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RS-01-319

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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Clinton Power Station, Unit 1
Facility Operating License No. NPF-62
NRC Docket No. 50-461

Subject: Additional Information Supporting the License Amendment Request to Revise Plant System Requirements During Fuel Handling Based on Alternative Source Term

Reference: Letter from J. M. Heffley (AmerGen Energy Company, LLC) to U.S. NRC, "Request for Amendment to Technical Specifications that Revise Plant System Requirements During Fuel Handling Based on Alternative Source Term," dated July 5, 2001

In the above reference, AmerGen Energy Company (AmerGen), LLC submitted a request for changes to Appendix A, Technical Specifications (TS), of Facility Operating License No. NPF-62 for Clinton Power Station (CPS). The proposed changes would revise requirements that apply during the movement of irradiated fuel and during Core Alterations. The NRC requested additional information in a telephone conversation regarding the proposed changes in the above reference. The attachment to this letter provides the NRC requested information.

Should you have any questions related to this information, please contact Mr. Timothy A. Byam at (630) 657-2804.

Respectfully,

J.W. Simpkin

for K. R. Jury
Director – Licensing
Mid-West Regional Operating Group

A001

Attachments:

Affidavit

Attachment: Additional Information Supporting the License Amendment Request to Revise Plant System Requirements During Fuel Handling Based on Alternative Source Term

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Clinton Power Station
Office of Nuclear Facility Safety – Illinois Department of Nuclear Safety

STATE OF ILLINOIS	)	
COUNTY OF DUPAGE	)	
IN THE MATTER OF	)	
AMERGEN ENERGY COMPANY, LLC	)	**Docket Number**
CLINTON POWER STATION, UNIT 1	)	**50-461**

SUBJECT: **Additional Information Supporting the License Amendment Request to Revise Plant System Requirements During Fuel Handling Based on Alternative Source Term**

AFFIDAVIT

I affirm that the content of this transmittal is true and correct to the best of my knowledge, information and belief.

T. W. Simpkin

T. W. Simpkin
Manager – Licensing
Mid-West Regional Operating Group

Subscribed and sworn to before me, a Notary Public in and for the State above named, this 28 day of December, 2001.

Anese L. Grigsby
Notary Public

OFFICIAL SEAL
ANESE L. GRIGSBY
NOTARY PUBLIC, STATE OF ILLINOIS
MY COMMISSION EXPIRES 3-13-2005

Question 1

In Attachment "A" to your submittal, you stated that your radiological consequence analysis was based on the GE-14 fuel bundle type that was conservatively assumed to have been operated at a reactor power level of 3542 MWt. Provide the assumed fission product inventory in the reactor core for each radionuclide of interest (noble gases and halogens). State the corresponding maximum fuel burnups assumed for the average-core assembly and peak-power assembly that bound those achievable from the Clinton Power Station (CPS) core management plans.

Response 1

A calculation was performed (Reference 1) to analyze the dose consequences at the site boundary and in the main control room following a Fuel Handling Accident using an Alternative Source Term (AST). Table 1 below contains the assumed fission product inventory in the reactor core. Provided in this table is the core activity in curies (Ci)/Megawatt-thermal (MWt) for a burnup of 42 Gigawatt-days per metric ton (GWd/MT), which represents the inventory used for the Fuel Handling Accident (FHA) assuming the reactor has operated at 3542 MWt for 1605 days. The core activity was developed using a linear regression technique. This approach provides a reasonable approximation of the core inventory at 42 GWd/MT.

Table 1

Clinton Power Station
Core Activity by Noble Gas, Iodine and Parent Isotope

Activity (Ci/MWt) at Burnup (GWd/MT)

Nuclide	Activity (Ci/MWt) for fuel Burnup of 42 GWd/MT
Se-83	1.237 E+03
Se-83m	1.825 E+03
Se-85	2.924 E+03
Se-87	4.618 E+03
Se-88	1.759 E+03
Se-89	5.139 E+02
Br-83	3.135 E+03
Br-85	6.350 E+03
Br-87	1.025 E+04
Br-88	1.073 E+04
Br-89	7.356 E+03
Kr-83m	3.149 E+03
Kr-85	3.590 E+02
Kr-85m	6.444 E+03
Kr-87	1.217 E+04
Kr-88	1.710 E+04
Kr-89	2.054 E+04
Sn-131	8.475 E+03
Sn-132	4.505 E+03
Sn-133	1.367 E+03
Sn-134	2.001 E+02
Sb-131	2.270 E+04
Sb-132	1.331 E+04
Sb-133	1.518 E+04
Sb-134	2.640 E+03
Te-131	2.428 E+04
Te-131m	4.034 E+03
Te-132	3.878 E+04
Te-133	3.223 E+04
Te-133m	1.931 E+04
Te-134	4.375 E+04
Te-137	3.506 E+03
Te-138	8.843 E+02

Table 1 (continued)

Clinton Power Station
Core Activity by Noble Gas, Iodine and Parent Isotope

Activity (Ci/MWt) at Burnup (GWd/MT)

I-131	2.739 E+04
I-132	3.945 E+04
I-133	5.473 E+04
I-134	5.982 E+04
I-135	5.128 E+04
I-137	2.335 E+04
I-138	1.154 E+04
Xe-131m	3.073 E+02
Xe-133	5.346 E+04
Xe-133m	1.730 E+03
Xe-135	1.652 E+04
Xe-135m	1.099 E+04
Xe-137	4.765 E+04
Xe-138	4.418 E+04

Question 2

Summarize the total fission product activity assumed in the fuel rod gap that is available for release to the water surrounding the failed fuel assembly and provide the assumed amounts of fission product activities (in curies) released to the environment following the postulated fuel handling accident.

Response 2

Table 2 below provides the activity of noble gases and iodines released to the reactor cavity following a 24-hour decay period after shutdown. The 42 GWd/MT values identified in Table 1 above were then decayed using the ORIGEN-S computer code to develop the core inventory at 24 hours after shutdown. The values listed in Table 2 Column A were obtained from the ORIGEN-S computer code output. The core activity includes the activity released to the reactor cavity following a FHA in containment and is based on the estimated fuel damage, gap fractions, and radial peaking factors. The values in the last column represent the activity released to the reactor cavity following the 24-hour decay period which is input to the RADTRAD and PERC2 computer codes to estimate the dose consequences at the site boundary and in the CPS main control room.

Table 2

Activity Released to Reactor Cavity Due to FHA in Containment at T=24 hours[1,2]

Isotope	Core Inventory at 42 GWd/MT (Ci/MWt) A	Per Bundle Inventory (Ci/MWt) C=A/624	Radial Peaking Factor D	Fuel gap Fractions E	Activity Release during FHA (Ci/MWt) F=C*D*E* 172/92
I-131	2.55E+04	4.09E+01	1.7	0.08	1.04E+01
I-132	3.23E+04	5.18E+01	1.7	0.05	8.23E+00
I-133	2.52E+04	4.04E+01	1.7	0.05	6.42E+00
I-134	1.31E-03	2.10E-06	1.7	0.05	3.34E-07
I-135	4.08E+03	6.54E+00	1.7	0.05	1.04E-01
Kr-83m	1.42E+01	2.28E-02	1.7	0.05	3.62E-03
Kr-85	3.59E+02	5.75E-01	1.7	0.10	1.83E-01
Kr-85m	1.59E+02	2.55E-01	1.7	0.05	4.05E-02
Kr-87	2.57E-02	4.12E-05	1.7	0.05	6.54E-06
Kr-88	4.88E+01	7.82E-02	1.7	0.05	1.24E-02
Xe-131m	3.06E+02	4.90E-01	1.7	0.05	7.79E-02
Xe-133	5.16E+04	8.27E+01	1.7	0.05	1.31E+01
Xe-133m	1.55E+03	2.48E+00	1.7	0.05	3.95E-01
Xe-135	1.35E+04	2.16E+01	1.7	0.05	3.44E+00
Xe-135m	6.66E+02	1.07E+00	1.7	0.05	1.70E-01

Notes to Table 2:

1. Alkali metals released from the gap are not reported as they are retained in the reactor cavity water and there is no halogen or noble gas progeny produced from their decay.

2. There are 624 fuel bundles in the core, each bundle having 92 fuel rods. It is assumed that 172 fuel rods will be damaged following a FHA in the containment.

Question 3

Describe the plant operating procedures to close the open airlock doors in the event of the postulated FHA. State if the airlock doors are capable of being closed, any cables or hoses crossing the airlock doors have quick-disconnects to ensure that the doors are capable of being closed in a timely manner, and a designated individual is available outside each open airlock to close the door in the event of the postulated FHA.

Response 3

The dose calculation supporting this amendment request assumed that the two containment personnel hatches (i.e., airlocks) and the containment equipment hatch are open at the time of the event, i.e., no credit is taken for containment closure during the event. As stated in our submittal, CPS will administratively implement the provisions of Section 11, "Assessment of Risk Resulting From Performance of Maintenance Activities," subsection 3.6.5 of Nuclear Utilities Management and Resources Council (NUMARC) document NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 3, pertaining to the capability to restore secondary containment. Station procedure 4979.07, "Dropped Fuel Bundle," an off-normal procedure, provides the required actions following a dropped fuel bundle. This procedure contains both automatic actions as well as required operator actions. The current revision of this procedure contains requirements to take appropriate actions necessary to isolate the area, to reduce or redirect the released radioactivity, to stop the degradation of conditions and mitigate their consequences, and to initiate actions to verify or re-establish Secondary Containment, and if needed, Primary Containment, following a dropped fuel bundle. It is possible that hoses or cables could be routed through the airlocks or the equipment hatch. Quick-disconnects are typically utilized for cables or hoses that pass through a primary containment barrier.

During a typical refueling outage, a Drywell Coordinator is stationed at the drywell access control point, which is located in very close proximity to the equipment hatch, to coordinate work activities inside the drywell. A dropped fuel bundle warning system is required to be installed in the drywell prior to starting any refueling activities. The system has alarms that are responded to by the radiation protection personnel and the Drywell Coordinator at the access control point located near the 737'-0" elevation. The Drywell Coordinator is responsible for ensuring that actions are initiated in a timely manner, including notifying the control room, to quickly restore primary or secondary containment in the event of a fuel handling accident. All of the Drywell Coordinators receive training on the actions associated with a dropped fuel bundle.

Question 4

Table 2 of Attachment "A" lists four scenarios for control room HVAC system operation following the postulated FHA to assess the sensitivity of the control room operator doses. Describe each scenario in detail with flow diagrams. State the significance and bases for each parameter and assumption used (timing and flow rates.)

Response 4

Table 2 of Attachment A to the Reference 2 submittal describes the four scenarios that were considered for the Alternative Source Term (AST) analysis. The four scenarios were analyzed to assess the sensitivity of the dose to various control room ventilation system operation and filtration levels. Scenarios 1 and 2 selected for the control room ventilation system analysis are the same scenarios used in the evaluation of control room doses supporting the Technical Specification changes associated with the Feedwater Leakage Control System requested in Reference 3 and approved with the issuance of License Amendment 127 (Reference 6). A simplified control room ventilation system diagram is attached that addresses all four scenarios.

In scenario 1, upon an automatic initiation signal of the control room emergency ventilation system (i.e., detection of high radiation by the control room intake monitors), the control room is assumed to be automatically isolated. In other words, the only flow into the control room for the first 20 minutes is from unfiltered inleakage and no flow is assumed from either the makeup fan or the supply/return fans. During the first 20 minutes, the calculation conservatively utilizes the "isolated" control room model per Standard Review Plan 6.4, "Control Room Habitability Systems." It is assumed that the unfiltered inleakage into the control room is 1510 cubic feet per minute (cfm) (i.e., one-half of the flow needed to maintain pressurization of the control room plus an additional contribution of 10 cfm for ingress or egress) per Reference 5. After 20 minutes, it is assumed that the control room is put into the emergency filtration mode by manual operator action. As shown in the attached figure, the emergency filtration mode (i.e., the "hi rad" mode), makeup fan provides makeup air through the control room makeup air filter and supply filter train. The return fan takes suction on the control room envelope (i.e., control room habitability zone) and mixes 61,000 cfm flow with the makeup air and returns it to the control room through the supply filter train. To maximize dose, the analysis for this scenario assumes that after 20 minutes, the control room ventilation system is assumed to be fully operational in the high radiation mode with a control room ventilation intake filtered flow of 3300 cfm (i.e., 3000 cfm +10%) and a filtered recirculation flow of 54,900 cfm (i.e., 61,000 cfm –10%). Additionally, during this emergency pressurized ventilation mode, there is an assumed 650 cfm inleakage flow that enters the control room upstream of the supply filters (i.e., activity is filtered by recirculation system before entering the control room).

In scenario 2, the control room emergency ventilation system automatically initiates following an initiation signal. Therefore, scenario 2 is the same as scenario 1 after 20 minutes. A conservative filtered intake rate of 3920 cfm is assumed as a special worst case value intended to maximize beta/gamma doses due to submersion. In addition, unfiltered inleakage is assumed to be 10 cfm for ingress and egress and 650 cfm is an assumed inleakage upstream of the supply filters. The filtered recirculation flow rate is 54,900 cfm, which is the same as scenario 1.

In scenario 3, the impact of the FHA was evaluated assuming the control room emergency ventilation system *without* makeup and recirculation flow filtration. The more limiting of the previously analyzed scenarios (i.e., scenario 1) was selected to perform this evaluation. The timing and flow rates were assumed to be the same as in scenario 1 with the only difference being the lack of makeup and recirculation filtration.

In scenario 4, the impact of utilizing the control room unfiltered normal operation ventilation during a FHA was evaluated. The makeup air intakes are cross connected downstream of the outside air dampers. Therefore, either train of the control room ventilation system can take suction from either intake. This allows makeup air to be drawn from the intake with the lowest radiation levels following a high radiation initiation signal. For the purposes of this analysis, the control room ventilation model continues to take credit for the "dual intake" design and the ability to select the intake with the lower radiation levels. As a result, it is assumed that the normal operation unfiltered ventilation flow is supplied via both control room intakes, and that the control room intake radiation monitors are operable. Scenario 4 assumes that the control room intake flow is 4400 cfm (i.e., 4000 cfm +10%). The model assumes an unfiltered inleakage of 660 cfm which includes the 10 cfm inleakage due to ingress and egress. Similar to the conditions assumed in scenario 3, this scenario assumed no intake or recirculation filtration.

References:

1. Stone and Webster Calculation 086457022-UR(B)-002, Revision 0, "Site Boundary and Control Room Dose following a FHA in Containment using Alternative Source Term"
2. Letter U-603485 from J. M. Heffley (AmerGen Energy Company, LLC), "Request for Amendment to Technical Specifications that Revise Plant System Requirements During Fuel Handling Based on Alternative Source Term," to U. S. NRC, dated July 5, 2001
3. Letter U-603032 from W. G. MacFarland (Illinois Power Company), "Application for Amendment of Facility Operating License No. NPF-62 for Clinton Power Station (LS-97-006)," to U. S. NRC, dated October 23, 1998
4. USAR Section 9.4.1.2, Control Room HVAC System, System Description
5. Calculation C-020
6. Letter from U. S. NRC to M. Reandeau (AmerGen), "Issuance of Amendment - Clinton Power Station, Unit 1," dated April 25, 2000.

Simplified VC Diagram - A Train "Hi Rad" Mode