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D. C. COOK UNIT 2

POTENTIAL RADIOLOGICAL CONSEQUENCES OF INCIDENTS INVOLVING HIGH EXPOSURE FUEL

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1.0 INTRODUCTION

Current operation of D. C. Cook Unit 2 is based, in part, on the results of analyses which conservatively demonstrate that the off-site radiological consequences of incidents are within the limits specified in 10 CFR 100.11. Those analyses are documented in Chapter 14 of the Updated Final Safety Analysis Report (UFSAR) for D. C. Cook Units 1 and 2 dated July 1982.

Analyses presented in the UFSAR include consideration of fission product inventories (radiological source terms) that result from operation of Unit 2 at a power level of 3391 MW_t and an end-of-cycle average fuel exposure of approximately 25,000 MWD/MTU.

Since increases in power and fuel exposure are planned for future D. C. Cook 2 operating cycles, consideration was given to the effect of those changes on the potential off-site radiological consequences of incidents. The results of those analyses which support operation at a reactor power of 3,425 MW_t with an end-of-cycle average exposure of 30,000 MWD/MTU are presented herein.

2.0 SUMMARY AND CONCLUSION

Changes in the potential off-site radiological consequences of incidents were evaluated considering planned increases in reactor power and fuel exposure. Analyses included consideration of the 1) potential for increased fuel failures resulting from accidents/transients, 2) increases in gaseous and volatile fission product inventories within the fuel, 3) changes in the fractional release of gaseous and volatile fission products to the pellet-clad gap and plenum, and 4) differences in the isotopic mixture of fission products available for release from the fuel. As a result of those analyses, it is concluded that:

- No increase in fuel failures will occur as a result of accidents/transients due to the proposed increases in reactor power to 3425 MW_t and the batch average discharge exposure to 40,000 MWD/MTU.
- A slight increase (∿5%) in the total inventory of gaseous and volatile fission products exists within the core at end-of-cycle conditions which would result in a proportional change in off-site exposures under extreme (maximum hypothetical) LOCA conditions. The off-site doses, however, remain well within the limits specified in 10 CFR 100.
- The inventory of gaseous and volatile fission products which is most readily available for release (i.e., that contained in the pellet-clad gap and plenum) is lower in the ENC 17x17 fuel

when operating at the proposed increased core power and exposure than in the 15x15 fuel used as a basis for prior analyses which considered operation at the lower core power and average exposure conditions.

- Changes in the fission product isotopic mixtures that result from increases in power and exposure and which are available for release from the fuel are not of radiological significance (i.e., no changes occur to negate prior assumptions used for evaluating off-site radiological consequences of accidents).
- No changes occur (e.g., in fuel rod pressure, decontamination factors, etc.) which would invalidate prior assumptions made in evaluating the off-site radiological consequences of accidents.

The analyses performed and/or discussed herein conservatively demonstrate that the proposed increase in reactor power, accompanied by the planned increase in batch averaged discharge exposure, can be accomplished in continued compliance with the limits specified in 10 CFR 100.

3.0 EFFECTS OF FUEL EXPOSURE AND REACTOR POWER INCREASES

The off-site radiological consequences of incidents (i.e., potential radiation doses at the Exclusion Area Boundary and within the Low Population Zone) are a function of the following:

1. Gaseous and volatile fission product inventories available for release from the fuel (i.e., in the fuel matrix and/or within the fuel rod gap);
2. The quantity of gaseous and volatile fission products released from the fuel in an incident (i.e., the number of rods damaged and the number of curies within those rods available for release);
3. The mixture of isotopes released to the environment;
4. The fraction and rate of release of fission products to the environment (e.g., containment leak rate, pool decontamination factor, filter efficiencies, etc.);
5. Meterological conditions at the time of the incident;
6. Distances from the point of release to the point at which the dose is evaluated; and
7. Biological factors associated with the exposed population (e.g., breathing rate, biological pathways, etc.).

Of the above items, only items 1, 2 and 3* are affected by changes in the reactor power, fuel design, and fuel irradiation history (e.g., specific power and total exposure). For those factors, proportionality

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* The spent fuel pool decontamination factor for iodine is dependent on the fuel rod gap pressure. However, calculations indicate that the pressure remains below the threshold value of 1200 psig at which the assumed decontamination factor remains conservatively valid (see Regulatory Guide 1.25).

constants have been developed to relate the thyroid and whole body doses resulting from changes in the radiation source terms to doses previously determined for the various incidents considered in the DC Cook Unit 2 UFSAR. In particular, thyroid and whole body doses associated with incidents involving operations at higher power and exposure levels can be expressed as:

$$D_{2(TH)} = D_{1(TH)} A_I (DCF)_I, \text{ and}$$

$$D_{2(WB)} = D_{1(WB)} EC, \text{ where;}$$

$D_{2(TH)}$ = Thyroid dose resulting from a particular incident involving high exposure fuel and high power operations.

$D_{1(TH)}$ = Thyroid dose calculated for DC Cook Unit 2 existing operating conditions.

A_I = The ratio of the total number of curies of iodine available for release from fuel operating at increased power and exposure to the number of curies available for release from fuel operating at conditions considered in the UFSAR.

DCF_I = The ratio of the thyroid dose per unit concentration (Rem/μCi inhaled) of the iodine isotopic mixture available for release from fuels operating at increased power and

exposure to the dose per unit concentration of the iodine isotopic mixture available for release from fuels operating at conditions considered in the UFSAR.

$D_{2(WB)}$ = Whole body external dose resulting from a particular incident involving high exposure fuel and high power operation.

$D_{1(WB)}$ = Whole body external dose calculated for DC Cook Unit 2 existing operating conditions.

E = The ratio of the average energy per disintegration of the fission products released from fuels operating at high exposure and increased power to the average energy per disintegration of fission products released from fuels operating at conditions considered in the UFSAR.

C = The ratio of the total number of curies available for release from fuels operating at high exposure and increased power to the total number of curies available for release from fuels operating at conditions considered in the UFSAR.

The equation which relates thyroid doses is precise. For whole body doses, however, the relationship assumes that the ratio of β^- and γ doses remains unchanged when increasing the power and fuel exposure. While not exact, that assumption should be valid for reasonable changes in the power and exposure of fuels considered herein.

4.0 RADIATION SOURCES

4.1 FISSION PRODUCT INVENTORIES

Off-site radiological consequences of an incident involving irradiated fuel result from the release of fission product gases to the environs. Such releases would normally involve only those fission products which are contained in gaseous form (noble gases and volatile fission products) within the fuel rod gap. In some cases, however, consideration is also given to the total inventory of gaseous and volatile fission products to conversatively bound the possible off-site consequences of an incident.

Core and gap fission product inventories, computed for the current DC Cook Unit 2 17x17 fuel assembly design, are given in Table 14.1-3 of the Unit 2 UFSAR. Those data were calculated assuming full power (3391 MWt) operation for a period of 650 days. Similar calculations of core and gap fission product inventories for the DC Cook Unit 1 15x15 fuel assembly are shown in Table 14.A.2-1 in Appendix 14A of the Unit 1 UFSAR.

Comparison of the core and gap fission product inventories given in the UFSAR for the two fuel designs indicates that the total inventory remains virtually unchanged but the percent of the fission product gases released to the gap is lower for the current 17x17 design. As a consequence of the above observation, it was concluded that the

off-site radiological consequences of accidents cited in Chapter 14 of the DC Cook Unit 1 UFSAR are also applicable to Unit 2 (see Section 14.3.5 of the UFSAR for Unit 2).

For the purpose of evaluating the consequences of incidents involving irradiated fuel, consideration must be given to the total core fission product inventories and related gap activities or, as appropriate, to the same values for the worst case fuel assembly. The total core and worst case fuel assembly activities calculated for DC Cook Unit 1 are given in Tables 14.A.2-1 and 14.A.3-1 in Appendix 14A of the UFSAR. Examination of the data and assumptions used to obtain the core and gap activities given in Table 14.A.2-1 indicates that the activities are representative of operation at rated power for a period of 650 days (i.e., to a core average exposure of 24,560 MWD/MTU). Assumptions used to obtain the activities for the "highest rated discharged assembly" given in Table 14.A.3-2 are somewhat more obscure. Comparison of the iodine-131 and krypton-85 activities given for the "highest rated assembly" with those for the total core indicates that the highest fuel assembly power was 1.37 times the average value and the maximum assembly exposure was ~42,400 MWD/MTU (i.e., ~1.73 times the core average value).

To develop the relationship between off-site accident doses for the current operational limits and those proposed for future operation, it is important to determine the radiological source terms for both

conditions in a consistent manner. Consequently, fission product inventories and gap release fractions for the DC Cook Unit 1 15x15 fuel assembly were re-evaluated to establish a basis for comparison with the DC Cook Unit 2 ENC 17x17 fuel assembly designed to operate at an increased power and exposure. Basic assumptions used for computing the fission product inventories and gap release fractions for the total core and worst case fuel assembly are given in Table 1.

For each case identified in Table 1, the ORIGEN Code[1] was used to calculate the fission product inventories for the assumed operating conditions and exposures. Gap release fractions for the various isotopes were computed using the RODEX 2 Code[2] with fission gas release models described in the proposed ANS Standard 5.4. The results of the ORIGEN and RODEX calculations are given in Tables 2 and 3 for the core averaged and worst case fuel assemblies, respectively.

A summary of the total activity of various elements of interest is given in Table 4. These data support the previous estimate of the exposure (42,400 MWD/MTU) used to derive data given for the "highest rated" fuel assembly. Also, the data show that the total inventory of fission products of interest within the core and/or worst case fuel assembly increases slightly, but the quantity available for release from the pellet-clad gap decreases for the ENC 17x17 design even at the proposed higher power and exposure conditions. A similar trend in data is apparent

in Table 5 which presents a summary of source terms needed for evaluating the relative accident consequences associated with the proposed operation at higher power and fuel exposure.

4.2 ISOTOPIC COMPOSITION OF FISSION PRODUCTS

The effects of changes in the isotopic composition of fission products, released from the fuel or gaps, were examined using the DACRIN[3] code. In particular, the thyroid and whole body doses due to inhalation of the isotopic mixtures of fission products available for release from the 17x17 and design base fuels were evaluated. These calculations support conclusions drawn by Scherpelz and Desrosiers[4] relative to the contribution of various isotopes to the whole body, thyroid and other organ doses which would result from similar accidents (see page 198 of Reference 4). More importantly, there are no isotopic composition changes which result from operation at increased power and exposure which would negate prior assumptions relative to the importance of the thyroid and whole body external doses as the controlling factors in evaluating the consequences of accidents.

4.3 DOSE PROPORTIONALITY CONSTANTS

Assuming that the rate and fractions of fission products released from the fuel remain unchanged from those presented in the UFSAR, then the thyroid and whole body doses for fuels other than those considered in the UFSAR are related as shown in Section 3.0. Values for the constants

which relate the dose for high exposure fuel with equivalent doses for the design base fuel have been calculated from the data given in Tables 2 and 3, from DACRIN calculations of thyroid dose per μCi of iodine inhaled, and from the β^- and γ disintegration energies given on the RIBD library[5]. Those constants, necessary for the computation of off-site doses due to increased power and exposure, are given in Table 6.

5.0 ACCIDENT CONSEQUENCES

5.1 FUEL HANDLING ACCIDENT

Fuel handling accidents evaluated in the UFSAR consider accidental damage to a worst case fuel assembly. Section 14.2.1.1 describes the assumptions and results of analyses of such an accident within the Auxiliary Building and Section 14.2.1.2 addresses the occurrence of such an accident inside containment. In each case the accident is assumed to occur 100 hours after reactor shutdown.

5.1.1 Auxiliary Building

To evaluate the consequences of an accident in the Auxiliary Building it was assumed that the gap activities calculated to be present in the worst case fuel assembly would be released (see page 14.2.1-9, Section 4.2.1.1). Consequently, for the higher power and exposure fuels, the off-site thyroid and whole body doses are proportional to the ratio of gap inventories of the worst case ENC 17x17 fuel assembly to the worst case 15x15 fuel assembly. Calculated two-hour whole body and thyroid doses for the base case and proposed higher power and fuel exposure are given in Table 7. As can be seen, potential radiation doses at the site boundary remain only a small fraction of the limiting values specified in 10 CFR 100 for accidents involving fuel used at proposed higher power and fuel exposures.

5.1.2 Inside Containment

In considering a fuel handling accident inside containment, assumptions relative to the release of fission products were taken from Regulatory Guide 1.25 (conservative case) and Regulatory Guide 4.2 (realistic case). Those assumptions are specified in Section 14.2.1.2 of the UFSAR. In each case it is assumed that the gap activity available for release from the fuel rods is a fixed fraction of the total inventory of the rod. Consequently, the off-site dose for high exposure fuel operating at increased reactor power is proportional to the ratio of the total inventory of the isotopes within the worst case fuel assemblies. (For the realistic case, the fraction of the inventory available for release also decreases since one row of fuel rods in the 17x17 design is a smaller fraction of the total assembly than is one row of rods in the 15x15 design. That difference is conservatively ignored in calculating the doses for the high exposure fuel operating at an increased reactor power). Calculated thyroid and whole body doses due to the higher exposure fuel and higher reactor power are given in Table 8. Doses at the site boundary resulting from a fuel handling accident inside containment remain well below the limits specified in 10 CFR 100 for proposed operations at higher reactor power and with increased fuel exposure.

5.2 LOCKED ROTOR INCIDENT

An analysis of the locked rotor event is presented in Section 14.2.1.6 of the Unit 2 UFSAR, but releases of fission product activity which result from that incident are not separately addressed. The analysis

is based on conservative assumptions of the peaking factor at the time of the incident and, as a result, indicates that some fuel failures may occur as a result of the incident. (The assumed value of F_Q = 2.5 is greater than the limiting value given in the Technical Specifications for DC Cook Unit 2).

An analysis of the locked rotor incident was subsequently performed by Exxon Nuclear[6] assuming operation of the reactor at the proposed increased power level of 3425 MW_t and with the proposed increased total peaking factor (F_Q) of 2.1 and an $F_{\Delta H}$ of 1.60. This analysis demonstrates that an adequate margin in the MDNBR exists such that no fuel failures are expected to occur as a result of a locked rotor incident. Consequently, no further consideration of the potential radiological impact of proposed increases in reactor power and fuel exposure is warranted.

5.3 RCC ASSEMBLY EJECTION INCIDENT

Section 14.2.6 of the UFSAR presents an evaluation of the RCC assembly ejection incident. As for the locked rotor incident, conservative assumptions (greater than technical specification limits) were made relative to peaking factors in the core. Results of these conservative analyses indicate that less than ten (10) percent of the affected fuel rods enter DNB. (This is less than 2% of the total core). The off-site radiological consequences of this event were not expressly evaluated since it is clear that the resulting doses would be only a small fraction (∿2%)of those calculated for the double-ended coolant pipe break (LOCA) event (see pages 14.2.6-12, Section 14.2.6 of the Unit 1 UFSAR).

An analysis of the RCC assembly ejection incident has been performed by Exxon Nuclear[7] using procedures developed in the Generic Rod Ejection Analysis[8] and assuming operation of the reactor at a power level of 3425 MW_t and with an F_Q value of 2.1. This analysis demonstrates that the maximum pellet energy deposition is <170 cal/gram. Consequently, no failures of ENC fuel are expected to occur as a result of an ejected RCC assembly. Should failures occur in other fuels within the mixed core configuration as is indicated in the UFSAR, they would be limited to a small fraction of the total core and would, therefore, result in an off-site release of fission product activity which is only a small fraction of that which would occur under LOCA conditions (i.e., off-site doses would be far below the limits given in 10 CFR 100).

5.4 STEAM GENERATOR TUBE AND MAIN STEAM LINE RUPTURES

Analyses are presented in Sections 14.2.4 and 14.2.5 which show that no fuel rods will experience transition boiling as a result of a steam generator tube rupture or a main steam line break. Consequently, since no fuel failures are expected to result from these incidents, the off-site thyroid and whole body dose calculations are based on the assumption of operation with equilibrium fission product activity in the primary and secondary systems determined by assuming that 1% of the fuel cladding is defective.

Similar analyses for ENC 17x17 fuel[6], operating at the increased reactor power level, support the contention that no fuel failures will result from generator tube ruptures or main steam line breaks.

Since the technical specification limit for coolant activity remains unchanged and no fuel failures occur as a result of the incidents, the estimated thyroid and whole body doses given in Section 14.2.7 of the Unit 1 UFSAR as a function of the primary-to-secondary leak rate remain unchanged. In addition, since the inventory of iodine available in the fuel rod gaps of the ENC 17x17 design at proposed power and exposure levels is less than for the design base conditions, the effect of iodine spiking behavior will be less than for the cases considered in the UFSAR.

5.5 LOCA EVENT

Section 14.3.5 of the Unit 2 UFSAR notes that the environmental consequences of a loss-of-coolant accident for Unit 1 is conservatively valid for Unit 2. That analysis, presented in Section 14.3.5 of the Unit 1 UFSAR, includes evaluations of the off-site radiological consequences of 1) the design base accident, and 2) the maximum hypothetical accident.

The Design Base Accident (DBA) assumes that the entire inventory of volatile fission products contained in the pellet-cladding gap is released. That inventory, as given in Table 14.3.5-2 and Table 14A.2-1 of the Unit 1 UFSAR is based on an end-of-cycle core averaged exposure of 24,560 MWD/MTU with a rated reactor power of 3391 MW_t. Consequently,

the off-site dose for proposed changes in power and end-of-cycle average fuel exposure are proportional to the change in fission product inventory within the pellet-clad gap.

For the maximum hypothetical accident, the fission product source is assumed to be a fixed fraction of the total core inventory of volatile fission products. As for the DBA, the inventory is based on the end-of-cycle core average values. Consequently, for the hypothetical accident assumptions, the off-site dose resulting from operations at the proposed higher power and end-of-cycle fuel exposure is proportional to the change in the total core inventory of volatile fission products.

Off-site doses which could result from a loss-of-coolant accident involving fuel operating at the proposed reactor power and exposure levels are given in Table 9 and compared with values determined in the UFSAR. As can be seen, the estimated off-site radiological consequences of a LOCA are within the limits specified in 10 CFR 100.

6.0 REFERENCES

1) M. J. Bell, "ORIGEN - The ORNL Isotope Generation and Depletion Code", Oak Ridge National Laboratory, ORNL-4628 (May 1973).

2) XN-NF-81-58(P), "RODEX 2 Fuel Rod Thermal Mechanical Response Evaluation Model", Exxon Nuclear Company, Inc., August 1981.

3) J. R. Houston, et. al., "DACRIN - A Computer Program for Calculating Organ Dose from Acute or Chronic Radionuclide Inhalation", Battelle Pacific Northwest Laboratories, BNWL-B-389/UC-41 (December 1974, Reissued April 1976).

4) R. I. Scherpelz and A. E. Desrosiers, "Doses Received While Crossing a Plume of Radioactive Material Released During an Accident at a Nuclear Power Plant", Health Physics, Vol. 43, No. 2 (August 1982), pp. 187-203.

5) R. O. Gumprecht, "Mathematical Basis of Computer Code RIBD", Douglas United Nuclear, Inc., DUN-4136 (June 1968).

6) XN-NF-82-32(P), "Plant Transient Analysis for the Donald C. Cook Unit 2 Reactor at 3,425 MW_t", Exxon Nuclear Company, Inc., April 1982.

7) XN-NF-82-37, "D. C. Cook Unit 2, Cycle 4 Safety Analysis Report", Exxon Nuclear Company, Inc., April 1982.

8) XN-NF-78-44, "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors", Exxon Nuclear Company, Inc., January 1979.

Table 1
Assumptions for Calculating Fission Product Inventories

	Core Activity		Worst Case Fuel Assembly	
	Base Case 15x15	ENC 17x17	Base Case 15x15	ENC 17x17
Rated Power (MW_t)	3391*	3425**	3391*	3425**
Actual Power (102% of Rated)	3459	3494	3459	3494
Fuel Exposure (MWD/MTU)	24,560	30,000	42,400	43,000
Radial Peaking Factor ($F_{\Delta H}$)	N/A	N/A	1.55*	1.55*
Local Peaking Factor	1.0	1.0	1.0	1.0
Total Peaking Factor	1.99*	2.10**	1.99*	2.10**

* Limiting value from technical specifications.

** Proposed limiting value.

Table 2
CORE AND GAP FISSION PRODUCT ACTIVITIES

Isotope	Design Base 15x15 Fuel Assembly EOC Average Fuel Exposure of 24,560 MWD/MTU Rated Reactor Power 3391 MW_t			ENC 17x17 Fuel Assembly Design EOC Average Fuel Exposure of 30,000 MWD/MTU Rated Reactor Power 3425 MW_t		
	Curies In Core	Fraction Of Activity In Gap	Curies In Gap	Curies In Core	Fraction Of Activity In Gap	Curies In Gap
I-129	2.35	$.5113 \times 10^{-1}$	0.12	2.54	$.2317 \times 10^{-1}$	0.06
I-131	9.57×10^{7}	$.1336 \times 10^{-1}$	1.28×10^{6}	1.02×10^{8}	$.5057 \times 10^{-2}$	5.16×10^{5}
I-132	1.37×10^{8}	$.4114 \times 10^{-2}$	5.64×10^{5}	1.45×10^{8}	$.1569 \times 10^{-2}$	2.28×10^{5}
I-133	1.86×10^{8}	$.8279 \times 10^{-2}$	1.54×10^{6}	1.96×10^{8}	$.3142 \times 10^{-2}$	6.16×10^{5}
I-134	2.12×10^{8}	$.5214 \times 10^{-3}$	1.11×10^{5}	2.22×10^{8}	$.1982 \times 10^{-3}$	4.40×10^{4}
I-135	1.67×10^{8}	$.4643 \times 10^{-2}$	7.75×10^{5}	1.75×10^{8}	$.1767 \times 10^{-2}$	3.09×10^{5}
Total I	7.98×10^{8}		4.27×10^{6}	8.40×10^{8}		1.71×10^{6}
Kr-85	8.35×10^{5}	$.2123 \times 10^{-1}$	1.77×10^{4}	8.79×10^{5}	$.9053 \times 10^{-2}$	7.96×10^{3}
Kr-85m	2.36×10^{7}	$.2129 \times 10^{-2}$	5.02×10^{4}	2.39×10^{7}	$.8120 \times 10^{-3}$	1.94×10^{4}
Kr-87	4.67×10^{7}	$.1639 \times 10^{-3}$	7.65×10^{3}	4.71×10^{7}	$.6238 \times 10^{-4}$	2.94×10^{3}
Kr-88	6.83×10^{7}	$.1722 \times 10^{-2}$	1.18×10^{5}	6.90×10^{7}	$.6568 \times 10^{-3}$	4.53×10^{4}
Kr-89	8.61×10^{7}	$.2382 \times 10^{-3}$	2.05×10^{4}	8.66×10^{7}	$.9069 \times 10^{-4}$	7.85×10^{3}
Total Kr	2.26×10^{8}		2.14×10^{5}	2.28×10^{8}		8.35×10^{4}
Xe-131m	7.00×10^{5}	$.1312 \times 10^{-1}$	9.18×10^{3}	7.53×10^{5}	$.4986 \times 10^{-2}$	3.75×10^{3}
Xe-133	1.86×10^{8}	$.6255 \times 10^{-2}$	1.16×10^{6}	1.95×10^{8}	$.2387 \times 10^{-2}$	4.65×10^{5}
Xe-133m	5.02×10^{6}	$.1011 \times 10^{-1}$	5.08×10^{4}	5.29×10^{6}	$.3847 \times 10^{-2}$	2.04×10^{4}
Xe-135	3.86×10^{7}	$.1498 \times 10^{-2}$	5.78×10^{4}	3.44×10^{7}	$.5711 \times 10^{-3}$	1.96×10^{4}
Xe-135m	4.96×10^{7}	$.5259 \times 10^{-3}$	2.61×10^{4}	5.22×10^{7}	$.2003 \times 10^{-4}$	1.05×10^{4}
Xe-137	1.81×10^{8}	$.2322 \times 10^{-3}$	4.20×10^{4}	1.92×10^{8}	$.8839 \times 10^{-3}$	1.70×10^{4}
Xe-138	2.03×10^{8}	$.3797 \times 10^{-3}$	7.71×10^{4}	2.16×10^{8}	$.1446 \times 10^{-3}$	3.12×10^{4}
Total Xe	6.64×10^{8}		1.42×10^{6}	6.96×10^{8}		5.67×10^{5}

Table 2 Continued

Cs-134	1.35×10^{7}	$.2935 \times 10^{-1}$	3.96×10^{5}	1.71×10^{7}	$.1270 \times 10^{-1}$	2.17×10^{5}
Cs-134m	5.04×10^{6}	$.2476 \times 10^{-2}$	1.25×10^{4}	6.43×10^{1}	$.9437 \times 10^{-3}$	6.07×10^{3}
Cs-135	1.66×10^{1}	$.2935 \times 10^{-1}$	0.49	1.58×10^{1}	$.1270 \times 10^{-1}$	0.20
Cs-136	4.49×10^{6}	$.2574 \times 10^{-1}$	1.16×10^{5}	5.24×10^{6}	$.9809 \times 10^{-2}$	5.14×10^{4}
Cs-137	7.00×10^{6}	$.2935 \times 10^{-1}$	2.05×10^{5}	7.63×10^{6}	$.1270 \times 10^{-1}$	9.69×10^{4}
Cs-138	1.71×10^{8}	$.1067 \times 10^{-2}$	1.82×10^{5}	1.81×10^{8}	$.4060 \times 10^{-3}$	7.35×10^{4}
Cs-139	1.73×10^{8}	$.5789 \times 10^{-3}$	1.00×10^{4}	1.81×10^{8}	$.2203 \times 10^{-3}$	3.99×10^{4}
Cs-140	1.59×10^{8}	$.1974 \times 10^{-3}$	3.14×10^{4}	1.67×10^{8}	$.7510 \times 10^{-4}$	1.25×10^{3}
Cs-141	1.15×10^{8}	$.1190 \times 10^{-3}$	1.37×10^{3}	1.20×10^{7}	$.4527 \times 10^{-4}$	5.43×10^{3}
Cs-142	9.00×10^{7}	$.3688 \times 10^{-4}$	3.32×10^{3}	9.43×10^{7}	$.1403 \times 10^{-4}$	1.32×10^{2}
Cs-143	4.69×10^{7}	$.3435 \times 10^{-4}$	1.61×10^{3}	4.89×10^{7}	$.1306 \times 10^{-4}$	6.39×10^{2}
Total Cs	7.85×10^{8}		1.06×10^{6}	8.29×10^{8}		5.05×10^{5}
Te-125m	1.84×10^{5}	.1923	3.54×20^{4}	2.07×10^{5}	$.7935 \times 10^{-1}$	1.64×10^{4}
Te-127	7.23×10^{6}	$.1727 \times 10^{-1}$	1.25×10^{5}	7.96×10^{6}	$.6539 \times 10^{-2}$	5.21×10^{4}
Te-127m	1.48×10^{6}	.2451	3.63×10^{5}	1.63×10^{7}	.1078	1.76×10^{5}
Te-129	3.60×10^{7}	$.6035 \times 10^{-2}$	2.17×10^{5}	3.88×10^{6}	$.2296 \times 10^{-2}$	8.91×10^{4}
Te-129m	2.16×10^{6}	.2341	5.06×10^{5}	2.36×10^{6}	.1013	2.39×10^{5}
Te-131	8.34×10^{7}	$.3626 \times 10^{-2}$	3.02×10^{5}	8.91×10^{7}	$.1382 \times 10^{-2}$	1.23×10^{5}
Te-131m	1.43×10^{7}	$.3082 \times 10^{-1}$	4.41×10^{6}	1.52×10^{8}	$.1172 \times 10^{-1}$	1.78×10^{5}
Te-132	1.32×10^{8}	$.4955 \times 10^{-1}$	6.54×10^{5}	1.41×10^{7}	$.1890 \times 10^{-3}$	2.66×10^{6}
Te-133	5.18×10^{7}	$.2568 \times 10^{-2}$	1.33×10^{5}	5.47×10^{8}	$.9774 \times 10^{-2}$	5.35×10^{4}
Te-133m	1.43×10^{8}	$.5135 \times 10^{-2}$	7.34×10^{5}	1.51×10^{8}	$.1953 \times 10^{-2}$	2.95×10^{5}
Te-134	1.87×10^{8}	$.4707 \times 10^{-2}$	8.80×10^{5}	1.96×10^{8}	$.1790 \times 10^{-2}$	3.51×10^{5}
Te-135	3.92×10^{7}	$.1029 \times 10^{-2}$	4.03×10^{4}	4.11×10^{7}	$.3910 \times 10^{-3}$	1.61×10^{4}
Total Te	6.97×10^{8}		1.03×10^{7}	7.39×10^{8}		4.25×10^{6}

Table 3

FUEL ASSEMBLY AND GAP FISSION PRODUCT ACTIVITIES 100 HOURS AFTER SHUTDOWN

	Worst Case 15x15 Fuel Assembly EOL Exposure of 42,400* MWD/MTU Rated Reactor Power 3391 MW_t			Worst Case ENC 17x17 fuel Assembly EOL Exposure of 43,000 MWD/MTU Rated Reactor Power 3425 MW_t		
Nuclide	Ci Per Fuel Assembly 100 Hours After Shutdown	Fraction Of * Activity In Gap	Curies* In Gap	Ci Per Fuel Assembly 100 Hours After Shutdown	Fraction Of Activity In Gap	Curies In Gap
I-131	5.89×10^5	.2950	1.74×10^5	6.06×10^5	.2959	1.79×10^5
I-132	4.78×10^5	.2826	1.35×10^5	4.89×10^5	.2619	1.28×10^5
I-133	5.75×10^4	.2925	1.68×10^4	5.90×10^4	.2936	1.73×10^4
I-135	4.26×10^1	.0657	2.80	4.36×10^1	.0601	2.62
Total I	1.13×10^6		3.26×10^5	1.15×10^6		3.24×10^5
Kr-85	6.58×10^3	.2584	1.70×10^3	6.02×10^3	.2434	1.47×10^3
Total Kr	6.58×10^3		1.70×10^3	6.02×10^3		1.47×10^3
Xe-131m	6.49×10^3	.2949	1.91×10^3	6.26×10^3	.2956	1.85×10^3
Xe-133	1.02×10^6	.2882	2.94×10^5	1.07×10^6	.2806	3.00×10^5
Xe-133m	1.58×10^4	.2932	4.63×10^3	1.64×10^4	.2932	4.81×10^3
Xe-135	1.91×10^3	.1747	3.34×10^2	1.95×10^3	.1601	3.12×10^2
Total Xe	1.04×10^6		3.01×10^5	1.10×10^6		3.07×10^5
Cs-134	2.01×10^5	.2891	5.81×10^4	1.82×10^5	.2699	4.91×10^4
Cs-136	4.34×10^4	.3209	1.39×10^4	4.12×10^4	.3061	1.26×10^4
Cs-137	6.49×10^4	.2891	1.88×10^4	5.73×10^4	.2699	1.55×10^4
Total Cs	3.09×10^5		9.08×10^4	2.81×10^5		7.72×10^4
Te-125m	2.10×10^3	.3204	6.73×10^2	1.83×10^3	.3948	7.22×10^2
Te-127	4.48×10^4	.2982	1.34×10^4	4.56×10^4	.2983	1.36×10^4
Te-127m	1.45×10^4	.2987	4.33×10^3	1.47×10^4	.3689	5.42×10^3
Te-129	3.25×10^4	.2881	9.36×10^3	3.39×10^4	.2783	9.43×10^3
Te-129m	5.06×10^3	.3109	1.57×10^4	5.29×10^3	.3683	1.95×10^4
Te-131	2.26×10^4	.2817	6.37×10^2	2.31×10^4	.2601	6.01×10^2
Te-131m	1.24×10^4	.3576	4.43×10^3	1.27×10^4	.3106	3.94×10^3
Te-132	4.65×10^5	.3862	1.80×10^5	4.77×10^5	.3206	1.53×10^5
Total Te	6.24×10^5		2.29×10^5	6.41×10^5		2.06×10^5

* Using the ANS 5.4 fission gas release model results in a calculated internal rod pressure which exceeds system pressure for the 15x15 fuel when irradiated at average power to 42,400 MWD/MTU. Gap release fractions, therefore, were calculated at a more realistic case of 33,000 MWD/MTU exposure but applied to the fission product inventories which are consistent with those used in the UFSAR.

Table 4
COMPARISON OF COMPUTED FISSION PRODUCT ACTIVITIES

EOC CORE AVERAGED EXPOSURE

	Core Activity			Gap Activity		
	UFSAR	ORIGEN Base Case	ORIGEN Proposed	UFSAR	ORIGEN Base Case	ORIGEN Proposed
Iodine	7.86×10^8	7.89×10^8	8.40×10^8	4.79×10^6	4.27×10^6	1.71×10^6
Krypton	2.28×10^8	2.26×10^8	2.28×10^8	7.93×10^5	2.14×10^5	8.35×10^4
Xenon	2.90×10^8	6.64×10^8(a)	6.96×10^8(a)	3.72×10^6(a)	1.42×10^6(a)	5.67×10^5(a)
Cesium	----	7.85×10^8	8.29×10^8	----	1.06×10^6	5.05×10^5
Tellurium	----	6.98×10^8	7.39×10^8	----	1.03×10^7	4.25×10^6
Totals	1.30×10^9	3.17×10^9	3.33×10^9	9.30×10^6	1.73×10^7	7.12×10^6

WORST CASE ASSEMBLY AT TIME OF REACTOR SHUTDOWN

	Assembly Activity			Gap Activity		
Iodine	5.60×10^6(b)	6.55×10^6	6.66×10^6	2.87×10^4	1.50×10^6	1.49×10^6
Krypton	1.48×10^6	1.45×10^6	1.54×10^6	8.27×10^3	1.51×10^5	1.43×10^5
Xenon	1.73×10^6	4.89×10^6(c)	5.43×10^6(c)	2.22×10^4(c)	6.03×10^5(c)	6.16×10^5(c)
Cesium	----	6.47×10^6	6.51×10^6	----	4.51×10^5	4.10×10^5
Tellurium	----	6.88×10^6	6.63×10^6	----	1.87×10^6	1.64×10^6
Totals	8.81×10^6	2.62×10^7	2.68×10^7	5.92×10^4	4.58×10^6	4.30×10^6

(a) Includes short lived isotopes not included in the UFSAR total. Considering only those isotopes in the base case as were included in the UFSAR data, the total number of curies calculated by ORIGEN is 2.79×10^8 Ci which compares well with the UFSAR value of 2.90×10^8 Ci.

(b) Evaluated with an assumed radial peaking factor of 1.37 rather than the limiting value of 1.55 as was assumed in the ORIGEN calculations.

(c) Includes short lived isotopes which were not included in the UFSAR data since they decay prior to 100 hours after shutdown. ORIGEN calculated data equivalent to the xenon activity given in the UFSAR results in an activity of 1.75×10^6 Ci which is in excellent agreement with the UFSAR value of 1.73×10^6 Ci.

Table 5
Summary of Activities Considered in Accident Evaluations

	Core Activity (Curies)				Worst Assembly Activity -100 Hours After Shutdown (Curies)			
	Core Activity		Gap Activity		Core Activity		Gap Activity	
Material	Base Case	Proposed	Base Case	Proposed	Base Case	Proposed	Base Case	Proposed
Iodine	7.98×10^{8}	8.40×10^{8}	4.27×10^{6}	1.71×10^{6}	1.13×10^{6}	1.15×10^{6}	3.26×10^{5}	3.24×10^{5}
Krypton	2.26×10^{8}	2.28×10^{8}	2.14×10^{5}	8.35×10^{4}	6.58×10^{3}	6.02×10^{3}	1.70×10^{3}	1.47×10^{3}
Xenon	6.64×10^{8}	6.96×10^{8}	1.42×10^{6}	5.67×10^{5}	1.04×10^{6}	1.10×10^{6}	3.01×10^{5}	3.07×10^{5}
Cesium	7.85×10^{8}	8.29×10^{8}	1.06×10^{6}	5.05×10^{5}	3.09×10^{5}	2.81×10^{5}	9.08×10^{4}	7.72×10^{4}
Tellurium	6.98×10^{8}	7.39×10^{8}	1.03×10^{7}	4.25×10^{6}	6.24×10^{5}	6.41×10^{5}	2.29×10^{5}	2.06×10^{5}
Total	3.17×10^{9}	3.33×10^{9}	1.73×10^{7}	7.12×10^{6}	3.11×10^{6}	3.18×10^{6}	9.49×10^{5}	9.16×10^{5}

Table 6
Constants Relating Off-Site Doses From Accidents
Involving Design Base and Proposed Operations

Radiation Source	A_I*	DCF_I*	E*	C*
Worst Case Assembly (Gap Activity)	0.993	1.036	0.995	0.939
Worst Case Assembly (Total Activity)	1.026	1.003	0.982	1.022
Core Average Fuel (Total Activity)	1.053	1.010	0.998	1.050
Core Average Fuel (Gap Activity)	0.400	1.005	1.000	0.412

* Constants are as defined in Section 3.0.

Table 7
Fuel Handling Accident in the Auxiliary Building
Two Hour Site Boundary Doses

	Design Base Operation*		Proposed Operation		
	Expected Case-Rem	Conservative Case-Rem	Expected Case-Rem	Conservative Case-Rem	10 CFR 100 Limit-Rem
Whole Body	0.22	0.51	0.21	0.48	25
Thyroid	0.84	2.15	0.86	2.21	300

* See Table 14.2.1-4 of the Updated Final Safety Analysis Report for DC Cook Units 1 and 2.

Table 8
Fuel Handling Accident Inside Containment
Two Hour Site Boundary Doses

	Design Base Operation*		Proposed Operation		
	Conservative Case-Rem	Realistic Case-Rem	Conservative Case-Rem	Realistic Case-Rem	10 CFR 100 Limit-Rem
Whole Body	1.3	3.73×10^{-4}	1.3	3.74×10^{-4}	25
Thyroid	82.3	5.61×10^{-4}	84.7	5.77×10^{-4}	300

* See page 14.2.1-13 of Section 14.2.1.2 of the Updated Final Safety Analysis Report for DC Cook Units 1 and 2.

Table 9 Off-Site Doses Resulting From a Loss of Coolant Accident

	Design Base 15x15 Fuel*				ENC 17x17 High Exposure Fuel			
	Site Boundary 0-2 hours x = 610 meters		Low Population Zone 0-30 days x = 3218 meters		Site Boundary 0-2 hours x = 610 meters		Low Population Zone 0-30 days x = 3218 meters	
	Thyroid Dose (Rems)	Whole Body Dose (Rems)	Thyroid Dose (Rems)	Whole Body Dose (Rems)	Thyroid Dose (Rems)	Whole Body Dose (Rems)	Thyroid Dose (Rems)	Whole Body Dose (Rems)
10 CFR 100 Guideline Limit	300	25	300	25	300	25	300	25
Design Basis Accident (Zero Ice Condenser I_2 Removal Efficiency)	2.45	0.031	3.88	0.052	0.98	0.013	1.56	0.021
Maximum Hypothetical Accident (Zero Ice Condenser I_2 Removal Efficiency)	154	7.64	221.8	9.78	164	8.01	236	10.25
Design Basis Accident								
20% Ice Condenser I_2 Removal Efficiency	2.37	-	3.85	-	0.95	-	1.55	-
40% Ice Condenser I_2 Removal Efficiency	2.35	0.031	3.81	0.05	0.94	0.0013	1.53	0.02
Maximum Hypothetical Accident								
20% Ice Condenser I_2 Removal Efficiency	148	-	220.4	-	158	-	235	-
40% Ice Condenser I_2 Removal Efficiency	146	7.6	220	9.7	155	8.0	234	10.2

* See Table 14.3.5-7 of the Updated Final Safety Report for DC Cook Units 1 and 2.

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