ENCLOSURE 1

MFN 14-001

Markup of GESTAR II US Supplement

Non-Proprietary Information – Class I (Public)

$$E_2 = W_G H_G + \int_0^{H_B} \frac{W_b}{H_B} y dy = W_G H_G + \frac{1}{2} W_B H_B$$

$$E_2 = (619\ lb)\left(\frac{160}{12}\right) + \frac{1}{2}(562)\left(\frac{160}{12}\right) = 12{,}000\ ft - lb.$$

As before, the energy is considered to be absorbed equally by the falling assembly and the impacted assemblies. The fraction available for clad deformation is 0.510. The energy available to deform the unfailed cladding in the impacted assemblies is one–half the energy resulting from the second impact:

$$E_c = (0.5)\ (12{,}000\ ft\text{–}lb)\ (0.510) = 3{,}060\ ft\text{–}lb$$

and the number of failures in the impacted assemblies is:

$$N_F = \frac{3{,}060\ ft - lb}{200\ ft - lb} = 15\ rods.$$

Since the rods in the dropped 9x9 assembly are considered to have failed in the initial impact, the total failed rods resulting from both impacts is 125 + 15 = 140.

The above analysis was completed using the GE12, ~~and~~ GE14, and GNF2 10x10 fuel rod arrays (References S–77, ~~and~~ S–95, and S-111)~~. The analysis~~, all three of which resulted in 172 failed rods from both impacts.

This compares with 111 failed rods from the analysis for the 7x7 fuel rod array bundle presented in the individual plant FSAR.

Radiological Consequences Comparisons. For the purposes of this evaluation, it is conservatively assumed that the ~~fractional plenum~~relative activity for any 9x9 rod will be 49 rods/74 rods, or 0.66 times the activity in a 7x7 rod. Based on the assumption that 140 9x9 rods fail compared to 111 for a 7x7 core, the relative amount of activity released for the 9x9 fuel is (140/111) (0.66) = 0.83 times the activity released for a 7x7 core. The activity released to the environment and the radiological exposures for all GE 9x9 fuel designs will therefore be less than 83% of those values presented in the FSAR for a 7x7 core. As identified in the FSAR, the radiological exposures for the 7x7 fuel are well below those guidelines set forth in 10CFR100; therefore, it can be concluded that the consequences of this accident with the new NF–500 mast and the 9x9 fuel will also be well below these guidelines.

A fuel bundle damage analysis and the resulting radiological consequences for the new NF–500 mast and the 8x8 fuel shows that the activity released to the environment and the radiological exposures will be less than 84% of those values presented in the FSAR for a 7x7 core. Similar to the above evaluation, the activity released to the environment and the radiological exposures for ~~all~~ the GE 10x10 fuel designs, expressed as a fraction of the values presented in

the FSAR for a 7x7 core, are as follows: ~~will therefore be less than (172/111)(49/87.33) = 0.87 or 87% of those values presented in the FSAR for a 7x7 core.~~

- GE12: (172/111)(49/86.4) = 0.88 times the activity released for a 7x7 core.
- GE14: (172/111)(49/85.84) = 0.88 times the activity released for a 7x7 core.
- GNF2: (172/111)(49/85.6) = 0.89 times the activity released for a 7x7 core.

S.2.3 Analysis Initial Conditions and Inputs

Inputs to the models utilized to analyze the AOO events discussed in Section S.2.2 are plant unique. The specific inputs related to the plant pressure relief systems (i.e., safety valves, safety/relief valves, etc.) are listed in the supplemental reload licensing report for each plant. Inputs such as thermal power, dome pressure, etc. are given in the individual plant supplemental reload licensing report. The initial conditions for the GETAB analysis are listed in the supplemental reload licensing report for each specific plant. Because the AOO model establishes operating conditions, only licensing basis values are given in the supplemental reload licensing report.

Cycle–dependent initial conditions for the GETAB analysis and the resulting reload parameters are given in the plant FSAR or the supplemental reload licensing report.

S.3 Vessel Pressure ASME Code Compliance Model

The pressure relief system was designed to prevent excessive overpressurization of the primary system process barrier and the pressure vessel and thereby precludes an uncontrolled release of fission products.

Prior to 1967, the design capacities of the safety valves for BWRs were determined according to the requirements of Section I, *Power Boilers*, of the ASME Boiler and Pressure Vessel Code. Under the provisions of this code, safety valve capacities were established to prevent either a vessel or pressure rise greater than 6% above the maximum allowable working pressure. At least one safety valve was to be set at or below the maximum allowable working pressure; the highest safety valve setting could not exceed 103% of the maximum allowable working pressure. No credit was allowed for reactor scram as a complementary pressure protection device. Thus, the required safety valve capacities were sized assuming essentially instantaneous isolation of the pressure vessel with no pressure relief other than that from the safety valves. Nine Mile Point–1 and Oyster Creek are the only plants that were designed to these criteria.

In 1991 Oyster Creek updated its overpressurization analysis (Reference S–88) to ASME Boiler and Pressure Code, Section III to be consistent with later BWRs and reducing the number of safety valves.

S-111 *GNF2 Advantage Generic Compliance with NEDE-24011-P-A (GESTAR II)*, NEDC-33270P, Revision 5, May 2013.