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RS-03-193

October 10, 2003

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

Dresden Nuclear Power Station, Units 2 and 3
Facility Operating License Nos. DPR-19 and DPR-25
NRC Docket Nos. 50-237 and 50-249

Subject: Additional Information Regarding Request for Technical Specifications Changes Related to Main Steam Safety Valve Operability Requirements

Reference: Letter from P. R. Simpson (Exelon Generation Company, LLC) to U. S. NRC, "Request for Technical Specifications Changes Related to Main Steam Safety Valve Operability Requirements," dated October 10, 2002

In the referenced letter, Exelon Generation Company, LLC (EGC) requested changes to the Technical Specifications (TS) of Facility License Nos. DPR-19 and DPR-25 for the Dresden Nuclear Power Station (DNPS), Units 2 and 3. The proposed changes increase the number of main steam safety valves that are required to be operable from eight to nine. In a communication from Mr. L. W. Rossbach of the NRC on August 4, 2003, and in a teleconference between Ms. M. Banerjee and other members of the NRC and Mr. A. R. Haeger and other members of EGC on August 14, 2003, the NRC requested additional information regarding these proposed changes. The attachments to this letter provide the requested information.

Attachment 1 provides the response to NRC Question 1 in a proprietary version furnished by General Electric (GE) Company. The information in Attachment 1 contains proprietary information. GE, as the owner of the proprietary information, has executed the affidavit provided as Attachment 2, which identifies that the information has been handled and classified as proprietary, is customarily held in confidence, and has been withheld from public disclosure. The proprietary information has been provided to EGC in a GE transmittal that is referenced in the affidavit. The proprietary information has been faithfully reproduced in the response such that the affidavit remains applicable. GE has requested that the proprietary information be withheld from public disclosure in accordance with 10 CFR 2.790, "Public inspections, exemptions, requests for withholding," and 10 CFR 9.17, "Agency records exempt from public disclosure." Attachment 3 provides a non-proprietary version of the information in Attachment 1. Attachment 4 provides responses to NRC Questions 2, 3, and 4.

Should you have any questions concerning his letter, please contact Mr. Kenneth M. Nicely at (630) 657-2803.

A P01

I declare under penalty of perjury that the foregoing is true and correct.

Respectfully,

10/10/03
Executed on

Patrick R. Simpson
Patrick R. Simpson
Manager – Licensing

Attachment 1: Response to NRC Question 1 Regarding Request for Technical Specifications Changes Related to Main Steam Safety Valve Operability Requirements (Proprietary Version)
Attachment 2: General Electric Company Affidavit
Attachment 3: Response to NRC Question 1 Regarding Request for Technical Specifications Changes Related to Main Steam Safety Valve Operability Requirements (Non-Proprietary Version)
Attachment 4: Response to NRC Questions 2, 3, and 4 Regarding Request for Technical Specifications Changes Related to Main Steam Safety Valve Operability Requirements

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

ATTACHMENT 2

General Electric Company Affidavit

General Electric Company

AFFIDAVIT

I, Ron Engel, state as follows:

(1) I am Technical Lead, Plant 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 the GE letter dated Octover 3, 2003, "Dresden SSV RAI Response." The proprietary information is delineated by a double underline inside double square brackets. Figures and large equation objects are identified with double square brackets before and after the object. In each case, the superscript notation{3} refers to Paragraph (3) of this affidavit, which provides the basis for the proprietary determination.

(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), and 2.790(a)(4) for "trade secrets" (Exemption 4). The material for which exemption from disclosure is here sought 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 aspects of past, present, or future General Electric customer-funded development plans and programs, resulting in potential products to General Electric;

d. 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 paragraphs (4)a., and (4)b, above.

(5) To address 10 CFR 2.790 (b) (4), 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 detailed methods and processes, which GE has developed and applied to analysis for Abnormal Transient Without Scram (ATWS) events for the BWR over a number of years. The development of the BWR ATWS analysis was achieved at a significant cost, on the order of 1 million dollars, to GE.

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.

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information, and belief.

Executed on this 3rd day of October 2003.

Ron Engel

Ron Engel
General Electric Company

ATTACHMENT 3

Response to NRC Question 1 Regarding Request for Technical Specifications Changes Related to Main Steam Safety Valve Operability Requirements (Non-Proprietary Version)

NRC RAI 1

The GENE licensing methodology confirms that the anticipated transients without scram (ATWS) analysis-of-record continues to bound the ATWS response for each cycle by evaluating the plant's cycle and core-specific ASME overpressure response. The Dresden amendment states that the core design changes necessary to meet the operating energy requirements of Dresden Units 2 and 3 affect the pressurization response for the units. In order to meet the ASME vessel overpressure requirement of 1375 psig and the corresponding technical specifications dome pressure requirement of 1345 psig, the licensee needs to take credit for all nine safety valves. However, the amendment request did not provide any ATWS evaluations to confirm that the Dresden units would be able to maintain vessel integrity. The following questions address this issue.

(a) Since the Dresden units have low ATWS peak pressure margin (1492 psig for GE14 and 1499 psig for the Legacy fuel), confirm that the ATWS analysis-of-record remains bounding in terms of peak pressure. Discuss and provide the bases for your conclusion.

GE Response

[[

]] The ATWS analysis is set up to be conservative relative to its nominal basis. Conservative SRV inputs, RPT setpoint and bounding initial conditions all assure a bounding analysis compared to the nominal basis, which uses realistic values for these inputs. Therefore, the Dresden Units continue to meet the ATWS vessel integrity criteria.

NRC RAI

(b) Explain if the ATWS analysis-of-record was based on an equilibrium GE14 core and equilibrium Atrium-9 core or if a core-configuration specific (GE14 and ATRIUM-9) ATWS was performed at the uprated conditions. If a core-configuration specific ATWS analysis was performed, provide the results of the ATWS analysis-of-record.

GE Response

To demonstrate the discussion in part (a) of this response, a cycle specific analysis was performed for Dresden 3 Cycle 18. This analysis used all of the same inputs as the EPU GE14 analysis except that the nuclear conditions for Dresden 3 Cycle 18 are used. The methods used in this analysis are the NRC approved methods described in Reference 1. The limiting ATWS case for Dresden is the Pressure Regulator Failure Open (PRFO). This event produces the limiting overpressure and peak suppression pool temperature. Therefore, this event was re-analyzed for this study. [[

]] The results of these analyses are presented in Table 1 below.

The results show that the Dresden 3 Cycle 18 core produced very similar results to the EPU GE14 calculations. The pressure is just slightly lower and the peak suppression pool temperature is very similar to the calculations for the legacy fuel core. Although the void coefficient is very similar to the legacy core the axial power shape is not as bottom peaked, which leads to some small variations in the result. The results from the void coefficient study show small changes in the peak pressure and peak suppression pool temperature results. It is noted that the peak pressure increased slightly for the BOC exposure and decreased slightly for EOC. A larger (more negative) void coefficient tends to make for a more severe power increase before the high pressure RPT, however, a larger void coefficient causes a greater reduction in power after the RPT. Therefore, the actual response may be slightly worse or slightly improved depending on the specific characteristics of the case. [[

]] The cases with a more nominal basis for the SRV inputs show a large reduction in the peak vessel pressure. The key conclusions here are:

1. EPU analyses represent a conservative analysis
2. [[ ]].
3. The ATWS Nominal Basis calculations show approximately 65 psi margin to the overpressure limit [[ ]]
4. The peak suppression pool temperature showed [[ ]]

Table 1 ATWS Results for peak vessel pressure and peak suppression pool temperature

Event and Description	Exposure	Peak Vessel Pressure (psig)	Peak Suppression Pool Temperature (°F)
[[			
			]]

Table 2 below shows a comparison of the key BOC nuclear parameters for GE14 equilibrium core, legacy core and the Dresden 3 Cycle 18 core. This comparison shows very small differences in the axial power shape and void coefficient.

Table 2, Key Nuclear Parameters for the BOC Overpressure Calculations

Node Number	Power Shape (Nodal Relative Power)		
	GE14 Equilibrium Analysis	Dresden 3 Cycle 18	Legacy Core
1	0.37	0.37	0.38
2	1.28	1.25	1.26
3	1.60	1.57	1.59
4	1.68	1.65	1.66
5	1.66	1.61	1.62
6	1.60	1.55	1.54
7	1.53	1.48	1.46
8	1.46	1.42	1.39
9	1.40	1.38	1.35
10	1.34	1.36	1.34
11	1.28	1.31	1.30
12	1.21	1.25	1.23
13	1.14	1.17	1.15
14	1.05	1.06	1.05
15	0.88	0.91	0.91
16	0.81	0.83	0.84
17	0.77	0.78	0.78
18	0.70	0.72	0.73
19	0.63	0.66	0.67
20	0.55	0.58	0.60
21	0.46	0.49	0.51
22	0.36	0.38	0.39
23	0.15	0.16	0.17
24	0.08	0.09	0.09
VOID COEF (¢/%Void)	-23.7	-22.1	-22.2

Reference 1
GE Nuclear Energy, "Qualification of the One-Dimensional Core Transient Model (ODYN) for Boiling Water Reactors (Supplement 1 – Volume 4)," Licensing Topical Report NEDC-24154P-A, Supplement 1, Class III, February 2000

ATTACHMENT 4

Response to NRC Questions 2, 3, and 4 Regarding Request for Technical Specifications Changes Related to Main Steam Safety Valve Operability Requirements

Question 2

Evaluate the safety analyses that are not performed on cycle-specific bases (e.g. ECCS-LOCA), and confirm that current core designs would not invalidate the safety analyses of record. Provide the bases for your conclusions.

Response

The NRC-approved GESTAR methodology specifies the analyses that are performed on a cycle-specific basis. There are cycle-independent safety analyses such as loss-of-coolant accident (LOCA), anticipated transient without scram (ATWS) (long term response), containment response, and 10 CFR 50 Appendix R that are potentially affected by cycle-specific changes in the core decay heat. The general practice is to generate a decay heat table for an equilibrium cycle consisting entirely of a single fuel product line based on input parameters that are bounding for the core and fuel design. The two most significant parameters affecting decay heat are average irradiation time of the fuel and the average initial enrichment of the fuel. All fuel types in the core were considered for development of the decay heat table for extended power uprate (EPU). The table is expected to be conservative, not only for the equilibrium cycle, but also for the transition cycles that lead up to equilibrium. The decay heat assumed in the cycle-independent safety analyses is not affected by core design changes resulting from a reload, as long as fuel type, power level, and fuel burnup assumptions are within the parameter values used in developing the cycle-independent decay heat table. Thus, the cycle-independent safety analyses remain applicable to each reload core.

The LOCA analysis for Dresden Nuclear Power Station (DNPS), Units 2 and 3, and Quad Cities Nuclear Power Station (QCNPS), Units 1 and 2, considers the most limiting large break size, break location, and single failure. It demonstrates that the acceptance criteria of 10 CFR 50.46 and requirements of the NRC safety evaluation on the SAFER/GESTR-LOCA application method are met. The key inputs to this analysis are emergency core cooling system (ECCS) parameters that incorporate values for ECCS performance that are consistent with the current Technical Specifications requirements. The licensing basis peak cladding temperature (PCT) is calculated using the assumptions in 10 CFR 50 Appendix K, which include the conservative decay heat model required by Appendix K. The LOCA analysis remains applicable with respect to core design changes as long as the ECCS parameters and operating conditions remain unchanged and the core radial and axial peaking factors are verified to be within the assumptions of the original analysis.

The short-term ATWS evaluation has been addressed in response to Question 1 (i.e., Attachment 1). The long-term ATWS evaluation for suppression pool temperature response is based on a conservative decay heat model, which is not affected by the core design changes.

The DNPS station blackout (SBO) evaluation that was performed using the guidelines of NUMARC 87-00 (i.e., Reference 1) except, where NRC Regulatory Guide 1.155 (i.e., Reference 2) takes precedence. The plant responses to and coping capabilities for a SBO event are not affected by changes due to core design. The SBO evaluation is only affected by the changes to the systems and equipment used to respond to or if the required coping time changed. Reactor makeup requirements depend on the isolation condenser's ability to remove decay heat without loss of reactor pressure vessel (RPV) inventory through the relief valves.

Since the isolation condenser has sufficient capacity to remove the decay heat, the SBO analysis is not affected by the small changes in decay heat resulting from cycle-specific core designs.

The short-term design basis accident (DBA) LOCA containment response during the reactor blowdown is governed by the blowdown flow rate and inventory in the RPV downcomer. This blowdown flow rate is dependent on the reactor initial thermal-hydraulic conditions, such as RPV dome pressure and the mass and energy of the RPV fluid inventory. These parameters are not affected by the core design. Therefore, the short-term DBA LOCA containment response is not affected by cycle-specific core designs. The long-term heatup of the suppression pool following a LOCA or a transient is governed by the ability of the Low Pressure Coolant Injection (LPCI)/Containment Cooling (CC) system to remove decay heat. The long-term response is affected by decay heat; however, as described above, a conservative decay heat table that bounds the cycle-specific core designs is used in the long-term containment analyses. Therefore, the containment response analyses are not affected by cycle-specific core designs.

The 10 CFR 50 Appendix R analysis has been performed to evaluate compliance with the requirements of 10 CFR 50.48 and 10 CFR 50 Appendix R for DNPS at EPU conditions. Specifically, the RPV coolant inventory, the peak suppression pool temperature, and containment pressure have been evaluated at EPU conditions for a postulated fire event to ensure RPV inventory and containment integrity. These analyses use a nominal decay heat formulation. Because there is no significant change in decay heat and no change in operating parameters with respect to core design changes, the Appendix R analysis of record remains applicable.

Question 3

The amendment request states that, for Cycle 18, Dresden Unit 2 does not need to take credit for all 9 S/RVs because DNPS Unit 2 has less top peaked axial power shapes than Unit 3. Based on the Cycle 18 reload analyses for both Units, please explain the differences in the type of core power distribution performed for each unit. Also confirm that the axial power distribution and burnup used for both units are comparable.

Response

The axial power distribution and core burnup used for both units are comparable. The DNPS Unit 2 Cycle 18 reload analysis was based on a nominal core power distribution at end of cycle (EOC). The DNPS Unit 3 Cycle 18 reload analysis was based on a top-peaked power distribution at EOC. The assumption of a nominal power distribution for DNPS Unit 2 Cycle 18 is being verified during the cycle by core monitoring.

ATTACHMENT 4

Response to NRC Questions 2, 3, and 4 Regarding Request for Technical Specifications Changes Related to Main Steam Safety Valve Operability Requirements

Question 4

Were any main steam safety valves assumed to be inoperable in the EPU ATWS analysis? If so, why were the Technical Specifications not updated as part of the EPU license amendment process?

Response

As noted in the EPU Power Uprate Safety Analysis Report, Section 9.4.1, "Anticipated Transients Without Scram," (Reference 3), no main steam safety valves were assumed to be out of service for the EPU ATWS analysis. Based on this analysis, during the EPU license amendment process, EGC considered the need to update Technical Specification Limiting Condition for Operation (LCO) 3.4.3, "Safety and Relief Valves," to require that all nine safety valves be operable. EGC determined that updating this LCO was not required.

In accordance with 10 CFR 50.36, "Technical specifications," paragraph (c)(2)(ii)(C), limiting conditions for operation are required for structures, systems, or components that are part of the primary success path to mitigate a DBA or transient. The DNPS licensing basis for the main steam safety valves does not consider the ATWS event to be a design basis event for the purposes of establishing Technical Specification requirements. At DNPS, the licensing and design basis for the main steam safety valves was established prior to promulgation of 10 CFR 50.62, "Requirements for the reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants." 10 CFR 50.62 did not require utilities to change the licensing basis of systems already installed that aid in ATWS mitigation. Thus, the DNPS Technical Specification Bases, Section 3.4.3, state that the design basis for the main steam safety valves is to mitigate the most severe expected overpressure transient, which is the closure of all main steam isolation valves, followed by reactor scram on high neutron flux. This is consistent with the improved standard Technical Specification Bases, issued as NUREG-1433, Rev. 1, "Standard Technical Specifications, General Electric Plants, BWR/4 Specifications." This is also consistent with the DNPS Updated Final Safety Analysis Report, Section 15.8, "Anticipated Transients Without Scram," which states that the ATWS event is not considered a design basis event. Further, this approach is consistent with the approach taken in the Technical Specification Bases description of the Standby Liquid Control system, which was also installed prior to promulgation of 10 CFR 50.62.

As stated in Reference 4, EGC requested the license amendment when it was determined that the pressurization transient analysis for the second half of the next operating cycle for DNPS, Unit 3 Cycle 18, shows that nine safety valves are required to mitigate the design basis overpressure transient.

References

1. NUMARC 87-00, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors," Revision 1, dated August 1991

2. NRC Regulatory Guide 1.155, "Station Blackout," dated August 1988

ATTACHMENT 4

Response to NRC Questions 2, 3, and 4 Regarding Request for Technical Specifications Changes Related to Main Steam Safety Valve Operability Requirements

3. Letter from R. M. Krich (Commonwealth Edison Company) to U. S. NRC, "Request for License Amendment for Power Uprate Operation," dated December 27, 2000

4. Letter from P. R. Simpson (Exelon Generation Company, LLC) to U. S. NRC, " Request for Technical Specifications Changes Related to Main Steam Safety Valve Operability Requirements," dated October 10, 2002