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RS-01-305

December 20, 2001

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 Reactor Systems Information Supporting the License Amendment Request to Permit Uprated Power Operation at Clinton Power Station

References: (1) Letter from J. M. Heffley (AmerGen Energy Company, LLC) to U.S. NRC, "Request for License Amendment for Extended Power Uprate Operation," dated June 18, 2001

(2) Letter from K. R. Jury (Exelon Generation Company, LLC) to U.S. NRC, "Additional Reactor Systems Information Supporting the License Amendment Request to Permit Uprated Power Operation at Clinton Power Station," dated November 21, 2001

In Reference 1, AmerGen Energy Company (AmerGen), LLC submitted a request for changes to the Facility Operating License No. NPF-62 and Appendix A to the Facility Operating License, Technical Specifications (TS), for Clinton Power Station (CPS) to allow operation at an uprated power level. The proposed changes in Reference 1 would allow CPS to operate at a power level of 3473 megawatts thermal (MWt). This represents an increase of approximately 20 percent rated core thermal power over the current 100 percent power level of 2894 MWt. The NRC, in conference calls on December 3, 2001 and December 4, 2001, requested additional information regarding the proposed response in Reference 2. The attachment to this letter provides the requested information.

A portion of the information in Attachment A is proprietary to the General Electric Company, and AmerGen requests that it be withheld from public disclosure in accordance with 10 CFR 2.790, "Public inspections, exemptions, requests for withholding," paragraph (a)(4). The proprietary information is indicated with sidebars. Attachment B provides the affidavit supporting the request for withholding the proprietary information in Attachment A from public disclosure, as required by 10 CFR 2.790, paragraph (b)(1). Attachment C contains a non-proprietary version of Attachment A.

A001

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

Respectfully,

Keith R. Jury

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

Attachments:

Affidavit
Attachment A: Additional Reactor Systems Information Supporting the License Amendment Request to Permit Uprated Power Operation at Clinton Power Station (Proprietary version)
Attachment B: Affidavit for Withholding Portions of Attachment A from Public Disclosure
Attachment C: Additional Reactor Systems Information Supporting the License Amendment Request to Permit Uprated Power Operation at Clinton Power Station (Non-Proprietary version)

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 Reactor Systems Information Supporting the License Amendment Request to Permit Uprated Power Operation at Clinton Power Station**

AFFIDAVIT

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

Keith R. Jury

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

Subscribed and sworn to before me, a Notary Public in and

for the State above named, this 20 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

ATTACHMENT A

Additional Reactor Systems Information Supporting the License Amendment Request to Permit Uprated Power Operation at Clinton Power Station (Proprietary)

Supplemental Question 3.9

ELTR1 and ELTR2 include a requirement to perform an evaluation for an EPU equilibrium core of the stability Option III OPRM amplitude trip setpoint. Please explain why it is not necessary for the OPRM setpoint to be calculated for the EPU equilibrium core and why it does not need to be included in the PUSAR submittal.

Response 3.9

The Oscillation Power Range Monitor (OPRM) instrumentation setpoint depends upon cycle-specific parameters. It must be calculated on a cycle-specific basis per the NRC-approved licensing basis methodology for reload analysis contained in Reference 1. Therefore, the value used in the plant when extended power uprate (EPU) is implemented is the value calculated for the corresponding cycle. Previously, an OPRM setpoint was calculated for an equilibrium EPU cycle to demonstrate the adequacy of the methodology for EPU. However, this setpoint was based on an equilibrium cycle that would never actually be used in the plant. Even when an equilibrium cycle is achieved, a cycle-specific calculation is still required. Multiple applications for EPU have shown that it is not necessary for the EPU equilibrium cycle calculation to be performed in order to assure that the methodology is adequate for the required cycle-specific setpoint evaluation.

To summarize, the OPRM instrumentation setpoint calculation is not performed for an EPU equilibrium cycle for the following reasons.

- It has been fully demonstrated that the OPRM setpoint methodology is adequate and acceptable for an EPU performed in accordance with Reference 2 (i.e., ELTR1) and Reference 3 (i.e., ELTR2)
- An OPRM setpoint calculated for an EPU equilibrium core would never be used in the plant
- The setpoint is always calculated on a cycle-specific basis

Supplemental Question 3.11

Discuss how the limited CPPU analysis builds on the full scope SAFER analysis for the plant.

Response 3.11

The response below describes how the constant pressure power uprate (CPPU) loss of coolant accident (LOCA) analysis builds on the full scope SAFER analysis for the plant. This description is followed by a brief summary of how this limited approach was applied to the Clinton Power Station (CPS) EPU.

CPPU LOCA Analysis Approach

The LOCA analysis for CPPU builds on the existing SAFER/GESTR-LOCA analysis for the plant. The basic break spectrum response is not affected by CPPU. There are two limiting points on the break spectrum: the full sized recirculation line break, and the worst small recirculation line break with failure of the high pressure emergency core cooling system (ECCS). The break spectrum response is determined by the ECCS network

General Electric Company

AFFIDAVIT

I, **George B. Stramback,** being duly sworn, depose and state as follows:

(1) I am Project Manager, Regulatory Services, General Electric Company ("GE") and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in Attachment 1 to letter GE-CPS-AEP-088, *Response to NRC Audit Request Regarding EPU – LOCA*, dated December 10, 2001. The proprietary information in Attachment 1 (*GE-CPS-AEP-088, GE Responses to NRC Audit Requests EPU – LOCA,* (GE Company Proprietary)), is identified by bars marked in the margin adjacent to the specific material.

(3) In making this application for withholding of proprietary information of which it is the owner, GE relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), 2.790(a)(4), and 2.790(d)(1) for "trade secrets and commercial or financial information obtained from a person and privileged or confidential" (Exemption 4). The material for which exemption from disclosure is here sought is all "confidential commercial information", and some portions also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by General Electric's competitors without license from General Electric constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

c. Information which reveals cost or price information, production capacities, budget levels, or commercial strategies of General Electric, its customers, or its suppliers;

d. Information which reveals aspects of past, present, or future General Electric customer-funded development plans and programs, of potential commercial value to General Electric;

e. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in both paragraphs (4)a. and (4)b., above.

(5) The information sought to be withheld is being submitted to NRC in confidence. The information is of a sort customarily held in confidence by GE, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GE, no public disclosure has been made, and it is not available in public sources. All disclosures to third parties including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within GE is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his delegate), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GE are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information identified in paragraph (2), above, is classified as proprietary because it contains further details regarding the GE proprietary report NEDC-32989P, *Safety Analysis Report for Clinton Power Station Extended Power Uprate,* Class III (GE Proprietary Information), dated June 2001, which contains detailed results of analytical models, methods and processes, including computer codes,

which GE has developed, obtained NRC approval of, and applied to perform evaluations of transient and accident events in the GE Boiling Water Reactor ("BWR").

The development and approval of these system, component, and thermal hydraulic models and computer codes was achieved at a significant cost to GE, on the order of several million dollars.

The development of the evaluation process along with the interpretation and application of the analytical results is derived from the extensive experience database that constitutes a major GE asset.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GE's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GE's comprehensive BWR safety and technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods.

The research, development, engineering, analytical and NRC review costs comprise a substantial investment of time and money by GE.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

GE's competitive advantage will be lost if its competitors are able to use the results of the GE experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to GE would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive GE of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing these very valuable analytical tools.

STATE OF CALIFORNIA)
) ss:
COUNTY OF SANTA CLARA)

George B. Stramback, being duly sworn, deposes and says:

That he has read the foregoing affidavit and the matters stated therein are true and correct to the best of his knowledge, information, and belief.

Executed at San Jose, California, this 10th day of December 2001.

George B. Stramback
George B. Stramback
General Electric Company

Subscribed and sworn before me this 10th day of December 2001.

TERRY J. MORGAN
Commission # 1304914
Notary Public - California
Santa Clara County
My Comm. Expires May 18, 2005

Terry J. Morgan
Notary Public, State of California

ATTACHMENT C

Additional Reactor Systems Information Supporting the License Amendment Request to Permit Uprated Power Operation at Clinton Power Station (Non-Proprietary)

Supplemental Question 3.9

ELTR1 and ELTR2 include a requirement to perform an evaluation for an EPU equilibrium core of the stability Option III OPRM amplitude trip setpoint. Please explain why it is not necessary for the OPRM setpoint to be calculated for the EPU equilibrium core and why it does not need to be included in the PUSAR submittal.

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To summarize, the OPRM instrumentation setpoint calculation is not performed for an EPU equilibrium cycle for the following reasons.

- It has been fully demonstrated that the OPRM setpoint methodology is adequate and acceptable for an EPU performed in accordance with Reference 2 (i.e., ELTR1) and Reference 3 (i.e., ELTR2)
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- The setpoint is always calculated on a cycle-specific basis

Supplemental Question 3.11

Discuss how the limited CPPU analysis builds on the full scope SAFER analysis for the plant.

Response 3.11

ATTACHMENT C

Additional Reactor Systems Information Supporting the License Amendment Request to Permit Uprated Power Operation at Clinton Power Station (Non-Proprietary)

Additional Reactor Systems Information Supporting the License Amendment Request to Permit Uprated Power Operation at Clinton Power Station (Non-Proprietary)

Supplemental Question 3.14

Discuss the SLCS system configuration especially the injection path to the RPV. Clarify why the SLCS system pump discharge relief valves will not open during an ATWS event with loss of offsite power at CPS. Discuss the relief valve setpoint margin. Also explain the differences between CPS and the Dresden and Quad Cities plants, where the pump discharge relief valves are postulated to open during an ATWS event.

Response 3.14

Standby Liquid Control System (SLCS) Injection Configuration

At CPS, the SLCS injects through the core spray sparger instead of a SLCS control standpipe located in the reactor lower plenum. The SLCS control standpipe configuration is used in all of the BWR/2 and BWR/3 plants and most of the BWR/4 plants. The core spray configuration is used in many of the BWR/5 and BWR/6 plants and provides for a significantly lower system injection pressure when operating both SLCS pumps. This is primarily due to the high-pressure losses associated with the control standpipe compared with the significantly lower pressure losses associated with injection through the core spray sparger. The length of the SLCS injection pipe run also has an impact on the injection pressure.

The power-actuated relief function for the CPS safety/relief valves (SRVs) is designed as safety related. The limiting reactor transient pressures would be associated with either the main steam isolation valve (MSIV) closure ATWS event or the pressure regulator failed open ATWS event. The loss of offsite power (LOOP) ATWS event is not the limiting event for CPS because the air supply to the SRVs remains available for this event. The 11 SRVs equipped to operate in the ADS mode and/or the low-low-set (LLS) mode have sufficient pneumatic supply capacity to meet the reactor pressure relief requirements during a LOOP ATWS event. The mix of ADS and LLS valves does not result in any change in the peak or minimum reactor pressures during SLCS operation when compared to the assumptions used in the ATWS analysis.

Operation of Pump Discharge Relief Valves

The calculated maximum required pump discharge pressure, based on the peak reactor pressure during the limiting ATWS event, is below the lowest calculated nominal opening pressure for the SLCS pump relief valves. Consequently, the pump discharge relief valves will not open during system injection.

The peak reactor pressure during SLCS operation is obtained from the results of the ATWS analysis. The long-term transient reactor pressure is dependent on the reactor power level, and the number, type, and capacity of the safety relief valves credited for reactor pressure control during ATWS events. The ATWS analysis assumes that the operator will manually initiate the SLCS about two minutes after occurrence of the event. The analysis also assumes continuous SLCS injection at rated pump flow rates.

The analysis is based on the calculated system flow and head losses with two pump operation (83 psi), and a pump relief valve tolerance of $\pm$ 3 percent of the nominal setpoint (1400 psig). These valves are periodically tested to maintain this tolerance. In addition, a 30-psi pressure margin is applied to the calculations to account for any pressure pulsations originating from the cyclic increase in the volumetric output of the positive displacement pumps.

Relief Valve Setpoint Margin

The relief valve setpoint margin is defined as the pressure difference between the maximum required pump discharge pressure and the minimum pressure needed for the pump discharge relief valves to open. The maximum required pump discharge pressure is based on the pump discharge pressure required for rated injection into the reactor at the peak reactor pressure expected during the time SLCS injection is required following an ATWS event. The minimum relief valve pressure setpoint is based on the lower setpoint limit for the valves, plus an allowance for the cyclic pressure pulsations developed by the positive displacement triplex pumps. For the CPS SLCS, this lower pressure would be 72 psi (0.03 x 1400 + 30) below the nominal setpoint for the pump discharge relief valve.

The operation of the SLCS also is analyzed to confirm that the pump discharge relief valves will reclose in the event that the system is initiated before the reactor pressure recovers from the first transient peak. The evaluation compares the calculated maximum reactor pressure needed for the pump discharge valves to reclose with the lower reactor pressures expected during the time the SRVs are cycling open and closed.

The pump discharge relief valves are required to have a reseat pressure not more than 10% below the nominal setpoint pressure for the relief valves (1400 x 0.9 = 1260 psig). Based on this requirement and the system flow and head losses for full injection, the maximum reactor pressure to permit reclosure can be determined (1176 psig). The cyclic reduction in pump volumetric output (typical of triplex positive displacement pumps) is expected to result in a cyclic reduction in the pressure at the inlet of the relief valves, causing the relief valves to reclose at a nominal valve inlet pressure above 1260 psig. Because there is some uncertainty in regard to the response of the relief valves to these cyclic pressure pulsations, no credit has been taken for this pressure effect.

Because the calculated maximum reactor pressure of 1176 psig is greater than the reactor pressure at the time the SRVs reclose (reported as 1009 psig in the ATWS analysis), the SLCS pump discharge relief valves will reclose, thus permitting continuous full injection flow to the reactor.

Differences Between CPS and other BWRs
The major differences between the CPS and certain previous EPU plants are identified in the following bullets and summarized in Table 3.14-1:

- Injection is through the core spray sparger instead of the higher pressure loss control stand pipe. System injection flow losses are 97 psi compared with 160 to 258 psi.
- The SLCS pump discharge relief valve nominal setpoint is 1400 psig instead of the 1500 psig.
- The peak ATWS reactor pressure during SLCS operation is 1185 psig compared to 1270 psig.

ATTACHMENT C

Additional Reactor Systems Information Supporting the License Amendment Request to Permit Uprated Power Operation at Clinton Power Station (Non-Proprietary)

Table 3.14-1

SLCS Differences

	CPS	Previous EPU Plants
Injection location	Core spray sparger	Control stand pipe in lower reactor plenum
Number of SLCS pumps required to operate for ATWS compliance	2	2
2-pump - Design injection rate	86 gpm	90 gpm
SRV description	Dikkers dual mode SRVs (i.e., solenoid and spring actuated): 1 at 1133 psig and 8 at 1143 psig	Dresser SRV: 1 at 1135 psig, Dresser relief valves: 2 at 1115 psig and 2 at 1135 psig, Dresser Spring Safety Valves: 2 at 1252 psig
SLCS pump nominal relief valve setpoint, psig	1400	1500
Pump relief valve setpoint tolerance, %	± 3 (± 42 psig)	± 3 (± 45 psig)
Pump relief valve pulsation margin, psi	30	30
Pump relief valve lower pressure for opening - including valve tolerance and pump pulsation allowance, psig	1328	1425
Peak ATWS reactor pressure during SLCS operation, psig	1185	1270
Total SLCS injection flow losses, psi	97	160 / 258 / 220
Injection elevation head, psi	- 14	- 24 / - 24 / - 21
Maximum pump discharge pressure at the peak ATWS reactor pressure during SLCS operation, psig	1268	1406 / 1504 / 1469
SLCS pump relief valve pressure margin, psi	60	19 / - 79 / - 44

Additional Reactor Systems Information Supporting the License Amendment Request to Permit Uprated Power Operation at Clinton Power Station (Non-Proprietary)

Supplemental Question 3.19

Previous information provided to the NRC indicated that the results of the ATWS instability analysis documented in NEDO-32047-A and NEDO-32164 are applicable to the CPS 120% power uprate. Please explain how the impact of Final Feedwater Temperature Reduction, Feedwater Heater Out-of-Service, and flatter radial power distribution are addressed by this analysis.

Response 3.19

Final Feedwater Temperature Reduction (FFWTR) - The purpose of FFWTR is to enable a reactor to remain at full power as fuel reactivity is used up near the end of an operating cycle. FFWTR inserts reactivity to compensate for the reduced fuel reactivity. Thus, the initial pre-runback condition is no more limiting considering total core reactivity content than the limiting initial full power operating condition prior to entering the FFWTR operating state.

In addition, there is a feedwater temperature reduction associated with the recirculation flow runback event, which occurs prior to the onset of the reactor instability. The analysis of the ATWS instability response assumes that all feedwater heating is lost during the event and feedwater temperature approaches the main condenser hotwell temperature. The temperature reduction from the full power operating condition to the hot well temperature will be less for an initial FFWTR operating state than for normal full power operation. Therefore, there will be less reactivity insertion from the feedwater temperature reduction for the FFWTR initial condition than from the normal full power operating condition. Thus, there will be a smaller total core reactivity for FFWTR conditions following the recirculation runback and feedwater temperature equilibration as compared to the limiting operating condition previously evaluated.

Feedwater Heater Out-of-Service (FWHOOS) - Occurrence of an ATWS has been judged to be of very low probability. In addition, ATWS is not designated as a design basis event. Therefore, the licensing basis for ATWS events is that they may be performed with best-estimate models and assumptions. One example of this is use of nominal feedwater temperature for all ATWS analysis, even if a plant has the FWHOOS flexibility option. Note that this is different than the ATWS analysis assumption for plants that have the safety/relief valves out-of-service (SRVOOS) flexibility option. Since SRVOOS may be a condition which (a) exists for a long period of reactor operation, (b) has a significant impact on the plant response for a limiting ATWS event (with reactor isolation), and (c) impacts reactor vessel integrity (i.e., peak reactor pressure), the SRVOOS condition is included in ATWS analyses. This may be contrasted with the FWHOOS option which (a) would not be expected to persist throughout a cycle due to fuel utilization economic incentives, (b) is expected to impact the propensity to oscillate more than the magnitude of the oscillation, and (c) does not impact vessel integrity. In addition, FWHOOS would not impact the effectiveness of mitigation strategies to insert control rods, lower reactor water level, or inject liquid boron.

When a plant is operating with FWHOOS, increased reactivity insertion from colder feedwater is compensated for by establishing a control rod pattern which has less fuel reactivity, so that the total core power remains in the licensed operating domain. Limiting ATWS evaluations assume that all feedwater heating is lost after a reactor recirculation system flow control valve runback and the temperature of feedwater injected following a runback is at the main condenser hot well temperature. The condenser hot well temperature is not affected by FWHOOS. Thus, as compared to the normal feedwater heating condition, the post runback FWHOOS condition will have reduced fuel reactivity since it is for a less limiting control rod pattern but the same reactivity insertion due to feedwater injection at the hot well temperature, which will produce reduced total power generation when feedwater temperature equilibrates. For these reasons, the FWHOOS flexibility option is not considered in ATWS analysis.

Flatter Radial Power Distribution - Fuel bundle design limitations result in core designs with a flatter radial power distribution in order to implement power uprate conditions. The flatter power distribution means that more bundles will have power generation near to the highest power bundle in the core. Since higher power bundles have a greater pressure drop and corresponding lower channel flow, the flatter power profile means that there is also a flatter core flow profile. This is beneficial for the ATWS instability mitigation strategies of lowering reactor water level and injecting borated water since there is a more uniform flow distribution across the core. However, more fuel rods could experience extended dryout if there were no ATWS instability mitigation actions performed.

The studies performed for Reference 4 (i.e., NEDO-32047) and Reference 5 (i.e., NEDO-32164) were performed for 8x8 GE fuel designs and limiting full power operating conditions. These conditions are limiting for many plant-specific power uprate conditions. However, they were not limiting for the CPS EPU maximum extended load line limit analysis (MELLLA) initial condition. Sensitivity studies have been performed for GE14 fuel (a 10x10 design) and for a more limiting full power operating condition than the CPS EPU MELLLA initial condition. The condition used in the sensitivity studies also had a flatter radial power distribution. These sensitivity studies show a similar fully coupled neutronic/thermal-hydraulic reactor instability power and flow response to the cases reported in References 4 and 5. However, the GE14 fuel has a lower heat flux per rod than the 8x8 or 9x9 fuel bundle designs and is less susceptible to extended fuel rod dryout than previously reported in Reference 4. Therefore, it is expected that the extent of fuel damage for an ATWS instability event without mitigation, from a condition more limiting than CPS EPU MELLLA conditions, with GE14 fuel and a flatter radial power distribution, is bounded by the fuel damage calculation reported in Reference 4.

Additional Reactor Systems Information Supporting the License Amendment Request to Permit Uprated Power Operation at Clinton Power Station (Non-Proprietary)

References

1. General Electric Company, "Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications," NEDO-32465-A, August 1996

2. General Electric Company Licensing Topical Report, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate," NEDC-32424P-A, Class III, February 1999

3. General Electric Company Licensing Topical Report, "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate," NEDC-32523P-A, Class III, February 2000, and Supplement 1, Volumes I and II

4. General Electric Company, "ATWS Rule Issues Relative to BWR Core Thermal-hydraulic Stability," NEDO-32047-A, February 24, 1992

5. General Electric Company, " Mitigation of BWR Core Thermal-Hydraulic Instabilities in ATWS," NEDO-32164, Revision 0, December 1992

ATTACHMENT B

Affidavit for Withholding Portions of Attachment A from Public Disclosure