusive. Intrusive methods such as quick closing valves, impedance gauges and electromagnetic probes have the disadvantage of penetrating the piping structures and altering the flow itself. These make them inapplicable to the thick-wall pipes of high-pressure systems, even though they can gather vast quantities of direct information about two-phase flow parameters. For these reasons they are not discussed here.

The non-intrusive methods based on nuclear radiation can eliminate these problems. There are several types of radiation that can be used for the measuring the flow parameters, such as beta particles, gamma rays, X-rays and neutrons. The appropriateness of the radiation beam, however, depends upon many factors such as the type of fluid transported, the quantity of material contained, the pipe size and material and the complexity of the pipe systems. For instance, gamma and X-rays were not recommended for measuring the void fraction and phase distribution in rod bundles or packed beds (Banerjee *et al.*, 1978) due to the high attenuation caused by the pipe wall and somewhat the resulting low interaction with the fluid.

Neutrons, however, are not significantly affected by the high atomic number material (steel) of the pipes but are greatly affected by hydrogeneous material inside the pipe, which is steam and water in this work. Therefore, the most promising radiation method for void fraction measurement in a high-pressure system is the neutron technique, which is discussed below.

2.2 Neutron Techniques

Neutron techniques are based on the interaction of neutrons with nuclei. They can be divided into two main techniques, which are scattering and transmission as shown in Figure 2.1. Neutrons are classified into four energy ranges; cold (10^{-4}-10^{-3}eV), thermal (0.025 eV at room temperature), epithermal (0.5eV to kV) and fast (MeV).

Early developers of the neutron technique such as Harms *et al.* (1971), Frazzoli and Magrini (1979) and Younis *et al.* (1981) concentrated on utilizing thermal neutron transmission. However, the most obvious problem in the application was the availability of a suitable neutron source since the neutron beam can be extracted only from either a nuclear reactor or a bulky thermalization assembly containing an isotopic source. Thus, it is difficult to produce a compact or portable device. In order to eliminate this problem, attention was changed to developing fast neutrons for measuring the void fraction since they are readily produced from available inexpensive neutron sources. As summarized in Table 2.1, fast neutron techniques are reviewed with the specified details to help understand the chronological development.

In addition, Hussein (1995) reviewed fast-neutron techniques in order to encourage their use in applications in which conventional techniques are difficult to apply. It was also shown that fast neutrons can be used to distinguish between two phases that are similar in their physical properties by introducing a neutron-sensitive additive to one of the phases.

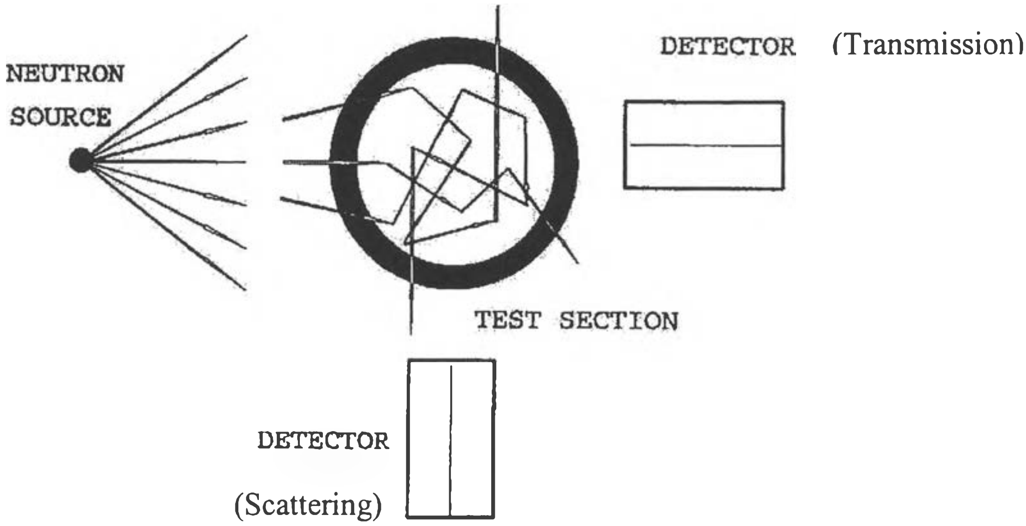

Figure 2.1 Schematic of neutron technique device.

Table 2.1 Summary of existing fast neutron techniques for diagnosing two-phase flow parameters.

Reference	Technique	Neutron source	Detector	Test section
Sha and Bonilla, 1965	Fast neutron moderation and thermalized neutron absorption	Sb-Be (0.03 MeV to 0.625 eV) 9×10^5 n/s	BF_3 proportional counter	Thin wall OD = 88.9, 152.4, and 219.1 mm At ambient environment
Jackson *et al.*, 1968	Fast neutron moderation and/or absorption	Tritium target reaction T(D,n) (14MeV) 1×10^{11} n/s	Three boron-lined proportional counters	Schedule-160 steel pipe with 1 in. thick wall 302-324 °C, 8717-10785 kPa and volumetric flow rate up to 4.25 m^3/min
Rousseau *et al.*, 1976	Fast neutron scattering	Reactor (5MW) Epithermal neutrons 1×10^7 n/s	Boron-lined proportional counters	ID = 102.3 mm OD = 114.3 mm 200-230 °C
Banerjee *et al.*, 1978	Fast neutron scattering and transmission	Reactor (5MW) Fast/epithermal neutrons 1×10^{11} n/s	^3He	Aluminum pipe ID = 25.4, 38.1 and 50.8 mm

Table 2.1 Summary of existing fast neutron techniques for diagnosing two-phase flow parameters (continued).

Reference	Technique	Neutron source	Detector	Test section
Frazzoli *et al.*,1978	Fast neutron scattering	0.5 μg of moderated ^{252}Cf (2 MeV) 1×10^{6} n/s	BF_3 proportional counter	Iron tube ID = 10 cm with 1 cm thick wall and length = 30 cm and nineteen Plexiglas tubes
Banerjee *et al.*,1979	Fast/epithermal neutron scattering and transmission	Reactor (5MW) Fast/epithermal neutrons	^{3}He	Aluminum rod ID = 13 mm in bundles of 102 mm
Banerjee *et al.*,1979	Fast/epithermal neutron scattering and transmission	Reactor (5MW) Fast/epithermal neutrons	^{3}He ID = 25 and 55 mm	ID = 25.4 and 50.8 mm
Hussein, 1988	Fast neutron scattering	Various source	BF_3	ID = 12.5-100 mm
Yuen *et al.*, 1988	Fast neutron scattering	$^{241}Am/Be$ (0.4 MeV) ^{252}Cf (2.8 MeV) $^{241}Am/Be$ (5.1 MeV)	Two ^{3}He detectors sub-cadmium	ID = 51 and 127 mm OD = 70 and 133.4 mm

Table 2.1 Summary of existing fast neutron techniques for diagnosing two-phase flow parameters (continued).

Reference	Technique	Neutron source	Detector	Test section
Hussein and Waller, 1990	Fast neutron scattering	2 μg of ^{252}Cf (2.2 MeV)	BF_3 proportional counter	Carbon steel OD = 127 mm, 7.1 mm thick and 200 mm height
Waller and Hussein, 1990	Fast neutron scattering	$^{241}Am/Be$	^3He	Carbon steel ID = 22 mm and 1.5 mm thick 50°C and atmospheric pressure
Han *et al.*, 1994	Fast neutron scattering	2 μg of ^{252}Cf moderated by 40 mm thick paraffin wax slab	^3He	Pressure tube ID = 44.8 mm OD = 57.2 mm Seven heating rods ID = 13.1 mm
Hussein and Han, 1995	Fast neutron scattering and transmission	$^{241}Am/Be$	^3He	Carbon steel ID = 130 mm OD = 142 mm

2.3 Concept of Fast Neutron Scattering Techniques

This technique relies on measuring the thermalization of fast neutrons incident on the pipe. The number of thermalized neutrons is used to estimate the amount of liquid present and, in turn, the void fraction or steam quality can be deduced. A device based on neutron scattering techniques is called a scatterometer, which is basically composed of a fast neutron source, a test section and a neutron detector typically located perpendicular to both the incident fast neutron beam and the test section as shown in Figure 2.1. In the scatterometer, the incident fast neutron beam is assumed to have a relatively uniform intensity. It will lose a significant amount of energy (its speed will be reduced) to the extent that it has approximately the same average kinetic energy as the atoms of the scattering medium. A neutron therefore reaches a thermal equilibrium state after several scattering collisions with nuclei of a hydrogeneous material such as water and paraffin wax; the energy loss due to interactions with the nuclei of heavy metals can be ignored. As a result, the scattered thermal fluxes depend on the amount of scattering material in the cross section. For the steam-water flow system, most of the thermalized neutrons result from the hydrogen atoms in the liquid phase rather than in the vapor phase because those in the vapor phase are too few to cause significant slowing down of the neutrons. Moreover, elastic scattering by the metallic container hardly affects the slowing-down process.

The void fraction can be predicted from the following equation:

$$\hat{\varepsilon} = \frac{N(0) - N(\varepsilon)}{N(0) - N(1)} \tag{2.1}$$

where $\hat{\varepsilon}$ is estimated void fraction and $N(\varepsilon)$, $N(0)$ and $N(1)$ are the detector count rate corresponding to the test section containing steam-water mixture, containing water and containing vapor, respectively.

This relationship is assumed to be linear in order to cheack the proportionality of the scatterometer response to the liquid fraction. Also, as the void fraction decreases the neutron count rate increases due to the increasing liquid fraction. Linearity can be achieved by matching the neutron source energy range to the pipe size (Hussein, 1988). If the two-phase flow in the test section produces a high degree of thermalization of the incident fast neutrons, the number of scattered neutrons provides a linear response to void fraction and fairly small depends on the flow regime. Additionally, two neutron detectors were recommended in order to minimize the flow-regime dependence (Banerjee, 1978).

2.4 Concept of Fast Neutron Transmission Techniques

The fast neutron transmission for steam quality measurement is similar to gamma-ray attenuation techniques because it is mainly based on the number of neutrons that can pass through the test section without collision. The components of this technique are the same as the fast neutron scattering technique, except that the detectors are placed co-axial with the neutron beams as shown in Figure 2.1. The number of uncollided neutrons reaching a detector increases exponentially as the density of the hydrogeneous material in the test section decreases and, therefore, it can be used to measure the void fraction in two-phase flows if there is knowledge of the flow patterns (Frazzoli and Magrini, 1979). However, some collided neutrons can also reach the detector. As the density of hydrogeneous flow decreases, the number of collided neutrons, which depends on the energy of the incident neutrons, the test section size, the detector aperture and the distance between the test section and the detector, also decrease.

Therefore, the performance of this technique can be improved by either minimizing or accounting for the contributions from collided neutrons. The easiest way is to decrease the collision probability, which can be done by increasing the energy of the neutron beam or by reducing the size of the test section and the amount of materials used. These methods are better than decreasing the detector aperture, which reduces the detector angle of view and, consequently, only the fluid density within this angle can be estimated. Even though increasing the distance between the detector and the test section is probably the simplest way, it is usually at the expense of the count rate. Another method is to discard all neutrons of energy less than that of the neutron beam, since collided neutrons lose part of their energy during the scattering process.

The size of the test section is restricted by the source energy. For example, if a low energy source is applied to a large pipe, the contribution of collided neutrons to the detector count rate becomes significant and gives a higher value of water content than the actual value. Therefore, higher energy neutrons must be used for larger pipes.

2.5 Components of Fast Neutron Techniques

2.5.1 Fast Neutron Source

Fast neutrons can be emitted from radioisotopes by spontaneous fission, obtained from the (α,n) reaction, photoneutron reactions and reactions from accelerated charged particles, or extracted from a nuclear research reactor. During the early development of neutron techniques, neutrons extracted from nuclear reactors were extensively used. However, they were not ideal because they give higher fluxes of thermal neutrons and lower fluxes of fast neutrons in the keV range. Moreover, a reactor cannot be used as a portable source for practical applications. Therefore, this leaves radioisotope sources as the more important. Some of the many commercially available isotopic sources are shown in Table. 2.2.

Table 2.2 Summary of commercial isotropic sources.

	Cf-252	Am-241/Be	Am-241/Li	Am-241/B	Ra-239/Be
Neutron emission (neutron per second)	2.3×10^6 /μg*	60/MBq	1/MBq	14/MBq	37/MBq
Half-life (year)	2.65	433	433	433	24000
Fraction of neutrons below 1.5 MeV	0.46	0.23	1.0	0.06	0.33
Gamma exposure rate at 1m (per 10^6 neutrons/s, μG/h)	1	10	625	50	-

* Specific activity of Cf-252: ~20 MBq/μg.

Because of the difference properties of each source, it is essential to select the appropriate source for a specific system. The two most important parameters that must be considered first were the energy spectrum of the source and the average energy because they may affect the performance of fast neutron techniques in terms of linearity and flow regime independence (Yuen, 1985). Moreover, it was found that the required energy spectrum depended on the size of the test section, its wall thickness and the amount of metal content for each system. The larger the test section and wall thickness, the higher the source energy required.

In general, there are two procedures to determine the most suitable source energy. First, Yuen (1985) suggested employing a general-purpose Monte Carlo code. However, this is sophisticated and requires experience from users. Moreover, the Monte Carlo code is expensive, time consuming to execute and requires a large computer storage space. Therefore, Hussein (1988) developed a simple model based on collision probability (CP) theory to determine the source energy range. Later, a simplified Center-Of-Mass (COM) Monte Carlo program was developed to verify the results of the CP model at different flow regimes.

The most widely used isotropic neutron source is ^{252}Cf, although its half-life is low. It has a relatively high neutron emission and produces a low gamma-ray yield per neutron produced, which reduces the amount of material required for gamma ray shielding. However, commercially available sources sometimes cannot provide the required source energy for a given test section. Therefore, it is imperative to modify the source spectrum in order to achieve the required energy. The source energy spectrum can be softened by using a moderator to lower the source energy as shown in Figure 2.2 or hardened by using cadmium sheets wrapped around the source to cut off the subcadmium (<0.5eV) or slow neutrons.

The moderators, which are often hydrogeneous materials such as water, wax, Lucite or plexiglas, are placed around the source to lower its energy. Moderator thickness was important to the detector's response, especially for a point source (Knoll, 1989). However, even if the moderator thickness is properly selected and the source intensity reduced, the number of low-energy neutrons is not necessarily decreased due to an insignificant absorption of low-energy neutrons while the fast neutrons are slowing down.

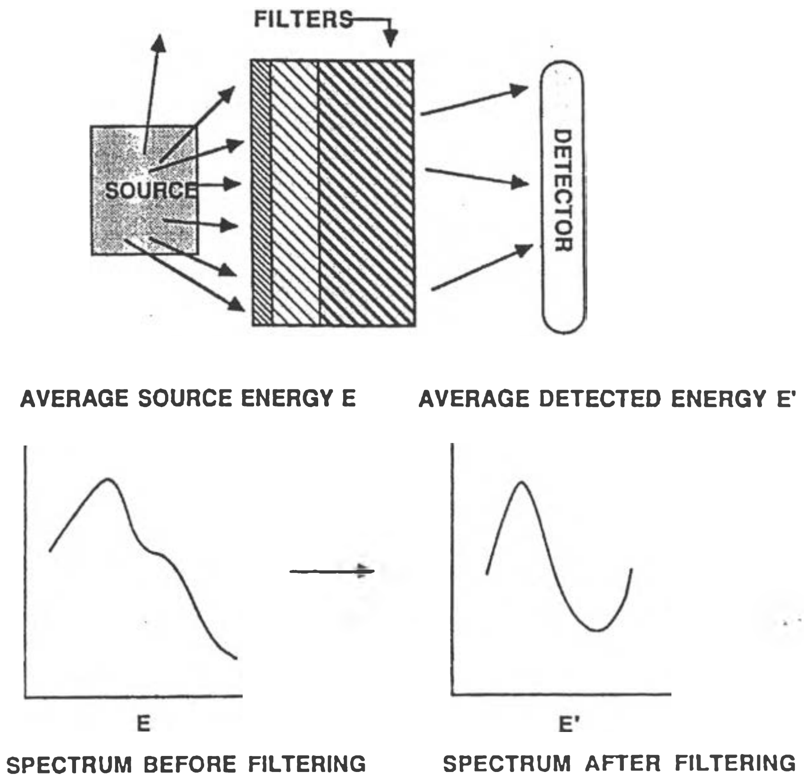

Figure 2.2 Filter concept.

Hussein (1988) found that the collimation of the neutron source did not affect the performance of the scatterometer because fast neutrons require several collisions in the process of thermalization before hitting the detector; consequently, they forget their origin point. On the other hand, it is essential for a transmission technique to collimate the neutron source and the detector in order to produce a well-defined path for neutrons. In addition, the source strength was found to be one of the key parameters especially when the measurement was taking place in a transient flow system, which required a higher count rate in order to reduce the statistical error (Hussein, 1988). However, employing a strong source is usually precluded because of radiation protection considerations.

2.5.2 Test Section

The test section can be considered as a secondary neutron source due to the cloud of thermal neutrons that is created by the slowing-down of source neutrons (Hussein, 1991). The distribution of the cloud, which is an important parameter that can affect the detector response, depends on the size, geometry and material of the test section as well as the source energy. For instance, the larger the test section was, the higher the source energy required for good linearity for a minimum flow regime dependent (Yuen, 1985 and Hussein, 1988). This was because the high source energy will cause a low concentration of neutron interactions in an area close to the source and therefore most of the test section will be more involved in providing the signal. On the other hand, if high-energy neutrons are applied to a small test section, they will pass through with few interactions, resulting in a low slowing-down probability.

For the effect of the geometry and material of the test section, Yuen (1988) found that a thick metallic wall absorbed some incident neutrons emitted from the source and, resulting in a reduction in the intensities of both incident neutron and out-going scattered neutrons. As a result, the void fraction tended to be overestimated. Moreover, the metallic test sections are not absolutely transparent to neutrons, particularly at lower energy, since as the ratio of thermal-neutron cross-section for hydrogen to that for iron becomes smaller as the neutron energy increases.

Radiological shielding around the test section is used to absorb neutrons that escape from the region of the test section. It may increase the thermalization probability and reflect lower energy neutrons towards the test section because it was usually made of hydrogeneous materials like the moderators (Yuen *et al.*,1988 and Hussein, 1988). Therefore, if it is not properly designed, it may significantly increase the background counting rate.

2.5.3 Detector

Neutrons are detected indirectly by the production of a photon and a charged particle (the result of a neutron interaction with a nucleus) because they cannot directly ionize atoms. Therefore, it is necessary to know the mechanism of the neutron interaction in order to estimate the number of neutrons that hit the detector by counting the products of the reaction. The number of particles that enter the detector is used to estimate the detector efficiency. For measuring the scattered slowed-down neutrons, helium (^3He) filled detectors and boron trifluoride (BF_3) detectors are the two main types used.

(1) ^3He detector

The ^3He detector is a proportional counter generally filled with helium in a cylindrical metal tube. It undergoes a (n, p) reaction as shown below (Knoll, 1989):

$$He^3 + n \longrightarrow H^3 + p + 765 \text{ keV}$$

Its efficiency for thermal neutron detection is very high due to a large cross section starting with 5400 barns for thermal neutrons and dropping to about 1 barn for neutrons of 2 MeV energy (Knoll, 1989), although a significant number of greater-than-thermal neutrons will be counted. It can also be used to measure sub-cadmium neutrons, which have energy below the cadmium cut-off of 0.5 eV, by measuring the neutron count rates obtained from the detector wrapped with cadmium foil and subtracting them from those obtained from an unwrapped detector.

(2) BF_3 detector

The BF_3 detector is a proportional counter filled with BF_3 gas. It detects alpha particle and lithium atoms produced by the B (n, α) reaction as follows:

$$^{10}_{5}B + ^{1}_{0}n \longrightarrow ^{4}_{2}He + ^{7}_{3}Li + 2.78 \text{ MeV}$$

It was widely used in early work for thermal neutron detection because boron has a large cross section-3480 barns at thermal energy. However, the BF_3 detector has a lower cross section for epithermal and fast neutrons than the ^3He detector.

The other parameter that should be considered is the geometric location of the detector defined by the relative directions and distances between detector, source and test section. For neutron scattering techniques, the detector location is not an important factor because the thermal neutrons are isotropically scattered out of the test section. It mainly affects the absolute count rate, but not the relative response to variation in void fraction. Moreover, the closer the detector got to the test section, the lower the average energy of neutron needed to be incident upon the test section (Waller, 1990). Nevertheless, the change in the detector location may affect the efficiency of fast and epithermal neutron detection. This provides more flexibility in the design of the scatterometer. For example, the orientation of the detector may be arranged to compensate for the effect caused by a hard or soft source for a specific test section.

In the situation where there is much metal but little hydrogenous fluid, the location of the detector (detecting forward, side or backscattered neutrons) can affect the performance of the scatterometer. This is because not all the incident neutrons can be fully thermalized so that the test section cannot be treated as an isotropic source of monoenergetic neutrons. In this case a ^{3}He detector, which is sensitive to epithermal neutrons, should be utilized (Han, 1994).

On the other hand, the direction is important in transmission for obtaining different chordal information. Associated electronic devices must be optimized because they may affect the overall characteristics of the system, especially the amplification coefficient. Electronic noise that results from the thermal motion of charge carriers in the components of the detection system such as cables, resistors and the detector itself manifests itself as a large number of low-level pulses that should be distinguished from the background pulses. In fact, early experimenters (Yuen, 1985 and Waller, 1990) introduced procedures for electronic optimization.

2.6 Thermodynamic Effects

Thermodynamic effects such as the temperature and pressure of the hydrogeneous flow medium may affect the performance of fast neutron techniques by altering either the neutron macroscopic or the neutron microscopic cross section. This is because both temperature and pressure affect the flow density or the number of the hydrogen atoms per unit volume. As the temperature increases, the probability that the hydrogeneous flow medium was slow down or scatters neutrons reduces because the flow density decreases. The probability of neutron interaction with a nucleus (i.e., the microscopic cross section) also decreases because it is inversely proportional to the square root of the thermal energy of neutrons and therefore of the energy at which neutrons reach equilibrium with the medium defined by Equation 2.2;

$$E_{th} = kT \qquad (2.2)$$

where E_{th} is the thermal energy of the neutron, T is the temperature of the medium and k is Boltzmann's constant. Some preliminary of temperature effect analyses (Yuen, 1985 and Waller, 1990) reported that the water fraction can be underestimated as temperature increases because of the decrease in the detector response for the same void fraction - especially in the low void fraction range. This means that detector response can vary significantly with temperature.

The pressure directly affects the density of the hydrogeneous flow medium and consequently the number of hydrogen atoms per unit volume. Therefore, when pressure increases, the probability of neutron interaction is higher due to the higher density of hydrogen atoms. However, Rousseau *et at* (1976) indicated that pressure may have only a slight effect and can be ignored for values changes less than 5MPa due to the small change in density of the liquid phase when pressure changes.

Hussein and Waller (1990) studied the effects of both temperature and pressure on the detector response and included them in the CP model. There is a difference in detector response operating conditions of 6.3MPa with a corresponding saturated temperature of 552 K and atmospheric conditions. As a result, the value of the water fraction under operating conditions, predicted using a device calibrated under atmospheric conditions, may be significant particularly as the water fraction increases. This can, however, be solved by introducing a correction procedure to incorporate the effect of the measured pressure and temperature. However, the predictions of temperature and pressure effects from CP models should be confirmed by experiment.

2.7 Monte Carlo Method

The Monte Carlo method approximately solves mathematical and physical problems by a random simulation. It is different from deterministic transport methods, which solve the transport equation for average particle behavior. Monte Carlo does not solve an explicit equation, but rather obtains answers by simulating individual particles and recording some aspects (tallies) of their average behavior. In other words, Monte Carlo solves a transport problem by simulating particle histories rather than by solving an equation. No transport equation need ever be written to solve a transport problem. Each particle history involves the generation of a source particle, its random walk as its moves through and collides with the transporting medium and finally its termination. Probability distributions are randomly sampled using transport data to determine the outcome at each step of particle life. Figure 2.3 represents the random history of a neutron incident on a slab of material that can undergo fission.

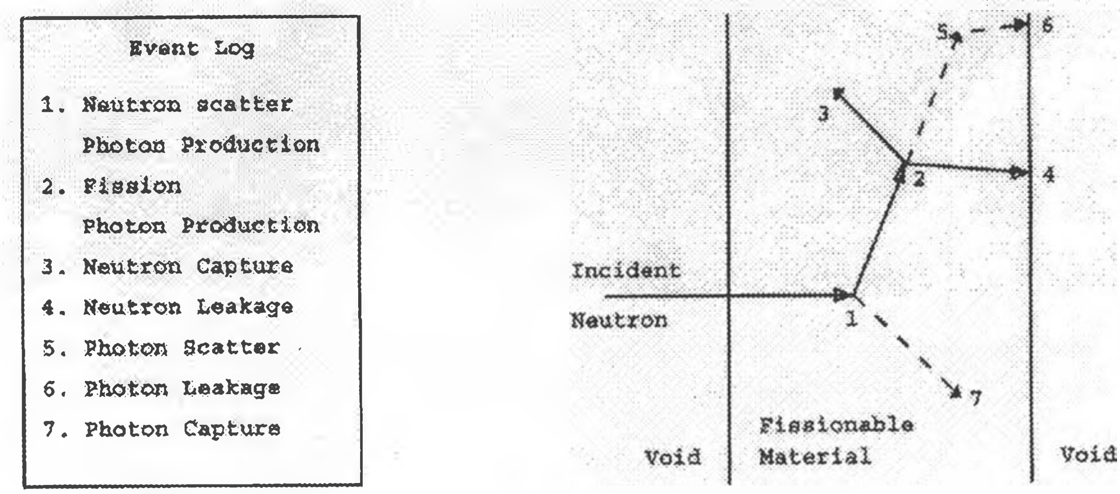

Figure 2.3 History of neutron incident on a slab.

In this particular example, a neutron collision occurs at event 1. The neutron is scattered in the direction shown, which is selected randomly from the physical scatter distribution. A photon is also produced and is temporarily stored, or banked, for later analysis. At event 2, fission occurs, resulting in the termination of the incoming neutron and the birth of two outgoing neutrons and one photon. One neutron and the photon are banked for later analysis. The first fission neutron is captured at event 3 and terminated. The banked neutron is now retrieved and, by random sampling, leaks out of the slab at event 4. The fission-produced photon has a collision at event 5 and leaks out at event 6. The remaining photon generated at event 1 is now followed with a capture at event 7. This neutron history is now complete. As more and more such histories are followed, the neutron and photon distributions become more accurate (Los Alamos National Laboratory, 2000).

The Monte Carlo method is well suited to solving complicated three-dimensional, time-dependent problems such as scatterometry, which is a complex process and cannot be described by simple mathematics. The general purpose Monte Carlo codes such as MORSE and TRIPLOI were developed by early workers (Banerjee *et al.*, 1979) to calculate parameters that are difficult to evaluate in the laboratory and to provide some intrinsic information useful for optimizing the system design. The main drawbacks of these codes are expensive to use, need for large computer storage, demand for access to a neutron cross-section library and require most of extensive experience. Hussein (1988) successfully developed a simple analytical model based on collision probability theory and a special simplified Monte Carlo code called the Center-Of-Mass (COM) computer program in order to facilitate the design and optimization of source strengths for various sizes of cylindrical pipe.

2.7.1 MCNP

MCNP stands for Monte Carlo N-Particle, which is a general-purpose, generalized-geometry and time-independent Monte Carlo transport code for calculating radiation transport. The Monte Carlo group in the applied theoretical physics division at the Los Alamos Laboratories developed MCNP between 1973 and 2000. The energy regimes for the neutron transport mode is from 10 -11 MeV to 20 MeV. The code allowed one to specify a wide variety of source conditions without having to make modifications. In order to input probability distributions for source variables, certain built-in functions are available. These include various analytic functions for fission and fusion energy spectra, such as Watt, Maxwellian and Gaussian spectra. The various tallies related to particle current, particle flux and energy deposition can be determined. MCNP tallies are normalized per starting particle and are functions of time and energy.

2.7.2 MCNP Geometry

The geometry of MCNP treats an arbitrary three-dimensional configuration of user-defined materials in geometric cells in a right-handed Cartesian coordinate system. The cells are defined by the intersections, unions, and complements of the regions bounded by the surfaces. Surfaces are defined by supplying the appropriate coefficients to satisfy the analytic surface equations or by specifying known geometrical points on certain types of surface that are rotationally symmetric about a coordinate axis. The surfaces are designated by mnenomics such as C/Z for a cylinder parallel to the z-axis.

2.7.3 MCNP Input File

By using a basic knowledge of UNIX (UNiplexed Information and Computing System), the user can create the input file in order to run MCNP. Basically, this file is mainly composed of a cell card, a surface card and a data card, respectively. The cell card is used to specify the geometric cells and the material descriptions of the problem. It comprises the list of the cell number, material number and material density followed by a list of operators and signed surfaces that bound the cell. The sign denotes the sense of the regions defined by the surfaces. The regions are combined with the Boolean intersection and union operators. The surface card, a list of surface numbers and alphabetic mnemonic, is used to indicate the surface type and the numerical coefficients of the surface in the proper order.

Finally, the data cards are used to determine the simulation mode in the mode cards (MODE), the relative cell importance in the cell and surface parameter cards (IMP:N), the starting particles and, the location and characteristics of the neutron, photon, or electron source in the source specification cards (SDEF), the type of answers or tallies and any variance reduction techniques used to improve efficiency in the tally specification cards (Fn, En), the isotopic composition of the materials and the cross-section evaluations in the material specification cards (Mn), and finally the parameters for some of the ways to terminate execution of MCNP in the problem cutoff cards (NPS).

2.7.4 Estimation of MCNP Errors

For the error estimation, an estimated relative error (R), which is defined to be one estimated standard deviation of the mean ($S_{\bar{x}}$) divided by the estimated mean ($\bar{x}$) is shown in the MCNP tallies. For a well-behaved tally, R will be proportional to the inverse of the square root of the number of histories, for a poorly-behaved tally, R may increase as the number of histories increases. There are four classes of variance reduction technique. First, the simplest one is the truncation method, which will speed up calculations by truncating parts of phase space that do not contribute significantly to the solution. Second, the modified sampling method alters the statistical sampling of a problem to increase the number of tallies per particle. The sampling is done from distributions that send particles in desired directions or into other desired regions of phase space such as time or energy, or change the location or type of collision. Next, the most complicated one is the partially deterministic method because it circumvents the normal random walk process by using deterministic-like techniques, such as next-event estimators, or by controlling the random number sequence. The last one is the variance reduction technique. It should be carefully used because it can result in a wrong answer with good statistics and few clues that anything is amiss.