20th Annual Conference of the Canadian Nuclear Society
Montreal, Quebec, Canada / May 30 - June 2, 1999

SIGNIFICANCE OF BETA AND GAMMA DOSE ON ENVIRONMENTAL QUALIFICATION OF COMPONENTS

by

K. M. Aydogdu and K.T. Tsang

Atomic Energy of Canada Limited
Sheridan Science and Technology Park
Mississauga, Ontario
L5K 1B2

ABSTRACT

Safety-related systems and components that are required to perform safety functions during accident conditions must be designed to withstand the harsh environmental conditions that occur as a consequence of the accident. Where these conditions are "harsh", and equipment operability can potentially be affected by the post-accident environment, environmental qualification of the equipment must be conducted to demonstrate that the required safety function can be maintained. It is also understood that non-safety-related equipment that affects, or prevents, the satisfactory operation of a safety-related system should also withstand the "harsh" environmental conditions caused by an appropriate design-basis accident.

There are essentially two types of requirements that must be satisfied to qualify equipment or components to withstand radiation damage, namely economic requirements and safety requirements. The general objective of the economic requirement is to reduce maintenance cost and to maximize component life during reactor operation. The general objective of the safety requirement is that the equipment should be qualified to withstand the harsh post-accident environmental conditions and should function properly for the appropriate length of time after a design-basis accident has occurred.

To address the economic factors - i.e., to reduce maintenance costs and to maximize component life - the radiation dose rates to equipment are calculated throughout the reactor building and the service building during reactor operation. These are also used for the safety requirement purpose, to assess radiation ageing of safety-related components caused by degradation of material properties with time at radiation exposure.

To address the safety requirement, the dose-rate estimates and accumulated doses after a LOCA coincident with loss-of-emergency-core cooling (LOECC) are provided.

The harsh post-accident environmental conditions defined for environmental qualification of components include conditions that deviate from those established for normal operation and those that could adversely affect the components, such as temperature, pressure, radiation, humidity etc. This paper deals with the radiation aspect for environmental qualification, and thus presents methods used to calculate dose rates received by various components for a typical CANDU® 6 reactor during normal reactor operation and after a LOCA-LOECC scenario.

---

1. INTRODUCTION

During reactor operation, the dose rates received by components and equipment are dictated mainly 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) 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.

The gamma and beta radiation emitted by fission products that are released into containment in accident scenarios constitute the major radiation source that contributes to the dose received by exposed components during accident conditions.

The assessment of radiation fields for environmental qualification purposes are normally based only on the absorbed dose from gamma radiation. For many components, the gamma radiation will be more dominant than beta radiation because of the short range of the beta particles. For example, the mechanical equipment - such as pumps and valves - is not expected to be affected by beta radiation. Electrical equipment is also generally installed in metal casings or is covered, both of which effectively attenuate beta particles.

Although the environmental qualification dose is usually dictated by gamma radiation, beta radiation may also become important for some components - such as airlock door seals, epoxy liner, gaskets in plate-type ECC heat exchangers, electrical components that are not encased in protective metal casings, etc. - and may therefore be directly exposed to fission products. It should be recognized, however, that gamma radiation is more penetrating than beta radiation, and thus results in a uniform dose to be received by the component, whereas beta radiation is essentially received by a few-millimetres-thick layer of that component. Nevertheless, for radiation-ageing purposes, the total integrated dose (TID) must include both the gamma and beta radiation doses for accident conditions and the gamma dose during normal operation. Thus, the beta radiation damage to equipment must be assessed on a case-by-case basis.

The dose-rate distributions throughout the reactor and service building have been generated during normal operation and after a loss-of-coolant accident (LOCA) coincident with a total loss of loss-of-emergency-core cooling (LOECC), and this information was used to calculate dose received at selected components that could potentially be exposed to beta radiation, i.e., airlock door seals, epoxy liner of containment, electrical junction boxes, cables and gaskets of the plate-type ECC heat exchangers. The total dose received by these components has been evaluated, and the methodology used for the dose calculations is briefly described.

2. SOURCES OF RADIATION DURING REACTOR OPERATION AND AFTER ACCIDENTS

The reactor building contains areas and rooms which are inaccessible during reactor operation because of high radiation fields from the operating core, fuelling machine, heat transport system (HTS) and moderator system components. Some areas have a restricted access, e.g., reactivity mechanism deck, and areas that are accessible at all times, such as airlocks and lobby areas. Other areas receive radiation from scatter or shine because of operating equipment, e.g., dousing tank, cat walks, etc. The induced activity in the D_2O coolant of the HTS and moderator system results in dose rates around the associated equipment that is normally shielded.

The fuelling machine vaults and their maintenance locks and the moderator room receive radiation from the operating reactor, the HTS components, the moderator system heat exchangers and pumps and piping. Therefore, the gamma and neutron fields are high in these areas during reactor operation, and any equipment located here must withstand high radiation doses during its lifetime.

Elsewhere in the reactor building, there are rooms where some of the auxiliary system components are located, e.g., the HTS auxiliaries including its purification system, the delay tank of the liquid zone control system and fuelling machine auxiliaries. The radiation fields in these rooms are dictated by the induced activity and deposited corrosion and fission-product activities.

The service building contains conventional service facilities - such as stores, workshops, change rooms, decontamination centre and laboratories as well as pieces of equipment for many auxiliary systems -, most of which are located in the service building basement. The ECC system components, including the ECC heat exchangers are located in areas with low (ambient) radiation fields during reactor operation.

The gamma and beta radiation emitted by fission products that are released into containment in accident scenarios constitute the major radiation source that contributes to the dose received by exposed components during accident conditions.

3. METHODOLOGY USED TO CALCULATE NORMAL OPERATION DOSE

ANISN[1] was used to calculate gamma and neutron dose rates from the operating reactor core at areas outside the radial shield systems, e.g., in accessible and inaccessible areas of the reactor building at power (e.g., at airlock doors that are located in accessible areas; epoxy liner of containment); similarly the DOT 4.2[2] code was used to calculate the corresponding dose rates in the fuelling machine vaults (e.g., epoxy liner, fuelling machine components, local air coolers). The QAD-CGGP-A[3] code was used to calculate dose rates from the fuelling machine loaded with spent fuel during its duty cycle.

Analytical techniques, the ANISN and QAD-CGGP-A codes, were also employed to calculate gamma dose rates from induced activities in the HTS and moderator system (e.g., valves, pump motors, epoxy liner of containment, local air coolers). ANISN or other analytical techniques were also used to calculate photoneutron dose rates around piping for these systems.

The gamma and neutron (where applicable) dose rates during reactor operation were established in every room of the reactor building of a typical CANDU 6 reactor, by combining the calculated dose rates with those from radiation surveys, where available. The dose rates received at the locations chosen for this study were obtained from that analysis.

4. METHODOLOGY AND ASSUMPTIONS USED TO CALCULATE ACCIDENT DOSE

The design-basis accident for the analysis of radiation fields for environmental qualification of components is a large-break LOCA, coincident with a total LOECC. An inventory of fission products was developed by assuming none of the three stages of the ECC (high, medium and low pressure) are initiated. This scenario generates the largest release of fission products of any design-basis event in a CANDU 6 reactor.

The LOCA-LOECC scenario does credit operation of the dousing system so that most of the radioiodines escaping from the break would be washed out of the containment atmosphere to the water pool created at the bottom of the containment. Only those fission products that remain volatile during transport along the fuel channels of the broken loop, through the end-fittings and along the length of the feeder pipes to a broken header would escape to containment and reach the water pool. Many of the semi-volatile and non-volatile fission products that escape from the fuel would be trapped on the surfaces of the HTS. These fission products do not contribute to the environmental qualification dose after an accident.

The source term is calculated for a total loss of ECC, and gamma and beta doses for environmental qualification assume that the waterborne fission products are uniformly distributed in the coolant volume of

one HTS loop, plus the volume of the water stored in the dousing tank (minus the ECC portion). For qualification of the ECC heat-exchanger gasket, it is postulated that an operator would perform manual initiation of at least low-pressure ECC at some time after the accident. That would allow the fission products trapped on the surfaces of the HTS components to be washed to the floor. This action by the operator would also transport fission products into the ECC heat exchangers. Nevertheless, this extra dilution that occurs because of that the low-pressure ECC water, was not credited.

The analysis of accident doses is divided into two sections, one to calculate the source terms from fission products released into containment after the accident and the other to calculate the gamma and beta doses resulting from releases into the air and water phases.

4.1 Source-term Calculations and Assumptions

The isotope generation and depletion code ORIGEN-S[4] was used in the fuel re-processing mode to calculate the source terms that are due to fission products released from the fuel after a LOCA-LOECC scenario. The BETA-S[5] code was run together with ORIGEN-S to calculate the fission-product decay beta spectra of the fission products that are in the air and water phases. The corresponding fission-product decay gamma spectra were calculated by the ORIGEN-S code.

Hundreds of radioactive fission products and actinides are produced in a natural- uranium fuel after irradiation in the reactor core. This number involves approximately 46 elements. These elements can be itemized in nine categories, as shown below. The categories group the elements as proposed in Reference 6 and the categories comprise elements whose chemical form has similar properties and exhibit similar behaviour.

i) Halogens (I, Br)

ii) Noble Gases (Xe, Kr)

iii) Alkali Metals (Rb, Cs)

iv) Alkali Earth Metals (Sr, Ba)

v) Tellurium Group (Te, Se, Sb, As)

vi) Zinc Group (Zn, Cd)

vii) Aluminum Group (In, Ga)

viii) Noble Metals (Ge, Mo, Tc, Ru, Rh, Pd, Ag, Sn)

ix) Rare-earth Metals (Y, Zr, Nb, La, Ce, Pr, Nd, Pm, Sm, Eu, Gd, Tb, Dy, Ho, Th, Pa, U, Np, Pu, Am, Cm).

The safety analysis calculations shows that the total release for ^{131}I is about 5.0×10^5 TBq, i.e., ~22% of the total core inventory for ^{131}I after a LOCA-LOECC scenario. The same percentage was also assumed for all iodine and bromine radionuclides, group (i) and all noble gases, group (ii). Although it takes 2400 s for the release of this inventory, the analysis assumes the activity was instantaneously released at time zero.

Before calculating the inventories of the fission products in groups (iii) to (ix), the relative inventories, half-lives, gamma yields given in ORIGEN-S, and the release fractions involving accidents with fuel melting given in References 6 and 7 were reviewed. Only 18 elements were considered to be significant from the radiation dose-rate point of view. These elements are shown in Table 1.

The next step is to partition these elements in air, on surfaces and in water sumps and pools. The partitioning of the radioactivity released into the air phase and water phases is shown in Table 2. The noble gases were assumed to be released into the air atmosphere of the reactor building instantaneously. About 1% of the remaining elements including iodines and bromines was assumed to be present in the air phase. The radionuclide concentration on water film formed on surfaces of components was assumed to be the same as in the sump. All elements, except noble gases, were assumed to be present in the water.

Two coupled ORIGEN-S and BETA-S calculations were made to obtain source terms to be used for releases to the airborne and waterborne phases respectively. The decay of all nuclides and decay products was followed by means of decay chains in ORIGEN-S, and the gamma source spectra versus time were obtained. The beta source spectra were calculated by BETA-S, using the inventories of fission-products calculated as a function of time by ORIGEN-S.

For the dilution of activity in the containment atmosphere, a free air-space volume of 48,000 m^3 was used. The total amount of water available was taken to be the coolant volume from the HTS (80 m^3), i.e., 50 m^3 (broken loop) and about 30 m^3 (volume in the pressurizer plus the dousing system portion of the dousing tank volume of 1560 m^3, a total of 1640 m^3. Note that the ECC portion of the dousing tank (500 m^3) was not credited.

4.2 Gamma and Beta Dose Calculations

4.2.1 Gamma Dose Calculations

The QAD-CGGP-A code was used to calculate the gamma dose rates at three representative points inside the reactor building after a LOCA-LOECC scenario. These points are located at the centre of the reactor building to simulate the dose received by components and equipment away from internal walls and containment wall, e.g., cables; a few centimetres from the containment wall to simulate dose received by pieces of equipment located near walls, which includes a contribution from a water film that may be formed on surfaces of components and equipment, e.g., airlock door seals, upper sections of epoxy liner of containment; and the contaminated water of the flooded reactor building basement to simulate dose received by the immersed portion of the epoxy liner.

To calculate the dose received at the first two dose points, two calculations were made. In the first calculation, the reactor building was modelled in cylindrical geometry (a 20.9 m radius and 36 m high) to simulate the free air space volume inside the reactor, around 48,000 m^3. The air contains those radionuclides that were assumed to go into the air phase. In the second calculation, the water-borne source was represented by a 2-mm-thick water film assumed to be formed on the containment wall and outer surface of other components.

The third calculation was made to calculate dose received by points that are immersed in contaminated water in the reactor building basement, e.g., immersed portion of the epoxy liner. This calculation is also used to estimate the dose received by gaskets of the plate-type ECC heat exchangers.

The water (source) region was modelled as a cylinder, about 1 m high and 20.9 m radius. The dose point is located directly over the source region on its axis; the variable source divisions were used to take account of geometric and material attenuation (self absorption) properly between the source point and the dose points.

For dose rate calculation purposes, twelve gamma energy groups and a gamma source strength of 10^{10} ≠/(cm^3.s) in each group were considered. The calculated dose rates for each energy group were

subsequently normalized to the gamma spectra from the ORIGEN-S calculations to obtain the dose rates. These dose rates were then integrated over time to calculate the accumulated dose after the accident.

4.2.2 Beta Dose Calculations

Using the decay-beta spectra calculated by BETA-S, the Monte Carlo N-Particle (MCNP) 4B[8] transport code simulated the transport of beta particles from the partitioned source in the air and water phases to the target material. The code was used to evaluate the absorbed dose in the epoxy liner, airlock door seals and electrical junction box containing cable wires, connectors and terminal blocks etc.

The Monte Carlo simulations were done in sets of nine, calculating the absorbed dose in the target material originating in nine beta-energy groups of BETA-S (default energy-group structure), i.e., 100, 200, 400, 700 keV, 1, 2, 3, 5, and 8 MeV). Table 3 shows typical beta spectra at 1 d after the accident calculated by the BETA-S code. This methodology permits the calculation of the absorbed dose at different decay times by prorating the MCNP results with the actual source from BETA-S. A set of MCNP calculations was needed for each target material (epoxy liner, airlock door seals, an electrical junction box). This calculation is required because the spectral change of beta particles in the target materials depends spatially on the arrangement of the source and the receptor.

For the epoxy liner exposed to contaminated air and covered with a water film, four sets of nine MCNP calculations were performed. The beta-particle sources in the air and water film were treated in separate calculations, and two water-film thicknesses were considered. The model included a finite air region, a 0.5-mm or 2-mm-thick water film, a 6-mm-thick epoxy liner and concrete. Contribution to dose rates from the radioactivity in the air phase and the water films was calculated.

The reason for using a water film of two different thicknesses is to show the effect of film thickness on the beta dose received by a given surface. For dose calculation purposes, such a film was taken to be present inside containment throughout 90 d after the accident. It was shown that a 2-mm-thick water film gives 25% more beta dose than a 0.5-mm-thick film does, but only 10% more than a water column. In addition, the beta dose received from the activity released into the air phase is greater with a 0.5-mm-thick film than for a 2-mm-thick film because of self-shielding effect of the water film with the latter thickness. Table 4 shows that the overall effect is a slight increase in the beta dose with a 0.5-mm-thick water film.

If no water film has formed on the surface of the epoxy liner, then the dose received would be due to radioactivity released into the air phase only. To evaluate this dose, one set of MCNP calculations was made using an air thickness of 5000 mm because, the beta-particle range is long, and this thickness of air would ensure that high-energy betas from deep inside the source would not be missed.

For the immersed portion of the epoxy liner, two water-film thicknesses were simulated, one with 10 mm and the other with 100 mm. The difference in water thickness is to cater for low- and high-energy beta particles. The thinner layer is used to model beta source at 100 and 200 keV, and the thicker layer is used with 400 keV and higher beta energies. Low-energy beta particles have shorter ranges, and particles released deeper into the source would not contribute to dose received by the epoxy. Modelling a thinner water thickness eliminates unnecessary tracking of unimportant beta particles and reduces computing time. Similarly, the thickness of water was increased for higher energy beta particles released deeper into the water pool would be considered. One set of MCNP calculations was completed.

For the airlock door seals, four sets of nine MCNP calculations were performed in order to calculate dose received from the airborne radioactivity inside containment with water film on surface of the door seals and the two water-film thicknesses. The door seals are made of silicone rubber and 6.35 mm thick with a density of 1.3 g/cm^3.

There are many sizes of electrical junction boxes. A typical junction was modelled as an enclosed box with a water droplet at the centre for dose tally. The box was taken to be 230 mm x 280 mm x 130 mm in size, which is relatively large, and thus would give a limiting dose for components inside the box; that is, it contains a larger amount of containment air inside. The radioactivity in air outside the box and any water film which may be formed on the box, would not contribute to beta dose received by components inside the box, e.g., cable wires, connectors, terminal blocks.

5. RESULTS

Figures 1 and 2 show the integrated gamma, beta and total (gamma + beta) energy releases over time (up to 3 months) into the air and water phase after a LOCA-LOECC respectively.

Figure 3 shows the gamma, beta and total dose received by the airlock door seals (a silicone rubber, about 6.35 mm thick, ρ=1.3 g/cm^3) with a 0.5-mm-thick water film versus time after a LOCA-LOECC.

Figure 4 shows the dose received by the immersed portion of the epoxy liner (ρ=1.3 g/cm^3). Figure 5 shows the beta energy deposition from a 2-m-thick water film as a function of beta energy through the 6-mm-thick epoxy liner and the concrete wall. That is, this figure shows how quickly the dose rate drops through these materials because of the small range of the beta particles. Figure 6 shows dose received by components inside a typical junction box.

Table 4 shows the effect of a 0.5-mm and a 2-mm-thick water film thickness on the beta dose received by the airlock door seals and epoxy liner from the air and water 3 months after the accident. Overall, there is a slight increase in the total beta dose with a thinner film. This is because the beta dose received from the air becomes larger with a thinner film, but the beta dose from the water film becomes lower with a thinner film. Table 4 also shows the gamma dose received by the airlock door seals for a 2-mm-thick water film. Note that the beta dose is highest without any water film because of the beta source in the air.

Inspection of Figures 3, 4 and 6 shows that the accumulated dose increases by a factor of about 2 from day 1 to 3 months after a LOCA-LOECC for those components exposed to air and a water film, but by a factor of 5 for the immersed portion of the epoxy liner over the same time period.

Table 5 shows the total accumulated dose during 30 years operation at 80% capacity factor plus the LOCA-LOECC dose received over 3 months for the airlock door seals, epoxy liner and electrical junction box.

The total dose received by the airlock door seals was estimated to be 1.9 Gy over 30 years of reactor operation at 80% capacity factor (based on 0.6 $\geq$Gy/h). The maximum dose received by the epoxy liner (opposite the fuelling machine on reactor face) was estimated to be 6.4 x 10^4 Gy over the same time period. Most of this dose is due essentially to radiation from spent fuel during refuelling operation (the fuelling machine) followed by the feeders and headers of the HTS components, and the core radiation (gamma and neutron) leaving the end shields. Note that the fuelling machine operates intermittently (5% of the time) but nevertheless dictates the dose received by this portion of the liner inside containment.

The dose received by other components and equipment depends on their locations. The fields would be higher in the fuelling machine vaults, the moderator enclosure, and the fuelling machine maintenance locks, whereas fields would be lower in accessible and restricted access areas. Thus, the doses were estimated to be as low as 1.9 Gy (accessible areas) and as high as ~1.0 x 10^4 Gy (the fuelling machine vaults). These dose rates were also applied to cables and cable splices, and connectors during reactor operation.

The total integrated dose (last column of Table 5) was calculated to be as high as 1.1 x 10^5 Gy for the immersed portion of the epoxy liner. Note that the beta dose was assumed to be an average dose received by that component.

The dose received by the plate-type ECC heat exchanger gaskets was assumed to be the same as that of the dose received by the portion of the epoxy liner immersed in water. Similarly, the dose received by cables was assumed to be the same as that of the dose received by the epoxy liner exposed to contaminated air given in Table 5.

Table 6 shows the surface dose received by a thin surface layer (0.01 mm thick) of the airlock door seals, epoxy liner, gaskets of the plate-type ECC heat exchangers, and cables. These surface doses are larger than those given in Table 5 for which the beta dose was assumed to be received by the whole component (i.e., averaged over that material), as stated above. The surface dose received by the first 0.01-mm-thick epoxy liner without any water film was calculated to be as high as ~1.7 x 10^5 Gy. However, it is not realistic to assume that there will be no water film on the containment wall during the whole duration of the accident. The surface dose for the immersed portion of the epoxy was estimated to be 1.3 x 10^5 Gy over 3 months after a LOCA-LOECC scenario.

It should be noted that special cables are associated with the fuelling machine components and the reactivity mechanisms that are located in the reactor core. These cables receive a much larger dose during their life-time prior to replacement, e.g., 2.0 x 10^6 Gy. The replacement interval for certain components and materials range from 5 to 10 years. Thus, the total dose received was based on such a time period, rather than the station life-time of 30 or 40 years. The dose received is added to the design-basis accident dose in order to evaluate the total integrated dose (TID) as the environmental qualification dose for that component or equipment.

In addition, the "mission times" would be different for many components, i.e., not necessarily 3 months. Thus, an appropriate TID needs to be used for that component. The mission time is that time period during which the component or equipment or both must perform its safety function. It should also be recognized that most of the accident dose is received within 1 d after the accident scenario.

6. CONCLUSIONS

The dose received by the airlock door seals, epoxy liner of containment, electrical junction box, cables and the plate-type ECC heat exchanger gaskets were calculated for environmental qualification of these components against radiation damage. The gamma dose received during reactor operation and after accidents constitute most of the dose received by most components. Nevertheless, the beta dose must be included for some components. The beta dose is a surface dose and is delivered over the first-few-millimetres-thick layer of that component. This becomes important for those components that are not encased in protective covers. A protective sacrificial cover of 3-mm-thick steel would be sufficient to reduce beta dose to insignificant levels.

For components such as the immersed portion of the epoxy liner, the ECC system components and airlock door seals, the accident dose dictates EQ; for the epoxy liner in air, both radiation-dose components contribute. Thus, if the accident dose dictates the qualification dose for a given equipment and components, then there would be no increase in the test dosage whether the station life-time has been extended from 30 to 40 years. It should also be noted that some components need to be replaced on regular intervals (e.g., every 5 to 10 years) and they are not affected by such extensions.

7. REFERENCES

1. Engle Jr., W.W., "A User's Manual for ANISN - A one-dimensional Discrete Ordinates Transport code with Anisotropic Scattering", K-1673, Union Carbide Corporation, 1967 March.

2. Rhoades, W.A., "DOT IV, Two-Dimensional Discrete Ordinates Radiation Transport Code System", RSIC Computer Code Collection, Oak Ridge National Laboratory, Oak Ridge, Tennessee, CCC-320, 1979.

3. QAD-CG, "Combinatorial Geometry Version of QAD-P5A, A Point Kernel Code for Neutron and Gamma Ray Shielding Calculations", CCC-307, RSIC Computer Code Collection (May,1979).

4. O.W. Hermann and R.M. Westfall, "ORIGEN-S: SCALE System Module to Calculate Fuel Depletion, Actinide Transmutation, Fission-Product Buildup and Decay, and Associated Radiation Source Terms", Oak Ridge National Laboratory (November, 1993).

5. I.C. Gauld, S.G. King, "BETA-S" A Code to Calculate Multigroup Beta Spectra", COG-96-33-1, RC-1564 (April, 1996).

6. Reactor Safety Study, United States Nuclear Regulatory Commission Report WASH 1400, Appendix 7, (October, 1975).

7. PWR Degraded Core Analysis, Committee chaired by J. H. Gittus, UKAEA report, ND-R-610S, (February 1982).

8. J.F. Briesmeister, Editor, "MCNP - A General Monte Carlo N-Particle Transport Code", Version 4B, Los Alamos National Laboratory Report, LA-12625-M (March, 1997).

TABLE 1
Fission Products Considered for Source-term Calculations by ORIGEN-S and BETA-S

Category	Fission Products
Halogens	Kr, Xe
Noble Gases	I, Br
Alkali Metals	Rb, Cs
Alkali Earth Metals	Sr, Ba
Tellurium Group	Te, Sb
Noble Metals	Mo, Tc, Ru, Rh
Rare-earth Metals	Y, Ce, Pr, Eu

TABLE 2
Percent of Total Core Inventory Included in ORIGEN-S Reprocessing Calculations

Element	Air Phase	Water Phase/Surface
Kr, Xe	22	0.0
I, Br	0.22	22
Rb, Cs	0.22	22
Sr, Ba	0.032	3.2
Te, Sb	0.22	22
Mo, Tc, Ru, Rh	0.0064	0.64
Y, Ce, Pr, Eu	0.0034	0.34

TABLE 3
Beta Source in Air and Water Phase at 1 d after a LOCA-LOECC Scenario, calculated by BETA-S

BETA-S Energy Group #	Upper Energy Limit (MeV)	Releases into Air Phase (beta/s)	Releases into Water Phase (beta/s)
1	8.0	0.0[i]	0.0[i]
2	5.0	2.91 E+14	2.46 E+11
3	3.0	2.69 E+14	9.28 E+14
4	2.0	9.17 E+14	5.07 E+16
5	1.0	2.11 E+15	1.98 E+17
6	0.7	4.17E +15	4.04 E+17
7	0.4	6.14 E+15	5.47 E+17
8	0.2	2.93 E+17	4.97 E+17
9	0.1[ii]	4.84 E+17	8.90 E+17

[i] The beta source in the 8 MeV group is insignificant after 15 minutes.

[ii] The lower energy limit is zero.

TABLE 4

Accumulated Beta, Gamma and Total Dose Received by Airlock Door Seals, Epoxy Liner of Containment and Electrical Junction Box over 3 Months After a LOCA-LOECC Scenario

Dose Point Location	Beta Dose (Gy)	Gamma Dose (Gy)	Total Dose (Gy) (Gamma+Beta)
Airlock Door Seals:			
Equipment Airlock and Personnel Airlock (Exposed to contaminated air and covered with 2 mm or 0.5 mm thick water film)	6.95 E+03 (8.50 E+03)	7.10 E+03 (~7.10 E+03)	1.4 E+04 (~1.7 E+04)
Epoxy Liner of Containment Wall:			
Epoxy Liner exposed to contaminated air without water film; and with 2.0- or 0.5-mm-thick water film	3.26 E+04; (2.77 E+04 or 2.89 E +04)	~7.10 E+03	~4.0 E+04 (~3.5 E+04 or ~3.6 E+04)
Epoxy Liner completely immersed in water in the reactor building basement	1.80E+04	9.10 E +04	~1.1 E +05
Electrical Junction Box	1.60E+04	9.50 E+03	2.6 E+04

TABLE 5

Accumulated Normal Operation Dose (for 30 years at 80% c.f.) and LOCA-LOECC Dose (after 3 months) Received by Airlock Door Seals, Epoxy Liner of Containment and Electrical Junction Box

Dose Point Location	Normal Operation Dose (Gy)	LOCA+LOECC Dose (Gy)	Total Dose (Gy)
Airlock Door Seals:			
Equipment Airlock and Personnel Airlock	1.9[a]	~1.4 E+04 (1.7 E+04)[b]	~1.4 E+04 (1.7 E+04)[b]
Epoxy Liner of Containment Wall:			
Epoxy liner exposed to contaminated air without water film and with water-film after LOCA-LOECC (opposite the fuelling machine on reactor face)	6.4 E+04	~4.0 E+04 (3.5 E+04 or 3.6 E+04)[b]	1.0 E+05 (1.0 E+05 or 1.0 E+05)[b]
Epoxy liner completely immersed in water in the reactor building basement after LOCA-LOECC	1.9[a]	1.1 E +05	1.1 E +05
Electrical Junction Box	1.9 – 1.0E4 [c]	2.6 E+04	3.6 E +04

[a] Based on 6 µGy/h during 100% full-power operations.

[b] With 2-mm or 0.5-mm-thick water film on surface of containment wall following LOCA-LOECC.

[c] Varies depending on the location of junction boxes, e.g., estimated to be as low as 1.9 Gy and as high as about 1.0E+04 Gy for any junction boxes located in the fuelling machine vaults.

TABLE 6

Estimated Surface Dose Received by a 0.01-mm-thick Layer of Airlock Door Seals and Epoxy Liner of Containment (Normal Operation and LOCA-LOECC)

Dose Point Location	Total Surface Dose (Gy)
Airlock Door Seals:	
Equipment Airlock and Personnel Airlock	~5.4 E+04 (~5.3 E+04)[i]
Epoxy Liner of Containment Wall:	
Epoxy liner exposed to contaminated air without water film and with water-film after LOCA-LOECC (opposite the fuelling machine on reactor face)	~1.7 E+05 (~1.2 E+05 or ~1.2 E+05)[i]
Epoxy liner completely immersed in water in the reactor building basement after LOCA-LOECC	~1.3 E +05
Plate-type ECC Heat Exchanger Gaskets	[ii]
Cables	[iii]

[i] With 2-mm or 0.5-mm-thick water film on surface of containment wall after a LOCA-LOECC.

[ii] Assumed to be the same as that of the dose received by the immersed portion of the epoxy liner.

[iii] Assumed to be the same as that of the dose received by the epoxy liner exposed to air with water film.

Figure 1 : Gamma and Beta Energy inside Containment Air, Integrated over Time after a LOCA-LOECC

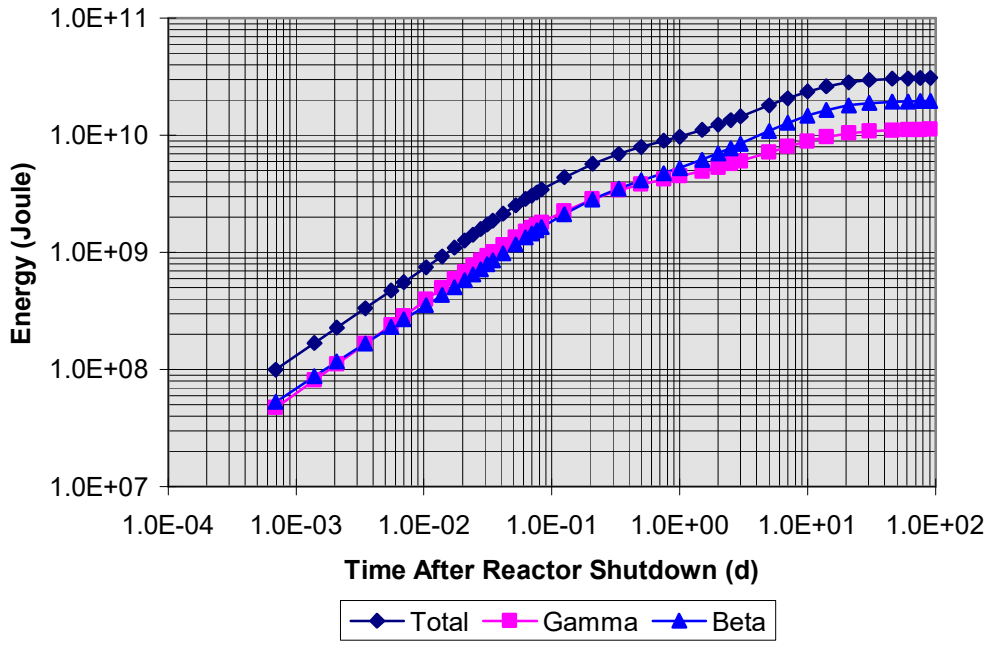

Figure 2: Gamma, Beta and Total Energy in the ECC Water, Integrated over Time after a LOCA-LOECC

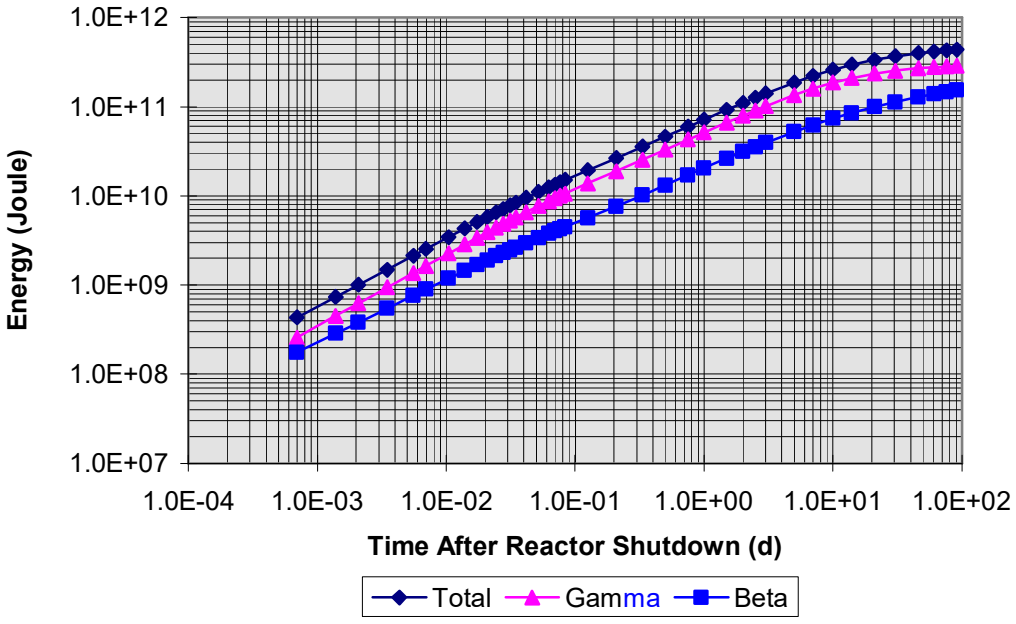

Figure 3: Radiation Dose Received by Airlock Door Seals after a LOCA-LOECC (air + 0.5-mm water film)

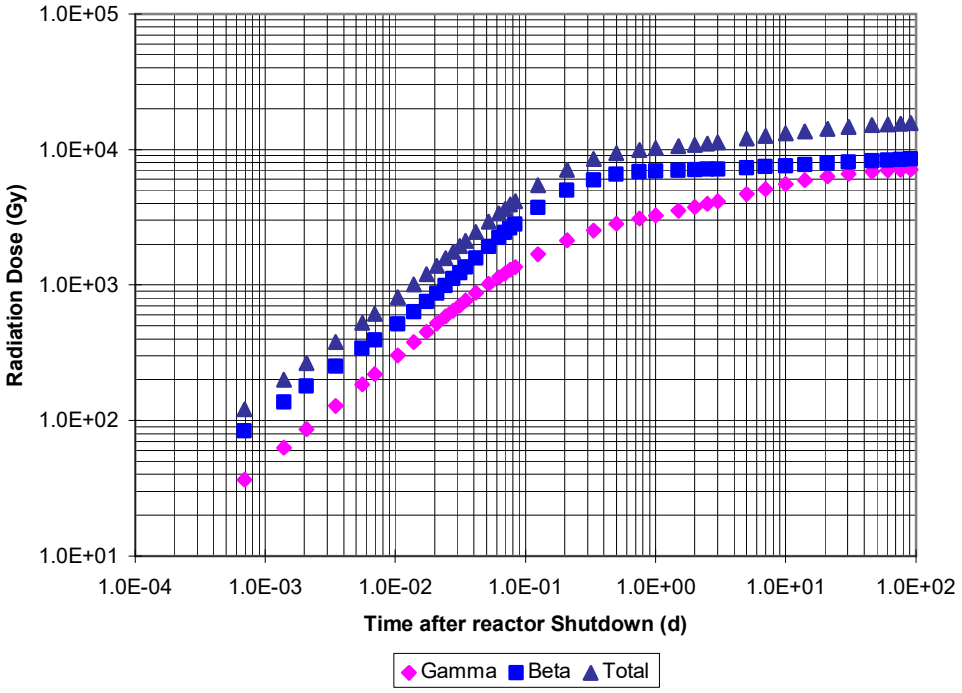

Figure 4: Radiation Dose Received by Epoxy Liner of Containment Immersed in Water after a LOCA-LOECC

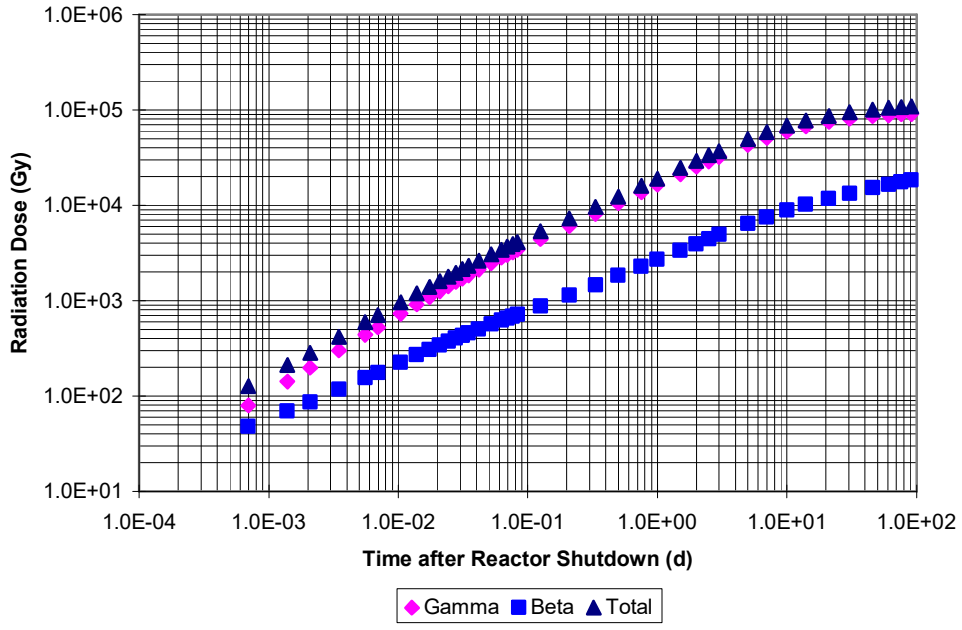

Figure 5: Beta Energy Deposited in Epoxy and Concrete by a 2-mm-Thick Water Film on Liner Surface after a LOCA-LOECC

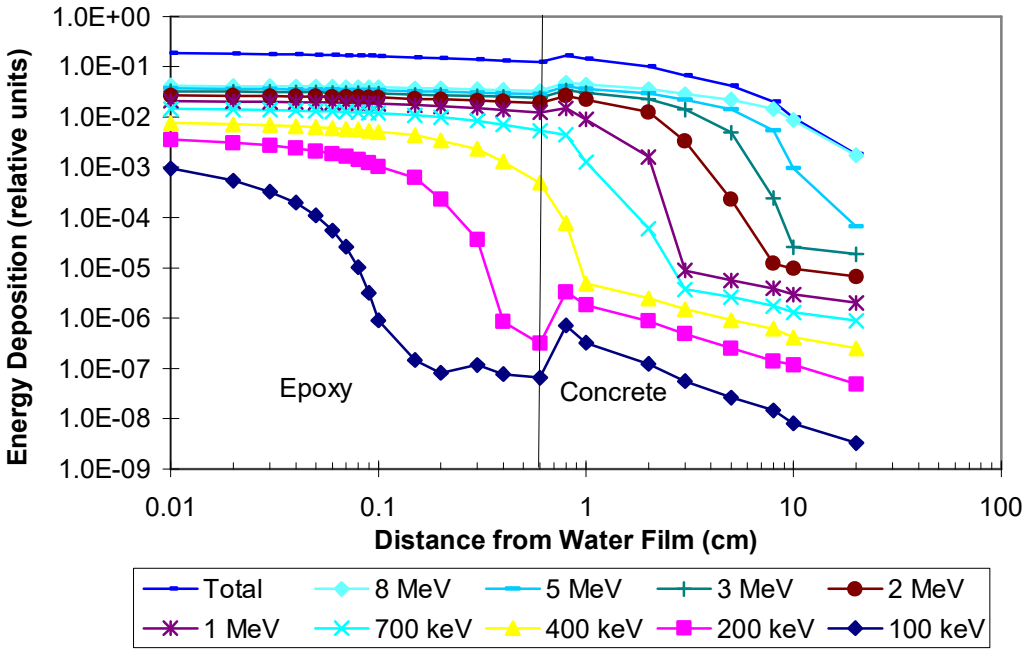

Figure 6: Radiation Dose Received by Junction Box inside Containment after a LOCA-LOECC