DISCRETE ORDINATES ANALYSIS OF THE IRIS INTERNAL SHIELDS

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ABSTRACT

IRIS is an advanced ~1000 MWth (350 MWe) light water reactor that is being developed by an international consortium led by Westinghouse, with shielding support by the Oak Ridge National Laboratory (ORNL) in the USA and the Politecnico di Milano in Italy. The presence of the primary steam generators and downcomers inside the reactor vessel can have the effect of reducing the fluence on the vessel, the dose rates outside the vessel, and the activation of the vessel itself. This last feature is attractive from a decommissioning viewpoint. This study focuses on the deterministic shielding analyses performed at ORNL, including (1) the initial 2-D shielding analysis at full power, (2) activation analyses for the carbon steel reactor vessel and downcomer shields, and (3) the 2-D decay gamma shielding analysis after shutdown and the removal of the fuel. At power, the dose rate at the midplane of the vault (162 μSv/h) was found to be about double that predicted by earlier 1-D analyses due to streaming above and below the five thick carbon steel baffles forming the downcomer region adjacent to the core. Post-shutdown decay gamma dose rates outside the empty vessel are negligible (< 0.12 μSv/h) but may be as high as 40 μSv/h if the downcomer shields are left inside the vessel. Except for the bottom head, activation levels in the reactor vessel itself appear to be well below regulatory limits for waste disposal. Computationally, the most unique aspect of this work was the semi-automated way in which the neutron fluxes from the first shielding calculation were coupled to separate irradiation/decay activation analyses for each of the 9810 mesh locations in the reactor vessel and downcomer regions, with the resulting spatially-distributed decay gamma source terms then being directly coupled to the second shielding analysis.

Key Words: IRIS, shielding, activation

1. INTRODUCTION

IRIS is an advanced ~1000 MWth (350 MWe) light water reactor that is being developed by an international consortium led by Westinghouse, with shielding support by the Oak Ridge National Laboratory (ORNL) in the USA and the Politecnico di Milano in Italy. Details related to the IRIS design are provided elsewhere [Refs. 1, 2, 3, 4], while Fig. 1 shows a general overview of the system configuration. IRIS has an integral vessel which houses the reactor core and support structures, the core barrel, upper internals, control rod guides and drive lines, the steam generators, a pressurizer located in the upper head, the reactor coolant pumps, and the downcomer shield plates in the radial region beyond the core. The vessel has a height of ~22 m and an outside diameter of ~6.7 m. Hot coolant rising from the reactor core to the top of the vessel is pumped downward into the steam generator annulus by eight pumps. Moreover, the presence of the steam generators inside the vessel creates a large downcomer region which allows for extra shielding between the core and the vessel. This additional internal shielding leads to a reduced external dose which improves safety, to reduced activation of the vessel which reduces waste, and to simplified maintenance and D&D which improves the overall economics.

In the following sections we will focus on the shielding and activation analyses for the lower part of the system – from the vault below the reactor vessel to a point about 1.6 meters above the top of the core, with an emphasis on the downcomer region.

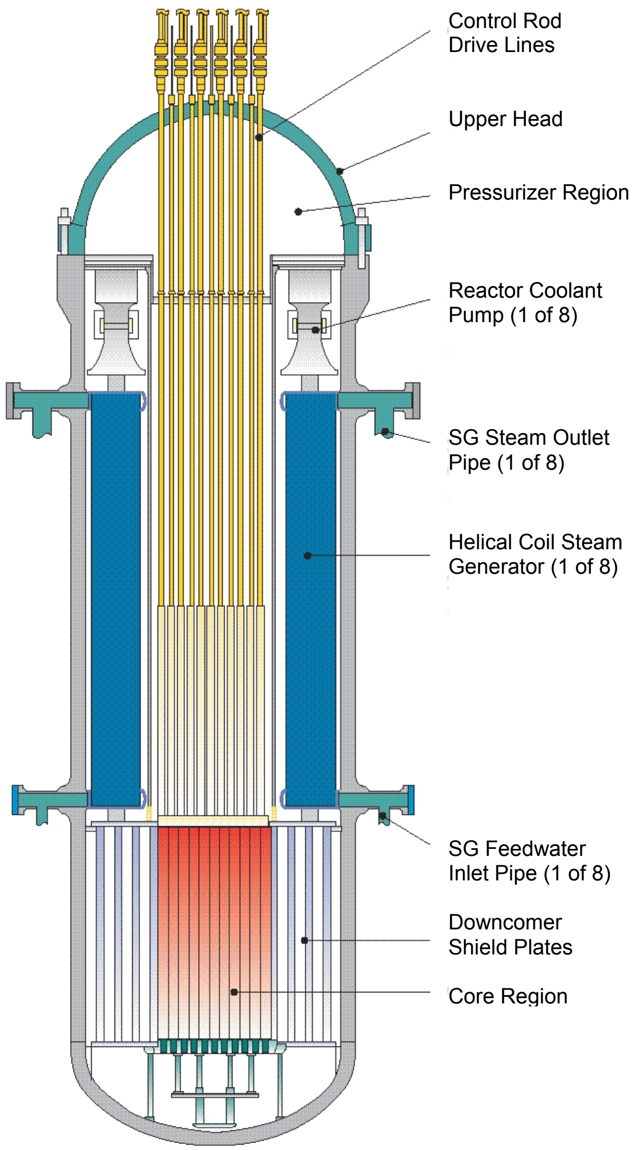

Figure 1. IRIS integral vessel layout.

2. INITIAL DETERMINISTIC 2-D SHIELDING ANALYSIS AT FULL POWER

The 2-D RZ DORT model used to represent the lower portion of the IRIS system is shown in Fig. 2. True to the initial design, this model included an elliptically-shaped lower head on the vessel. The 89 fuel assemblies in the (487.7-cm-long) active core region were homogenized and had an equivalent radius of 120.08 cm, with the core barrel extending radially from 132 cm to 137 cm, and the carbon steel reactor vessel extending from 305 cm to 336 cm. Previous parametric studies [Ref. 5] considered no downcomer shield plates, then 4 and 5 plates of uniform thickness, then 5 plates of different thicknesses, whose thicknesses and spacing were optimized based on 1-D shielding analyses. In this 2-D analysis, there were 5 carbon steel downcomer shield plates between the core barrel and reactor vessel, with thicknesses of 15.03, 12.54, 10.86, 9.63, and 8.69 cm, as shown in Fig. 2. The model also includes the concrete vault beyond the vessel.

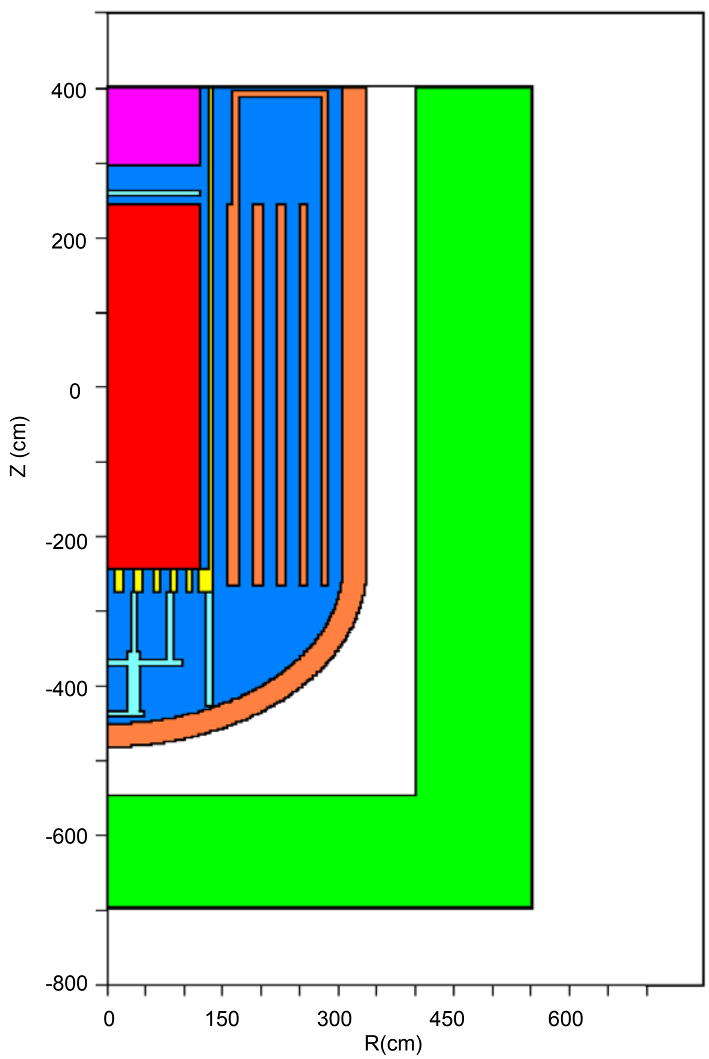

Figure 2. DORT model of lower portion of IRIS system.

This initial 2-D RZ $S_{12}P_3$ full-power fine-mesh DORT calculation [Ref. 6] assumed a total source of 7.5573×10^{19} n/s (corresponding to 1000 MWth), with a flat distribution over the active core. A total of 80195 mesh intervals were used (215 radially, and 373 axially), with a typical mesh size of about 2.5 cm in each direction. Cross sections were based on the BUGLE-96 library [Ref. 7] with 47 neutron groups and 20 gamma groups. The resulting neutron and gamma dose rates during normal full-power operation are shown in Figs. 3a and 3b. Because of the wide water annulus and the downcomer shield plates adjacent to the core region, the fast neutron flux (> 1 MeV) on the inner wall of the vessel has an extremely low value of 475 n/s/cm^2, with an attendant 30-year fast fluence of 3.6×10^{11} n/cm^2, assuming a duty factor of 0.8. For comparison, 10 CFR 50 App. H specifies that a reactor vessel surveillance program is not required if the vessel fluence is conservatively estimated to not exceed 1×10^{17} n/cm^2. As might be expected, the external dose rates are also very low. In the vault across from the midplane of the core, the external dose rate is only 162 μSv/h. Interestingly, this is about twice as high as one would estimate with a 1-D code due to radiation streaming around the top and bottom of the downcomer shield plates. In the vault, just above the downcomer region, for example, the total dose rate is about 478 μSv/h. In the vault near and below the lower head, the dose rates during full power operation vary from a few mSv/h, to a maximum of 0.1156 Sv/h near the centerline. More recent studies [Ref. 8] by Dr. Carlo Lombardi et al. of the Politecnico di Milano in Italy show the mitigating effect of additional shielding that has since been placed in the lower plenum.

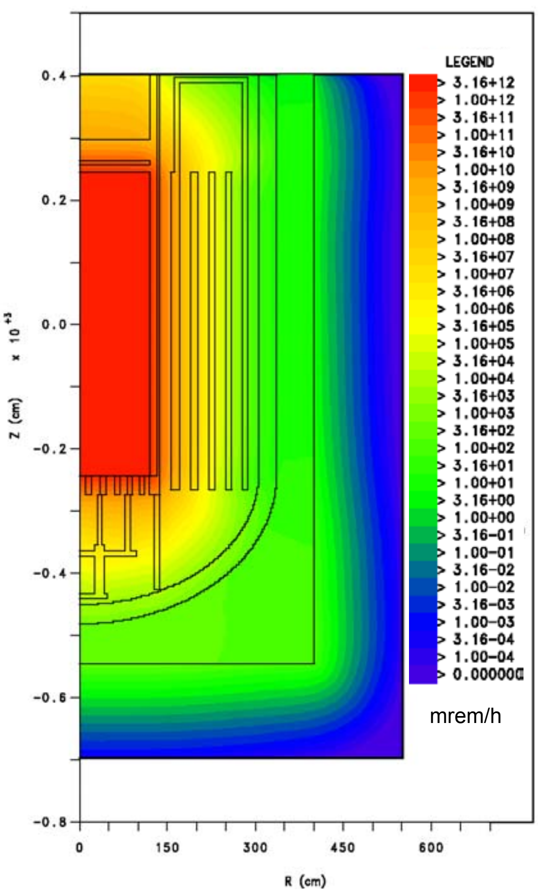

Figure 3a. Neutron dose rates at full power.

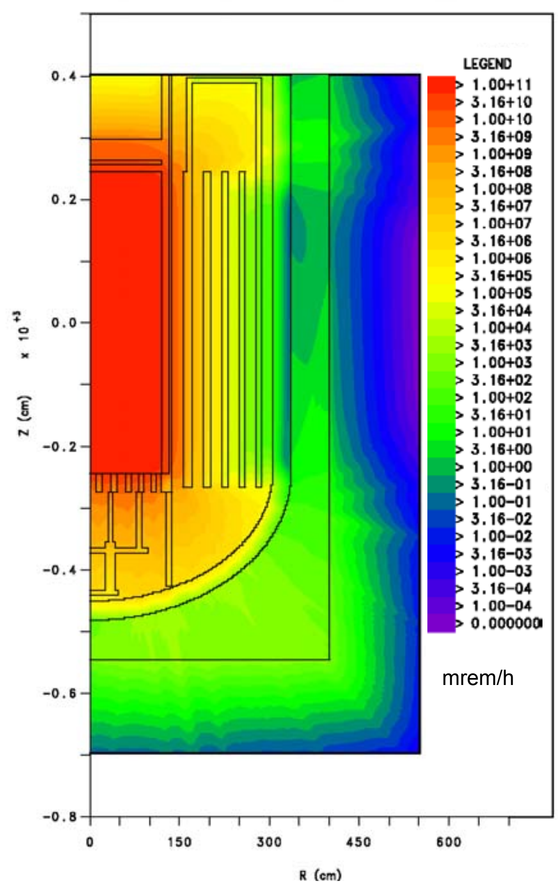

Figure 3b. Gamma dose rates at full power.

3. ACTIVATION ANALYSES FOR THE CARBON STEEL VESSEL AND DOWNCOMERS

Detailed activation analyses for the carbon steel vessel and downcomers, and the subsequent shielding analyses, were needed for three reasons: First, we needed to know the external dose rate coming from the empty reactor vessel (by itself) after 30 years of operation. This would be important for decommissioning and ultimate disposal. Programmatically, it was hoped and anticipated that this would be a non-issue in the case of the IRIS reactor. Secondly, there was great interest in knowing what the external dose rates would be if the highly activated downcomer shield plates were left inside the vessel. If sufficiently low, this would further simplify the ultimate shipping and waste disposal issue. Lastly, it was important to determine the actual activation levels in these components – partly so as to perform the necessary shielding analyses, but also because many waste disposal limits also involve specific activation levels (Bq/g) in the material, in addition to (or instead of) dose rates.

This was a formidable challenge since the activation levels and associated decay gamma source terms varied greatly from one location to another. As such, neutron fluxes from the initial shielding analysis (at power) would have to be accessed automatically and used to perform separate activation analyses at literally thousands of locations within the vessel material and downcomer material, with the gamma source terms at each of those 9810 locations then being piped back into a second (decay gamma) shielding analysis. To facilitate such a large number of activation calculations in a semi-automated fashion, an intermediate code module called ACTIVATE was written. This new module: (1) reads the printed DORT output file and scans for all mesh intervals having carbon steel, then (2) extracts the 47-group neutron fluxes for each such mesh interval from the DORT flux moment file. (3) It will then fold those multigroup fluxes with the key nuclide-dependent cross section data (described below and obtained from MCNP) to obtain the necessary reaction rates in each mesh interval, and (4) perform the necessary activation analysis for the 30-year irradiation period and a specified decay period (typically one week) before (5) writing out the resulting mesh-dependent gamma sources in a DORT-ready format for use in the decay gamma shielding analysis.

Figure 4 shows the subset of nuclides and nuclear processes considered in the simplified activation analysis performed by the present version of the ACTIVATE code. In this case, the four key radioactive nuclides of ultimate interest are Fe-59, Co-60, Mn-56, and Mn-54, but to reach that point, 12 different sets of multigroup nuclear cross section data are required, as well as the standard decay constants. Moreover, this simplified scheme for carbon steel was chosen because it had previously been used in similar activation analyses for the carbon steel rotary shutters on the HFIR beam tubes [Ref. 9], and found to yield excellent results when compared to post-shutdown measurements of the resulting gamma dose rates.

Neutron activation cross sections for the nuclides and processes shown in Fig. 4 were obtained in the BUGLE 47-group energy structure, *a priori*, with the aid of the MCNP4b code [Ref. 10], the associated pointwise cross section libraries [Ref. 11], and "thin disk" models of foils involving only the isotope in question. Using that code, each thin foil was exposed to a monodirectional beam of neutrons having a standard "fission-1/E-Maxwellian" spectrum, while energy-dependent

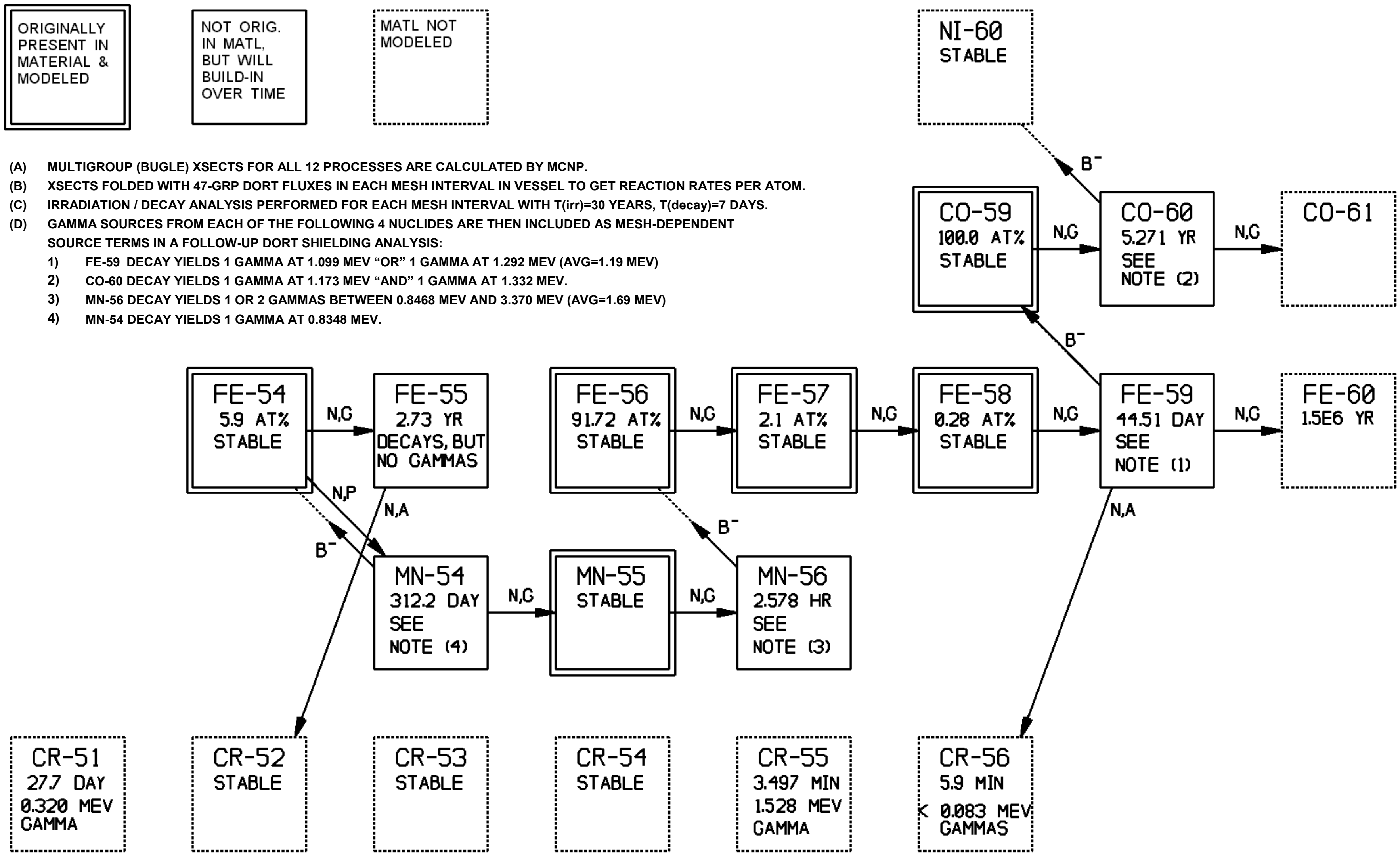

Figure 4. Nuclear processes leading to the production of four key radioactive nuclides (Fe-59, Co-60, Mn-56, and Mn-54) in Type 1020 carbon steel. Identified here are 12 individual reaction rates that must be known to perform the necessary activation analysis.

source biasing was used to sample uniformly and adequately from each of the 47 BUGLE energy groups. The tabulated reaction rates in each energy group were then divided by the tabulated neutron fluxes in each energy group to obtain the multigroup activation cross sections needed by the ACTIVATE code.

Prior to any irradiation, the carbon steel was assumed to initially contain 0.45 wt% Mn-55 and 0.015 wt% Co-59 since these impurity levels are typical of the initial carbon steel used in the HFIR reactor vessel at Oak Ridge. The actual activation analysis for each DORT mesh used timesteps of one week throughout the 30-year irradiation period at full power, followed by timesteps of one day for most nuclides (with micro-steps of only one hour for Mn-55 and Mn-56) during the one-week post-shutdown decay period. While the resulting activities and gamma source terms at each of the 4733 mesh points in the downcomer plates and the 5077 mesh points in the reactor vessel were computed and used in the subsequent decay gamma shielding analysis, the results at six key points of interest are summarized in Fig. 5. On the innermost surface of the innermost downcomer plate, the gamma source terms varied from a maximum of 1.96×10^8 p/s/cc near the midplane (point a), to 4.34×10^7 p/s/cc near the bottom (point b). On the inner surface of the reactor vessel, the gamma source terms varied from a very low value of 0.854 p/s/cc near the midplane (due to the downcomer shield plates), to 0.631 p/s/cc at point (d) near the bottom of the downcomer region, to a peak of 98.2 p/s/cc at point (e) on the inner surface of the lower head, and 2.01 p/s/cc at the corresponding point (f) on the outer surface of the lower head. These last two points (e and f) are located about 150 cm from the reactor centerline, and are actually somewhat higher than at the centerline due to multi-dimensional effects.

Because this simplified pointwise activation analysis and the subsequent 2-D decay gamma shielding analysis are both very fast (requiring only a few minutes), there are two important benefits to the designers: (1) it provides a useful vehicle for determining local hot spots or problem areas, and (2) it provides a quick tool whereby a designer can easily change the impurity concentration specifications (say on the cobalt) and immediately assess the effect on the contaminated waste material and dose rates throughout all or part of the system. Moreover, this last capability is useful when trying to determine if more restrictive impurity specifications would be past the point of diminishing returns from an economic viewpoint.

4. THE 2-D DECAY GAMMA SHIELDING ANALYSIS AFTER SHUTDOWN AND REMOVAL OF THE FUEL

Using the spatially-distributed gamma source terms computed in this fashion, two decay gamma shielding calculations were then performed using the DORT code. These are illustrated, along with a few key results, in Figs. 6a and 6b. In the first case (Fig. 6a), we have only the empty (air-filled) reactor vessel still sitting in the concrete vault. On the outer surface of the reactor vessel, the dose rate varied from 0.0063 μSv/h near the midplane (point a), to 0.022 μSv/h near the bottom of the downcomer region (point b), to a maximum of 0.121 μSv/h at point (c) close to the centerline. As initially anticipated, these dose rates are all very low. In the second case (Fig. 6b) we were asked to assess the effect of keeping (and shipping) the highly-activated downcomer

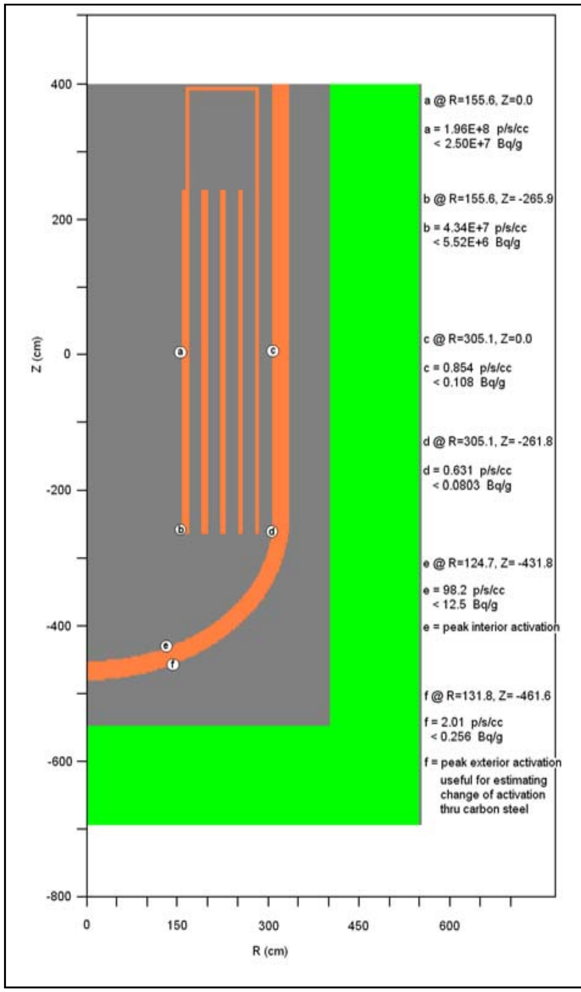

Fig. 5. Decay gamma sources in the downcomers and vessel, based on T(irr)=30 years and T(decay)=1 week.

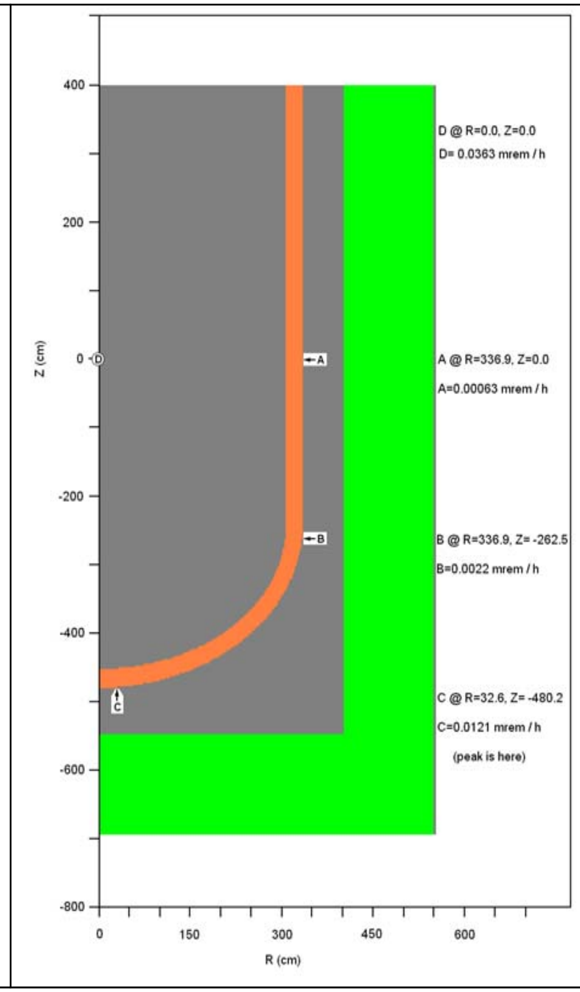

Fig. 6a. External gamma dose rates from the empty (air-filled) vessel only, based on T(irr)=30 years and T(decay)=1 week.

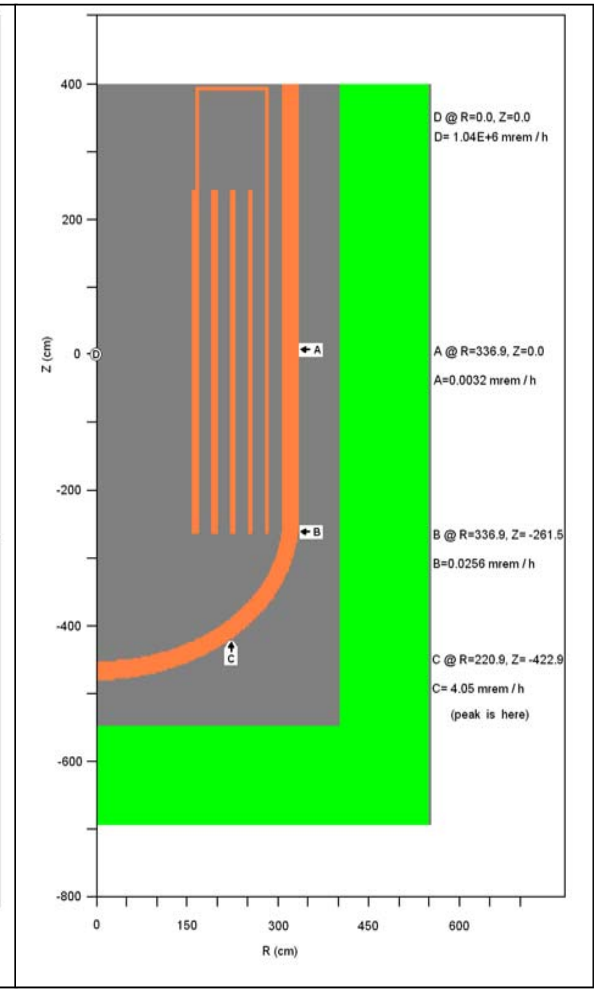

Fig. 6b. External gamma dose rates with the downcomers left in the vessel, based on T(irr)=30 years and T(decay)=1 week.

shield plates inside the otherwise empty vessel. In this case, the dose rates outside the vessel were considerably higher – especially on the lower head, below the downcomer plates. On the outer surface of the vessel, the dose rate was still only 0.032 μSv/h at the midplane (point a), and only 0.256 μSv/h near the bottom of the downcomer region (at point b) where the outer downcomer plates still provide significant shielding; but at point (c), on the outer surface of the vessel below the second and third downcomer plates, the dose rate was now about 40.5 μSv/h. Because of this, some additional shielding material may have to be placed inside the vessel prior to shipping if the downcomers shield plates are to be shipped inside the vessel.

5. CONCLUSIONS

The IRIS design concept is unique insofar as the entire primary cooling loop is contained within the reactor vessel. The wide water annulus in the downcomer region adjacent to the core, in combination with a series of thick downcomer shield plates, greatly reduces both the fast flux and the thermal flux impingent on the inner wall of the reactor vessel. Moreover, the fast flux on the vessel is so small that there is no regulatory need for a reactor vessel surveilance program, while the low levels of activation due to the low thermal neutron flux means that most of the reactor vessel (except perhaps the lower head) can, for regulatory purposes, be disposed of as non-radioactive waste.

At power, the dose rate at the midplane of the vault (162 μSv/h) is about double that predicted by 1-D calculations due to radiation coming from above and below the heavily shielded downcomer region adjacent to the core. In the portion of the vault below the core, the dose rate at power is now about 0.1156 Sv/h near the centerline. Because the vault is filled with argon during operation, manned access at that time is not possible and that dose rate is not considered a problem. To reduce activation levels in the lower head region, however, a revised design has been developed with additional shielding in the lower plenum. These analyses are now underway in Italy.

Shutdown decay gamma dose rates outside the empty vessel are negligible (< 0.12 μSv/h), but may be as high as 40 μSv/h if the highly activated downcomer shield plates are left inside.

To facilitate the thousands of mesh-dependent activation calculations required in this study, a new activation code called ACTIVATE was written which: (1) automatically reads the mesh-dependent neutron fluxes from the first "at power" 2-D DORT analysis, (2) performs the necessary mesh-dependent activation analyses, and then (3) writes out the resulting mesh-dependent decay gamma source terms in a DORT-compatible format for the follow-up "decay gamma shielding" analysis. This automation allows the designers to easily change impurity specification limits and immediately see the consequences of those changes. This has proven useful in addressing decommissioning and disposal issues.

One useful and important extension would be to expand the nuclear cross section database in the ACTIVATE code so as to be able to treat stainless steel and other materials in addition to the carbon steel presently allowed by the code. Such an extension would greatly facilitate the determination of internal maintenance dose rates due to structural components. In that regard, one important caveat is that, in many portions of the upper head, the majority of the dose during maintenance operations may well be due to "plate out" source terms deposited by the coolant, rather than the activation of structural components in that area. Still, both issues must eventually be addressed.

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