RADIATION PHYSICS AND SHIELDING CODES AND ANALYSES APPLIED TO DESIGN-ASSIST AND SAFETY ANALYSES OF CANDU® AND ACR™ REACTORS

by

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1. ABSTRACT

This paper discusses the radiation physics and shielding codes and analyses applied in the design of CANDU and ACR reactors. The focus is on the types of analyses undertaken rather than the inputs supplied to the engineering disciplines. Nevertheless, the discussion does show how these analyses contribute to the engineering design.

Analyses in radiation physics and shielding can be categorized as either design-assist or safety and licensing (accident) analyses. Many of the analyses undertaken are designated "design-assist" where the analyses are used to generate recommendations that directly influence plant design. These recommendations are directed at mitigating or reducing the radiation hazard of the nuclear power plant with engineered systems and components. Thus the analyses serve a primary safety function by ensuring the plant can be operated with acceptable radiation hazards to the workers and public.

In addition to this role of design assist, radiation physics and shielding codes are also deployed in safety and licensing assessments of the consequences of radioactive releases of gaseous and liquid effluents during normal operation and gaseous effluents following accidents. In the latter category, the final consequences of accident sequences, expressed in terms of radiation dose to members of the public, and inputs to accident analysis, e.g., decay heat in fuel following a loss-of-coolant accident, are also calculated.

Another role of the analyses is to demonstrate that the design of the plant satisfies the principle of ALARA (as low as reasonably achievable) radiation doses. This principle is applied throughout the design process to minimize worker and public doses. The principle of ALARA is an inherent part of all design-assist recommendations and safety and licensing assessments.

The main focus of an ALARA exercise at the design stage is to minimize the radiation hazards at the source. This exploits material selection and impurity specifications and relies heavily on

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experience and engineering judgement, consistent with the ALARA philosophy. Special care is taken to ensure that the best estimate dose rates are used to the extent possible when applying ALARA.

Provisions for safeguards equipment are made throughout the fuel-handling route in CANDU and ACR reactors. For example, the fuel bundle counters rely on the decay gammas from the fission products in spent-fuel bundles to record the number of fuel movements. The International Atomic Energy Agency (IAEA) Safeguards system for CANDU and ACR reactors is based on item (fuel bundle) accounting. It involves a combination of IAEA inspection with containment and surveillance, and continuous unattended monitoring. The spent fuel bundle counter monitors spent fuel bundles as they are transferred from the fuelling machine to the spent fuel bay. The shielding and dose-rate analysis need to be carried out so that the bundle counter functions properly.

This paper includes two codes used in criticality safety analyses. Criticality safety is a unique phenomenon and codes that address criticality issues will demand specific validations. However, it is recognised that some of the codes used in radiation physics will also be used in criticality safety assessments.

2. RADIATION PHYSICS ANALYSES

2.1 Design Assist

Generally in designing a CANDU reactor, design-assist analyses can be classified into four categories:

1. Radiation Protection and Shielding (e.g., contained sources, access control and zoning, primary, secondary, auxiliary and special shielding),
2. Nuclear Heating (e.g., the moderator and shield cooling system heat loads, station heat balance),
3. Radiation Effects (e.g., radiation damage, radiolysis, environmental qualification of safety-related equipment), and
4. Radiation Detection and Monitoring (e.g., in-core and out-of-core instrumentation, effluent monitoring).

Within each of these categories, different types of analyses are undertaken using, primarily, third-party computer codes, and these are discussed in the paper.

The design-assist work provides input to many engineering disciplines. The guidance and inputs supplied to these engineering disciplines usually demand a sequence of calculations. For example, the thickness of the concrete walls around the fuelling machine vaults in CANDU units can only be specified after the dose rates from the nitrogen-16 sources in the end fittings, feeders and headers have been calculated. However, the irradiated fuel in the fuelling machine and end fittings and radiation escaping from the core through the end shield will also make contributions to the dose rate through that concrete. Thus calculations of energy spectra and intensity of the three different sources (induced activities in the coolant, the reactor core and the spent fuel during refuelling operations) must be undertaken before the transport of radiation from these sources through the concrete can be assessed. Consequently, six different pieces of analysis (three radiation-source calculations and three radiation-transport calculations) are necessary to specify one wall thickness.

2.2 Safety and Licensing

The main role of radiation physics and shielding in safety and licensing is the prediction of radiation doses to members of the public living at or outside the exclusion area boundary. These projected radiation doses are calculated during normal operation from gaseous-effluent releases and liquid-effluent discharges from the plant and gaseous-effluent releases[1] following a postulated accident. Radiation exposures of plant workers following an accident are also required.

Worker dose following a postulated accident is addressed in a post-LOCA habitability study. The analysis calculates the exposures to workers from external radiation from fission-product sources accumulated outside the reactor building. These sources include the Emergency Core Cooling (ECC) heat exchangers and ECC piping, the Reactor Building Ventilation System (RBVS) exhaust, upstream of the isolation damper, the airlocks and any leakage from containment.

The calculation of doses to members of the public following a postulated accident is limited to the calculation of the dose from external radiation emitted by the cloud of contaminants, the dose from inhalation of the contaminants, and the dose from contaminants deposited on the ground as the cloud passes. These calculations demand a complete description of the releases in terms of timing of releases, energy spectra of the resulting radiation, location of releases and amount of releases. These details are used in combination with the weather data for a specific site to assess the consequences of an accident.

2.3 ALARA

The objectives of ALARA (social and economic factors taken into consideration) are first applied at the design stage, since this is the stage that the most cost-effective changes can be made to reduce radiation exposures. Dose reduction initiatives continue during the operation stage via an active ALARA program that reviews dose reductions measures during operation, maintenance and repair activities.

2.4 Safeguards

In addition to these safety-and-licensing analyses, some analyses support the engineering features installed to support reactor safeguards to show compliance with the international non-

[1] For the design-basis accidents, most of the releases are airborne. Liquids released from the heat transport and moderator systems are usually contained within the reactor building. Potential doses for the members of the public from this pathway will be small compared with doses from airborne releases."

proliferation treaty. These analyses include the development of fuel-bundle counters and similar devices to monitor the compliance of other signatory countries to which AECL has sold CANDU reactors.

2.5 Criticality Safety

The slightly enriched uranium in the fuel bundles of ACR introduces a criticality hazard throughout the fuel-handling route. The analyses undertaken to date have assessed the design of the fresh and spent fuel storage systems, to avoid a criticality safety hazard.

3. RADIATION PHYSICS AND SHIELDING CODES

The radiation physics and shielding codes used by AECL are, primarily, third-party codes developed by the worldwide nuclear industry and distributed by the Radiation Safety Information and Computational Center (RSICC) at Oak Ridge National Laboratory (ORNL). The codes reflect many years of development by several organizations and the application of the codes to a variety of problems. At AECL, the radiation-physics and shielding codes can be categorized into four types:

1. Characterizing radiation source

Computer codes include ORIGEN-S [1], which is part of the SCALE 4.4 code package and MATLAB [2].

2. Modelling the transport of radiation, including point-kernel approximations

Computer codes include ANISN [3], DORT [4] and TORT [5], which are part of the DOORS code package, and MCNP [6], QAD-CGGP-A [7], and MICROSHIELD [8].

3. Assessing doses (on-site and off-site) from effluents released.

Computer codes include ADDAM [9], XOQDOQ [10] and IMPACT [11]. The methodologies in codes like PAVAN [12], ARCON96 [14], RADTRAD [15] and HABIT [16] have been used for CANDU 6 reactors. IMPACT has been used for analyses of public dose from radionuclide releases in gaseous and liquid effluent pathways for ACR.

4. Analyzing configurations of fissile material for criticality safety.

Computer codes KENO Va [17] and MCNP are used in assessments of slightly enriched-uranium fuel bundles.

Most of the codes are run on HP/UNIX machines but some PC versions of the codes have been used. In the future, standardization of the codes to operate on PC/LINUX machines will become a feature of configuration management.

4. APPLICATION EXAMPLES

This section shows three examples of how the codes are applied to design-assist and safety-analysis questions, and elaborates on what remedies are used when the capability of the code is challenged for a given problem. Most of the time multiple computer codes have to be used to tackle a given problem, streamlined methods are needed for preparing inputs to avoid unnecessary time spent and interfacing codes are also needed to speed up analyses.

4.1 Analysis of CANDU 6 End Shields

The CANDU 6 end shields are approximately 6.8-m-diameter discs at either end of the calandria. Each end shield has a 0.787-m region of carbon-steel balls and light water sandwiched between a 50.8-mm-thick calandria side tubesheet and a 76.2-mm-thick fuelling-machine tubesheet, a total thickness of 0.914 m.

There are 380 fuel-channel penetrations in each end shield to allow primary coolant to flow through the end shield to the core and permit refuelling operations. The fuel-channel penetrations are arranged on a 285.8-mm square-lattice pitch of the reactor core and are formed by thin-walled lattice tubes. The fuel-channel penetration is shown in Figure 1.

The penetrations so formed are plugged by stainless steel end fittings that are thick-walled tubes, the liner tube, the shield plug, and finally a closure plug. Except for a narrow gas annulus between the lattice tube and the end fitting, the remaining volumes are filled with heavy-water coolant.

It is important to note that the coolant feeder pipes were ignored in the analysis. These feeder pipes are routed in the gaps between the end-fittings outboard of the end shield. They are clustered more deeply (i.e., closer to the fuelling machine tubesheet) away from the reactor axis. These feeder pipes have a significant attenuation on radiation leaving the fuelling machine tubesheet. Near the reactor axis they are less deeply clustered and their shielding effect is small.

4.1.1 DORT Analysis

The repeating array of the CANDU 6 fuel channels allows the analysis to model one channel in a square lattice and simulate the effect of the adjacent channels by using reflective boundary conditions.

Using the two-dimensional discrete-ordinates code DORT, the fuel channel can be modelled in a two-dimensional RZ geometry with R representing the radial distance from the channel axis and Z representing the axial distance along the channel. An equivalent radius of 161.3 mm can be used in the model to represent the square lattice.

The analysis is run in the fixed-source mode with a fission neutron/photon source defined in the end two bundles of a fuel channel. The code will be run with in-house multi-group coupled neutron (46 groups) and photon (20 groups) cross sections to calculate prompt neutron and gamma, plus fission-product decay gamma, flux distributions throughout the model.

An example of the three-group neutron fluxes (fast flux, $E_n \geq 0.82$ MeV, intermediate flux, 0.82 MeV $> E_n >$ 0.414 eV and thermal flux, $E_n \leq 0.414$ eV) calculated by DORT is plotted in Figure 2 through the end shield along the lattice-cell edge.

The neutron fluxes throughout the end shield can be used to calculate the cobalt activation reaction rates in all meshes using the response functions embedded in the in-house 66-group coupled cross-section library as pseudo material cross sections.

Dose rates outside the end shield are dominated by the ^{60}Co build-up over the operating time of the reactor, as cobalt is present in the steel components of the end shield as impurities. Based on the cobalt activation reaction rates, a separate DORT analysis is run, inputting two gammas of energies 1.17 MeV and 1.33 MeV, for each activation reaction rate, as fixed sources. The reaction rate distribution is used to simulate the distribution of decay gammas and to calculate the dose-rate profile at the fuelling machine tube sheet and at the end plane of the end fittings.

4.1.2 Activation Analysis of Fuel-Channel Components

Results of the activation dose rates can be used to characterize the radioactive wastes from the CANDU 6 reactors during refurbishment. The three-group neutron fluxes from the DORT code can be used in an ORIGEN-S calculation to evaluate the activation gamma source that includes ^{60}Co. The gamma spectra calculated by the ORIGEN-S code can be then input into the QAD-CGGP-A code to determine the shielding requirements for the flasks that hold the fuel-channel components, or to calculate the unshielded dose rates of the fuel channel components, e.g., end fittings, shield plugs, liner tubes, calandria tube inserts, calandria tubes, and pressure tubes.

If the ^{60}Co activity in the steel materials of the end-shield components is the dominant radiation source, then the gamma-source distribution can be calculated using the DORT code, i.e., ORIGEN-S is not used.

4.2 Specific Activities of Coolant and Moderator at Core Exit

In a CANDU reactor, radioactive species that are of importance in the coolant and moderator are the ^{16}N, ^{19}O, ^{14}C, ^{3}H, ^{17}F, and ^{18}F activities. These activities can be estimated based on the Primary Heat Transport (PHT) System and moderator inventory and circuit flow rates. The following provides a brief description of the activity calculations for ^{16}N, ^{19}O, ^{17}F and ^{18}F as examples. These activation products, and the corrosion and fission products are considered in shielding and dose-rate calculations.

The MCNP code can be used to calculate directly the reaction rates leading to the production of ^{16}N, ^{14}C and $^{2}H(n,\gamma)^{3}H$ production of ^{3}H. Neutron fluxes can be tallied in the run to calculate ^{19}O and ^{17}F activities. The calculation can be done in an infinite lattice cell model solving the system multiplication factor in a normal operating condition. For the PHT and Moderator Systems, the core-exit activities are calculated using equations for a circulating system given in Section 2 of Reference 18.

The ^{16}N is produced by the $^{16}O(n,p)^{16}N$ reaction, while ^{19}O is produced by $^{18}O(n,\gamma)^{19}O$ reaction. The circuit times of ^{16}N and ^{19}O inside the PHT and moderator systems are short and comparable to their half-lives, 7.1 seconds and 26.9 seconds, respectively; these are used to account for the buildup and decay of the activities over each cycle in the reactor.

The calculation of ^{17}F activities is done analytically since it is generated by $^{16}O(d,n)^{17}F$ reactions. To calculate the ^{17}F activities, one needs to calculate the deuteron fluxes that originate from fission neutrons scattering in D_2O. In a heavy-water reactor, the neutrons from fission scatter elastically from the heavy-water molecules. The energy given up by the neutron is much greater than the chemical-bond energy holding the deuterium atom or deuteron in the heavy water molecule. Thus the deuteron recoils from the molecule to create a deuteron flux. Only the fission neutrons that have a minimum energy of 2.23 MeV (the binding energy of the neutron and proton in the deuteron nucleus) were considered. The 19-energy group fast-neutron fluxes, the deuteron energies, the group-averaged deuteron cross sections and the reaction rate cross sections were used to calculate the total ^{17}F production rate in the D_2O coolant and moderator. The core-exit activities were calculated [18] using the reaction rates, core-transit and total circuit times (see Table 1).

Likewise, the calculation of ^{18}F is also done analytically since it is generated in the H_2O coolant of ACR by $^{18}O(p,n)^{18}F$ reaction. Proton fluxes in the core originate from elastic scatters of fission neutrons with the hydrogen atoms. The threshold energy for proton-generating scatter reactions is 2.5 MeV.

An example of the calculated coolant and moderator activities for the gamma-emitting radionuclides are summarized in Table 1.

Table 1 Typical Coolant and Moderator Activities at Core Exit for ACR

Reaction	H_2O coolant (core exit) Bq/g	D_2O moderator (core exit) Bq/g
$^{16}O(n,p)^{16}N$	2.54E+06	1.19E+07
$^{18}O(n,\gamma)^{19}O$	1.00E+05	2.75E+06
$^{16}O(d,n)^{17}F$	-	1.31E+06
$^{18}O(p,n)^{18}F$	2.44E+04	-

4.3 Environmental Qualification of Equipment against Radiation Damage

During reactor operation, the dose rates received by components are dictated by the penetrating gamma radiation from the reactor core, induced activities in the coolant and moderator heavy water (e.g., ^{16}N, ^{19}O, ^{17}F for CANDU and ACR and ^{18}F for ACR), or spent fuel bundles in the fuelling machine during its operation. The deposited corrosion and fission-product activities also contribute to dose rates around process-system components.

During accident conditions, the major radiation source that contributes to dose received by exposed components is the gamma and beta radiation emitted by fission products that are released into containment in reactor accident scenarios.

The assessment of radiation fields for environmental qualification purposes is normally based only on the absorbed dose from gamma radiation. For many components the gamma dose will be more important, because of the short range of the beta particles. For example, the mechanical equipment such as pumps and valves will not be affected by beta radiation. Electrical equipment is also generally installed in metal casing or cover, which effectively stops beta particle fluxes. Although the environmental qualification dose is usually dictated by gamma radiation, beta radiation may also become important for some components, e.g., airlock door seals, epoxy liner, gaskets in plate-type ECC heat exchangers, electrical components that are not encased in protective metal casing, and can be directly exposed to fission products. The harsh environmental conditions were taken to be a loss-of-coolant accident with emergency core cooling impaired.

The dose rates during normal operation are normally obtained from calculations using computer codes ANISN, DORT and QAD-CGGP-A, and from analytical methods and station measurements.

For the accident scenarios, the isotope generation and depletion code ORIGEN-S can be used to calculate the gamma-source terms due to fission products released from the fuel following a Loss of Coolant Accident with Loss of Emergency Core Cooling System (LOCA + LOECC) scenario. Note that no fission product transport calculations are done using ORIGEN-S. The code was used to calculate gamma spectra as a function of time using the inventory of fission products released into containment after an accident. ORIGEN-S results were also used with an in-house code to calculate the fission-product decay beta spectra in the air and water phases.

The point kernel integration code QAD-CGGP-A can be used to calculate the gamma dose received at many dose-point locations in the reactor and service buildings. The beta dose inside containment can be calculated by MCNP for various components, e.g., electrical junction box, epoxy liner of containment (CANDU 6 reactor). The beta doses are generally calculated for activity released into the air inside containment, from water film that can be present on equipment surfaces, and epoxy liner that is immersed in water.

Figure 3 shows an example of the gamma and beta dose inside the junction box after a LOCA+LOECC of a CANDU 6 reactor. The gamma dose was calculated by the QAD-CGGP-A code. The beta dose was calculated by the MCNP code. The junction box was modelled as an enclosed box with a water droplet at the centre for dose tally. The box was 22.5 cm x 27.6 cm x 12.7 cm in size to maximize the radiation source inside the box.

Figure 4 shows an example of the MCNP model that can be used to calculate beta dose in an epoxy liner. The source consists of the contaminated air, and a water film that may form on surfaces above a flood level. To simulate this scenario, an infinitely long, finite thickness water film can be modelled on the surface of the epoxy liner, which is 0.6 cm thick. Two water-film thicknesses can be modelled to see the effect of dose received by beta radiation. Figure 5 shows relative beta-energy deposition in the epoxy liner with a 2-mm-thick water film as a function of beta energies. The *F8 tally in MCNP was used to calculate energy depositions in tally cells and overall thickness of the material, e.g., 0.6-mm-thick epoxy liner. The MCNP results were normalized to the actual beta spectra from BETA-S calculations to calculate the beta dose as a function of time after the accident.

4.4 Other Applications

Computer codes like ORIGEN-S and QAD-CGGP-A can be used in other applications like designing fuel-bundle counters for safeguards, estimating dose rates for post-LOCA habitability, and atmospheric dispersion and dose calculations for the main control room (MCR). At AECL, the codes MCNP and KENO Va are also being used to assess the criticality conditions of different fissile assemblies.

4.5 Summary

Sections 4.1 to 4.3 gave specific examples of how various radiation-shielding codes are applied and the sequence of their application. Examples included the shielding assessment of the primary end shields using a two-dimensional discrete ordinates codes DORT, and the calculation of induced activities of ^{16}N, ^{19}O, ^{17}F, and ^{18}F in the process-fluids using MCNP for the ACR. Other examples include the assessment of gamma and beta dose received after an accident using the ORIGEN-S, QAD-CGGP-A and MCNP codes. These codes were also used for Point Lepreau Generating Station (PLGS) refurbishment.

5. CONCLUSION

This paper presents examples of where and how radiation physics and shielding codes are deployed and focuses on the types of analyses undertaken, and shows how these analyses contribute to the engineering design.

The radiation physics and shielding engineers and scientists in AECL manage to address all design-assist and safety- and licensing-related questions, recognizing the capabilities and limitations of the codes used.

6. REFERENCES

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Figure 1 CANDU 6 Fuel Channel Assembly

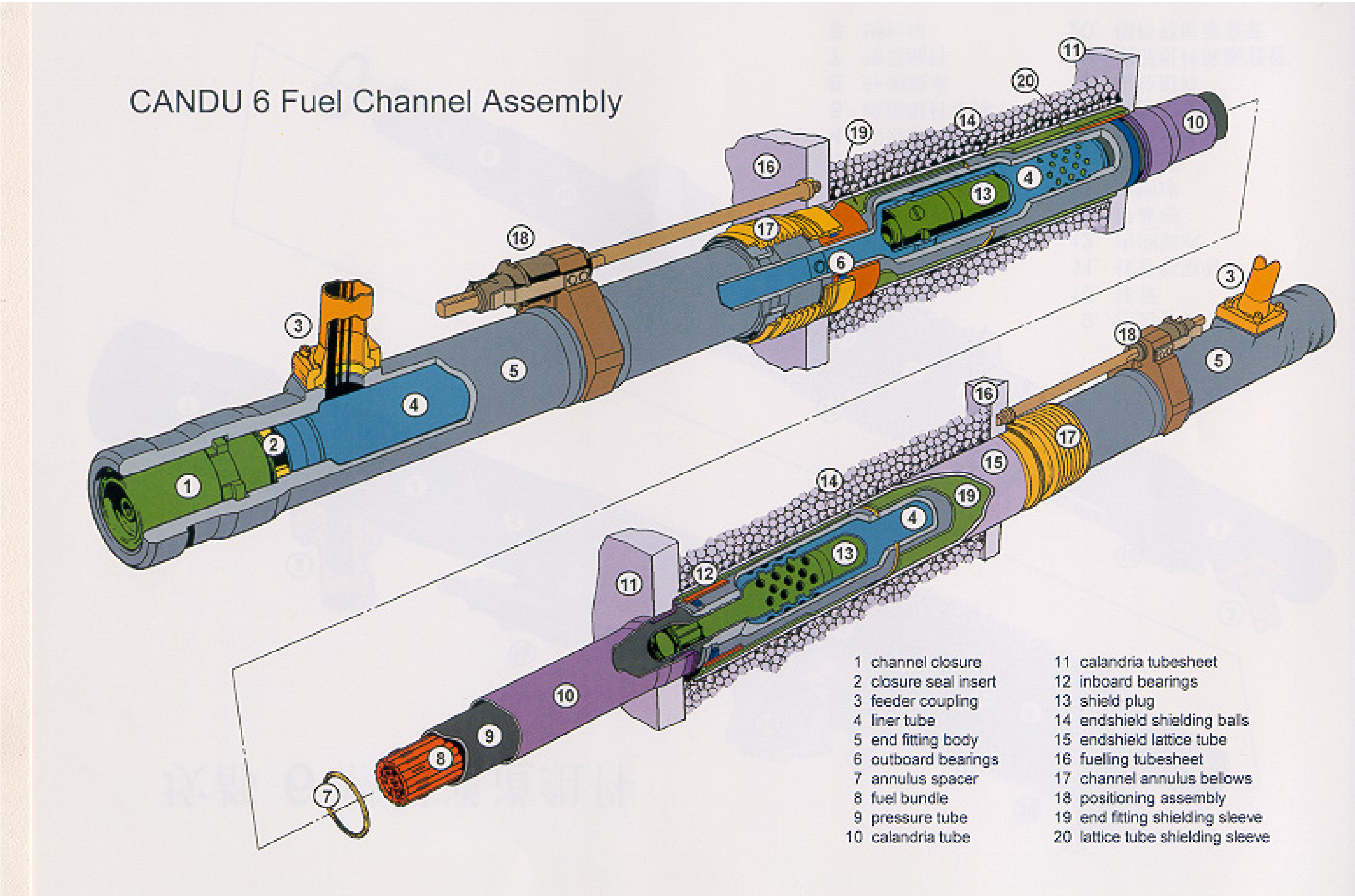

Figure 2 Neutron Fluxes Through CANDU 6 End Shield

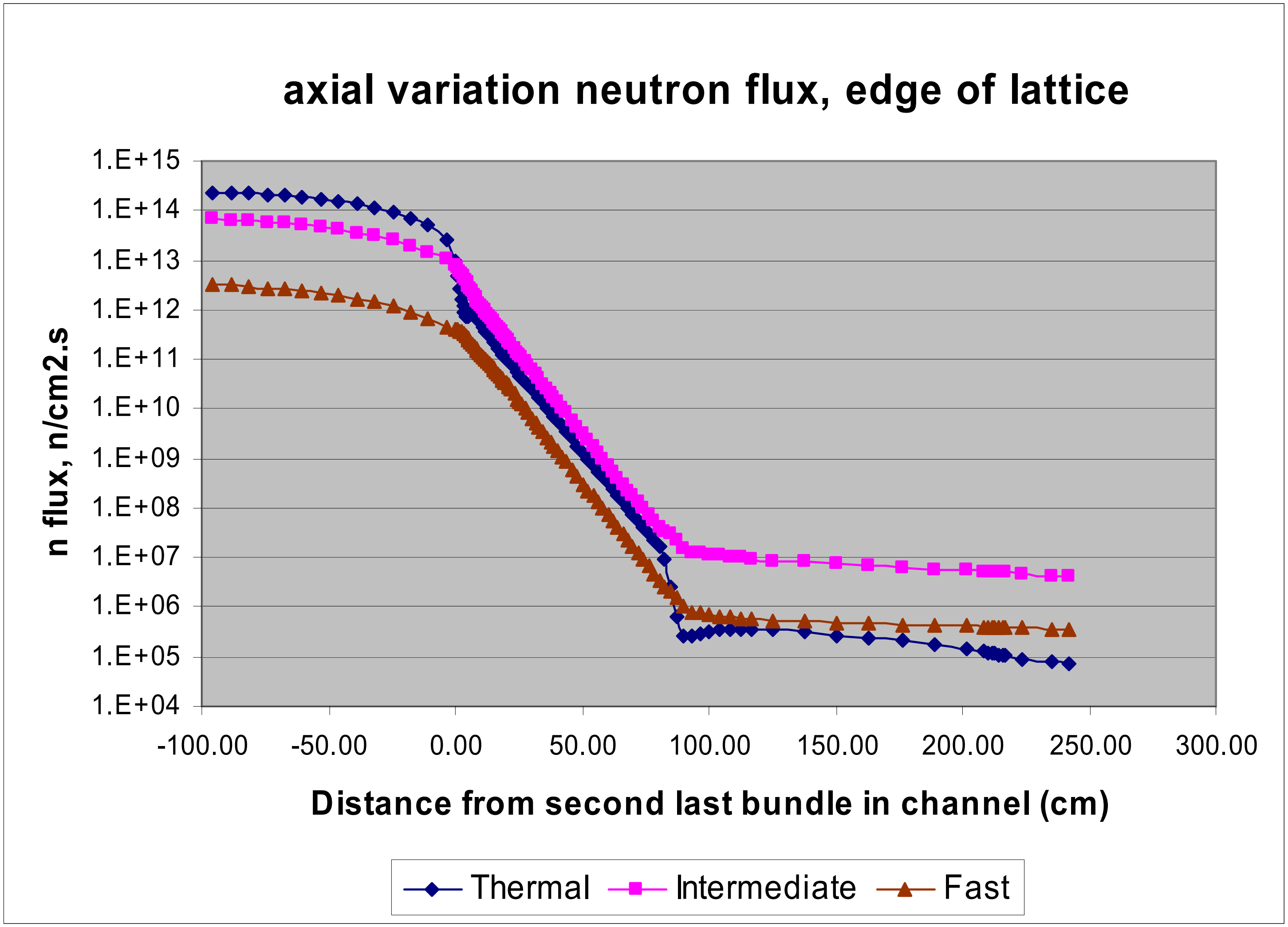

Figure 3 Radiation Dose Received by Junction Box Inside Containment after a LOCA+LOECC (CANDU 6)

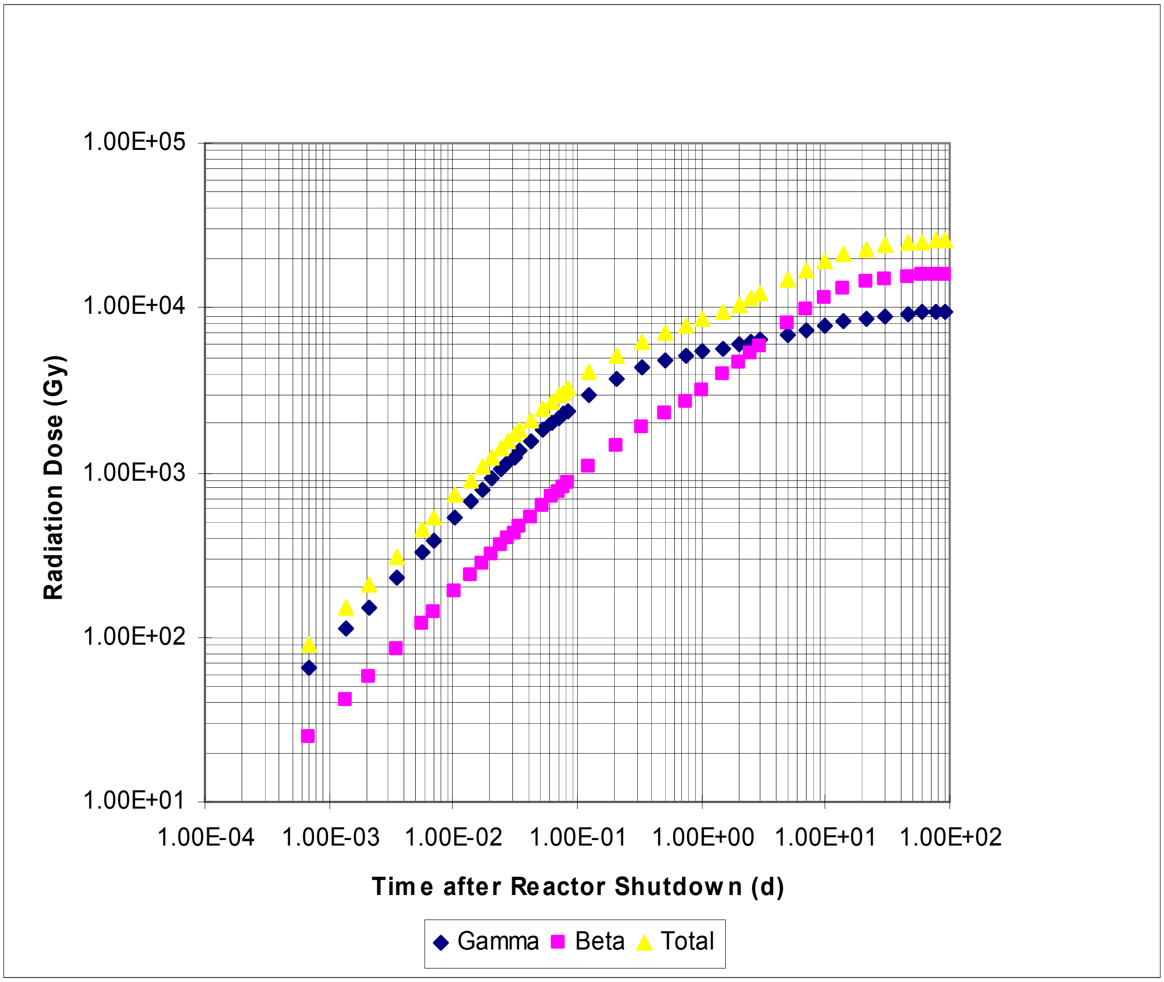

Figure 4 MCNP Model of Water Film on Epoxy Liner of CANDU 6 Containment Wall

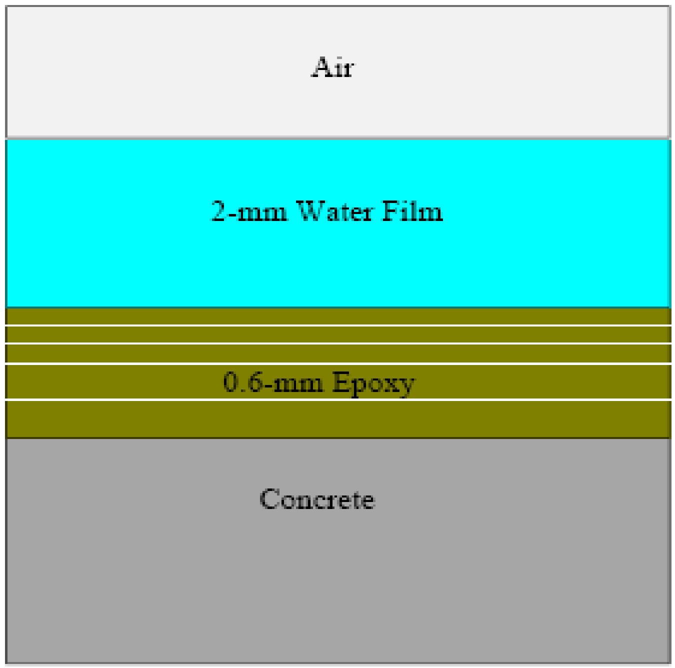

Figure 5 Beta-Energy Deposition in Epoxy Liner and Concrete by a 2-mm-thick Water Film on Liner Surface after a LOCA+LOECC Scenario for CANDU 6