RESPONSES TO NRC QUESTIONS
ON CEN-122(B)-P ENTITLED
"ENVIRONMENTAL IMPACT
OF EXTENDED BURNUP FUEL
CYCLES IN CALVERT CLIFFS
UNITS 1 AND 2" DATED
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CEN-186(B)-NP

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Question 1

Does BG&E intend to submit a safety analysis in the near future? Is it necessary to review the safety analysis report concurrent with the environmental report?

Response

A reload safety evaluation will be submitted concurrent with Unit 1 Cycle 6 reload license application by February 15, 1982. We do not believe that it is necessary to concurrently review the safety evaluation for the reasons which follow.

The accident analysis does not provide estimates of releases during anticipated operational occurrences and postulated accidents on a best estimate basis. Rather, operating conditions and analysis assumptions are conservatively bounded for the purpose of demonstrating that all applicable regulations and acceptance criteria are met for any allowable operating condition or mode. The safety analysis for Calvert Cliffs Unit 1 Cycle 6 will demonstrate that all applicable regulations, including those applicable to site boundary doses, are met. Further, it is not anticipated that the calculated site boundary dose for Cycle 6 will differ significantly from those reported in Cycle 5 because of the small extension in discharge burnup, the lower power density of high burnup fuel, similarities in radioactive source terms, and the bounding nature of the Cycle 5 analysis.

Anticipated Operational Occurrences (AOO) are constrained by the criteria that fuel failure during the event is precluded. For these events, residual coolant activity (from corrosion product activation and residual fuel failures) serves as the source term for site boundary dose calculations. Coolant activity levels are limited by the Technical Specifications, and it is assumed that coolant activity levels are at the Technical Specification limits when calculating site boundary doses. Since no changes to these Technical Specifications will be prepared for Cycle 6, calculated site boundary doses will not be altered.

For postulated accidents, fuel failure constitutes an added radiological source term, in addition to the residual coolant activity previously mentioned. Fuel which is predicted to fail during these events is that which operates at the highest power densities. Since high burnup fuel operates at relatively low power densities, failure is not predicted. Also, as discussed in response to Question 5, the gas gap activity in high burnup fuel will be significantly lower than that present in lower burnup/higher power density assemblies used in the analysis to calculate site boundary dose.

In summary, the site boundary doses for anticipated operational occurrences and other postulated accidents for Cycle 6 are not expected to be significantly different from those of the reference cycle, and consequently should not affect the environmental assessment. The Cycle 6 safety analysis will, of course, be reviewd by the NRC to insure that all appropriate criteria and regulations are met.

Question 2

Will the increased burnup consider the increased probability of fuel failure?

Response

C-E's experience with the operation of PWR fuel rods to extended burnups has shown a reduced frequency of fuel failure with increased burnups. The primary reason for the improved performance at high burnups is probably the reduced linear heat ratings associated with the fuel which operates at the extended burnups. Data from lead test assemblies and data from the overall performance of C-E's fuel will be discussed below to support the position that increased burnup does not necessarily increase the probability of failure.

A program conducted in cooperation with EPRI and Baltimore Gas and Electric provided an opportunity to extend the irradiation of a lead assembly to 43,000 MWd/T. The assembly was examined, in detail, after each of its four operating cycles in Calvert Cliffs-I. The performance of the assembly was excellent, and the specific measurements taken have provided a valuable benchmark for the design equations supporting the extended irradiation of standard 14x14 fuel. Eight test rods from that lead assembly have been transplanted to another assembly which had three prior cycles. These rods are expected to achieve burnups ranging from 55,000 to 57,000 MWd/T at the end of the current cycle.

Another program is underway at Ft. Calhoun and is being performed in cooperation with the U.S. Department of Energy and Omaha Public Power. In the initial stage of that program, a sub-batch of 17 standard fuel assemblies were examined after a fourth cycle with burnups up to 37,000 MWd/T. All seventeen assemblies were found to be intact and no anomalies were observed. One of the assemblies was selected for an additional cycle in the center position of the core, and has recently completed that cycle with an assembly average burnup of 45,000 MWd/T. The lead rod has achieved a burnup of 49,000 MWd/T. There has been no indication of any problems with this lead assembly, and a detailed examination will be conducted during October of 1981. By mutual agreement of all parties involved, the assembly will be returned to the core for still higher exposures, as long as the surveillance measurements confirm design projections for a successful sixth cycle. The calculated targets for that cycle are 52,000 MWd/T for the assembly and 56,000 MWd/T for the lead fuel rod.

Data from all eight of C-E's plants has been evaluated to assess any relationship of fuel failure to burnup level. In most cases, it was necessary to rely on the activity levels of the coolant, as sipping of the fuel assemblies has been unnecessary. [

] for purposes of the statistics which follow. In cases where the iodine levels changed to some higher level, the escape rate coefficient was used to estimate the number of failed rods. In those cases where sipping was performed, examinations of the fuel assemblies permitted a more direct count of the number of failed rods. Table 1 shows the number of failed rods resulting from the operation of fuel in various cycles and the burnups for the various cycles. Although the data are somewhat limited, the trend is dramatic and very encouraging. Table 1 shows that [ ]

[ ]

C-E believes that this operating experience supports the operation of fuel to higher exposures, without necessarily increasing the number of fuel failures.

Lead test assembly programs, of the type discussed above, are being continued to add confidence to the reliable operation of PWR fuel to extended burnups. The fundamental decline in linear heat ratings which accompanies the high burnup assemblies is the reason to expect very low incidence of fuel failure at high burnups. The experience cited above, supports this and the statistical validity of the observation will grow gradually.

TABLE 1

FUEL PERFORMANCE STATISTICS

CYCLES OF EXPOSURE	1	2	3	4	5
Fuel Assembly Burnup Range (GWd/MTU)	0-20	10-30	23-35	34-43	44-45
Number of Fuel Assemblies (Fuel Rods)					
Discharged	690 (131,143)	851 (144,678)	500 (84,592)	21 (3,676)	---
Operating	368 (67,556)	439 (83,022)	377 (68,156)	14 (2,458)	1 (176)
Total	1058 (198,699)	1290 (227,700)	877 (152,748)	35 (6,134)	1 (176)

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Question 3

Provide a comparison of fission product inventories (using applicable codes) for the normal and extended burnup cases.

Response

Table 2 compares fission product inventories at the same power level calculated using the ORIGEN code for burnups of 28,800 and 43,200 MWD/MTU.

Table 2

Fission Product Inventories from ORIGEN (Gram-atoms/MTU)

Isotope	28,800 MWD/MTU	43,200 MWD/MTU
Br-84	2.79-5	2.58-5
Kr-85m	3.05-4	2.80-4
Kr-85	2.56-1	3.59-1
Kr-87	1.56-4	1.40-4
Kr-88	4.93-4	4.45-4
Rb-88	5.29-5	4.79-5
Rb-89	5.88-5	5.28-5
Sr-89	3.08-1	2.76-1
Sr-90	5.47+0	7.85+0
Y-90	1.49-3	2.15-3
Sr-91	2.93-3	2.69-3
Y-91	4.60-1	4.20-1
Zr-95	7.52-1	7.30-1
Mo-99	3.49-2	3.49-2
Ru-103	4.79-1	5.10-1
Ru-106	1.49+0	1.90+0
Te-129	1.12-4	1.18-4
I-129	1.32+0	2.04+0
I-131	5.66-2	5.76-2
Xe-131m	5.92-4	6.03-4
Te-132	3.26-2	3.30-2
I-132	9.72-4	9.85-4
I-133	1.21-2	1.21-2
Xe-133	7.43-2	7.43-2
Te-134	2.96-4	2.80-4
I-134	5.47-4	5.43-4
Cs-134	1.21+0	2.23+0
I-135	3.57-3	3.55-3
Xe-135	9.40-4	8.80-4
Cs-136	5.17-3	8.16-3
Cs-137	8.27+0	1.22+1
Xe-138	1.08-4	1.06-4
Cs-138	2.62-4	2.56-4
Ba-140	1.53-1	1.51-1
La-140	2.07-2	2.08-2
Ce-144	2.47+0	2.58+0
Pr-143	1.38-1	1.35-1

Question 4

Will the accident analyses consider items such as increased fission gas pressure? The iodine DF for the spent fuel pool is a function of fission gas pressure, and this is a fuel handling accident consideration.

Response

As discussed in the following, an increase in the burnup of the fuel will not result in an increase in the fuel handling accident doses. The current analysis for standard burnup fuel employs a bubble size and a bubble rise time which are based on a design maximum pressure. Since this design pressure will not be exceeded for increased burnup fuel, this aspect of the analysis remains conservative. Further, the radioactive gas released from a fuel assembly that has attained high burnup will be less than the radioactive gas release assumed for the fuel handling accident in the FSAR. The FSAR accident analysis assumes that the radioactive gas is released from the hottest fuel assembly in the core which is also assumed to be at the maximum power. The high burnup fuel assemblies will be operating at a much lower power and temperature. Since the significant radioiodine fission products have a half-life less than 8.1 days, the gas gap inventory is determined by the operating history just prior to discharge. Measurements show that the gas release from UO_2 is extremely temperature dependent. Even considering enhancement due to high burnup, the radioactive iodine inventory in the gas gap is expected to be over a factor of ten lower than the gas gap inventory assumed in the fuel handling accident.

Question 5

Are there any changes expected with respect to iodine spiking behavior.

Response

Changes in the iodine spiking behavior are not expected to occur due to extended fuel burnup. This conclusion is based on studies of the spiking phenomenon and analysis of plant data. Correlations have been made to identify the initiating events and relationships between the magnitude and rate of increase of the spike and various plant parameters and core characteristics. The studies referred to herein are those described in References 1 through 8.

The primary initiating events of iodine spikes are power transients (both increasing and decreasing), and pronounced reactor coolant pressure decreases. The mechanisms which cause the iodine spikes to occur may differ slightly for power increases, power decreases, initial depressurization and final depressurization, but the studies indicate:

1. The iodine spiking is not a result of generation of new fuel cladding defects, but is directly related to the number of defective fuel rods already present; and

2. The magnitude and rate of increase of the spike is a function of the iodine inventory deposited within the rod gas gap which becomes accessible to reactor coolant which has leaked into the gap via cladding defects.

As indicated in the response to Question 3 and 4, the extended fuel burnup will result in a negligible (less than 2%) increase in total core iodine inventory. The gas gap iodine inventories will actually decrease relative to those attained during the annual cycle due to the lower fuel rod temperatures associated with the extended burnup. Therefore, the extended fuel burnup will not result in increased iodine spiking behavior. Furthermore, operation of the reactor will continue to be based on the same radiological technical specification limits as are presently being employed.

References (Question 5)

1. "Fission Product Release after Reactor Shutdown", N. Eickelpasch and R. Hock, IAEA Symposium, Vienna 1973. Paper IAEA-SM-178/19, 1974

2. "Shutdown Release of Fission Products in LWRs", G.G. Eigenwillig and R. Hock, Trans, Am. Nucl. Soc. 23 (1976)

3. "The Iodine Spiking Phenomenon in Pressurized Water Reactor Coolants", W. Chubb, E.J. Piplica, R.J. Lutz, Trans. Am. Nucl. Soc. 26 (1977)

4. "Iodine Spiking-Cause and Effect", R.J. Lutz and W. Chubb, Trans. Am. Nucl. Soc. 28 (1978)

5. "Iodine Spiking in PWRs: Origin and General Behavior", K.H. Neeb and E. Schuster, Trans. Am. Nucl. Soc. 28 (1978)

6. "Iodine Release Mechanism and Its Verification in Plant Operation", N. Eickelpasch, R. Seepolt, and R. Hock, Trans. Am. Nucl. Soc. 28 (1978)

7. "Iodine Behavior Under Transient Conditions in the Pressurized Water Reactor", WCAP-8637, R.J. Luntz, November, 1975

8. "Iodine Spiking: Radioiodine Behavior in the Reactor Coolant System during Transient Operations", CENPD-180-P, G.F. Caruthers and P.K. Green, March, 1976