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U. S. Nuclear Regulatory Commission
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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 Supporting the License Amendment Request to Permit Uprated Power Operation at Dresden Nuclear Power Station

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

In the referenced letter, Commonwealth Edison Company, now Exelon Generation Company (EGC), LLC, submitted a request for changes to the operating licenses and Technical Specifications for Dresden Nuclear Power Station (DNPS), Units 2 and 3, and Quad Cities Nuclear Power Station, Units 1 and 2, to allow operation with an extended power uprate. In a September 18, 2001, telephone conference call between representatives of EGC and Mr. S. N. Bailey and other members of the NRC, the NRC requested additional information regarding these proposed changes for DNPS. The attachment to this letter provides the requested information.

Should you have any questions concerning this letter, please contact Mr. A. R. Haeger at (630) 657-2807.

Respectfully,

K. A. Ainger

K. A. Ainger
Director – Licensing
Mid-West Regional Operating Group

A001

Attachments:

Affidavit
Additional Information Supporting the License Amendment Request to Permit Uprated Power Operation at Dresden Nuclear Power Station

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

STATE OF ILLINOIS	)
COUNTY OF DUPAGE	)
IN THE MATTER OF	)
EXELON GENERATION COMPANY, LLC	) **Docket Numbers**
DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3	) **50-237 AND 50-249**

SUBJECT: Additional Information Supporting the License Amendment Request to Permit Uprated Power Operation at Dresden Nuclear Power Station

AFFIDAVIT

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

K. A. Ainger

K. A. Ainger
Director – Licensing
Mid-West Regional Operating Group

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

for the State above named, this 26th day of

September, 20 01.

Betty Fox

Notary Public

OFFICIAL SEAL
BETTY FOX
NOTARY PUBLIC, STATE OF ILLINOIS
MY COMMISSION EXPIRES:05/14/05

Additional Information Supporting the License Amendment Request to Permit Uprated Power Operation at Dresden Nuclear Power Station

Question

1. *Provide additional detail concerning the ability to shut down the Dresden Unit 2 and 3 reactors following a dam failure. Include details regarding the seismic design of the isolation condenser, the assumed makeup water sources and their seismic capabilities, and the volumes required and available. Discuss changes for the above for the extended power uprate.*

   Also provide additional detail regarding the previously submitted core damage frequency estimate for the dam failure scenario, including the seismic hazard spectrum considered, the equipment failure rates for the spectrum considered, and the assumed operator actions and human error probabilities.

Response

The current Dresden Nuclear Power Station (DNPS), Units 2 and 3 licensing basis is that the units can be simultaneously brought to hot shutdown and maintained for 30 days following a failure of the Dresden lock and dam, using the isolation condenser (IC).

The IC has been evaluated to the review level earthquake of 0.3 g, including the piping and connections from the IC shell side out to the first isolation valve. These isolation valves include either normally closed isolation valves or check valves.

The sources of makeup water to the IC shell side are not seismically qualified, but given the redundancy and diversity of these sources, there is a high confidence that at least one source will be available following a seismic event. The current sources include initial makeup from on-site tanks and the Unit 1 fire pump, and makeup from the ultimate heat sink (UHS). The DNPS response to the Individual Plant Examination of External Events (IPEEE) (Reference 1) included a commitment to provide a seismic makeup path to the IC by November 2003.

IC Shell Inventory

Prior to EPU, the IC shell has water inventory for 20 minutes of IC operation, including allowance for moisture carryover. DNPS Technical Specifications ensure this inventory is present. The design heat removal capacity for the IC was originally based on the decay heat rate generated at five minutes into a loss of heat sink event. This is the point at which the IC begins to reduce reactor pressure and reactor relief valves cease operation. During the five minute interval, the relief valves cycle to remove the decay heat that is in excess of the IC capacity. The IC heat removal rate is controlled by surveillance and adjustment of the condensate return valve. The EPU analysis used the original design heat removal capacity, thus maintaining the same rate of inventory depletion from the IC shell. This analysis showed that the time at which the IC begins to reduce reactor pressure occurs at about 530 seconds (i.e., 8.8 minutes) into the loss of heat sink event. Thus, the 20 minutes of initial operating time of the IC is not exceeded post-EPU.

Initial Makeup Sources

Initial makeup sources can be lined up within twenty minutes both pre and post EPU. The preferred source is the Clean Demineralized Water Tank which has a typical operating volume of 130,000 gallons and a low level alarm at 84,000 gallons. A separate source, with a separate makeup line to the IC, is contaminated condensate from the "2/3A," "2/3B," and "1A" condensate storage tanks (CSTs). These tanks have a combined capacity of 700,000 gallons and are administratively controlled to contain an inventory of at least 130,000 gallons per operating unit. At EPU conditions, these sources will supply several hours of makeup. An estimate performed for EPU conditions shows 40,000 gallons is needed per unit for two hours of IC operation.

In addition, the Unit 1 diesel driven fire pump is supplied from the Unit 1 intake canal. The water level in this canal will drop to an elevation of 495' following a dam failure. The fire pump has its suction at approximately elevation 492' and therefore is available immediately for makeup. The fire pump makeup line joins the makeup line from the CSTs to the IC shell.

Thus, there are diverse initial makeup sources with sufficient inventory for at least two hours of operation of the IC following a dam failure.

Makeup from the UHS

The UHS at DNPS consists of water sources in the retaining structures and the canals connecting the water sources with the intake and discharge structures. The evaluations presented below take credit for the water in the Unit 2/3 intake canal. The water in the Unit 2/3 intake canal will drain to an elevation of 495' following dam failure. The water in the discharge canal and the hot and cold canals to the cooling lake is also likely to be available.

Within two hours of dam failure, makeup to the IC can be supplied from the UHS via the Unit 2/3 diesel driven fire pump, which has its suction at approximately elevation 498'. The loss of level in the intake bays causes a reactor scram on low vacuum and also initiates procedures which direct operators to install stop logs and re-flood the intake compartment that provides a suction source for the Unit 2/3 diesel driven fire pump. The reflooding is accomplished using the refuse pit pumps, which are powered from motor control centers that can be fed from the emergency diesel generators. The refuse pit pumps suction is at approximately elevation 480', which is the bottom of the intake bay. The action to install the stop logs and reflood the intake compartment is proceduralized and the two hour performance time has been validated. Thus, makeup to the IC from the UHS will be available prior to depletion of the initial makeup sources.

The volume of water in the Unit 2/3 intake canal is not sufficient to provide makeup for 30 days to maintain either or both reactors in hot shutdown following a dam failure, both pre and post EPU. This was previously evaluated in the Systematic Evaluation Program (SEP) for DNPS (Reference 2), which estimated that the available volume in the intake canal would provide cooling for several days to a week or more. Based upon the SEP evaluation, the NRC concluded that this volume was acceptable, since, based upon the time available, replenishment of the UHS can be effected to ensure the continuous capability of the sink to perform its safety function.

A recent evaluation of this conclusion for the EPU conditions was performed. Given the complex geometry of the intake canal, the volumes listed below are estimates made for the purposes of assessing available time to accomplish further actions. The volume available in the intake canal is in excess of 3 million gallons. This includes essentially the entire volume of the intake canal, which is assumed to be available to supply the refuse pit pumps, which take suction from the bottom of the intake bay at 480'. The portion of this volume available in the intake canal to supply the emergency diesel generator (EDG) cooling water pumps, which take suction at approximately elevation 487', is approximately 2 million gallons.

An estimated additional 3 million gallons is available in the discharge canal, for a total estimated volume of 6 million gallons, as noted in Reference 3. Water in the discharge canal can be returned to the intake canal through a de-icing line, which is isolated by a manually operated slide gate valve. If the slide gate is rendered inoperable by a seismic event, the canal flow regulating station or circulating water pump discharge valves can also be used to return flow from the discharge canal to the intake canal.

The volume of water required to achieve hot shutdown and maintain it for 30 days was determined to be 2.5 million gallons per unit prior to EPU. As noted in Reference 3, EPU raises this required volume to 2.9 million gallons per unit. This volume does not account for moisture carryover in the IC shell.

A recent evaluation was performed to include the effects of moisture carryover. This assessment also included the impact of running the EDG cooling water pumps during the dam failure scenario. This evaluation determined the time available before suction will be lost to the EDG cooling water pumps, since the EDGs provide power to the refuse pit pumps when offsite power is unavailable. No other significant sources beyond the IC and the EDG cooling water pumps are currently assumed to deplete the intake canal inventory. This evaluation credited the volume of the intake canal above the elevation of the suction to the EDG cooling water pumps (i.e., 2 million gallons).

Running the EDG cooling water pumps will add water continuously to the discharge canal. This water will recirculate by gravity feed to the intake through the deicing line. Control of recirculating flow with the deicing line slide gate is addressed in the DNPS dam failure procedure. Credit was taken for the water used by the EDG cooling water pumps recirculating from the discharge canal back to the intake canal via the deicing line or the alternate methods noted above. No additional credit was assumed for the volume of water initially trapped in the discharge canal.

This evaluation determined that the boil off of water in the IC, including moisture carryover effects, will deplete the available intake canal inventory of 2 million gallons in approximately four days, assuming a concurrent loss of offsite power and shutdown of both units. For pre-EPU conditions, this same inventory will last for approximately five and one-half days. If offsite power is restored prior to four days following the event, this time is increased, since the entire volume of the intake canal can be used if diesel generator cooling is not required.

<u>Replenishment of the UHS</u>
Before depletion of water in the intake canal, makeup to the intake canal can be provided from the Kankakee River using portable low head, high volume, engine-driven pumps, which can be readily obtained from other power stations or via purchase or rental. As noted above, the NRC concluded as part of the SEP that this was acceptable, given the projected time of several days to a week available for the action. As demonstrated above, under EPU conditions, at least four days are available before this replenishment would be necessary. Therefore, the basis for this conclusion is unaffected by EPU.

Core Damage Frequency Estimate

In Reference 4, EGC provided an estimated core damage frequency (CDF) due to a seismically-induced dam failure without a LOCA for which IC makeup would be required. This CDF is estimated at 9E-6/yr prior to EPU. The increase in this estimated CDF due to EPU is 1E-8/yr and was provided in Reference 5.

The scenario assessed in this analysis is as follows.

- Seismic-induced failure of the Dresden lock and dam, resulting in loss of intake level at the crib house
- Loss of offsite power
- Successful EDG operation
- Successful reactor scram
- Use of the isolation condenser and associated shell makeup options from on-site tanks as the safe shutdown path. Makeup to the IC from the UHS was not credited.

Other shutdown paths and equipment (e.g., high pressure coolant injection, automatic depressurization, and containment cooling service water (CCSW), with support from refuse pit pumps) are not credited as part of this analysis. Therefore, the calculated CDF for this scenario is conservative.

The components and structures applicable to this assessment and their seismic capacities, expressed as a high confidence low probability of failure (HCLPF), are as follows.

- Dresden Lock/Dam (0.10g HCLPF)
- Isolation Condenser (0.30g HCLPF)
- Clean Demineralized Water Tank (0.15g HCLPF)
- Contaminated Condensate Storage Tank "1A" CST (0.15g HCLPF). This is the limiting seismic capacity of the three CSTs. The "2/3A" and "2/3B" CSTs have seismic capacities of 0.2 g HCLPF.

Table 1 lists the seismic initiators and frequencies assumed in the analysis. These were obtained from NUREG-1488, "Revised Livermore Seismic Hazard Estimates for Sixty-Nine Nuclear Power Plant Sites East of the Rocky Mountains," April 1994. Table 2 provides the seismic-induced failure probabilities (i.e., fragilities) of equipment and structures associated with the seismic initiators. These were calculated using NUREG/CR-2300, "A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants," January 1983.

Operator Actions

Three human error probabilities (HEPs) are explicitly included in this assessment as noted below.

- Operator manually aligns IC makeup from clean demineralized water tank in approximately 20 minutes (HEP of 2.8E-3)
- Operator manually aligns IC makeup from contaminated CSTs in approximately 20 minutes (HEP of 2.8E-3). Complete dependence is assigned between the two short term actions.
- Operator manually aligns IC makeup from CSTs in several hours (HEP of 5E-4 pre-EPU and 1E-3 post EPU)

The shorter time frame actions align makeup from either the clean demineralized water tank (primary source) using the IC makeup pumps or the CSTs (alternative source) using the condensate transfer pumps. Alignment of either IC makeup option in the first 20 minutes is quantified in the seismic analysis using the HEP derived in the DNPS human reliability analysis (HRA). The alignment process and the associated human action performance shaping factors are essentially identical for either makeup source alignment (i.e., remotely opening a motor-operated valve and starting a pump). The same human error probability is used in the seismic scenario assessment for either alignment in the short term. In addition, complete dependence is assigned between the two short term actions.

The longer time frame action aligns makeup from the alternative source (i.e., CSTs). This action is considered in the seismic scenario risk assessment after the clean demineralized water tank makeup alignment using the IC Makeup Pumps has been successfully completed. On a best estimate basis, the clean demineralized water tank inventory will be exhausted in many hours, requiring alignment of another IC shell makeup source in order to meet the 24 hour mission time of the risk assessment. The DNPS PRA does not include alternative IC shell makeup from the CSTs. A base HEP of 5E-4 is calculated for this action. This HEP is conservatively doubled in this assessment to model the EPU condition.

<u>Procedural Guidance and Equipment Accessibility</u>
The operator actions related to the IC are proceduralized in DNPS procedure DOP 1300-02, "Automatic Operation of Isolation Condenser." No action is required to initiate the IC system, which will automatically initiate on receipt of a high reactor pressure vessel (RPV) pressure (>1070 psig).

Both tank alignments are performed from within the Control Room and would not require manual equipment manipulations or other actions outside the Control Room to initiate makeup flow to the IC shell.

The analyzed seismic scenario includes the following key procedural entry conditions.

- Seismic event
- Seismic-induced Dresden lock and dam failure, resulting in reduced intake level
- Seismic-induced loss of offsite power and subsequent reactor scram

These events will create symptoms which will direct the operators to enter and follow concurrently abnormal and emergency operating procedures.

A seismic event will trigger entry into abnormal operating procedure DOA 0010-03, "Earthquakes." Typical of industry earthquake abnormal procedures, DOA 0010-03 directs operators to inspect for equipment and structural damage.

Failure of the downstream Dresden Lock and Dam will be identified to DNPS operating personnel by one or more of the following.

- Report of dam failure by the U. S. Army Corp of Engineers
- Report of substantial reduction in river level by site personnel
- Intake canal level approaching or below elevation 501'-6"

These symptoms trigger entry into abnormal operating procedure DOA 0010-01, "Dresden Lock and Dam Failure." This procedure directs maintaining suction level to the CCSW pumps by installing stop logs around Bay 13 of the crib house and using the refuse pit pumps to fill the bay. While activities to refill Bay 13 will be pursued in parallel to other procedurally directed activities, the seismic scenario risk assessment does not credit these activities. The assessment also does not include makeup from the Unit 1 fire pump.

The loss of offsite power and subsequent reactor scram triggers entry into the abnormal operating procedure, DGA-12, "Partial or Complete Loss of AC Power," and the Dresden Emergency Operating Procedures (DEOPs). DGA-12 directs activities with regard to electrical requirements, while the governing directions for safe shutdown of the reactor are covered by the DEOPs.

As discussed earlier, the IC will automatically initiate on sustained high RPV pressure. The Reactor Control legs of the DEOPs direct cooling down the RPV using the IC. Operating procedure DOP 1300-02, "Automatic Operation of Isolation Condenser," directs operation of the IC with manual action required to align IC shell makeup.

<u>Question</u>

2. *Provide additional discussion regarding the results of the study to confirm the adequacy of the isolation condenser to provide suppression pool cooling following a small break LOCA with a dam failure, and the acceptability of proceeding with the power uprate based on the results of this study.*

<u>Response</u>

The study for the small break loss of coolant accident (SBLOCA) coincident with a dam failure has been completed for EPU conditions. The study assumed a one inch small break, consistent with the guidance in EPRI NP-6041-SL, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin (Revision 1)." EPU decay heat was used in the analysis. The analysis demonstrates that the IC and available emergency core cooling systems (ECCS) (i.e., high pressure coolant injection (HPCI) and low pressure coolant injection (LPCI)) are sufficient to mitigate a seismically induced SBLOCA for a 24-hour period. The study shows that additional equipment, specifically a cooling water supply to the CCSW heat exchangers, will be required 24 hours after the onset of the event to supply suppression pool cooling.

DNPS has developed a conceptual design using large portable pumps that would be used to restore the required CCSW cooling flow via suction from the intake canal. These pumps would be stored in an area that could withstand the postulated seismic event, and would be staged with hose connections to the CCSW piping. The necessary fittings will be installed on the existing CCSW piping. Power for the portable pumps will be supplied either by portable diesel engines or by temporary power connections to the available existing electrical buses. Procedures will be developed to ensure that the necessary actions will be taken within the 24 hour period to establish suppression pool cooling flow. These actions will provide the capability to mitigate the seismically induced SBLOCA for the 72 hour time frame given in EPRI NP-6041-SL. These actions will be completed on the same schedule as the modification to provide a seismically qualified makeup path to the IC as described in Reference 1.

The risk impact was assessed for the small break LOCA with dam failure prior to installation of the seismic makeup path to the IC and the modification to provide suppression pool cooling. The methods used to develop this estimate are similar to those described above for the dam failure with normal shutdown. The SBLOCA is postulated to have a probability of occurrence as determined in NUREG/CR-4550, "Analysis of Core Damage Frequency: Peach Bottom, Unit 2 External Events," December 1990. This probability as a function of "g" loading is included directly in the model calculation and is shown in Table 2. Failure at this mode is assessed to result in a condition that will defeat the IC from effective operation and result in eventual containment overpressurization. The base CDF resulting from this assessment for the pre-EPU condition is 1.9E-6/yr. The EPU was determined to have negligible impact on this CDF.

As noted in the response to Question 1 above, operation of the IC at the pre-EPU design heat removal rate will result in an increase in the time during which the RPV relief valves will cycle open following a loss of heat sink. This time increases from 5 minutes pre-EPU to 8.8 minutes post-EPU. The increased potential for a stuck open relief valve was modeled in the original EPU risk assessment for transient sequences, but was not modeled in the CDF estimate for the dam failure with small break LOCA provided above. A sensitivity study was performed to determine the impact of a stuck open relief valve for this assessment. The results of this study show that the stuck open relief valve with a failure of the dam increases the base CDF for the SBLOCA scenario with a failure of the dam from 1.9E-6/yr to 2.1E-6/yr with an EPU delta of 4.6E-8/yr.

The results of these assessments indicate that operation following EPU, but prior to correcting the seismic margins outlier related to the IC makeup path and the additional modification to provide CCSW flow, has a very small effect on risk.

<u>References</u>

1. Letter from P. Swafford (Commonwealth Edison Company) to U. S. NRC, "Request for Additional Information Regarding Individual Plant Examination of External Events," dated March 30, 2000

2. Letter from P. W. O'Connor (U. S. NRC) to L. DelGeorge (Commonwealth Edison Company), "Dresden 2 Nuclear Generating Station, Safety Evaluation of Hydrology SEP Topics II-3.A, II-3.B, II-3.B.1 and II-3.C," dated June 21, 1982

3. Letter from K. A. Ainger, (Exelon Generation Company, LLC) to U. S. NRC, "Additional Plant Systems Information Supporting the License Amendment Request to Permit Uprated Power Operation, Dresden Nuclear Power Station and Quad Cities Nuclear Power Station," dated August 13, 2001

4. Letter from K. A. Ainger (Exelon Generation Company, LLC) to U. S. NRC, "Additional Risk Information Supporting the License Amendment Request to Permit Uprated Power Operation at Dresden Nuclear Power Station and Quad Cities Nuclear Power Station," dated September 14, 2001

5. Letter from K. A. Ainger (Exelon Generation Company, LLC) to U. S. NRC, "Additional Risk Information Supporting the License Amendment Request to Permit Uprated Power Operation at Dresden Nuclear Power Station and Quad Cities Nuclear Power Station," dated September 5, 2001

Table 1
SEISMIC INITIATOR FREQUENCIES

SEISMIC INITIATOR	DESCRIPTION	INPUT FREQ. (1/yr)
G1	Seismic Event at Dresden Site (<0.1g)	Note 1
G2	Seismic Event at Dresden Site (0.1 - <0.2g)	1.1E-04
G3	Seismic Event at Dresden Site (0.2 - <0.3g)	2.1E-05
G4	Seismic Event at Dresden Site (0.3 - <0.4g)	1.2E-05
G5	Seismic Event at Dresden Site (0.4 - <0.5g)	4.2E-06
G6	Seismic Event at Dresden Site (0.5 - <0.6g)	2.4E-06
G7	Seismic Event at Dresden Site (0.6 - <0.7g)	1.1E-06
G8	Seismic Event at Dresden Site (0.7 - <0.9g)	1.0E-06
G9	Seismic Event at Dresden Site (>0.9g)	8.7E-07

Note 1: The G1 seismic initiator is not explicitly quantified. Such low magnitude seismic events have a very low likelihood of resulting in seismic-induced failures. The plant will likely remain at power and not experience a trip or a shutdown.

Table 2
STRUCTURAL AND EQUIPMENT FRAGILITIES

Component/ Structure	HCLPF (g)	Am (g)	Seismic Magnitude (g)								
			0.05 (Interval G1)	0.15 (Interval G2)	0.25 (Interval G3)	0.35 (Interval G4)	0.45 (Interval G5)	0.55 (Interval G6)	0.65 (Interval G7)	0.80 (Interval G8)	1.00 (Interval G9)
Dresden Lock/Dam	0.10	0.27	3.64E-05	8.41E-02	4.31E-01	7.32E-01	8.87E-01	9.54E-01	9.81E-01	1.00E+00	1.00E+00
Isolation Condenser	0.30	0.81	epsilon	3.64E-05	2.86E-03	2.44E-02	8.41E-02	1.83E-01	3.05E-01	4.91E-01	6.93E-01
Clean Demin. Water Tank	0.15	0.40	4.27E-07	9.81E-03	1.29E-01	3.68E-01	6.01E-01	7.67E-01	8.69E-01	9.47E-01	9.84E-01
1A CST	0.15	0.40	4.27E-07	9.81E-03	1.29E-01	3.68E-01	6.01E-01	7.67E-01	8.69E-01	9.47E-01	9.84E-01
RPV Small LOCA	NA	NA	2.0E-07	7.0E-05	5.0E-03	2.5E-02	6.0E-02	1.2E-01	2.0E-01	3.5E-01	6.0E-01

NOTES :

1. *epsilon* = seismic fragility less than or equal to 1E-7.
2. At a calculated value of >=0.99, the value 1.00 is used.
3. Fragilities calculated using NUREG/CR-2300, January 1983. Induced small LOCA fragility values based on NUREG/CR-4550, Vol. 4, Rev. 1, Part 3 (Figure 4.20)