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SUBJECT: Forwards response to NRC 830502 request for addl info re assumptions & methodology used in estimating post-accident radiation environs for equipment items.
SEE REPTS.

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November 3, 1983

AEP:NRC:0578I

Donald C. Cook Nuclear Plant Unit Nos. 1 and 2
Docket Nos. 50-315 and 50-316
License Nos. DPR-58 and DPR-74
EQUIPMENT ENVIRONMENTAL QUALIFICATION PROGRAM;
CALCULATION OF SPECIFIED RADIATION DOSES

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Denton:

This letter responds to Mr. S. A. Varga's letter dated May 2, 1983, which requested additional information on the environmental qualification program for electric equipment important to safety at the Donald C. Cook Nuclear Plant Unit Nos. 1 and 2. More specifically, Mr. S. A. Varga requested that we provide a reference for the assumptions and methodology used in estimating post-accident radiation environments for equipment items of concern. Furthermore, we were requested to include a description of the Donald C. Cook Nuclear Plant containment type and a discussion regarding the source terms which have been used in the qualification program.

Attachment 1 to this letter provides a general description of the ice condenser reactor containment system utilized by the Donald C. Cook Nuclear Plant. Please note that the Final Safety Analysis Report for our facility contains additional information which we have not included in this abbreviated description.

Attachment 2 to this letter addresses the basic methodology used in calculating radiation environments both inside and outside the ice condenser containment. In general, we utilize a computer code which calculates gamma and beta doses due to a cylindrical source which is modeled as a set of line sources. Versions of this code (e.g., NSLSHL3 and SHL1GG) have been used in a number of applications, such as calculating equipment radiation doses for equipment items which are near the recirculation flow path piping systems following a Loss-of-Coolant Accident (LOCA). We have also applied this methodology to airborne and submerged source term dose calculations for many components inside containment.

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The application of this methodology to equipment items inside containment is described in additional detail in Attachment 3 to this letter. That attachment provides information on the assumed source term (comprised of fifty-four radioisotopes), the studies which led to the selection of bounding accumulated dose values as a function of time for components inside containment (both above and below the containment flood-up level), and sample calculations.

Attachment 4 to this letter presents the assumptions which led to similar bounding accumulated dose values as a function of time for components outside containment. It is noted that the computer code which was used in the outside containment studies formed the basis for the computer code utilized in our work on equipment within containment. The methodology employed by each of these codes is, however, similar enough to ensure that no appreciable differences exist between the basic calculational methods used in calculating radiation doses either inside or outside containment. The results of sample calculations are also included in Attachment 4 for numerous equipment items located near recirculation flow paths in a post-LOCA environment.

It should be noted that we have not enclosed calculations for every equipment item identified in response to either IE Bulletin No. 79-01B or 10 CFR 50.49. Rather, this submittal is intended to provide your staff with an understanding of the basic methodology and assumptions which are used in determining environmental qualification radiation specification doses for many electric equipment items. We also note that page 2-2 of Attachment 2 discusses certain differences between the specification doses presented in Attachments 3 and 4 of this submittal and the specification doses listed on the System Component Evaluation Worksheets (SCEW sheets) presented in Attachments 4 and 5 to our letter No. AEP:NRC:0578B, dated June 11, 1982. The reasons for these differences are also described in Attachment 2 to this submittal. Since the specification doses for many electric equipment items were revised during the process of preparation of this letter, we have not yet had time to apply our quality assurance procedures to the new calculational results. A quality assurance review of these results and an ongoing review of the radiation specification doses for the Donald C. Cook Nuclear Plant will be conducted following transmittal of this submittal. If these reviews uncover the need for any additional changes, we will advise you of those changes by separate letter.

This document has been prepared following Corporate Procedures which incorporate a reasonable set of controls to ensure its accuracy and completeness prior to signature by the undersigned.

Very truly yours,

M. P. Alexich
Vice President

MPA/dam
cc: (attached)

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R. C. Callen
G. Charnoff
E. R. Swanson, NRC Resident Inspector - Bridgman

ATTACHMENT 1 TO AEP:NRC:0578I
GENERAL DESCRIPTION OF ICE CONDENSER CONTAINMENT
DONALD C. COOK NUCLEAR PLANT UNIT NOS. 1 AND 2

Control # 8311110065

General Description Of Ice Condenser Containment

The ice condenser reactor containment system is divided into three major compartments -- the reactor coolant system or lower compartment, the upper compartment, and the ice condenser compartment. Figures 1-1 through 1-3 present the general boundaries of these compartments. These Figures also show the dead-ended compartments within containment whose air volumes are not displaced by steam into the upper compartment during a Loss-of-Coolant Accident (LOCA).

The lower compartment completely encloses the reactor coolant system equipment and associated auxiliary systems equipment. The upper compartment contains the refueling canal, refueling equipment, and the polar crane which is used during refueling and maintenance operations. The upper and lower compartments are separated by a low leakage barrier (e.g., the operating deck) to minimum steam bypass between the compartments during a LOCA. The dead-ended volumes are adjacent to the lower compartment and include the auxiliary pipe tunnel, the accumulator compartments, and the instrument room.

The ice condenser compartment, which contains the borated ice provided to quench the energy released during a LOCA, is in the form of a completely enclosed and refrigerated annular compartment which is located radially between the reactor coolant system compartment and the outer wall of the containment and, in elevation, generally above the operating deck. That portion of the ice condenser which extends into the lower compartment has a series of hinged doors exposed to the atmosphere of the lower containment compartment. For normal plant operation, these doors are designed to remain closed. At the top of the ice condenser is another set of doors exposed to the atmosphere of the upper containment compartment. These doors are also designed to remain closed during normal operation. The ice bed is held within the ice condenser in baskets arranged to promote heat transfer from steam to ice should the condenser be needed to serve its function. A refrigeration system maintains the ice in the solid state. Suitable insulation surrounding the ice condenser compartment minimizes heat transfer to the ice condenser enclosure.

In the event of a Design Basis Accident (DBA) such as a LOCA or a Main Steam Line Break (MSLB) within containment, the door panels located below the operating deck open due to the pressure rise in the lower compartment. This allows air and steam to flow from the lower compartment into the ice condenser. The resulting buildup of pressure in the ice condenser causes the door panels at the top of the condenser to open, allowing for air flow into the upper compartment. The steam condenses quickly upon entering the ice condenser, thus limiting the peak pressure in the containment building. The major factor which sets the peak pressure reached within containment as a result of a DBA is the compression of air displaced by steam as it flows from the lower and ice condenser compartments to the upper compartment. Therefore, since the peak pressure is related to the ratio of volumes for the various containment compartments, judicious selection of compartment volumes during the containment design process can alter the predicted peak pressure until any desired value is achieved.

It is also noted that condensation of steam within the ice condenser causes a pressure differential between the upper and lower compartments which results in a continual flow of steam from the lower compartment to the condensing surface of the ice. This helps reduce the time that the containment is at elevated pressure.

Performance Criteria For The Ice Condenser Containment

The performance of the ice condenser containment is demonstrated by results and analysis of ice condenser tests performed on a full-scale section test at the Westinghouse Waltz Mill Site. These tests confirmed the ability of the ice condenser to perform satisfactorily over a wide range of conditions, exceeding the range of conditions that might be experienced in an accident inside the containment building.

The ice condenser containment performance has been evaluated by testing the effect of certain important parameters. A partial list of parameters tested include blowdown rate, blowdown energy, deck leakage, compression ratio, drain performance, ice condenser hydraulic diameter, dead-ended volumes, and long term performance. Analytic models have been developed to correlate and supplement these test results in the evaluation of the containment design. The results indicate that the analytic models are conservative and that the performance of the ice condenser containment is predictable relative to these variables.

The energy absorption capacity of the ice condenser is at least twice that required to absorb all of the energy that can be released during the initial blowdown phase of the reactor coolant system for all reactor coolant pipe break sizes up to and including the hypothetical double-ended severance of the reactor coolant piping, or for any steam system pipe break size up to and including the hypothetical severance of the main steam line inside containment, without exceeding the containment design pressure.

Steam bypass of the ice condenser during a postulated reactor coolant system blowdown is to be avoided. The operating deck and any other leakage paths between the lower and upper compartments are reasonably sealed to limit bypass steam flow to a low value previously approved during the design phase of the Donald C. Cook Nuclear Plant containment design process. For the containment, the analysis considered bypass area as composed of two parts -- a conservatively assumed leakage area around the various hatches in the operating deck, and a known leakage area through the deck drainage holes for containment spray, located at the bottom of the refueling canal.

Flow distribution to the ice condenser for any reactor coolant system pipe rupture that opens the ice condenser lower inlet doors, up to and including the hypothetical double-ended severance case, is limited such that the maximum energy input into any section of the ice condenser does not exceed its design capability. The door port flow resistance and size provides for flow distribution for breaks that fully open the inlet doors. For breaks that only partially open the inlet doors to the ice condenser, the lower inlet doors act to proportion flow into the ice bed to limit maldistribution effects.

Analysis of the ice condenser reactor containment performance has shown that the ice condenser alone is capable of preventing containment overpressure during the initial blowdown of the reactor coolant system or secondary side system within containment, such that containment spray is not a requirement for overpressure protection. However, extremely small blowdown rates would not generate a differential pressure sufficient to open the ice condenser lower inlet doors. In this case, the energy release (even at an assumed small rate) would eventually require containment spray operation to prevent overpressure.

Another case has been examined where it is postulated that a Small Break LOCA (SBLOCA) precedes a larger break accident. The larger break accident is assumed to occur before all of the coolant energy is released by the SBLOCA (i.e., a double accident). During the SBLOCA blowdown, some quantity of steam and air will bypass the ice condenser and enter the upper compartment via leakage through the operating deck. The important design requirement for the case of the double accident is that the amount of steam leakage into the upper compartment be limited during the SBLOCA phase of the double accident so that only a small increase in final peak pressure results for the second part of the double accident. In general, the resultant peak pressure will be a function of both break sizes, the time between breaks, and the steam bypass during both phases of the double accident. The containment spray system is used to limit the partial pressure of steam in the upper compartment due to deck bypass. The key elements which determine the double accident performance of the ice condenser are the lower inlet doors, which open at low differential pressure to admit steam to the ice condenser and limit bypass flow of steam to the upper compartment, and the containment sprays which condense the bypass flow of steam and limit the partial pressure of steam in the upper compartment to a low value. The containment spray set point actuation pressure has been set at 3 psig to limit steam partial pressure to less than 2 psia in the upper compartment for the double accident case.

After a LOCA, the ice condenser has sufficient remaining heat absorption capacity such that, together with the containment spray system, subsequent assumed heat loads are absorbed without exceeding the containment design pressure. The subsequent heat loads considered include reactor core and coolant system stored heat, residual heat, substantial margin for an undefined additional energy release, and consideration of steam generators as active heat sources.

The primary purpose of the containment spray system is to spray cool water into the containment atmosphere in the event of a LOCA, thereby ensuring that containment pressure cannot exceed the containment design pressure. Protection is afforded for all pipe break sizes up to and including the hypothetical instantaneous circumferential rupture of a reactor coolant pipe. Adequate containment heat removal capability for the ice condenser containment is provided by two separate full capacity containment spray systems. The containment spray system is designed based on the conservative assumption that the core residual heat is continuously released to the containment as steam, eventually melting all ice in the ice condenser. The heat removal capability of each spray system is sized to keep the containment pressure below design pressure after all the ice has melted and residual heat generated steam continues to enter the containment.

Additional information on the design and function of the Donald C. Cook Nuclear Plant ice condenser containment can be found in the Final Safety Analysis Report (FSAR). In particular, the FSAR provides a copious amount of information on structural adequacy during seismic events, containment integrity during postulated blowdowns of the primary and secondary coolant systems, details regarding how containment isolation is achieved during DBAs, etc.

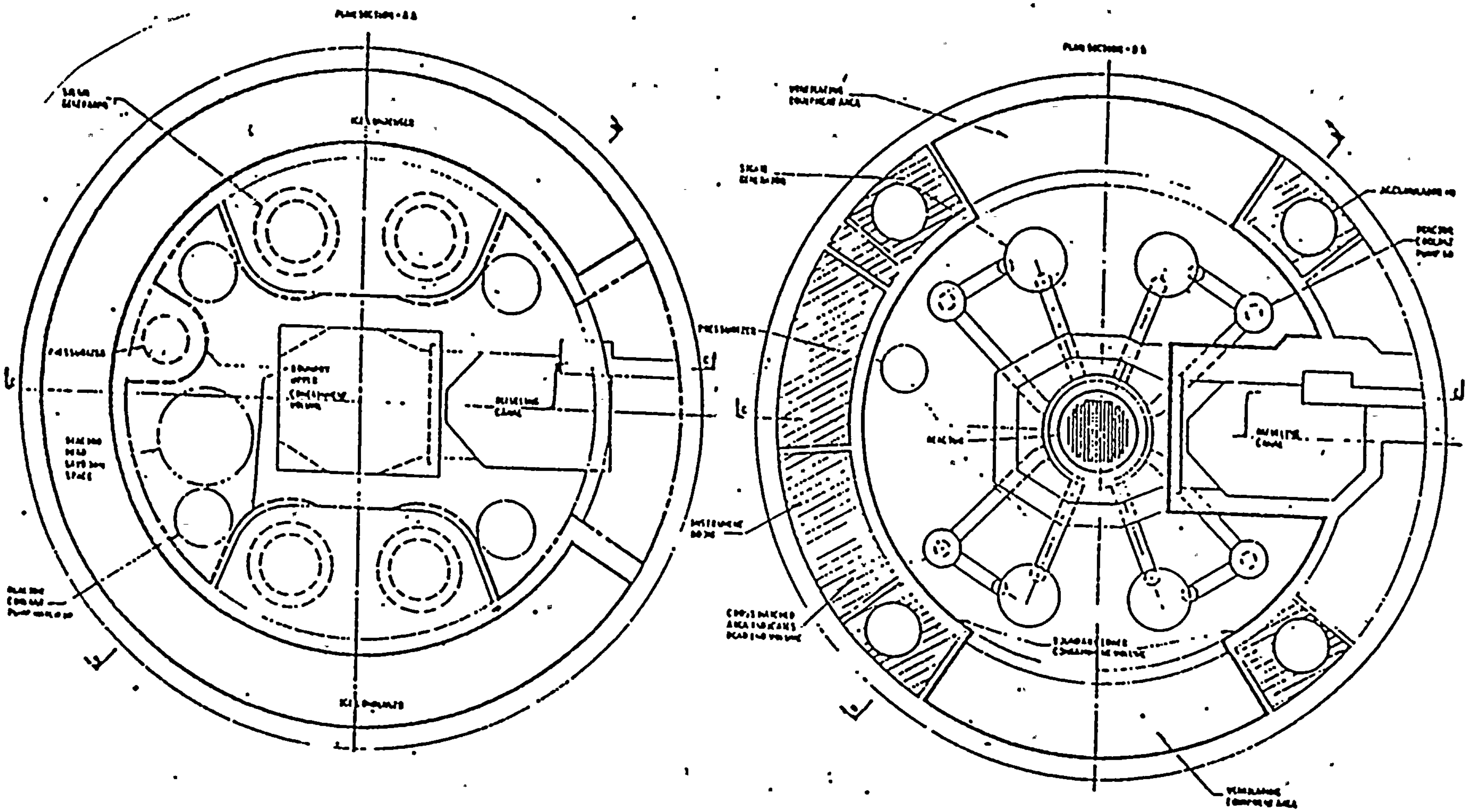

Figure 1-1 Ice Condenser Containment Volume Boundaries

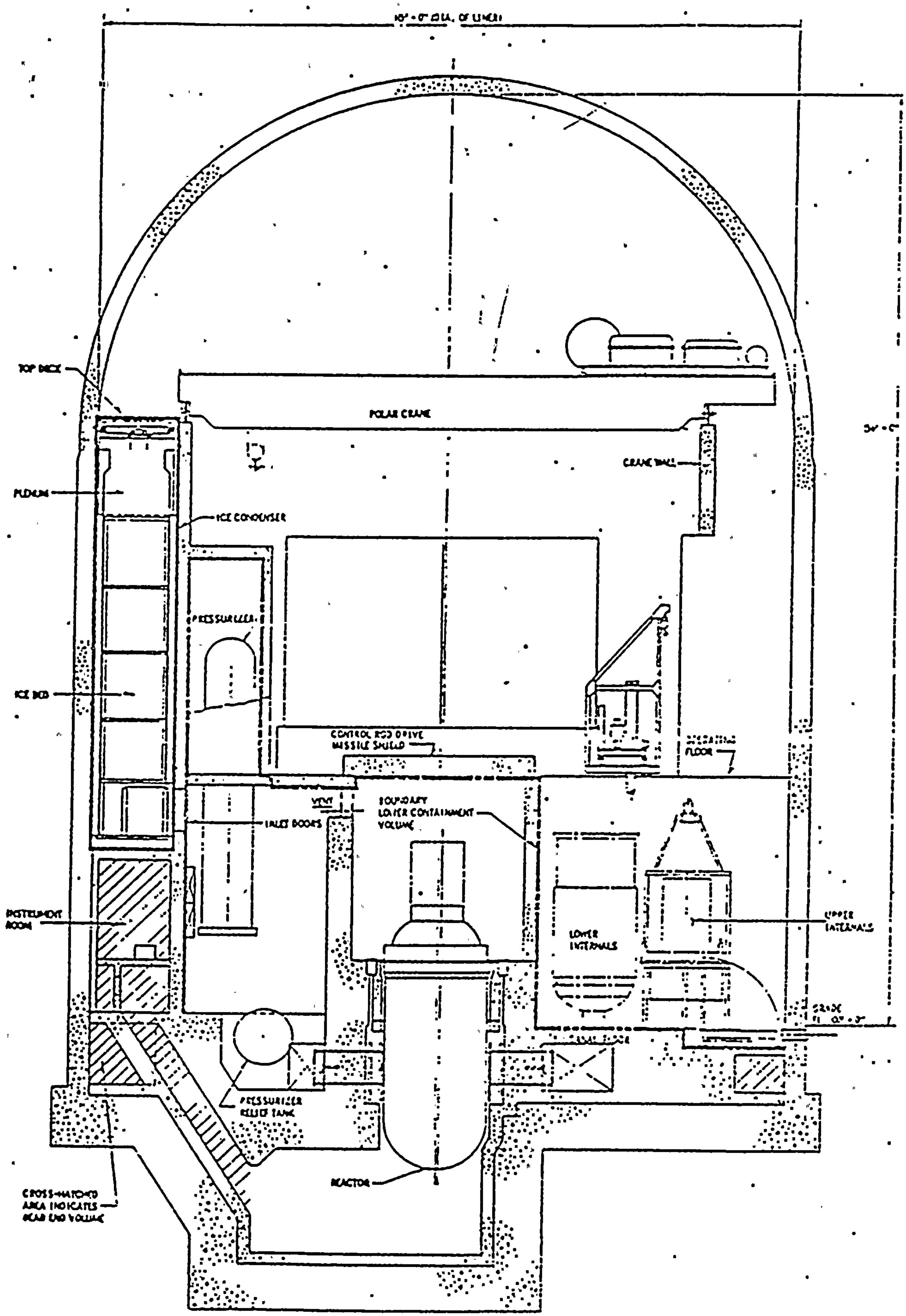

Figure 1-2   Sectional Elevation Plan "C-C"
Ice Condenser Containment

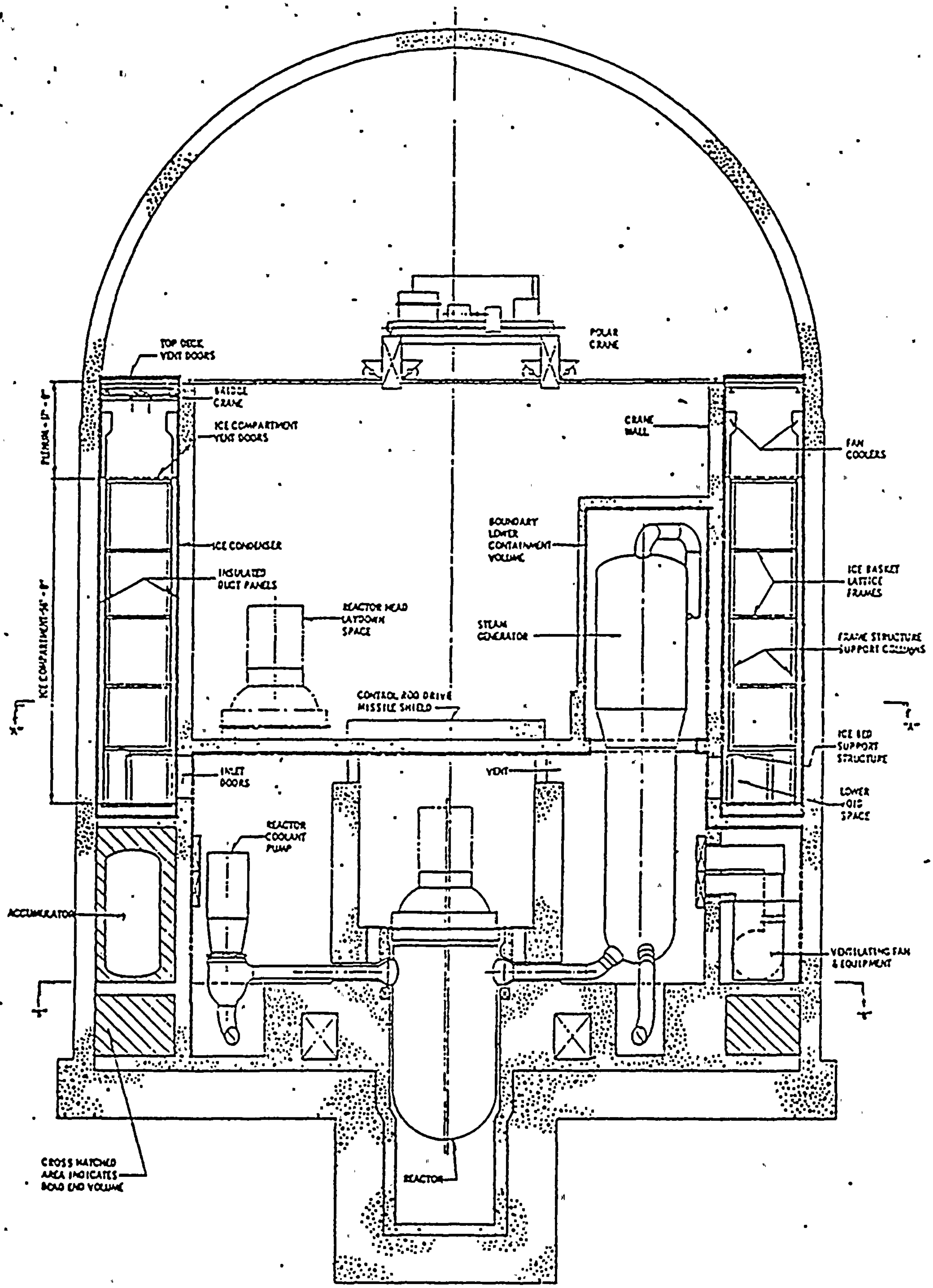

Figure 1-3 Sectional Elevation Plan "D-D"
Ice Condenser Containment

ATTACHMENT 2 TO AEP:NRC:0578I
DESCRIPTION OF SHIELDING CODE FOR CYLINDRICAL SOURCES
DONALD C. COOK NUCLEAR PLANT UNIT NOS. 1 AND 2

This attachment contains a copy of a user's manual for a computer program entitled "SHL1GG". SHL1GG was programmed, debugged, benchmarked, and utilized in the electric equipment environmental qualification program for the Donald C. Cook Nuclear Plant Unit Nos. 1 and 2 by personnel of the American Electric Power Service Corporation.

The user's manual describes in detail the basic models which SHL1GG uses in computing gamma and/or beta doses at an observer/detector located inside a cylindrical source, outside a cylindrical source, or in intervening shields (including an assumed pipe wall if a pipe containing recirculating sump fluid in the recirculation phase of a LOCA is the assumed cylindrical source). In general, SHL1GG may be used for applications in which the dose rate at an observer is desired and the geometry of the source is other than a cylinder. More specifically, for the case of equipment radiation qualification calculations, the subcompartments of the ice condenser containment were modeled as cylindrical sources within which equipment (i.e., observer/detector) was located. Selections for subcompartment cyclindrical model dimensions (e.g., length or height and radius of an equivalent cylinder for each subcompartment) are described in more detail in Attachment 3 to this letter.

Although the user's manual correctly describes the SHL1GG program in use today, we have made minor modifications to the actual coding since the time the user's manual was issued. In particular, a calculational error was discovered in late 1982 with regard to the utilization of a Simpson's Rule approximation for the evaluation of the Sievert Integral (a factor in the dose rate equation). This has been corrected and taken into account in computing the beta attenuation factors presented in Attachment 3 to this submittal. The specified doses for equipment inside containment, as given in Attachment 3 to this submittal, are therefore slightly different than those presented in Attachments 4 and 5 to our letter No. AEP:NRC:0578B, dated June 11, 1982.

Furthermore, as noted in Attachment 4 to this letter, SHL1GG was not used in computing radiation doses for equipment items outside containment. Rather, a computer code entitled "NSLSHL3" was used in those calculations. NSLSHL3 is, in effect, an earlier version of SHL1GG, and thus utilized much of the same methodology. We note, however, that review of the earlier work on equipment outside containment has indicated that some of the NSLSHL3 output was misinterpreted in computing radiation doses (i.e., doses from a 14" outside diameter pipe were used for some equipment items, rather than the limiting doses from an 8.625" outside diameter pipe). The specified doses for some equipment items outside containment, as given in Attachment 4 to this submittal, are therefore higher than those presented in Attachments 4 and 5 to our letter No. AEP:NRC:0578B, dated June 11, 1982.

The radiation qualification doses for electric equipment important to safety (both inside and outside containment) are undergoing continual review for the Donald C. Cook Nuclear Plant.

Revision No. 0

Date: December 31, 1980

SHL1GG: A Shielding Code for Cylindrical Sources

Developed By: G. Garner 12/31/80

G. Garner Date

Verified By: V. P. Manno 12/31/80

V. P. Manno Date

Approved By: Castresana 2/9/81

J. I. Castresana Date

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SHLIGG: A Shielding Code for Cylindrical Sources

I. Abstract

SHL1GG calculates dose rate and time integrated dose due to a cylindrical pipe containing a gamma and/or beta source. The source is distributed in water, steam, a steam-water mixture, or air. Gamma dose is calculated inside the pipe, outside the pipe, or in the pipe wall. Beta dose is only calculated inside the pipe; beta dose beyond the pipe wall inner surface is expected to be small compared to the corresponding gamma dose.

For beta dose inside the pipe, SHL1GG decides whether the source may be treated as an infinite cloud for each beta particle. If so, infinite cloud results are applied. If not, SHL1GG divides the source into a number of prismatic elements. Each element is treated as a line source, and a dose rate is calculated. The individual dose rates are summed to produce a total dose rate for the entire source. Beta and gamma doses are tabulated separately.

For gammas, buildup in the source, pipe wall, shield, and surrounding air is accounted for.

II. Introduction

SHL1GG is a FORTRAN IV computer program for calculating dose rate due to a cylindrical pipe containing a gamma and/or beta source. The source is distributed in water, steam, a steam-water mixture, or air. Gamma dose is calculated inside the pipe, outside the pipe, or in the pipe wall. Beta dose is only calculated inside the pipe; beta dose beyond the pipe wall inner surface is expected to be small compared to corresponding gamma dose.

For gamma dose outside the pipe, a number of shields may be present between the source and observer. The shields must be parallel to the cylinder axis and perpendicular to the perpendicular line joining the observer and this axis. The shields must be large enough that the observer is fully shielded (ie., a line drawn from an arbitrary source point to the observer must pass through the shields).

In all cases, self-shielding of the source is considered. Shielding due to the pipe wall is considered where appropriate. Buildup of gamma flux in the source, and, if appropriate, in the pipe wall, shields, and surrounding air is accounted for using a Taylor buildup factor (Reference 3, p. 415).

For beta dose inside the pipe, SHL1GG decides whether the source may be treated as an infinite cloud. This decision is made for each individual beta source. If so, a dose for that particular beta is immediately calculated using infinite cloud formulation. If not, the source is divided into prismatic elements. Each element has sectors of two concentric cylinders and two radial planes forming its boundaries. The division is done uniformly; the user specifies the number of radial divisions and the number of angular divisions. Each prismatic element is treated as a line source. Gamma dose rates are calculated for individual elements for each energy and summed to obtain total gamma dose rate. Beta dose rate is calculated, where appropriate (ie., when dose inside the pipe is desired and the source may not be treated as an infinite cloud), in a similar manner for individual beta particles and summed (each beta particle has a given energy probability density, maximum energy, and average energy). Formulation for beta attenuation analogous to that for gamma attenuation is used as described in Reference 1. For both beta and gamma doses relaxation length is based on that portion of the perpendicular distance from the particular prismatic element to the observer passing through each material.

Dose rate is initially obtained at time zero. Dose rate at any later time step is obtained by decaying the dose rate due to each isotope at the previous time step by the amount the respective isotope decays in the time interval. In addition, time integrated dose is calculated at each time step. A simple exponential decay is assumed; dose due to product isotopes is neglected. According to Reference 2, this effect may be accounted for by multiplying the above doses by 1.3. Additional work has shown that this factor is valid for gamma dose, but not necessarily for beta dose (Reference 13).

III. Model

A. Geometry

The geometry for the observer/detector point inside the pipe, inside the pipe wall, and outside the pipe is shown in Figures 1 to 3 respectively. For the case of the observer/detector outside the pipe, shields may be present and are perpendicular to line CO. All remaining space outside the pipe is assumed to be filled with air.

The j^{th} prismatic element is shown, at location (R_j, θ_j). For a single beta or gamma energy E_k, assume this prismatic element has a line intensity $S_{L,jk}$ (in Ci/cm). From Reference 3, p. 348, the dose rate D_{jk} due to this element and energy is:

$$D_{jk} = B_{jk} \, (DF)_k \cdot \frac{S_{L,jk}}{4\pi a_j} \cdot K \cdot$$

$$\left[F(\theta_{2j}, b_{jk}) - F(\theta_{1j}, b_{jk})\right] \qquad (1)$$

where K is a conversion factor equal to 3.7 x 10^{10} dis/Ci-sec and:

FORM GE-8

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AMERICAN ELECTRIC POWER SERVICE CORP.
2 BROADWAY
NEW YORK

DATE 12/31/80 BY [illegible] CK. [illegible]
COMPANY
G.O.
PLANT

SUBJECT

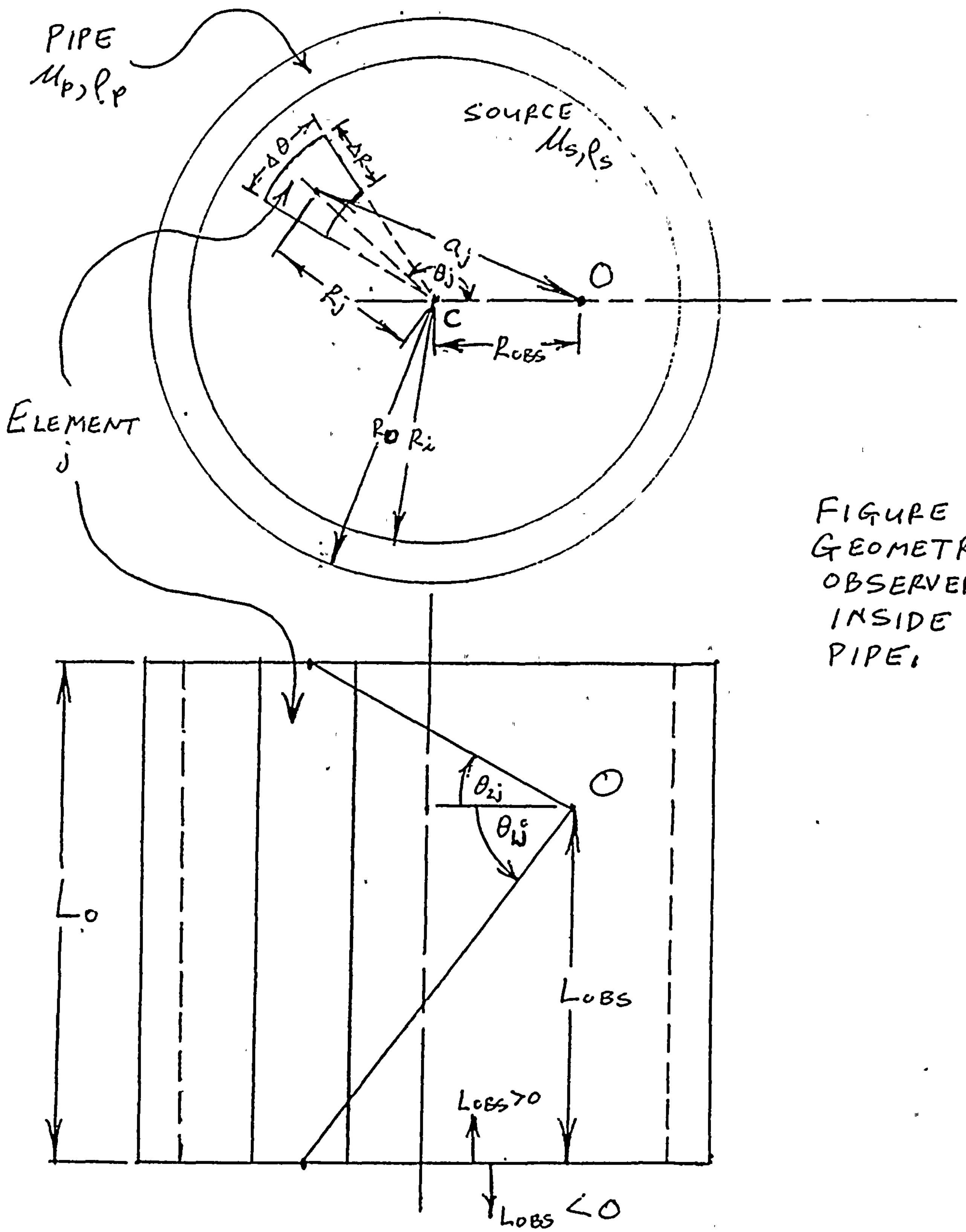

FIGURE 1. GEOMETRY — OBSERVER INSIDE PIPE.

FORM GE-6

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NEW YORK

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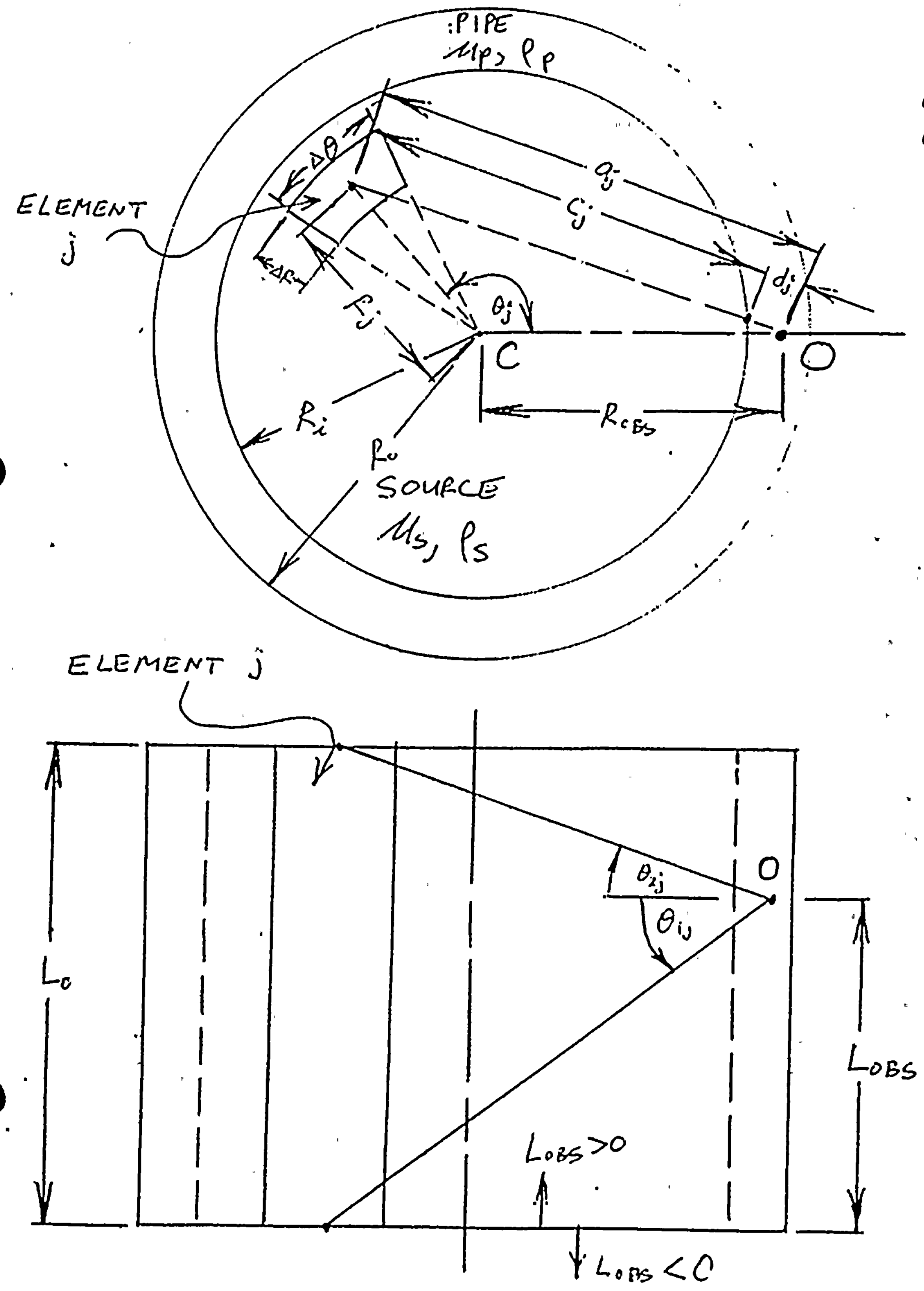
FIGURE 2, GEOMETRY— OBSERVER INSIDE PIPE WALL.



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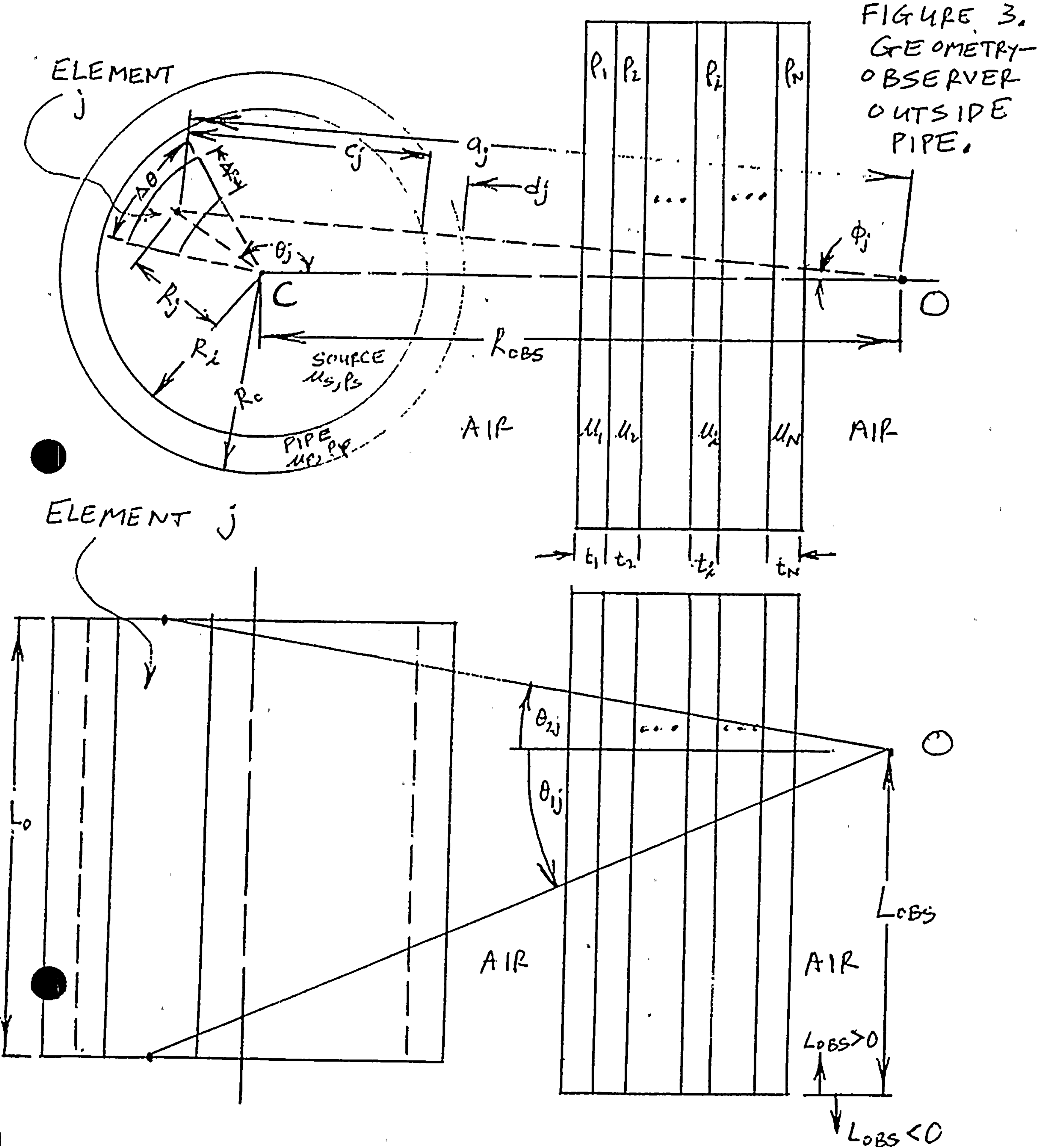

FIGURE 3. GEOMETRY— OBSERVER OUTSIDE PIPE.

b_{jk} = dimensionless distance from element j to observer, for energy E_k, expressed as a multiple of relaxation length in the various regions.

B_{jk} = buildup factor for energy E_k and dimensionless distance b_{jk}.

$(DF)_k$ = dose conversion factor at energy E_k in Rem-cm^2-sec/hr for gammas, and Rad-cm^{2}sec/hr for betas.

$F(\theta,b) = \int_0^{\theta} e^{-b \sec \theta'} d\theta'$.

The dimensionless distance b_{jk} is given by

$$b_{jk} = \mu_{s,k} a_j \qquad (2a)$$

observer inside pipe

$$b_{jk} = \mu_{s,k} c_j + \mu_{p,k} d_j \qquad (2b)$$

observer inside pipe wall

$$b_{jk} = \mu_{s,k} c_j + \mu_{p,k} d_j + \mu_{air,k} d_{air} + \sum_{i=1}^{N} \mu_{i,k} t_i \sec \phi_j \qquad (2c)$$

observer outside pipe (with N shields)

where: $\mu_{i,k}$ = linear attenuation coefficient of shield i at energy E_k in cm^{-1}

$\mu_{s,k}, \mu_{p,k}, \mu_{air,k}$ = linear attenuation coefficients of source, pipe, and surrounding air respectively, at energy E_k.

$d_{air,j}$ = distance traveled through surrounding air for observer outside pipe

$$= a_j - c_j - d_j - \sum_{i=1}^{N} t_i \sec \theta_j$$

Remaining quantities are defined in Figures 1-3.

If R_o, R_i, L_o, R_{OBS}, L_{OBS}, R_j, and θ_j are given, then a_j, c_j, d_j, θ_{1j}, θ_{2j} and ϕ_j may be calculated. In all cases, a_j, θ_{1j}, and θ_{2j} are given by:

$$a_j = \left(R_j^2 + R_{OBS}^2 - 2 R_j R_{OBS} \cos \theta_j \right)^{1/2} \tag{3}$$

$$\theta_{2j} = \tan^{-1} \frac{L_o - L_{OBS}}{a_j} \tag{4}$$

$$\theta_{1j} = \tan^{-1} \left[- \frac{L_{OBS}}{a_j} \right] \tag{5}$$

Note that $L_{OBS} > 0$ measured along the cylinder, and $L_{OBS} < 0$ measured antiparallel to the cylinder (see Figures 1-3). Then θ_{1j}, $\theta_{2j} > 0$ measured clockwise, and θ_{1j}, $\theta_{2j} < 0$ measured counter clockwise, as required (see Figures 1-3).

c_j is obtained by reference to Figure 4. The coordinates of E are $(R_j \cos \theta_j, R_j \sin \theta_j)$. The coordinates of H are given by the solution to

$$x^2 + y^2 = R_i^2 \tag{6a}$$

$$y = \frac{R_j \sin \theta_j}{R_j \cos \theta_j - R_{OBS}} (x - R_{OBS}) \tag{6b}$$

$$\equiv m(x - R_{OBS})$$

where

$$m = \frac{R_j \sin \theta_j}{R_j \cos \theta_j - R_{OBS}} \tag{6c}$$

The solution (x_H, y_H) is

$$x_H = \frac{m^2 R_{OBS} \pm \sqrt{R_i^2 (1+m^2) - m^2 R_{OBS}^2}}{1 + m^2} \tag{7a}$$

$$y_H = \pm \sqrt{R_i^2 - x_H^2} \tag{7b}$$

SUBJECT

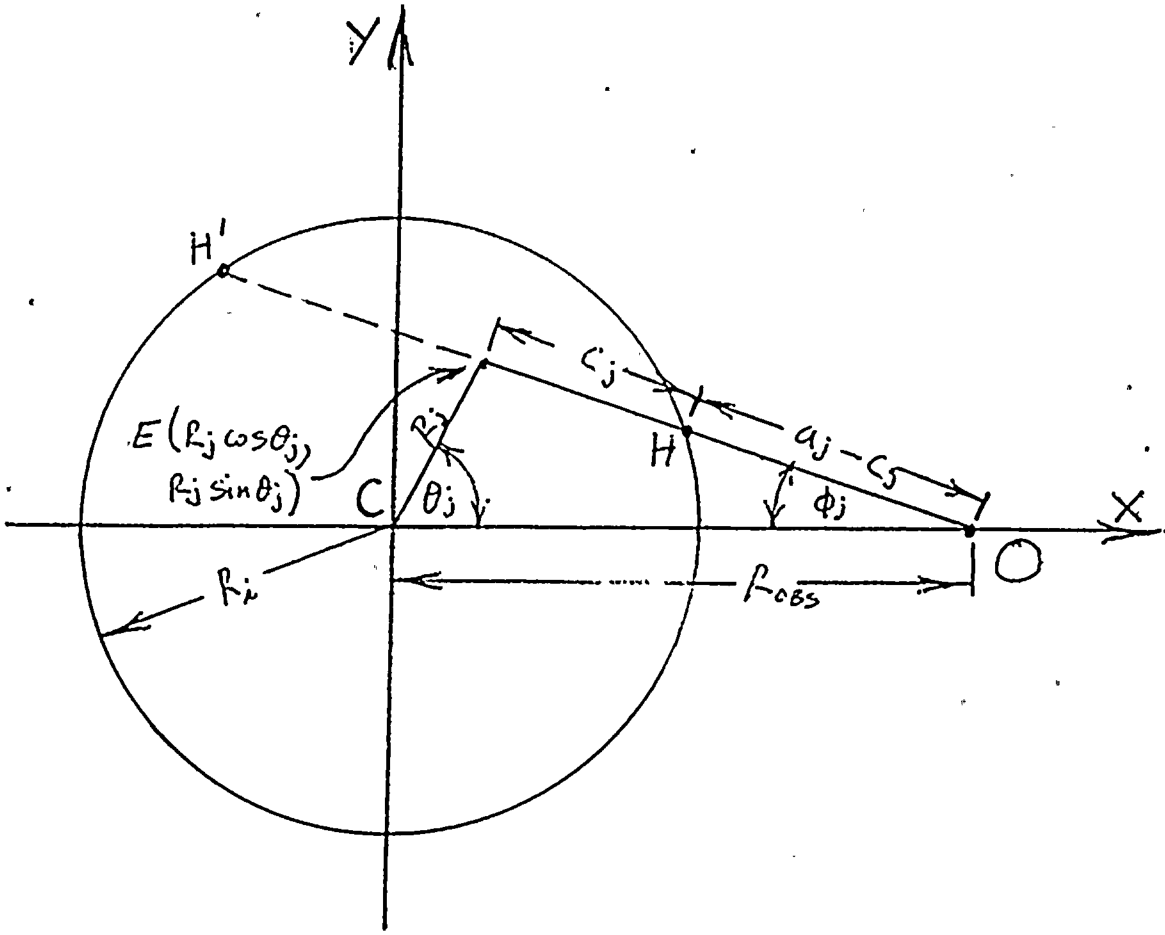

FIGURE 4. CALCULATION OF C_j.

In (7a), the positive sign is chosen, as the larger value of x_H is required (ie., the value corresponding to H, and not H', in Figure 4). In (7b), the positive sign is chosen for $0 \leq \theta_j \leq \pi$, and the negative sign is chosen for $\pi \leq \theta_j \leq 2\pi$. Then c_j is given by

$$c_j = \left[(x_H - R_j \cos \theta_j)^2 + (y_H - R_j \sin \theta_j)^2 \right]^{1/2} \quad (8).$$

To calculate d_j, the quantity $c_j + d_j$ is obtained by replacing R_j by R_o in Figure 4 and equations (6) - (8). Knowing c_j and $c_j + d_j$, d_j is obtained.

To obtain ϕ_j, it is seen from Figure 4

$$R_j \sin \theta_j = a_j \sin \phi_j \quad (9a)$$

$$\sin \phi_j = \frac{R_j}{a_j} \sin \theta_j \quad (9b)$$

$$\cos \phi_j = \left[1 - \left(\frac{R_j}{a_j} \sin \theta_j \right)^2 \right]^{1/2} \quad (9c)$$

$$\sec \phi_j = \left[1 - \left(\frac{R_j}{a_j} \sin \theta_j \right)^2 \right]^{-1/2} \quad (9d).$$

B. Gamma Dose

Gamma dose is calculated for each gamma energy in accordance with equation (1). Linear attenuation coefficients for various materials as a function of energy are given in Reference 4, p. 82.

Buildup factors for energy E_k are

$$B_{jk} = B_{air}\, B_{s,jk}\, B_{p,jk} \prod_{i=1}^{N} B_{ijk} \tag{10a}$$

outside pipe

$$B_{jk} = B_{s,jk}\, B_{p,jk} \tag{10b}$$

inside pipe wall

$$B_{jk} = B_{s,jk} \tag{10c}$$

inside pipe .

where:

$B_{s,jk}$ = buildup factor in source (11a)

$$\doteq \begin{cases} A_k e^{\alpha_k \mu_{s,k} c_j} + (1 - A_k) e^{\beta_k \mu_{s,k} c_j}, & \text{observer outside pipe or inside pipe wall} \\ A_k e^{\alpha_k \mu_{s,k} a_j} + (1 - A_k) e^{\beta_k \mu_{s,k} a_j} & \text{(11b) observer inside pipe} \end{cases}$$

$B_{p,jk}$ = buildup factor in pipe wall (11c)

$$= A_k e^{\alpha_k \mu_{p,k} d_j} + (1 - A_k) e^{\beta_k \mu_{p,k} d_j}$$

observer inside pipe wall or outside pipe

$B_{ijk} =$ buildup factor in shield i

$$= A_k e^{\alpha_k \mu_{ik} t_i \sec \phi_j} + (1-A_k) e^{\beta_k \mu_{ik} t_i \sec \phi_j} \qquad (11d)$$

observer outside pipe, and shields are present.

A_k, α_k, and β_k, for various materials as functions of energy, are given in Reference 3, pp. 416-423.

B_{air} is approximated as 1.05 (this may be verified to be conservative for most problems of interest using Reference 4, p. 527). For steam, B is also taken to be 1.05. For steam-water mixtures, B is calculated for steam and for water and is weighted by quality.

Dose conversion factor as a function of energy is given in Reference 3, p.19 (final dose is in R/hr). Curve fits to this figure for various energy ranges are used in SHL1GG.

C. Beta Dose

Beta dose is calculated inside the pipe only; beta dose beyond the pipe wall inner surface is expected to be small compared to corresponding gamma dose.

Beta dose calculation may be formulated analogously to gamma dose calculation (Reference 1, pp. 625-629). This is due to the particular analytical forms of beta spectra and electron scattering and absorption cross sections.

Beta dose is calculated as in equation (1). Linear attenuation coefficient is (Reference 1, p. 628)

$$\frac{\mu}{\rho} = \frac{17}{E_{max}^{1.14}} \tag{12}$$

where E_{max} is the maximum possible energy for the given beta particle, and ρ is the material density. Average beta energy (for a given beta particle) is approximated as one-third maximum beta energy (Reference 1, p. 540). Then

$$\frac{\mu}{\rho} = \frac{4.859}{E_{av}^{1.14}} \tag{13}$$

In this formualtion, there is no buildup of beta flux. Buildup factor in equation (1) is ignored.

Dose conversion factor is given by

$$(DF)_k = \left(\frac{\mu}{\rho}\right) E_k \tag{14}$$

where $\frac{\mu}{\rho}$ for beta attenuation is evaluated using (13) with average energy E_k. This is conservative, as this $\frac{\mu}{\rho}$ is for total attenuation (absorption plus scattering); actually only $\frac{\mu}{\rho}$ for absorption should be used. $\frac{\mu}{\rho}$ for absorption is not available. To obtain beta dose in Rads/hr

$$(DF)_k = \frac{\mu}{\rho} E_k \left[\frac{\text{Mev-cm}^2}{\text{g}}\right] \left[\frac{1.602 \times 10^{-6}\ \text{erg}}{1\ \text{Mev}}\right] \cdot \left[\frac{1\ \text{Rad}}{100\ \text{erg/g}}\right] \left[\frac{3600\ \text{sec}}{1\ \text{hr}}\right]$$

$$= 5.767 \times 10^{-5} \frac{\mu}{\rho} E_k \ \frac{\text{Rad-cm}^2\text{-sec}}{\text{hr}} \tag{15}$$

For a given beta average energy, if infinite cloud assumptions apply, dose is given by (Reference 5)

$$D_{K,air} = 0.457\, E_K\, S_{v,l} \tag{16}$$

instead of equation (1).

In equation (16),

$D_{k,air}$ = air dose rate in Rads/sec

E_k = beta particle average energy in MeV

$S_{v,1}$ = activity concentration in Ci/m^3

For an arbitrary infinite cloud source medium

$$D_K = D_{K,air} \frac{\rho_{air}}{\rho_s} \tag{17}$$

where:

ρ_{air} = density of air = 1.293×10^{-3} g/cm^3 at standard conditions (Reference 3, p. 19)

ρ_s = density of source in g/cm^3

D_K = dose in source in Rads/sec.

Inserting ρ_{air} and altering the units so that D_k is in Rads/hr and $S_{v,1}$ is in Ci/cm^3 produces

$$D_K = 2.127 \times 10^6 \frac{E_K S_{v,l}}{\rho_s} \text{ Rads/hr} \tag{18}$$

The source may be treated as an infinite cloud for a particular beta energy if all beta particles of that energy produced at the observer point are effectively absorbed in the source. Relative to the observer, the source looks like an

infinite medium. The infinite cloud model as used here has meaning only for the observer point inside the source.

The criterion for treating the source as an infinite cloud is that a beam of beta particles of the respective average energy emitted at the observer point must be reduced to 1% of its initial intensity over the shortest distance between the observer and the pipe wall. In this case contributions to the beta dose at the observer point due to source points beyond this distance are negligible.

The shortest distance between the observer and the pipe wall is

$$RTS = R_i - R_{OBS} \tag{19}$$

Then the above criterion may be stated

$$\exp[-\mu_s(RTS)] \leq 0.01 \tag{20}$$

where μ_s = linear attenuation coefficient for betas as given in equation (13). This gives

$$RTS \geq \frac{4.605}{\mu_s} \tag{21}$$

When (21) is satisfied, the source may be treated as an infinite cloud for betas of the particular energy, and (18) may be used.

D. Source Strength

The line intensity in equation (1) is obtained from

$$S_{L,jk} = S_{V,\ell} f_{\ell k} \, \Delta A_j \tag{22}$$

where:

$S_{v,l}$ = activity density of isotope l (in Ci/cm^3)

f_{lk} = fraction of activity of isotope l comprising gamma energy or average beta energy E_k

A_j = cross-sectional area of element j = $R_j \Delta R \Delta \theta$

Activity density is obtained from

$$S_{v,l} = \frac{S_{T,l} f_l}{V} \tag{23}$$

where:

$S_{t,l}$ = total activity of isotope l (in Ci)

f_l = fraction of this total activity released

V = volume into which this activity is released.

Equation (1) is summed over elements (j) and energies (k) to get total dose rate. Note that in the output, those values labeled activities (both activities for each isotope and total activity for all isotopes) do not incorporate the fractions f_l. Those values labeled activity concentrations do incorporate the fractions f_l.

E. Decay of Isotopes and Integrated Dose

Simple exponential decay is assumed; dose due to product isotope is neglected. According to Reference 2, this effect may be accounted for by multiplying the obtained doses by 1.3. It is shown in Reference 13 that this is valid for gamma dose, but not necessarily for beta dose.

For each isotope, a decay constant must be input. Activity, activity concentration, gamma dose for a given energy, and beta dose for given energy at time t + Δt are obtained from corresponding quantities at time t via

$$X(t+\Delta t) = X(t)\, e^{-c\Delta t} \tag{24}$$

where c is the decay constant, and X is any of the above four quantities. Integrated beta or gamma dose for a given energy at time t + Δt is obtained from

$$\begin{aligned} Y(t+\Delta t) &= Y(t) + \int_0^{\Delta t} X(t)\, e^{-ct'}\, dt' \\ &= Y(t) + X(t)\left[\left(\tfrac{1}{c}\right)\left(1 - e^{-c\Delta t}\right)\right] \end{aligned} \tag{25}$$

where Y is integrated dose and X is dose rate for the respective gamma or beta energy.

IV. Code Description and Options

SHL1GG consists of a main program and the 11 subroutines PD1, PD2, PD3, DSR, F, GXP, BLDPF, ABSRP, SNTP, BLDUP, and DRV (in addition, there is a block data subroutine that initializes common block B3; this block is common to MAIN, PD1, PD2 and PD3). The main program reads and prints input, calculates decay of isotopes and dose rates over time, calculates integrated dose, and summarizes individual and total doses at each time by energy or isotope as desired by the user.

PD1 calculates gamma dose rate outside the pipe or inside the pipe wall at time zero for each gamma energy. PD2 calculates gamma dose rate inside the pipe at time zero for each gamma energy. PD3 calculates beta dose rate inside the pipe at time zero for each beta average energy. DSR(EN)

evaluates gamma dose conversion factor for energy EN. For this calculation, curve fits to various portions of Figure 2.1 of Reference 3 are used. F(THETA,B) evaluates the Sievert integral function $F(\theta,b)$. BLDPF(MATN,EO,Y, YL,YT) and BLDUP(MATN,B,Y,YL,YT,X) calculate buildup factor for material MATN, dimensionless length B, energy EO and quality X (in the case of steam-water mixtures). Y, YL, and YT are the quantities A_k, α_k, and β_k respectively in equation (11); these depend only on energy (and not on B). The calculation of buildup factor is divided into a portion which depends only on energy (BLDPF) and a portion which depends on distance traveled and energy (BLDUP). Execution time is reduced by calling BLDPF only once for each gamma energy. This portion of the calculation need not be done for each source element. BLDUP must be called once for each gamma energy and source element. ABSRP(MATN,EO) evaluates gamma $\left(\frac{\mu}{\rho}\right)_{total}$ for material MATN and energy EO. DRV(AIS,DTT,NISOT,FCTD) evaluates fractions by which istotpes decay over the given time step (ie., evaluates $e^{-\lambda_i \Delta t}$, i = 1, 2, ..., NISOT), where

DTT = time step size

AIS = array containing isotope information

AIS(8,J) = λ_J = decay constant for isotope J

NISOT = number of isotopes

FCTD(J) = $e^{-\lambda_J * DTT}$ (J = 1,NISOT)

SNTP(X,ARG,VAL,Y,NDIM) does linear interpolation. ARG(NDIM) and VAL(NDIM) are arrays of the independent and dependent variable respectively. X is the argument for which the output Y is desired. Y is obtained by linear interpolation. GXP evaluates the function $e^{-(b/\cos c)}$. If $\frac{b}{\cos c}$ is large enough to cause an underflow, GXP is set to zero.

The source, pipe, and shield materials are limited to water, air, iron (represents all steels); concrete, and lead. Water includes steam and steam-water mixtures; quality must be input (steam, x = 1; water, x = 0; steam-water mixture, $0 < x < 1$). When dose outside the pipe is calculated, the pipe is assumed to be surrounded by air (except at shield locations). This air need not have the same density as a possible air source material.

At each time step, the code will print out dose rate and cumulative dose for each individual beta average energy and gamma energy, totals for each isotope, or totals for all isotopes. A different option may be in effect in each print interval. A print interval is defined as a period of time with constant time step size and print option. There may be many print intervals, each with its own time step size and print option. At each time step, total activity and activity concentration, total dose rate and integrated dose, and ratio of activity concentration to dose rate are always printed. Beta and gamma doses are always printed spearately, as gamma doses are given in R/HR and beta doses are given in RADS/HR. When dose is calculated inside the pipe, the user may specify beta dose is to be evaluated. Beta dose is calculated inside the pipe only.

The code may be run under two options. With one option a single case is run, while with the other option a number of cases are run. A case is defined as a single problem with a specified set of physical and geometrical parameters. For the former option, any number of shields may be present. For the latter option, a single shield is present, and the shield thickness is incremented by a given amount for each case. The initial thickness may be zero (if no shield is present in the first case). In all cases, the user must insure that the shield is thin enough to fit between the pipe and the observer (if not, an error message is printed out and execution stops).

All calculations are done in double precision.

V. Input Description

A. Input Requirements

1) Card 1 (2 cards) (10A8)

   These cards contain information identifying the source reference of beta and gamma energies and decay fractions. This information is printed.

2) Card 2 (F10.5)

   SYSVOM

   SYSVOM = system volume in cm^3

3) Card 3 (2A8)

   SCMAT, PIPMAT

   SCMAT = source material name

   PIPMAT = pipe material name

   Names are limited to:

   WATER (includes steam-water mixture), IRON, CONCRETE, AIR, and must be left justified. Any other input will produce an error message, and execution will stop.

4) Card 4 (3F10.5)

   RO, RI, LO

   RO = pipe outer radius (cm)

   RI = pipe inner radius (cm)

   LO = pipe length (cm)

5) Card 5 (2I5)

   NISOT, BTOPT

   NISOT = number of isotopes comprising source (maximum of 100)

   BTOPT = beta option parameter

   $$= \begin{cases} 1 & \text{calculate beta dose} \\ 0 & \text{don't calculate beta dose} \end{cases}$$

.6) Card 5 (NISOT Cards)

Each Card:

i) Isotope name, activity (Ci) (total activity for this isotope), number of different gamma energies emitted, fraction of total activity of this isotope released into volume SYSVOM, decay constant (sec^{-1}), number of different average beta energies emitted - A8, F12.3, I5, 2F12.3, I5.

Isotope name is a character string identifying the isotope; it should be left justified.

ii) Gamma energy (MeV), fraction of gammas emitted with this energy, gamma energy, fraction of gammas emitted with this energy, ... 8F10.3. There are as many entries as different gamma energies for this isotope; information should be continued onto as many cards as required. The total number of gamma energies for all isotopes must be less than or equal to 1000.

iii) Beta average energy (MeV), fraction of betas emitted with this average energy, beta average energy, fraction ... 8F10.3. There are as many entries as different beta energies for this isotope; information should be continued onto as many cards as required. The total number of beta energies for all isotopes must be less than or equal to 500.

7) Card 7 (2I5)

NSHLD, SHLDOP

NSHLD = number of shields (20 maximum; may be zero)

SHLDOP = shield option

SHLDOP = 0 1 case, possibly many shields

SHLDOP = 1 possibly many cases, 1 shield.

For this option, NSHLD must be 1.

8) Card 8 - NSHLD cards (A8,2F10.3)

Included only if NSHLD > 0

Shield material name, shield thickness (cm), shield density (g/cm^3)

Shield material name must be chosen from the names given in Card Set 1 description, and must be left justified. If SHLDOP = 1, shield thickness is the initial thickness (case 1).

9) Card 9 - (I5, F10.3)

Included only if SHLDOP = 1

NOPT, SINC

NOPT = number of cases

SINC = shield thickness increment (cm)

10) Card 10 (4F10.3)

SCDEN, X, PIPDEN, AIRDEN

SCDEN = source material density (g/cm^3)

X = source quality (ignored unless source is water; $0 \leq X \leq 1.0$)

PIPDEN = pipe material density (g/cm^3)

AIRDEN = density of air surrounding pipe (g/cm^3)

Note that quality is specified only for the source.

If water is used as a shield, it must be liquid.

11) Card 11 - (2I5)

NRAD, NANGL

NRAD = number of radial divisions

NANGL = number of angular divisions

12) Card 12 - (2F10.3)

ROBS, LOBS

ROBS = perpendicular distance from source centerline to observer (cm)

LOBS = distance along source from one end of source to observer (cm). LOBS may be measured from either end of the source, as the entire system is symmetric with respect to the source midplane. LOBS > 0 indicates LOBS is measured parallel to the source, while LOBS < 0 indicates LOBS is measured antiparrallel to the source (see Figures 1-3).

13) Card 13 - NPRNT+1 cards

i) NPRNT I5

NPRNT = number of print intervals (20 maximum)

ii) NPRNT cards, one for each print interval - giving endtime for each print interval (sec), number of steps in each print interval, print option F10.3, 2I5

Print Option =
- 0 print doses for each energy
- 1 print only totals
- 2 print totals for each isotope

B. Criteria for Choosing NRAD and NANGL

NRAD and NANGL must be chosen such that, for the lowest energy gamma and beta, there is not appreciable attenuation over the extent of any element. This is because each element is treated as a line source concentrated at the element center and shielded by the element source material. If there is appreciable attenuation, lumping will cause radiation from portions of the element closest to the observer to be attenuated more than it should be. Resulting doses calculated will be non-conservative.

The largest elements will be at the periphery. They will be of extent

$$\Delta R = \frac{R_i}{NRAD} \tag{26a}$$

$$\Delta S = R_i \Delta\theta = \frac{2\pi R_i}{NANGL} \tag{26b}$$

where: Δs = arc length subtended by element mid-section and other quantities are defined in Figures 1-3. For attenuation to be small over the extent of an element

$$\mu_{max} \Delta R \ll 1 \tag{27a}$$

$$\mu_{max} \Delta S \ll 1 \tag{27b}$$

where: μ_{max} is the largest attenuation coefficient, corresponding to the smallest gamma and/or beta energy. Substituting (26) into (27) gives

$$NRAD \gg b_{max} \qquad (28a)$$

$$NANGL \gg 2\pi\, b_{max} \qquad (28b)$$

where $b_{max} = R_i \mu_{max}$

When beta doses are calculated, μ_{max} for betas will generally be the controlling factor, because beta attenuation is much greater than gamma attenuation. However, in many cases beta attenuation will be so large that infinite cloud results will apply; then gamma attentuation will be the controlling factor. The user should choose NRAD and NANGL based on (28) and the particular beta and gamma energies input. It is sufficient that the inequalities be satisfied by a factor of 10.

It is also necessary that the element size be small compared to the distance from the element to the observer. This requirement is significant when an observer is close to a large source.

As an example, assume a water source and 0.1 MeV smallest energy gamma. This is the lowest gamma energy the code will distinguish; $\frac{\mu}{\rho}$ is taken as a constant equal to $\frac{\mu}{\rho}$ at 0.1 MeV (=0.167 cm^{-1}) for lower engeries. This is conservative. Assume beta dose is not calculated. Then

$$\mu_{max} = 0.167 \ cm^{-1}$$

$$b_{max} = 0.167\ R_i$$

where R_i is in cm. Therefore

$NRAD \gg 0.167\ R_i$

$NANGL \gg 1.05\ R_i$

While satisfying (28) is sufficient for calculated doses to be conservative, it is not necessary. In particular, for the observer at the center of the pipe, NANGL can be 1 with no loss of accuracy, due to symmetry. For the observer slightly off center, (28b) need not be satisfied. For doses outside the pipe or in the pipe wall, (28a) and (28b) should both be satisfied.

C. Typical Densities

Typical densities for the various source, pipe, and shield materials are

Water (x=o)	1.0 g/cm^3
Air	1.293 $x10^{-3}$ g/cm^3
Iron	7.83 g/cm^3
Lead	11.3 g/cm^3

VI. Required JCL

SHL1GG source is stored in member SHL1GG of library SRCENSL. To run SHL1GG, the following JCL may be used:

```
JOB CARD
//STEP1 EXEC SRCE,LIB=NSL
//SRCE.DATA DD *
=M NSLKCHR
      VARIOUS =R,  =D,=A CARDS
      AND NEW DATA
```

```
/*
//STEP2 EXEC SSFLG,C=C,SRC=NSL
//GØ.SRCE DD *
=M SHL1GG
/*
//FT05F001 DD DSN=*.STEP1.SRCE.SYSUT2,DISP=(ØLD,DELETE)
//
```

Note: Ø = letter O

0 = zero

The member NSLKCHR contains SHL1GG input. A listing of this member is attached as Run 11. The isotopes and activities are taken from Reference 10, Table 5-29, and Reference 11, Appendix A, Table A.2-1. For Reference 10 activities, a U mass of 8.86×10^4 Kg is used (see Section VIII and Reference 11, Chapter 3 (Unit 1), Table 3.2.3-1). In cases where the two references conflict, Reference 10 activities are used. Decay constants, gamma and beta energies, and fractions of total isotope activity comprising each energy are given in Reference 12 (this is indicated in the first two input lines). It is assumed that 100% of the noble gases, 50% of the iodines, and 1% of the particulates are released into volume SYSVOM.

These items constitute the bulk of the input. With the exception of the fractional releases, these items generally do not change from run to run. Remaining input items (lines 100-600 and 37200-37700 of NSLKCHR) must be input by the user for the desired case(s). This may be accomplished by modifying NSLKCHR as necessary via =R, =A, and =D cards.

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VII. Test Cases

A number of test cases were run and compared with hand calculations. An attempt was made to cover as many code options as possible. Table 1 summarizes the cases, options, and runs. Run numbers are indicated on the printouts attached to this calculation.

In all cases the following input is assumed:

pipe material : IRON (steel)
pipe material density : $\rho_p = 7.83$ g/cm^3
pipe inner radius : $R_i = 10.250$ cm $= 4.035$ in
pipe outer radius : $R_o = 10.960$ cm $= 4.315$ in
pipe length : $L_o = 335.2$ cm $= 10.997$ ft
source material : WATER
source material density : $\rho_s = 1.00$ g/cm^3
source quality : $x = 0$
system volume : $V = 3.5679\times10^8$ cm$^3 = 1.26\times10^4$ ft^3
density of surrounding air : $\rho_{air} = 1.293\times10^{-3}$ g/cm^3
observer axial location : $L_{OBS} = 167.6$ cm $= 5.5$ ft

number of print intervals $= 1$
printout every 15 minutes for 1 hour

number of isotopes $= 2$
number of gamma energies $= 4$
number of beta energies $= 2$

Information on isotopes and energies used is in ~~Table~~ Table 2. Remaining parameters are given

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TABLE 1. SUMMARY OF TEST CASES RUN AND OPTIONS USED.

CASE	OPTIONS	RUN NUMBER	REMAINING PARAMETERS
1) DOSE INSIDE PIPE	NO SHIELDS { NSHLD=0, SHLDOP=0 } BETA DOSE CALCULATED BTOPT=1 PRINT OPTION=0	1	ROBS=0. NRAD=90 NANGL=1
		2	ROBS=0. NRAD=1000. NANGL=1
		3	ROBS=0. NRAD=5 NANGL=1
	NO SHIELDS { NSHLD=0, SHLDOP=0 } BETA DOSE CALCULATED BTOPT=1 PRINT OPTION=1	4	ROBS=0. NRAD=90 NANGL=1
	NO SHIELDS { NSHLD=0, SHLDOP=0 } BETA DOSE CALCULATED BTOPT=1 PRINT OPTION=2	5	ROBS=0. NRAD=90 NANGL=1
2) DOSE OUTSIDE PIPE	NSHLD=2 SHLDOP=0 BTOPT=0 PRINT OPTION=0	6	ROBS=22.960 cm =9.039 in NRAD=5 NANGL=18
DOSE OUTSIDE PIPE	NSHLD=1 SHLDOP=1 NUMBER OF CASES=2. BTOPT=0 PRINT OPTION=0	7	ROBS=16.960 cm =6.677 IN NRAD=5 NANGL=18

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TABLE 2. SUMMARY OF ISOTOPE INFORMATION

ISOTOPE	ACTIVITY (Ci)	FRACT. OF ACTIVITY RELEASED INTO SYSTEM VOLUME	DECAY CONSTANT (sec^{-1})	GAMMA ENERGY (MeV)	FRACTION OF ACTIVITY FOR GAMMA ENERGY	BETA ENERGY (MeV)	FRACT. OF ACTIVITY FOR BETA ENERGY
Kr-85m	4.3×10^7	1.0	4.376×10^{-5}	0.305 0.15	0.1 0.77	0.273	1.0
Nb-97	1.68×10^8	0.01	1.605×10^{-4}	0.665 1.00	1.0 0.03	0.423	1.0

in Table 1. Note that for case 1 only 1 angular division is required (due to symmetry), as the observer is at the pipe center.

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A. Case 1 Dose Calculation – Time Zero.

The uncollided flux at the centerline of a cylinder is (reference 3, p. 403):

$$\phi = \frac{S_V}{2\mu_s}\left[G(\mu_s h_1, b) + G(\mu_s h_2, b)\right]$$

where: μ_s = linear attenuation coefficient of source
$b = \mu_s R_i$
S_V = activity concentration
h_1, h_2 = distances from observer to cylinder ends
$G(x, b)$ = function tabulated in reference 3, pp. 366–367.

From above:

$$S_V = \frac{(\text{activity})(\text{fraction})}{\text{volume}}$$

Kr-85m :
$$S_V = \frac{(4.3\times10^7\ Ci)(1.0)}{3.5679\times10^8\ cm^3} = 0.1205\ Ci/cm^3$$
$$= 4.459\times10^9\ \frac{\text{DISINT}}{\text{SEC-cm}^3}$$

Nb-97
$$S_V = \frac{(1.68\times10^8\ Ci)(0.01)}{3.5679\times10^8\ cm^3} = 4.710\times10^{-3}\ Ci/cm^3$$
$$= 1.743\times10^8\ \frac{\text{DISINT}}{\text{SEC-cm}^3}$$

i) $E_\gamma = 0.305$ MeV . Kr-85m

from reference 3, p. 448 : $\left(\frac{\mu}{\rho}\right)_s = 0.1175\ \frac{cm^2}{g}$

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$$\therefore \mu_s = 0.1175 \ cm^{-1}$$

$$b = \mu_s R_i = (0.1175 \ cm^{-1})(10.250 \ cm) = 1.204$$

$$h_1 = h_2 = 167.6 \ cm$$

$$\therefore \mu_s h_1 = \mu_s h_2 = (.1175)(167.6) = 19.693$$

$$G(\mu_s h, b) = 0.795$$

$$\phi = \frac{4.459 \times 10^9 \ sec^{-1} \cdot cm^{-3}}{2 \ (0.1175 \ cm^{-1})} \left[2(0.795) \right] (0.1)$$

FRACT. OF Kr-85m ACTIVITY GOING INTO THIS γ

$$= 3.017 \times 10^9 \ cm^{-2} \ sec^{-1}$$

dose rate:

$$D_\gamma = (DF) B \phi$$

where: DF = dose conversion factor
B = buildup factor

from reference 3, pg. 419 (and from values used as input to code in subroutine BLDPF (see data statements))

$E = 0.305$ MeV
$A_1 = 29.85$ (by extrapolation)
$\alpha_1 = 0.171$ (by extrapolation)
$\alpha_2 = 0.0012$ (by extrapolation)

$$\therefore B \cong A_1 e^{\alpha_1 \mu x} + (1 - A_1) e^{-\alpha_2 \mu x}$$

Take $\mu x = \mu \bar{R}$, where $\bar{R}$ = volume weighted average radius $= \frac{1}{A}\int r\,dA$ (A = c.s. area). This will account for more source elements at larger distances from the center.

$$\bar{R} = \frac{1}{\pi R_i^2}\int_0^{R_i} r \cdot 2\pi r\,dr = \frac{1}{\pi R_i^2}\cdot 2\pi \frac{R_i^3}{3}$$

$$= \frac{2}{3} R_i$$

$$\therefore \mu x = \frac{2}{3} b = 0.8027$$

$$B = 29.85 e^{0.171(.8027)} - 28.85 e^{-(0.0012)(0.8027)}$$

$$= 5.42$$

from reference 3, p. 19:

$$DF = 6\times10^{-7}\ \frac{R/Hr}{cm^{-2}\ sec^{-1}}$$

$$\therefore D_\gamma = (6\times10^{-7})(5.42)(3.017\times10^{9}) = 9810\ R/Hr$$

ii) $E_\gamma = 1$ MeV Nb-97

$$\left(\frac{\mu}{\rho}\right)_s = 0.071\ cm^2/g$$

$$\mu_s = 0.071\ cm^{-1}$$

$$b = (0.071)(10.250) = 0.7278$$

$$h_1 = h_2 = 167.6\ cm$$

$$\mu_s h_1 = \mu_s h_2 = 11.9$$

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$$\phi = \frac{1.743 \times 10^8 \text{ cm}^{-3}\text{sec}^{-1}}{2(0.071 \text{ cm}^{-1})}\left[2(.63)\right](.03)$$

FRACT. OF Nb-97 ACTIVITY GOING TO THIS γ

$$= 4.640 \times 10^7 \text{ cm}^{-2}\text{sec}^{-1}$$

$$D_\gamma = (DF)\, B\, \phi$$

$$A_1 = 11.$$
$$\alpha_1 = 0.104$$
$$\alpha_2 = 0.028$$

$$\mu_s x = \frac{2}{3} b = \frac{2}{3}(0.7277) = 0.4852$$

$$B = 11\, e^{(0.104)(.4852)} - 10\, e^{-(0.028)(.4852)}$$

$$= 1.704$$

$$DF = 1.9 \times 10^{-6} \frac{R/hr}{\text{cm}^{-2}\text{sec}^{-1}}$$

$$\therefore D_\gamma = (4.64 \times 10^7)(1.704)(1.9 \times 10^{-6})$$

$$= 150.2 \quad R/hr$$

iii) $E_\gamma = 0.665$ MeV Nb-97

$$\left(\frac{\mu}{\rho}\right)_s = 0.086 \text{ cm}^2/g$$

$$\mu_s = 0.086 \text{ cm}^{-1}$$

$$b = (.086)(10.250) = 0.8815$$

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$\therefore \quad G(\mu_s h, b) \doteq 0.695$

$$\phi = \frac{(1.743 \times 10^{8})}{2(.086)} \left[2(.695)\right](1.0)$$

$$= 1.409 \times 10^{9} \quad cm^{-2} sec^{-1}$$

$$D_\gamma = (DF) B \phi$$

$A_1 \doteq 19$

$d_1 = 0.123$

$d_2 = 0.004$

$$\mu_s x = \tfrac{2}{3} b = \tfrac{2}{3}(.8815) = 0.5877$$

$$B = 19 e^{(.123)(.5877)} - 18 e^{-(.004)(.5877)}$$

$$= 2.470$$

$$DF = 1.35 \times 10^{-6} \ \frac{R/hr}{cm^{-2} sec^{-1}}$$

$$D_\gamma = (1.409 \times 10^{9})(2.470)(1.35 \times 10^{-6})$$

$$= 4698 \quad R/hr$$

iv) $\bar{E}_\beta = 0.273$ MeV $\qquad$ Kr-85m

$$\left(\frac{\mu}{\rho}\right)_s = 4.85882 \, \bar{E}_\beta^{\,-1.14}$$

$$= 4.75882 (0.273)^{-1.14}$$
$$= 21.35 \quad cm^2/g$$

$$\therefore \mu_s = 21.35 \quad cm^{-1}$$

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$b = 21.35\ (10.25) = 218.8$

$\mu_s h_1 = \mu_s h_2 \doteq 3578$

$G = 1.0$

$$\therefore \phi = \frac{(4.459 \times 10^9)}{2\mu_s} (2G)$$

$$= \frac{4.459 \times 10^9}{\mu_s} \quad cm^{-2}\ sec^{-1}$$

dose.: $D_\beta = (DF)\phi$

$DF = 5.767 \times 10^{-5} \left(\frac{\mu}{\rho}\right)_s \bar{E}_\beta$ (eqn. 15)

$$\therefore D_\beta = \frac{(4.459 \times 10^9)(5.767 \times 10^{-5})(.273)}{\rho_s} \ \frac{Rad}{hr}$$

$$= 7.02 \times 10^4 \ Rads/hr$$

v) $\bar{E}_\beta = 0.423$ MeV. Nb-97

$\left(\frac{\mu}{\rho}\right)_s = 4.85882\ (.423)^{-1.14} = 12.96 \ \frac{cm^2}{g}$

$\mu_s = 12.96 \ cm^{-1}$

$b = 132.8$

$\therefore G = 1.0$

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$$DF. = 5.767\times10^{-5}\left(\frac{\mu}{\rho}\right)_s \bar{E}_\beta$$

$$\therefore D_\beta = \frac{1.743\times10^{8}}{\mu_s}\left(5.767\times10^{-5}\right)\left(\frac{\mu}{\rho}\right)_s (.423)$$

$$= 4252 \text{ rads/hr}$$

Comparison of these results with runs 1-3 shows good agreement. The hand calculated result for $E_\gamma = 0.305$ MeV is 23% above the result in run 2. In lumping the source (in the hand calculations) for purposes of calculating buildup factor, more buildup is assumed for those source elements closest to the pipe center, and less buildup is assumed for those source elements furthest from the pipe center. The former outweighs the latter because $d^2B/d(\mu_s x)^2 < 0$ for all $\mu_s x$ (B = buildup factor; $\mu_s x$ = radiation length).

Hand calculated results for higher gamma energies are slightly below results in run 2. This is due to conservatism in the evaluation of $F(\theta, b)$ for $b<1$ and $\theta > 60°$ (this case arises for elements close to the pipe center) (see Appendix A). This effect outweighs the above buildup factor effect for higher gamma energies, because in this case buildup is small.

Beta dose results agree to within $\frac{1}{2}$%. Beta energies are low enough that the code uses the infinite cloud model.

Comparison of runs 1-3 shows that little accuracy is gained in going from NRAD = 90 (run 1) to NRAD = 1000 (run 2). In many cases NRAD = 5 (run 3) may be adequate; these results differ from the NRAD = 1000 results by no more than 4%. All this is in accordance with section IV. B.:

$$\mu_{max} = \mu_s (0.15 \text{ MeV}) = 0.149 \text{ cm}^{-1}$$

$$b_{max} = (0.149 \text{ cm}^{-1})(10.250 \text{ cm}) = 1.527$$

$$\therefore \quad NRAD \gg 1.527$$

B. <u>Case 1 Dose Calculation — Verification of Isotope Decay and Case Summations and Summaries</u>

i) $E_\gamma = 0.305$ MeV   Kr-85m

decay constant : $C = 4.376 \times 10^{-5}$ sec^{-1}
time step : $\Delta t = 900$ sec

$$\exp(-C\Delta t) = \exp[-(4.376 \times 10^{-5})(900)] = 0.9614$$

$$\frac{1 - \exp(-C\Delta t)}{C} = 882.5 \text{ sec}$$

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from runs 1–7:

$$\frac{\text{Activity } (t = 900 \text{ sec})}{\text{Activity } (t = 0)} = \frac{4.134 \times 10^7 \text{ Ci}}{4.30 \times 10^7 \text{ Ci}} = 0.9614$$

$$\frac{\text{Activity } (t = 3600 \text{ sec})}{\text{Activity } (t = 1800 \text{ sec})} = \frac{3.673 \times 10^7 \text{ Ci}}{3.974 \times 10^7 \text{ Ci}} = 0.9243$$

$$= (0.9614)^2$$

$$\frac{\text{Activity Concentration } (t = 2700 \text{ sec})}{\text{Activity Concentration } (t = 0)} =$$

$$\frac{1.071 \times 10^{-1} \text{ Ci/cm}^3}{1.205 \times 10^{-1} \text{ Ci/cm}^3} = 0.8888 = (0.9614)^3$$

$$\frac{D_\gamma (t = 3600 \text{ sec})}{D_\gamma (t = 2700 \text{ sec})} = \frac{6824 \text{ R/hr}}{7098 \text{ R/hr}} = 0.9614$$

$$\frac{D_{\gamma,\text{integ}} (t = 3600 \text{ sec}) - D_{\gamma,\text{integ}} (t = 2700 \text{ sec})}{D_\gamma (t = 2700 \text{ sec})}$$

$$= \frac{7390 \text{ R} - 5651 \text{ R}}{7098 \text{ R/hr}} = 0.2450 \text{ hr} = 882 \text{ sec}$$

where:

D_γ = gamma dose rate

$D_{\gamma,\text{integ}}$ = gamma integrated dose (see eqns (24) – (25))

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ii) $E_\beta = 0.423$ MeV Nb-97

decay constant : $c = 1.605 \times 10^{-4}$ sec^{-1}
time step : $\Delta t = 900$ sec

$$\exp(-c\,\Delta t) = 0.8655$$

$$\frac{1 - \exp(-c\Delta t)}{c} = 838.0 \text{ sec}$$

$$\frac{D_\beta\,(t=900 \text{ sec})}{D_\beta\,(t=0)} = \frac{3667. \text{ Rad/hr}}{4236 \text{ Rad/hr}} = 0.8657$$

$$\frac{D_{\beta,\text{integ}}\,(t=900) - D_{\beta,\text{integ}}\,(t=0)}{D_\beta(t=0)} = \frac{(986.2 - 0) \text{ Rad}}{4236 \text{ Rad/hr}}$$

$$= 0.2328 \text{ hr}$$
$$= 838 \text{ sec}$$

iii) It may be verified from runs 1-5 via hand calculations that doses are summed and dose and activity fractions are calculated correctly. In addition, initialization and results are printed correctly for the various print options.

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C. Case 2 Dose Calculation – Time Zero.

The uncollided flux outside a cylinder is (reference 3, p. 360):

$$\phi = \frac{S_v R_i^2}{4(a+Z)} \left[F(\theta_2, b) - F(\theta_1, b) \right]$$

where: S_v = activity concentration
R_i = pipe inner radius
a = distance from pipe wall inner surface to observer
Z = self-absorption distance (obtained from reference 3, pp. 361–363)
θ_1, θ_2 defined in figure 3

$$b = \mu_s Z + \mu_p t_p + \sum_{i=1}^{N} \mu_i t_i$$

μ_s, μ_p, μ_i = linear attenuation coefficients for source, pipe wall, and shield i respectively
R_o = pipe outer radius
$t_p = R_o - R_i$
N = number of shields

Shield 1 : concrete $\rho_1 = 2.35$ g/cm³ $t_1 = 4.0$ cm
Shield 2 : iron $\rho_2 = 7.83$ g/cm³ $t_2 = 2.0$ cm

ROBS = distance from source ℄ to observer
= 22.960 cm

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i) $E_\gamma = 1.0$ MeV   Nb-97

from Section VI.A.: $S_V = 4.710 \times 10^{-3}$ Ci/cm³

$= 1.743 \times 10^{8}$ $\frac{\text{DISINT}}{\text{SEC-cm}^3}$

$\left(\frac{\mu}{\rho}\right)_s = 0.071$ cm²/g

$\therefore \mu_s = 0.071$ cm⁻¹

$a = ROBS - R_i = 22.96 \text{ cm} - 10.825 \text{ cm} = 12.71 \text{ cm}$

$\theta_1 = \theta_2 = \tan^{-1}\left[\frac{167.6}{22.96}\right] = 1.435 = 82.2^\circ$

from reference 3, p. 447:

$\left(\frac{\mu}{\rho}\right)_p = \left(\frac{\mu}{\rho}\right)_2 = \left(\frac{\mu}{\rho}\right)_{Fe} = 0.059 \frac{cm^2}{g}$

pipe wall

$\therefore \mu_p = \mu_2 = (0.059)(7.83) = 0.462$ cm⁻¹

$\mu_p t_p = 0.3280$

$\left(\frac{\mu}{\rho}\right)_1 = \left(\frac{\mu}{\rho}\right)_{concrete} = 0.0635$ cm²/g

(from code - subroutine ABSRP - data statements)

$\therefore \mu_1 = (0.0635)(2.35) = 0.1492$ cm⁻¹

$\mu_s(R_i + a) = \mu_s(ROBS) = (0.071)(22.96) = 1.630$

$a/R_i = 12.71/10.25 = 1.240$

$b_1 = \sum_{i=1}^{2} \mu_i t_i = (0.1492)(4) + (0.462)(2) = 1.521$

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from curves on pp. 362-363 of reference 3:

$$m \cong 0.38$$

$$\frac{1}{m}(\mu_s Z) \cong 1.56$$

$$\mu_s Z = 0.38(1.56) = 0.5928$$

$$Z = \frac{0.5928}{0.071} = 8.349 \text{ cm}$$

$$b = \mu_s Z + \mu_p t_p + b_1$$

$$= 0.5928 + 0.3280 + 1.521 = 2.442$$

$$\therefore \phi = \frac{(1.743\times10^{8}\ sec^{-1}\ cm^{-3})(10.250\ cm)^2}{4(12.71\ cm + 8.349\ cm)} \cdot 2\,F(\theta_2, b) \cdot (3\times10^{-2})$$

FRACTION OF ACTIVITY OF Nb-97 GOING TO THIS γ

$$= 1.304\times10^{7}\ F(\theta_2, b)\ cm^{-2}\,sec^{-1}$$

from reference 3, pp. 385-390:

$$F(\theta_2, b) = 0.056$$

$$\therefore \phi = 7.304\times10^{5}\ cm^{-2}\ sec^{-1}$$

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dose rate:

$$D_\gamma = (DF)\, B\, \phi$$

where: DF = dose conversion factor
B = buildup factor

from reference 3, pp. 417, 419, 423 (and from values used as input to code in subroutine BLDPF (see data statements)):

$E = 1.0$ MeV

water: $A_{1,s} = 11.0$
$\alpha_{1,s} = 0.104$
$\alpha_{2,s} = 0.028$

iron: $A_{1,p} = A_{1,2} = 8.6$
$\alpha_{1,p} = \alpha_{1,2} = 0.088$
$\alpha_{2,p} = \alpha_{2,2} = 0.028$

concrete: $A_{1,1} = 9.9$
$\alpha_{1,1} = 0.088$
$\alpha_{2,1} = 0.0295$

$b_s = \mu_s Z = 0.5928$
$b_p = \mu_p t_p = 0.3280$
$b_1 = \mu_1 t_1 = (0.1492\ cm^{-1})(4\ cm) = 0.5968$
$b_2 = \mu_2 t_2 = (0.462\ cm^{-1})(2\ cm) = 0.9240$

$$B_s = 11\, e^{(0.104)(0.5928)} - 10\, e^{-(0.028)(0.5928)}$$

$$= 1.864$$

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$$B_p = 8.6\, e^{(0.088)(0.3280)} - 7.6\, e^{-(0.028)(0.3280)} = 1.321$$

$$B_1 = 9.9\, e^{(0.088)(0.5968)} - 8.9\, e^{-(0.0295)(0.5968)}$$

$$= 1.689$$

$$B_2 = 8.6\, e^{(0.088)(0.9240)} - 7.6\, e^{-(0.028)(0.9240)}$$

$$= 1.923$$

$$B_{air} \cong 1.05$$

$$\therefore\ B = B_s B_p B_1 B_2 B_{air} = (1.864)(1.321)(1.689)(1.923)(1.05)$$

$$= 8.397$$

from reference 3, p. 19:

$$DF = 1.95 \times 10^{-6}\ \frac{R/hr}{cm^{-2}\,sec^{-1}}$$

$$\therefore\ D_\gamma = \left[1.95 \times 10^{-6}\ \frac{R/hr}{cm^{-2}\,sec^{-1}}\right](8.397)(7.304 \times 10^{5}\ cm^{-2}sec^{-1})$$

$$= 11.96\ R/hr$$

This value for D_γ is 19.6% above the code calculated value of 9.996 R/hr (see run 6). It is stated in reference 3, p. 401 that values for D_γ obtained using the

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method there (i.e., the method used here) can be as much as 40% larger than the correct D_γ. Maximum error occurs when a/R_i is small (i.e., $a/R_i \ll 10$) and/or b is large.

D. Case 3 Dose Calculation — Time Zero.

$E_\gamma = 0.305$ MeV   Kr-85m

Shield thickness $= 0$

ROBS = distance from source ℄ to observer $= 16.96$ cm

from Section VI. A. ; $S_V = 0.1205\ Ci/cm^3$
$= 4.459 \times 10^9 \frac{DISINT}{sec\text{-}cm^3}$

$\left(\frac{\mu}{\rho}\right)_s = 0.1175\ cm^2/g$

$\therefore \mu_s = 0.1175\ cm^{-1}$

$a = ROBS - R_i = 16.96\ cm - 10.250\ cm$
$= 6.71\ cm$

$\theta_1 = \theta_2 = \tan^{-1}\left[\frac{167.6}{16.96}\right] = 84.22^\circ = 1.470$ Rad

from reference 3, p. 447 ;

$\left(\frac{\mu}{\rho}\right)_p = 0.1075 \frac{cm^2}{g}$ (assume Fe wall)

$\mu_p = (0.1075)(7.83) = 0.8417\ cm^{-1}$

$\mu_p t_p = (0.8417)(10.96 - 10.25) = 0.5976$

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$\mu_s(R_i + a) \doteq \mu_s(R_{OBS}) = 1.993$

$a/R_i = 0.6546$

from curves on pp. 362-363 of reference 3:

$m \cong 0.44$

$b_1 = \mu_p t_p = 0.5976$

$\therefore \frac{1}{m}\mu_s Z = \begin{cases} 1.89 & \text{at } a/R_i = 0.6 \\ 1.74 & \text{at } a/R_o = 0.8 \end{cases}$

$\therefore \frac{1}{m}\mu_s Z = 1.85$ by interpolation

$\mu_s Z = 0.8136$

$Z = \dfrac{0.8136}{0.1175 \text{ cm}^{-1}} = 6.924$

$b = \mu_s Z + \mu_p t_p = 0.8136 + 0.5976 = 1.411$

$\therefore \quad \phi = \dfrac{(4.459 \times 10^9 \text{ cm}^{-2}\text{sec}^{-1})(10.250 \text{ cm})^2}{4(6.71 \text{ cm} + 6.924 \text{ cm})} \cdot 2F(\theta_2, b)(0.1)$

FRACTION OF ACTIVITY OF Kr-85m GOING TO THIS γ

$= 1.718 \times 10^9 \, F(\theta_2, b) \ \text{cm}^{-2}\text{sec}^{-1}$

from reference 3, pp. 385-390:

$F(\theta_2, b) = 0.19$

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$$\therefore \quad \phi = 3.264 \times 10^{8} \ cm^{-2} sec^{-1}$$

dose rate:

$$D_\gamma = (DF) B\phi$$

from reference 3, pp. 417, 419 (and from values used as input to code in subroutine BLDPF (see data statements)) and from section VI. A.:

$$E = 0.305 \ MeV$$
$$A_{1,s} = 29.85$$
$$\alpha_{1,s} = 0.171$$
$$\alpha_{2,s} = 0.0012$$

$$A_{1,p} = 10.59$$
$$\alpha_{1,p} = 0.09645$$
$$\alpha_{2,p} = 0.00661$$

$$B_s = 29.85 e^{(0.171)(.8136)} - 28.85 e^{-(0.0012)(0.8136)}$$
$$= 5.484$$

$$B_p = 10.59 e^{(.09645)(.5976)} - 9.59 e^{-(.00661)(.5976)}$$
$$= 1.666$$

$$B = B_s B_p B_{air} = (5.484)(1.666)(1.05) = 9.594$$

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$$DF = .6 \times 10^{-7} \frac{R/hr}{cm^{-2}\,sec^{-1}}$$ from reference 3, p.19

$$\therefore D_\gamma = \left(6 \times 10^{-7} \frac{R/hr}{cm^{-2}\,sec^{-1}}\right)(9.594)\left(3.264 \times 10^{8}\ cm^{-2}\,sec^{-1}\right)$$

$$= 1879\ R/hr$$

This value for D_γ is 40% above the code calculated value (see run 7). This is explained by conservatism in the method of reference 3 (see p. 401). In addition, the lumping of the source at Z produces a larger buildup than would be calculated for the distributed source.

VIII. Benchmark Case

The beta and gamma dose rates inside a typical dry containment during the thirty day period following a LOCA are calculated by the code and compared with independently obtained results summarized in Reference 8, Appendix D (additional information is given in Reference 7, Appendix B.)

A reactor power of 3391 MWT is assumed. The containment volume is 2.52×10^6 ft^3 = 7.136×10^{10} cm^3 (Reference 7, p. B-2). The containment is modeled as an equivalent cylinder with height equal to inner diameter (Reference 8, p. D-7). This gives an inner radius of 2248 cm and a height of 4496 cm. All remaining dimensions are taken from Figure A-2 of Reference 9. The containment is assumed to be filled with air of density 1.293×10^{-3} g/cm^3.

The analysis in References 7 and 8 assumes the only fission products present are noble gases and iodines. Both references assume that 100% of the noble gases and 25% of the iodines are released into the containment. Total activities are given in Reference 10, Table 5-29, and Reference 11, Appendix A, Table A.2-1. In cases where the two references conflict, Reference 10 activities are used. For Reference 10 activities, a U mass of 8.86×10^4 Kg is assumed. This roughly corresponds to a UO_2 mass of 216,600 lbm (Reference 11, Chapter 3 (Unit 1), Table 3.2.3-1). Decay constants are given in Reference 12. Gamma and beta energies, and fractions of total isotope activity comprising each energy are given in Reference 12.

The source is divided into 5 radial division and 18 angular divisions. This may be shown to satisfy the criteria of section V.B. For an air source, the maximum attenuation coefficient used by the code is $(\mu/\rho)_{max}$ = 0.151 cm^2/g (see SUBROUTINE ABSRP listing). Then

$$b_{max} = (0.151 \frac{cm^2}{g}) (1.293 \times 10^{-3} \frac{g}{cm^3})(2248 \text{ cm}) = 0.4389$$

$$2\pi b_{max} = 2.758$$

Therefore, (28a) and (28b) are satisfied.

Dose is calculated at one hour intervals for one day, and then at one day intervals up to thirty days. Print option 2 (totals for each isotope) is used. All input is summarized in run 8.

Reference 8 accounts for iodine removal by plate-out. In addition, Reference 8 may not use the same initial source strengths as are input to the code. Therefore, iodine and noble gas doses calculated by the code are scaled by $\frac{A_{I,REF.8}}{A_{I,CODE}}$ and $\frac{A_{NG,REF.8}}{A_{NG,CODE}}$

respectively,

where,

$A_{I,REF.8}$ = iodine activity used in Reference 8

$A_{NG,REF.8}$ = noble gas activity used in Reference 8

$A_{I,CODE}$ = iodine activity used by code

$A_{NG,CODE}$ = noble gas activity used by code.

All activities are evaluated at the time step in question, and must be expressed in units such that the above ratios are dimensionless. $A_{I,CODE}$ must account for 25% of total iodine activity.

Scaled code results and Reference 8 results are compared in Tables 3 and 4. With the exception of the beta dose rate at t=0 and t=720 hrs (30 days), scaled code results are higher than Reference 8 results. Beta doses agree reasonably well; discrepancies are most likely due to discrepancies in input beta energies and decay fractions. Gamma doses agree well up to 24 hours. Discrepancies are most likely due to discrepancies in input gamma energies and decay fractions, and conservative approximations in the code calculation of gamma dose.

TABLE [illegible]

COMPARISON OF SCALED CODE RESULTS AND REFERENCE 8 RESULTS - GAMMA DOSE

TO BE COMPARED

Time (HR)	Iodine Concentration[1] (10^8Ci)	Noble Gas Concentration[1] (10^8Ci)	Iodine Dose Rate[2] (10^5R/HR)	Noble Gas Dose Rate[2] (10^5R/HR)	Scaled Iodine Dose Rate[3] (10^5R/HR)	Scaled Noble Gas Dose Rate[3] (10^5R/HR)	Total Scaled Dose Rate[4] (10^5R/HR)	Reference 8 Dose Rate[5] (10^5R/HR)
0	2.06	5.31	8.95	8.40	19.9	20.72	40.62	37.9
1	1.61	4.41	6.05	5.95	0.63	8.65	9.28	8.1
8	0.83	2.70	1.86	1.51	0.13	2.02	2.15	1.82
24	0.477	2.023	0.77	0.54	0.048	0.622	0.670	0.399
96	0.193	1.239	0.221	0.296	0.013	0.317	0.330	0.100
720	0.0184	0.0456	0.01971	0.00887	0.00114	0.01169	0.01283	0.00395

[1] Activities from Run 8. Iodine activities multiplied by 0.25.

[2] Dose rates from Run 8.

[3] Dose rates scaled using Reference 8 activities from Table D-2, Columns 2 and 6.

[4] Sum of Columns 6 and 7.

[5] Dose rates from Reference 8, Table D-5, Column 2. Dose rates are reduced by a factor of 1.3 for comparison with code results (see Reference 8, p. D-7).

TABL

COMPARISON OF SCALED CODE RESULTS AND REFERENCE 8 RESULTS - BETA DOSE

TO BE COMPARED

Time (HR)	Iodine Concentration[1] (10^8Ci)	Noble Gas Concentration[1] (10^8Ci)	Iodine Dose Rate[2] (10^5R/HR)	Noble Gas Dose Rate[2] (10^5R/HR)	Scaled Iodine Dose Rate[3] (10^5R/HR)	Scaled Noble Gas Dose Rate[3] (10^5R/HR)	Total Scaled Dose Rate[4] (10^5R/HR)	Reference 8 Dose Rate[5] (10^5R/HR)
0	2.06	5.31	20.79	43.09	46.23	106.3	152.53	182.5
1	1.61	4.41	15.08	30.54	1.57	44.40	46.0	25.9
8	0.83	2.70	6.42	8.60	0.45	11.50	11.95	11.8
24	0.477	2.023	3.28	4.68	0.204	5.39	5.59	5.44
96	0.193	1.293	0.909	2.714	0.053	2.907	2.96	2.40
720	0.0184	0.0456	0.07678	0.08565	0.0044	0.1117	0.11610	0.146

[1] Activities from Run 8. Iodine activities multiplied by 0.25.

[2] Dose rates from Run 8.

[3] Dose rates scaled using Reference 8 activities from Table D-2, Columns 2 and 6.

[4] Sum of Columns 6 and 7.

[5] Dose rates from Reference 8, Table D-6, Column 2. Dose rates are reduced by a factor of 1.3 for comparison with code results (see Reference 8, p. D-7).

References

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Appendix A

Evaluation of $F(\theta,b)$

The Sievert Integral Function, $F(\theta,b)$, is defined by

$$F(\theta,b) = \int_0^{\theta} e^{-b\sec\theta'}\, d\theta' \qquad \text{(A1)}$$

In all applications here, $b \geq 0$ and $-\pi \leq \theta \leq \pi$.

This function is evaluated by the function subprogram F(THETA,B). The following approximations are used:

1) $b < 1, \quad 0 \leq \theta \leq 60^{\circ}$

$$F(\theta,b) \cong \theta e^{-b} \qquad \text{(A2)}$$

This approximation is given in Reference 3, p. 408 for $\theta < 5^{\circ}$. However, examination of curves for $F(\theta,b)$ (Reference 3, pp. 385-390) shows that for $b < 1$, this approximation is good for θ up to 60°.

2) $b < 1, \quad 60^{\circ} < \theta \leq 90^{\circ}$

$$F(\theta,b) \cong \frac{\pi}{2} e^{-b} \qquad \text{(A3)}$$

This approximation is conservative for all (θ,b) in this range. $F(\theta,b)$ is overestimated by as much as 50% for $\theta = 60^{\circ}$ (Reference 3, pp. 385-390).

3) $b \geq 1, \quad 0 \leq \theta \leq 5^{\circ}$

$$F(\theta,b) \cong \theta e^{-b} \qquad \text{(A4)}$$

(Reference 3, p. 408)

4) $b \geq 1, \quad 5^\circ < \theta < 60^\circ$

For this range of (θ, b) a Simpson's Rule approximation is used with a step size equal to 0.01θ (Reference 6, p. 212)

5) $b \geq 1, \quad 60^\circ \leq \theta \leq 90^\circ$

$$F(\theta, b) \cong \sqrt{\frac{\pi}{2b}}\, e^{-b} \left(1 - \frac{3}{8b}\right) \qquad \text{(A5)}$$

(Reference 4, p. 439)

For $\theta < 0$, the result $F(-\theta, b) = F(\theta, b)$ is used (Reference 4, p. 439).

Appendix B

Version SHL3

An earlier version of the code, SHL3, has been used in several applications. This version differs from SHL1 in that beta dose is calculated based on an infinite cloud source regardless of the cylinder size or beta energy. For small cylinders or high energies this will be conservative. In addition, beta activity, energy, and dose input and output formats for SHL1 and SHL3 differ.

Run 9 reproduces case 1, run 1 using SHL3. Comparison of runs 1 and 9 shows that SHL1 and SHL3 results for gamma dose are identical. For beta dose (dose in water, run 9), run 9 is 0.3% larger. This is due to a constant in SUBROUTINE PD3 of SHL3 being 0.3% larger than the corresponding constant in SHL1. This error is insignificant, and doses inside a cylindrical source calculated by SHL3 are correct.

The purpose of this appendix is to verify SHL3 for previous applications. SHL3 is not presently used.

Appendix C

Final Modifications of November 20, 1980 and December 16, 1980

Several modifications were made to the main program. These changes relate to branching around certain DO loops.

An additional CONTINUE statement was added immediately following each respective DO loop. The branch around the loop was changed to a branch to this statement. Previously, the branch was to the DO loop terminal statement. This can have adverse effects in IBM FORTRAN.

All changes are labeled 11/20/80 in the updated source listing (run 10). Previous numerical results are unaffected.

Effective December 16, 1980, the source member name is changed from NSLSHL1 to SHL1GG. This is accomplished by the attached run 12.

ATTACHMENT 3 TO AEP:NRC:0578I

DESCRIPTION OF CALCULATIONAL PROCEDURE USED IN DETERMINING RADIATION DOSES TO ELECTRIC EQUIPMENT INSIDE CONTAINMENT

Introduction

In order to calculate the total integrated dose to selected electric equipment inside containment, the upper volume, lower volume, fan/accumulator rooms, pipe tunnel, instrument room, and total free volume were modeled into equivalent cylinders, based on volume information obtained from the Donald C. Cook Nuclear Plant Final Safety Analysis Report (FSAR), Appendix P. The computer program SHL1GG, described in Attachment 2 to this submittal, was then used with an appropriate source term to obtain the integrated doses and beta attenuation factors of interest. These values were then applied to equipment specific calculations, based upon required operating times and the actual shielding installed at the plant.

Volume Nodalization

For the overall containment, the total free volume's equivalent cylinder was determined to have a radius of 50.5 ft and a height of 145 ft (case 1). Calculational results for this case are assumed to bound those for the upper volume, which was determined to be equivalent to a cylinder 50.5 ft in radius with a height of 84.4 ft.

Three models were developed to describe the lower volume. First, the annulus around the reactor cavity was treated as an equivalent cylinder with a radius of 17 ft and a height of 162 ft, into which airborne radiation is released (case 2). A second approach which was deemed more likely to be conservative was to assume that the reactor cavity volume is a part of the lower volume. This equivalent cylinder had a radius of 41.5 ft and a height of 40 ft, into which airborne radiation was released (case 3). The third approach involved taking that portion of the lower volume which is expected to be submerged, and modeling it into an equivalent cylinder with a radius of 41.5 ft and a height of 18 ft (case 4).

With regard to the fan/accumulator rooms, they were modeled into equivalent cylinders of radius 10.5 ft and height 79 ft (case 5). The results obtained for these cylinders were assumed to bound those for the instrument room , which was determined to be equivalent to a cylinder of 10.5 ft in radius and 67 ft in height.

The pipe tunnel was described as an equivalent cylinder of 7.5 ft in radius and 141 ft in height. Calculational results were obtained for the pipe tunnel both as if the tunnel had been flooded (case 6) and as if it had not been flooded (case 7). For the earliest stages of a DBA, a source term in air or an air-steam mixture may be a more realistic assumption than that of instantaneous flooding.

Source Terms And Calculational Results

Activities released into the containment for the assumed DBA were evaluated using values specified in Table 3-1. This Table lists fifty-four (54) radioisotopes which have been identified as being characterisitic of the Donald C. Cook Nuclear Plant core. This list was compiled from two sources -- the Donald C. Cook Nuclear Plant FSAR, Appendix 14A, and the Westinghouse Standard Information Package, "Radiation Analysis Manual: Standard Plant Model 412," Revision 3, dated November 1978.

Using Table 3-1, it was a simple task to calculate the activities released into containment for the assumed DBA. Using NUREG-0588 as a basis of reference, 100% of the core inventory of noble gases, 50% of the core inventory of radioiodines, and 1% of the core inventory of particulates was assumed to be released into the containment free volume after flood-up (e.g., 1,157,408 ft^3). These activities are presented in Table 3-2. For submerged components, 50% of the core inventory of radioiodines and 1% of the core inventory of particulates was assumed to be distributed into approximately 715,500 gallons of water (assuming a maximum flood-up elevation of 614'). Halogen plate-out on containment surfaces and halogen removal by containment sprays and borated ice were not assumed for conservatism.

In order to examine radiation attenuation in concrete, SHL1GG was used to calculate the integrated gamma dose at the center, the inner edge, and the outer edge of a cylinder with an inner radius of 50.5 ft, an outer radius of 52 ft, and a height of 145 ft. These calculations indicated that an 18" thickness of concrete (density = 2.35 grams per cubic centimeter), which is a typical concrete thickness between compartments, would attenuate all of the beta particles and a substantial portion of the gamma rays. More specifically, integrated gamma doses were compared at various times between the three locations of concern. For the accumulated one-year dose, the value at the inner edge of the cylinder was approximately 52.0% that at the centerline of the cylinder, and the value at the outer edge of the cylinder was a mere 0.051% that at the centerline of the cylinder. Therefore, it was concluded that the dose contribution from one compartment would not have a significant effect on any adjacent compartments, and thus each compartment could be treated separately from the others.

For each of the nodalized volumes inside containment identified above, two runs were performed. In the first of each set of runs, the observer (e.g., equipment item) was assumed to be on the centerline of the cylinder of concern, midway between the cylinder's ends. In the second run, the observer was located at the midplane of the cylinder at the inner edge of the compartment.

Although much data was collected from these runs, the results of only two are of particular concern. These cases are those which give the greatest accumulated doses as a function of time for airborne radiation and for a source in water. More specifically, the greatest

accumulated doses as a function of time for airborne sources are obtained from case 1 (e.g., using the total free volume of containment after flood-up and the source term given in Table 3-2, with the observer at the center of the cylinder). Likewise, the greatest accumulated doses as a function of time for submerged sources are obtained from case 4, utilizing the flooded portion of the lower volume, again with the observer at the centerline of the cylinder. The accumulated gamma and beta doses for these cases are presented in Tables 3-3 and 3-4, respectively, where the gamma doses given in Roentgen by SHL1GG have been converted to Mrads.

In order to correct the beta dose values given in Tables 3-3 and 3-4 for attenuation near an equipment item (due to housings, jackets, etc.), the percentage of accumulated surface beta dose that is attenuated by various thicknesses of three different materials was evaluated. The result of this evaluation is presented in Tables 3-5 through 3-8. It should be noted that this evaluation is source specific, relying upon the relative activities of the different isotopes and the energies of the various betas emitted.

Application To The Equipment Qualification Process

The results of the calculations described above were applied to electrical components within containment through the following process: (a) the equipment was identified as either being submerged or in an airborne source; (b) the applicable gross gamma and beta accumulated doses were obtained from Tables 3-3 or 3-4, as appropriate, corresponding to a time equal to or greater than the equipment required operating time; (c) the beta dose was adjusted for shielding effects, if any, utilizing Tables 3-5 through 3-8, as appropriate; and (d) the corrected gamma was added to the gross gamma to give the equipment specified post-accident dose.

The following provides examples of how this process was applied to a few categories of equipment:

a) Category "A". Cables and cable terminations above flood level were assumed at the center of the total free volume after containment flood-up. The combined thickness of cable insulation and jacket is conservatively deemed to be 70 mils of unit density material. Splicing material which protects the cable terminations is also in excess of 70 mils of unit density material. Credit is taken, however, for only 70 mils of this material. This category covers Unit 1 System Component Evaluation Worksheets (SCEW sheets, as presented in Attachments 4 and 5 to our letter No. AEP:NRC:0578B, dated June 11, 1982) CC-1, CC-5, CC-6, CC-7, CC-8, CI-1, CI-2, CI-3, CI-5, CI-8, CI-9, CP-9, CP-11, CP-12, CP-13, TC-7, TC-8, TI-1, TI-4, and TP-2, and Unit 2 SCEW sheets CC-1, CC-2, CC-3, CC-4, CC-5, CC-6, CC-7, CC-8, CI-5, CI-7, CI-8, CI-9, CI-11, CP-4, CP-5, CP-11, TC-7, TC-8, TI-1, TI-2, TI-4, and TP-2.

b) Category "B". Same as in category "A", except that component will be submerged post-accident in flood-up tubes (Kapton insulated wire) which have a wall thickness of 16 mil of steel. Use Table 3-4 to obtain gross gamma and beta doses to the equipment. This category covers Unit 1 SCEW sheets CI-11 and CI-12, and Unit 2 SCEW sheets CI-15 and CI-16.

c) Category "C". Similar to Category "A", except that the sum of cable insulation and jacket material is less than 70 mils. This includes Unit 1 SCEW sheets CP-4 (63 mils), CP-7 (45 mils), and CP-8 (60 mils), and Unit 2 SCEW sheets CP-6 (63 mils), CP-9 (45 mils), and CP-10 (60 mils).

d) Category "D". Components are above flood level and are enclosed in steel enclosures greater then 70 mils thick which do not permit free air circulation. Use Table 3-3 and Table 3-6, assuming only 70 mils of steel. This includes Unit 1 SCEW sheets F-1, LS-1, TC-2, TC-4, TC-16, TC-17, TP-3, V-2, and V-4, and Unit 2 SCEW sheets F-1, LS-1, TC-2, TC-4, TC-16, TP-3, V-2, and V-4.

e) Category "E". The electric hydrogen recombiner is above flood level, and beta attenuation is attributed to 10 mils of steel enclosure. SCEW sheet H-1 (both Unit 1 and Unit 2).

f) Category "F". Cable terminations, inside stainless steel flood-up tubing (16 mils wall thickness) will be submerged and are assumed at the center of the lower volume. No credit is taken for splicing material which protects the terminations; however, credit for 10 mils of stainless steel is taken. This category includes SCEW sheets TC-6, TI-3, and TP-1 (both Unit 1 and Unit 2).

g) Category "G". Submerged components, assumed at the center of the flooded lower volume. No credit is taken for beta attenuation due to valve enclosures because they are not assumed to be watertight. This category includes SCEW sheets EP-01, EP-02, TC-1, TC-3, TI-5, TI-8, TP-5, V-1, and V-5 (both Unit 1 and Unit 2).

Tables 3-9 and 3-10 present, for the Donald C. Cook Nuclear Plant Unit Nos. 1 and 2, respectively, the results of the calculations for the specific SCEW sheets noted above in categories "A" through "G". It is noted that Unit 1 devices CI-1, CI-2, TC-2, and V-4, and Unit 2 devices TC-2 and V-4 are assumed to have an effective beta attenuation factor of 90%. This factor is considered to be conservative since these devices are protected by either unit density material or steel in excess of 70 mils thickness.

Table 3-1

Core Isotope Activities

Isotope	Activity (Ci)*	Isotope	Activity (Ci)*
Kr-85	8.860×10^{5}	Te-132	1.420×10^{8}
Kr-85m	4.300×10^{7}	I-131	9.750×10^{7}
Kr-87	7.790×10^{7}	I-132	1.420×10^{8}
Kr-88	1.060×10^{8}	I-133	1.950×10^{8}
Sr-89	8.060×10^{7}	I-134	2.190×10^{8}
Sr-90	7.270×10^{6}	I-135	1.700×10^{8}
Y-90	7.710×10^{6}	Xe-133	1.950×10^{8}
Y-91	1.060×10^{8}	Xe-133m	2.840×10^{7}
Zr-95	1.590×10^{8}	Xe-135	5.310×10^{7}
Zr-97	1.590×10^{8}	Xe-135m	5.220×10^{7}
Nb-95	1.590×10^{8}	Cs-134	2.300×10^{7}
Nb-95m	1.950×10^{6}	Cs-136	6.380×10^{6}
Nb-97	1.680×10^{8}	Cs-137	1.060×10^{7}
Mo-99	1.770×10^{8}	Ba-137m	9.750×10^{6}
Tc-99m	1.590×10^{8}	Ba-140	1.680×10^{8}
Ru-103	1.680×10^{8}	La-140	1.770×10^{8}
Ru-106	5.850×10^{7}	Ce-141	1.590×10^{8}
Rh-103m	1.680×10^{8}	Ce-143	1.420×10^{8}
Rh-105	1.060×10^{8}	Ce-144	1.240×10^{8}
Rh-106	5.850×10^{7}	Pr-143	1.420×10^{8}
Ag-110m	6.380×10^{5}	Pr-144	1.240×10^{8}
Ag-111	5.670×10^{6}	Nd-147	6.290×10^{7}
Sb-125	9.750×10^{5}	Pm-148	2.040×10^{7}
Sb-127	1.060×10^{7}	Pm-148m	8.860×10^{6}
Te-127	1.060×10^{7}	Pm-149	6.290×10^{7}
Te-129	3.280×10^{7}	Sm-153	6.290×10^{7}
Te-129m	8.860×10^{6}	Eu-156	3.280×10^{7}

*Activities of Kr-85m, Kr-87, Kr-88, I-134, I-135, Xe-135, and Xe-135m were obtained from FSAR Table 14.A.2-1 (see July 1982 Updated FSAR). Other activities were estimated from Table 5-29 of the Westinghouse Standard Information Package, "Radiation Analysis Manual: Standard Plant Model 412," Revision 3, dated November 1978. All values are rounded off.

Table 3-2

Case #1 Isotope Activities*
Total Free Volume -- Observer At Center

Isotope	Activity (Ci)	Isotope	Activity (Ci)
Kr-85	8.860×10^{5}	Te-132	1.420×10^{6}
Kr-85m	4.300×10^{7}	I-131	4.875×10^{7}
Kr-87	7.790×10^{7}	I-132	7.100×10^{7}
Kr-88	1.060×10^{8}	I-133	9.750×10^{7}
Sr-89	8.060×10^{5}	I-134	1.095×10^{8}
Sr-90	7.270×10^{4}	I-135	8.500×10^{7}
Y-90	7.710×10^{4}	Xe-133	1.950×10^{8}
Y-91	1.060×10^{6}	Xe-133m	2.840×10^{7}
Zr-95	1.590×10^{6}	Xe-135	5.310×10^{7}
Zr-97	1.590×10^{6}	Xe-135m	5.220×10^{7}
Nb-95	1.590×10^{6}	Cs-134	2.300×10^{5}
Nb-95m	1.950×10^{4}	Cs-136	6.380×10^{4}
Nb-97	1.680×10^{6}	Cs-137	1.060×10^{5}
Mo-99	1.770×10^{6}	Ba-137m	9.750×10^{4}
Tc-99m	1.590×10^{6}	Ba-140	1.680×10^{6}
Ru-103	1.680×10^{6}	La-140	1.770×10^{6}
Ru-106	5.850×10^{5}	Ce-141	1.590×10^{6}
Rh-103m	1.680×10^{6}	Ce-143	1.420×10^{6}
Rh-105	1.060×10^{6}	Ce-144	1.240×10^{6}
Rh-106	5.850×10^{5}	Pr-143	1.420×10^{6}
Ag-110m	6.380×10^{3}	Pr-144	1.240×10^{6}
Ag-111	5.670×10^{4}	Nd-147	6.290×10^{5}
Sb-125	9.750×10^{3}	Pm-148	2.040×10^{5}
Sb-127	1.060×10^{5}	Pm-148m	8.860×10^{4}
Te-127	1.060×10^{5}	Pm-149	6.290×10^{5}
Te-129	3.280×10^{5}	Sm-153	6.290×10^{5}
Te-129m	8.860×10^{4}	Eu-156	3.280×10^{5}

*The activity values listed in this Table are equivalent to 100% of the core inventory of noble gases, 50% of the core inventory of radioiodines, and 1% of the core inventory of particulates. Total activity upon release to containment free volume is 1.0012×10^{9} Ci. Activity concentration upon release to containment is 865.04 Ci per ft^3 of air.

Table 3-3

Case #1 Integrated Doses
Total Free Volume -- Observer At Center

Time	Integrated Gamma Dose (Mrads)	Integrated Beta Dose (Mrads)
0.0 sec	0.00	0.0
1.0 hour	4.31	16.0
12.0 hour	20.98	87.1
1.0 day	27.96	126.3
2.0 day	36.37	178.8
200.0 hour	61.11	338.5
400.0 hour	76.50	432.7
30.0 day	88.31	497.8
60.0 day	99.60	553.8
90.0 day	105.76	583.5
120.0 day	110.08	604.1
6.0 month	116.12	632.0
1.0 year	125.80	670.8

Table 3-4

Case #4 Integrated Doses
Submerged Lower Volume -- Observer At Center

Time	Integrated Gamma Dose (Mrads)	Integrated Beta Dose (Mrads)
0.0 sec	0.00	0.0
1.0 hour	0.67	0.1
12.0 hour	3.44	0.8
1.0 day	4.80	1.2
2.0 day	6.44	1.7
200.0 hour	11.2	2.9
400.0 hour	14.4	3.8
30.0 day	17.2	4.5
60.0 day	19.9	5.0
90.0 day	21.4	5.2
120.0 day	22.4	5.3
6.0 month	23.8	5.4
1.0 year	26.1	5.8

Table 3-5

Surface Beta Attenuation Factors*
For Unit Density Material
Assuming Airborne Source Term

	Thickness of Unit Density Material (mils)						
Time	10	20	30	40	50	60	70
1 hour	50.74	35.97	27.52	22.00	18.11	15.21	12.95
12 hours	35.56	21.57	14.69	10.72	6.96	5.48	4.46
1 day	31.92	18.52	12.09	8.50	5.39	4.13	3.29
2 days	28.08	15.74	9.95	6.78	4.25	3.18	2.48
1 month	18.19	9.51	5.82	3.83	2.42	1.73	1.28
2 months	19.26	10.26	6.31	4.14	2.62	1.84	1.34
3 months	19.99	10.79	6.66	4.36	2.77	1.93	1.39
4 months	20.42	11.11	6.88	4.50	2.86	1.98	1.42
½ year	20.80	11.40	7.07	4.62	2.94	2.03	1.44
1 year	20.66	11.33	7.03	4.58	2.92	2.01	1.42

*Note: Factors given are the fractions of accumulated beta dose <u>remaining</u> after a given time.

Table 3-6

Surface Beta Attenuation Factors*
For Stainless Steel
Assuming Airborne Source Term

Time	Thickness of Stainless Steel (mils) 10	20	30	40	50	60	70
1 hour	11.47	4.07	1.59	0.64	0.27	0.11	0.05
12 hours	3.83	1.27	0.51	0.21	0.09	0.04	0.02
1 day	2.78	0.89	0.35	0.15	0.06	0.03	0.01
2 days	2.08	0.63	0.25	0.10	0.04	0.02	0.01
1 month	1.03	0.25	0.09	0.04	0.02	0.01	0.00
2 months	1.06	0.23	0.08	0.03	0.01	0.01	0.00
3 months	1.09	0.22	0.08	0.03	0.01	0.01	0.00
4 months	1.11	0.21	0.08	0.03	0.01	0.01	0.00
½ year	1.12	0.21	0.07	0.03	0.01	0.01	0.00
1 year	1.10	0.20	0.07	0.03	0.01	0.01	0.00

*Note: Factors given are the fractions of accumulated beta dose remaining after a given time.

Table 3-7

Surface Beta Attenuation Factors*
For Aluminum
Assuming Airborne Source Term

Time	Thickness of Aluminum (mils) 10	20	30	40	50	60	70
1 hour	29.65	16.86	10.98	7.53	5.30	3.78	2.72
12 hours	16.33	6.30	3.63	2.39	1.65	1.18	0.85
1 day	13.61	4.82	2.63	1.69	1.16	0.82	0.59
2 days	11.31	3.76	1.95	1.22	0.83	0.59	0.42
1 month	6.68	2.10	0.95	0.53	0.34	0.23	0.16
2 months	7.24	2.27	0.98	0.52	0.32	0.21	0.15
3 months	7.64	2.39	1.00	0.52	0.31	0.20	0.14
4 months	7.88	2.46	1.01	0.51	0.30	0.20	0.13
½ year	8.10	2.53	1.02	0.51	0.29	0.19	0.13
1 year	8.05	2.51	1.00	0.49	0.28	0.18	0.12

*Note: Factors given are the fractions of accumulated beta dose remaining after a given time.

Table 3-8

Surface Beta Attenuation Factors For 10 Mils Of Stainless Steel Assuming Submergence Source Term

Time	Percentage Of Beta Dose Remaining
1 hour	2.88
12 hours	1.50
1 day	1.19
2 days	0.98
1 month	0.57
2 months	0.56
3 months	0.55
4 months	0.54
½ year	0.52
1 year	0.49

Table 3-9

Donald C. Cook Nuclear Plant Unit No. 1
Specified Radiation Doses Inside Containment

Device	Equipment Description	Specified Operating Time	Gamma Dose, (MRads)	Beta Dose, Gross, (MRads)	Beta Dose, After Attenuation, (MRads)	Effective Specified Dose, (MRads)
CC-1	Continental Cable 3119	14 days	76.50	432.7	10.73	87.23
CC-5	Continental Cable 3121	14 days	76.50	432.7	10.73	87.23
CC-6	General Electric 3121	14 days	76.50	432.7	10.73	87.23
CC-7	Continental Cable 3122	14 days	76.50	432.7	10.73	87.23
CC-8	General Electric 3121	14 days	76.50	432.7	10.73	87.23
CI-1	Boston Ins'd. Wire 3064	5 seconds	4.31	16.0	1.6	5.91
CI-2	Rockbestos 3064	5 seconds	4.31	16.0	1.6	5.91
CI-3	Samuel Moore 3075	4 months	110.08	604.1	8.58	118.66
CI-5	Boston Ins'd. Wire 3075	4 months	110.08	604.1	8.58	118.66
CI-8	Cerro 3077	4 months	110.08	604.1	8.58	118.66
CI-9	Samuel Moore 3077	4 months	110.08	604.1	8.58	118.66
CI-11	Kapton Insulated Wire	1 year	26.1	5.8	0.03	26.13
CI-12	Kapton Insulated Wire	10 seconds	0.67	0.1	0.1	0.68
CP-4	Okonite 399	3 months	105.76	583.5	11.26	117.02
CP-7	Anaconda 3116	14 days	76.50	432.7	29.34	105.84
CP-8	Essex 3116	14 days	76.50	432.7	13.76	90.26
CP-9	Kerite 3116	14 days	76.50	432.7	10.73	87.23
CP-11	Kerite 3127	1 year	125.80	670.8	9.53	135.33
CP-12	Cyprus 347	3 months	105.76	583.5	8.11	113.87
CP-13	Anaconda 347	3 months	105.76	583.5	8.11	113.87
EP-01	4 kV Penetrations	1 year	26.1	5.8	5.8	31.9
EP-02	Penetrations	1 year	26.1	5.8	5.8	31.9
F-1	Fan Motors	1 year	125.80	670.8	0.00	125.80
H-1	Hydrogen Recombiner	3 months	105.76	583.5	6.36	112.12
LS-1	NAMCO Limit Switch	14 days	76.50	432.7	0.04	76.54
TC-1	Control Cable Term.	30 minutes	0.67	0.1	0.1	0.77
TC-2	Control Cable Term.	30 minutes	4.31	16.0	1.6	5.91

Table 3-9 (continued)

Device	Equipment Description	Specified Operating Time	Gamma Dose, (MRads)	Beta Dose, Gross, (MRads)	Beta Dose, After Attenuation, (MRads)	Effective Specified Dose, (MRads)
TC-3	Control Cable Term.	30 minutes	0.67	0.1	0.1	0.77
TC-4	Control Cable Term.	1 day	27.96	126.3	0.01	27.97
TC-6	Control Cable Term.	14 days	14.4	3.8	0.04	14.44
TC-7	Control Cable Term.	14 days	76.50	432.7	10.73	87.23
TC-8	Cable Termination	14 days	76.50	432.7	10.73	87.23
TC-16	Control Cable Term.	1 day	27.96	126.3	0.01	27.97
TC-17	Seal Assembly	14 days	76.50	432.7	0.04	76.54
TI-1	Instrument Cable Term.	4 months	110.08	604.1	8.58	118.66
TI-3	Instrument Cable Term.	4 months	22.4	5.3	0.03	22.43
TI-4	Instrument Cable Term.	4 months	110.08	604.1	8.58	118.66
TI-5	Instrumentation Term.	5 minutes	0.67	0.1	0.1	0.77
TI-8	Instr. Cable Term.	32 seconds	0.67	0.1	0.1	0.77
TP-1	Power Cable Term.	1 year	26.1	5.8	0.00	26.1
TP-2	Termination	1 year	125.80	670.8	9.53	135.33
TP-3	Termination	1 year	125.80	670.8	0.00	125.80
TP-5	Power Cable Term.	30 minutes	0.67	0.1	0.1	0.77
V-1	Valve Motor Operator	30 minutes	0.67	0.1	0.1	0.77
V-2	Valve Motor Operator	1 day	27.96	126.3	0.01	27.97
V-4	Valve Motor Operator	30 minutes	4.31	16.0	1.6	5.91
V-5	Valve Motor Operator	60 seconds	0.67	0.1	0.1	0.77

Table 3-10

Donald C. Cook Nuclear Plant Unit No. 2
Specified Radiation Doses Inside Containment

Device	Equipment Description	Specified Operating Time	Gamma Dose, (MRads)	Beta Dose, Gross, (MRads)	Beta Dose, After Attenuation, (MRads)	Effective Specified Dose, (MRads)
CC-1	Continental Cable 3119	14 days	76.50	432.7	10.73	87.23
CC-2	Continental Cable 3120	14 days	76.50	432.7	10.73	87.23
CC-3	General Electric 3120	14 days	76.50	432.7	10.73	87.23
CC-4	Anaconda 3120	14 days	76.50	432.7	10.73	87.23
CC-5	Continental Cable 3121	14 days	76.50	432.7	10.73	87.23
CC-6	General Electric 3121	14 days	76.50	432.7	10.73	87.23
CC-7	Continental Cable 3122	1 day	27.96	126.3	4.16	32.12
CC-8	General Electric 3122	1 day	27.96	126.3	4.16	32.12
CI-5	Samuel Moore 3075	4 months	110.08	604.1	8.58	118.66
CI-7	Boston Ins'd. Wire 3075	4 months	110.08	604.1	8.58	118.66
CI-8	Cerro 3077	4 months	110.08	604.1	8.58	118.66
CI-9	Samuel Moore 3077	4 months	110.08	604.1	8.58	118.66
CI-11	Boston Ins'd. Wire 3077	4 months	110.08	604.1	8.58	118.66
CI-15	Kapton Insulated Wire	1 year	26.1	5.8	0.03	26.13
CI-16	Kapton Insulated Wire	10 seconds	0.67	0.1	0.1	0.68
CP-4	Cyprus 347	1 year	125.80	670.8	9.53	135.33
CP-5	Anaconda 347	1 year	125.80	670.8	9.53	135.33
CP-6	Okonite 399	3 months	105.76	583.5	11.26	117.02
CP-9	Anaconda 3116	14 days	76.50	432.7	29.34	105.84
CP-10	Essex 3116	14 days	76.50	432.7	13.76	90.26
CP-11	Kerite 3116	14 days	76.50	432.7	10.73	87.23
EP-01	4 kV Penetrations	1 year	26.1	5.8	5.8	31.9
EP-02	Penetrations	1 year	26.1	5.8	5.8	31.9
F-1	Fan Motors	1 year	125.80	670.8	0.00	125.80
H-1	Hydrogen Recombiner	3 months	105.76	583.5	6.36	112.12
LS-1	NAMCO Limit Switch	14 days	76.50	432.7	0.04	76.54

Table 3-10 (continued)

Device	Equipment Description	Specified Operating Time	Gamma Dose, (MRads)	Beta Dose, Gross, (MRads)	Beta Dose, After Attenuation, (MRads)	Effective Specified Dose, (MRads)
TC-1	Control Cable Term.	30 minutes	0.67	0.1	0.1	0.77
TC-2	Control Cable Term.	30 minutes	4.31	16.0	1.6	5.91
TC-3	Control Cable Term.	30 minutes	0.67	0.1	0.1	0.77
TC-4	Control Cable Term.	1 day	27.96	126.3	0.01	27.97
TC-6	Control Cable Term.	14 days	14.4	3.8	0.04	14.44
TC-7	Control Cable Term.	14 days	76.50	432.7	10.73	87.23
TC-8	Control Cable Term.	14 days	76.50	432.7	10.73	87.23
TC-16	Control Cable Term.	1 day	27.96	126.3	0.01	27.97
TI-1	Instrument Cable Term.	4 months	110.08	604.1	8.58	118.66
TI-2	Instrument Cable Term.	4 months	110.08	604.1	8.58	118.66
TI-3	Instrument Cable Term.	4 months	22.4	5.3	0.03	22.43
TI-4	Instrument Cable Term.	4 months	110.08	604.1	8.58	118.66
TI-5	Instrumentation Term.	5 minutes	0.67	0.1	0.1	0.77
TI-8	Instr. Cable Term.	32 seconds	0.67	0.1	0.1	0.77
TP-1	Power Cable Term.	1 year	26.1	5.8	0.00	26.1
TP-2	Power Cable Term.	1 year	125.80	670.8	9.53	135.33
TP-3	Termination	1 year	125.80	670.8	0.00	125.80
TP-5	Power Cable Term.	30 minutes	0.67	0.1	0.1	0.77
V-1	Valve Motor Operator	30 minutes	0.67	0.1	0.1	0.77
V-2	Valve Motor Operator	1 day	27.96	126.3	0.01	27.97
V-4	Valve Motor Operator	30 minutes	4.31	16.0	1.6	5.91
V-5	Valve Motor Operator	60 seconds	0.67	0.1	0.1	0.77

ATTACHMENT 4 TO AEP:NRC:0578I
DESCRIPTION OF CALCULATIONAL PROCEDURE
USED IN DETERMINING RADIATION DOSES TO
ELECTRIC EQUIPMENT OUTSIDE CONTAINMENT

Introduction

In order to obtain bounding estimates for radiation doses to equipment located near pipes which carry fluid recirculated from the containment (for the purpose of long-term post-LOCA heat removal), radiation calculations were performed for a number of piping systems outside containment. The results of these calculations and their application to the equipment qualification process are described in this attachment.

Piping Calculations

The original piping calculations of concern were performed in April of 1980. The calculations were performed using a computer program entitled "NSLSHL3" which modeled the source in a cylindrical pipe containing radioactive material as a set of line sources. This code is, in effect, an earlier version of the SHL1GG code described in Attachment 2 to this submittal.

The input data required included parameters to characterize the pipe material and dimensions, the target (i.e., observer or equipment) location, and the activities, gamma energies, and decay rates of the radioactive isotopes in the fluid. Because the calculations were concerned only with targets outside the pipe, beta radiation dose was assumed to be negligible when compared with the associated gamma dose. Furthermore, the only output of concern was assumed to be accumulated (integrated) gamma dose as a function of time.

For all calculated cases, the source in the pipe was assumed to be distributed in water; air was assumed to surround each pipe. All pipes were assumed to be made of iron, with no additional shielding present.

In sum, eleven (11) dose calculations for the pipes were made. For cases 1 through 11, the volume of water into which the source was distributed was assumed to be equal to the minimum volume available for recirculation at the time of switchover from the injection to the recirculation phase of a LOCA. This volume was defined as consisting of the difference between the minimum Refueling Water Storage Tank (RWST) content (350,000 gallons) and the RWST low level set point (131,962 gallons), plus the Reactor Coolant System inventory of 93,960 gallons. This total water volume is, therefore, 311,998 gallons.

For cases 1 through 10, the fraction of core radioactive inventory in the water was assumed to be 100% of the noble gases, 50% of the radioiodines, and 1% of the other fission products. Since it was deemed highly unlikely that the noble gases would actually be in the recirculating fluid (rather, they should remain in the containment atmosphere), case 11 was performed with no noble gases present. Since

the input for case 4 and case 11 were identical with the exception of the assumed noble gas source, a base was provided to assess the effect of the noble gas source term. Indeed, the one-year accumulated gamma doses of 10.2×10^6 R and 8.92×10^6 R for cases 4 and 11, respectively, indicated that inclusion of the noble gases in the calculations could provide a conservative margin on the order of 14% to the estimated doses.

The piping systems considered in the calculations involved portions of the Centrifugal Charging (CC), Safety Injection (SI), Residual Heat Removal (RHR), and Containment Spray (CS) systems. These systems are located outside containment and may be expected to carry fluid recirculated from the containment sump. The specific pipe sizes considered are presented in Table 4-1. Bounding cases were run for the largest diameter pipe (14.0" outside diameter) and for the pipes with the largest outside diameter-to-wall thickness ratios (OD/T = 58.28; OD/T = 49.44). All targets were assumed at a location equidistant from the ends of the length of pipe considered.

Cases 1 through 4 were run to test the sensitivity of the results to the choice of radial and azimuthal divisions of the source. A matrix of 20 radial and 36 azimuthal divisions was found to be satisfactory for even the largest diameter pipe. This method of division was therefore used in cases 4 through 11.

Cases 4 through 6 were run to compare piping doses as a function of OD/T. The dose was found to increase as OD/T increased, even though the pipe ODs varied. Apparently the pipe sizes considered were not of prime importance; it appeared as though, for the specific selection of pipes, the most important factor was whether the pipe wall was thin enough to minimize gamma attenuation.

Cases 4, 7, and 8 were run to test the effect of locating the target at various distances from the 14" OD pipe. In particular, these cases assumed the target at the pipe outer surface, 13" from the pipe centerline, and 31" from the pipe centerline. The resultant one-year accumulated doses expressed in MR for these cases were, respectively, 1.02, 0.52, and 0.21.

Cases 4, 6, 9, and 10 were run to assess the effect of an assumed pipe length on the calculations. There was no difference in integrated doses due to a 12.5 ft or a 25.0 ft length of pipe (assuming the target at the pipe outer surface).

In summation, the bounding case found as a result of this work was case 6. This case assumed 100% of the core inventory of noble gases, 50% of the core inventory of radioiodines, and 1% of the core inventory of other fission products to be distributed in approximately 312,000 gallons of water. The assumed pipe dimensions were 8.625" OD, 0.148" wall thickness, and 12.5 ft length. Utilizing 20 radial and 36 azimuthal divisions, a one-year integrated dose of 14.58×10^6 R was calculated at a point on the pipe outer surface, midway between the pipe ends. Multiplying this value by 1.3 to account for daughter sources and

by 0.875 to convert to rads yields a one-year accumulated dose of 16.58×10^6 rads for this bounding case. This value compares favorably to the one-year integrated dose of 12×10^6 rads specified for equipment outside containment (not subject to submersion) in information transmitted to us by Westinghouse Electric Corporation. The integrated dose at various times up to one year for case 6 were also determined to be as follows: at 2 hours, 1.02 Mrads; at 8 hours, 2.62 Mrads; at 1 day, 3.98 Mrads; at 1 month, 12.3 Mrads; at 4 months, 14.79 Mrads; and at one year, 16.58 Mrads.

Application To The Equipment Qualification Process

Based upon the results presented above for case 6, the accumulated gamma dose as a function of time up to one year was applied to various equipment items located outside containment. As an example, if an equipment item had a required operating time of one month, the accumulated gamma dose corresponding to that time frame as determined in case 6 (e.g., 12.3 Mrads) was applied to the item. If the equipment item was then determined to be located at a distance from the nearest pipe, a factor applicable to that distance (i.e., 1.0 for equipment at the pipe outer surface, 0.51 for equipment 6" from the pipe outer surface, and 0.21 for equipment 2 ft from the pipe outer surface) would then be multiplied by the dose determined from case 6 to yield the equipment specified post-accident dose. These correction factors were obtained by comparing the one-year accumulated doses for cases 4, 7, and 8 (see discussion on calculation findings above).

Using this calculational procedure and plant specific installation records, specified doses were obtained for most of the electric equipment of concern outside containment. In general, these equipment items were treated as belonging to any of the four following categories:

a) Cables and cable terminations at instruments and solenoids which are not subject to radiation are treated as category "A" items. Such items include those shown on Unit 1 SCEW sheets CI-4, CI-10, TC-15, TI-9, and TI-10, and Unit 2 SCEW sheets CI-6, CI-10, TC-15, TI-9, and TI-10.

b) Components assumed at the surface of the pipe are category "B" items. Such items include those on Unit 1 SCEW sheets CC-2, CC-3, CC-4, CC-9, CC-10, TC-10, TC-11, TC-13, TP-6, V-6, V-7, and V-11, and Unit 2 SCEW sheets CC-9, CI-12, CI-14, TC-10, TC-11, TC-13, TP-6, V-6, V-7, and V-11.

c) Components in category "C" are those located at distances greater than or equal to 6" from a recirculation line, but which are conservatively assumed to be at a 6" distance. This applies to Unit 1 SCEW sheets CP-1, CP-3, CP-5, CP-10, M-1, M-2, and TP-4, and Unit 2 SCEW sheets CP-1, CP-2, CP-8, CP-13, M-1, M-2, and TP-4.

d) Components in category "D" are those located at distances greater than or equal to 2 ft from a recirculation line, but which are conservatively assumed to be at a 2 ft distance. This applies to Unit 1 SCEW sheets CP-6, TC-14, and V-10, and Unit 2 SCEW sheets CP-7, CP-12, TC-14, and V-10.

Specified radiation doses for these equipment items in categories "A" through "D" are presented in Tables 4-2 (Unit 1) and 4-3 (Unit 2).

Table 4-1

Pipes Considered In
Calculating Outside Containment Doses

System	Designation*	Outside Diameter, OD (inches)	Wall Thickness, T (inches)	OD/T
CC	M-14	4.5	0.438	10.3
CC	M-14	3.5	0.438	7.99
CC	M-14	2.0	0.344	5.81
CC	B-14	6.625	0.134	49.44
CC	B-14	8.625	0.148	58.28
SI	K-14	4.5	0.337	13.35
SI	B-14	4.5	0.12	37.5
SI	B-14	6.625	0.134	49.44
SI	K-14	2.375	0.276	8.61
SI	K-14	1.9	0.2	9.5
SI	K-14	1.05	0.154	6.82
RHR	G-14	14.0	0.438	31.96
RHR	G-14	3.5	0.216	16.20
RHR	G-14	8.625	0.322	26.79
CS	E-14	10.75	0.365	29.45
CS	E-14	8.625	0.322	26.79
CS	E-14	6.625	0.28	23.66
CS	E-14	3.5	0.216	16.20
CS	E-14	1.315	0.133	9.89
CS	B-14	8.625	0.148	58.28
CS	G-14	12.75	0.406	31.40
CS	G-14	2.375	0.154	15.42

*Note: Pipe designation indicates materials of construction, seismic classification, and quality level. For further information, AEPSC specifications should be consulted.

Table 4-2

Donald C. Cook Nuclear Plant Unit No. 1
Specified Radiation Doses Outside Containment

Device	Equipment Description	Specified Operating Time	Specified Rads ($x 10^6$)
CC-2	Continental Cable 3120	1 day	3.98
CC-3	General Electric Cable 3120	1 day	3.98
CC-4	Anaconda Cable 3120	1 day	3.98
CC-9	Continental Cable 3123	1 day	3.98
CC-10	General Electric Cable 3123	1 day	3.98
CI-4	Continental Cable 3075	4 months	Not Applicable
CI-10	Continental Cable 3077	4 months	Not Applicable
CP-1	Okonite Cable 324	1 month	6.40
CP-3	Essex Cable 324	1 month	6.40
CP-5	Anaconda Cable 3102	1 year	8.62
CP-6	Okonite Cable 3102	1 month	2.58
CP-10	Anaconda Cable 3103	1 month	6.40
M-1	CVCS, SI, RHR (Pump Motors)	1 year	8.62
M-2	Containment Spray (Pump Motors)	1 month	6.40
TC-10	Term. At Valve Motor Operator	1 day	3.98
TC-11	Term. At Valve Motor Operator	1 day	3.98
TC-13	Termination At Terminal Block	1 day	3.98
TC-14	Term. At Valve Motor Operator	2 hours	0.21
TC-15	Termination At Solenoid	25 seconds	Not Applicable
TI-9	Term. At Barton Instruments	4 months	Not Applicable
TI-10	Term. At Foxboro Instruments	4 months	Not Applicable
TP-4	Termination At Pump Motor	1 year	8.62
TP-6	Termination At Valve Motor	1 day	3.98
V-6	Valve Motor Operator	1 day	3.98
V-7	Valve Motor Operator	1 day	3.98
V-10	Valve Motor Operator	2 hours	0.21
V-11	Valve Motor Operator	1 day	3.98

Table 4-3

Donald C. Cook Nuclear Plant Unit No. 2
Specified Radiation Doses Outside Containment

Device	Equipment Description	Specified Operating Time	Specified Rads ($x 10^6$)
CC-9	Continental Cable 3123	1 day	3.98
CI-6	Continental Cable 3075	4 months	Not Applicable
CI-10	Continental Cable 3077	4 months	Not Applicable
CI-12	Raychem Cable 3111	1 day	3.98
CI-14	Continental Cable 3069	1 day	3.98
CP-1	Essex Cable 324	1 month	6.40
CP-2	Cyprus Cable 324	1 month	6.40
CP-7	Cyprus Cable 3102	1 month	2.58
CP-8	Okonite Cable 3102	1 year	8.62
CP-12	Anaconda Cable 3102	1 month	2.58
CP-13	Anaconda Cable 3103	1 month	6.40
M-1	CVCS, SI, RHR (Pump Motors)	1 year	8.62
M-2	Containment Spray (Pump Motors)	1 month	6.40
TC-10	Term. At Valve Motor Operator	1 day	3.98
TC-11	Term. At Valve Motor Operator	1 day	3.98
TC-13	Termination At Terminal Block	1 day	3.98
TC-14	Term. At Valve Limit Switch	2 hours	0.21
TC-15	Termination At Solenoid	5 seconds	Not Applicable
TI-9	Term. At Barton Instruments	4 months	Not Applicable
TI-10	Term. At Foxboro Instruments	4 months	Not Applicable
TP-4	Termination At Pump Motor	1 year	8.62
TP-6	Termination At Valve Motor	1 day	3.98
V-6	Valve Motor Operator	1 day	3.98
V-7	Valve Motor Operator	1 day	3.98
V-10	Valve Motor Operator	2 hours	0.21
V-11	Valve Motor Operator	1 day	3.98