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RS-01-124

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U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 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

Quad Cities Nuclear Power Station, Units 1 and 2
Facility Operating License Nos. DPR-29 and DPR-30
NRC Docket Nos. 50-254 and 50-265

Subject: Additional Offsite Dose Information Supporting the License Amendment Request to Permit Uprated Power Operation

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 (ComEd) Company, now Exelon Generation Company (EGC), LLC, submitted a request for changes to the operating licenses and Technical Specifications for Dresden Nuclear Power Station, Units 2 and 3, and Quad Cities Nuclear Power Station, Units 1 and 2, to allow operation with an extended power uprate. In a discussion between representatives of EGC and Mr. L. W. Rossbach and other members of the NRC, the NRC requested additional information regarding these proposed changes. 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,

R. M. Krich
Director – Licensing
Mid-West Regional Operating Group

A009

Attachments:

Affidavit
Attachment A: Additional Offsite Dose Information Supporting the License Amendment Request to Permit Uprated Power Operation at Dresden Nuclear Power Station, Units 2 and 3
Attachment B: Additional Offsite Dose Information Supporting the License Amendment Request to Permit Uprated Power Operation at Quad Cities Nuclear Power Station, Units 1 and 2

cc: Regional Administrator - NRC Region III
NRC Senior Resident Inspector - Dresden Nuclear Power Station
NRC Senior Resident Inspector - Quad Cities 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** |
| **QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2** | **)** | **50-254 AND 50-265** |

**SUBJECT:** Additional Offsite Dose Information Supporting the License Amendment Request to Permit Uprated Power Operation

## AFFIDAVIT

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

RMKrich

R. M. Krich
Director – Licensing
Mid-West Regional Operating Group

Subscribed and sworn to before me, a Notary Public in and for the State above named, this 6th day of July, 2001.

OFFICIAL SEAL
JACQUELINE T EVANS
NOTARY PUBLIC, STATE OF ILLINOIS
MY COMMISSION EXPIRES:02/24/02

Jacqueline T. Evans

Notary Public

## Attachment A
## Additional Offsite Dose Information Supporting the License Amendment Request to Permit Uprated Power Operation at Dresden Nuclear Power Station, Units 2 and 3

Question

*The ComEd application cites Section 5.4 of the ELTR1, which states that the uprate application will provide justification that current radiological consequences are still bounding and within applicable criteria, or provide re-analysis of any areas adversely affected by power uprate. Appendix H of the ELTR1 describes the methodology and assumptions for these re-analyses. Appendix H does not provide specific assumptions, noting only that the analyses will be based on the methodology, assumptions, and analytical techniques described in the regulatory guides, the SRP, and previous safety evaluations. The staff's SER on ELTR1 dated September 14, 1998, notes that radiological consequences will be assessed on a plant-specific basis using NRC-approved methods.*

*Section 9.3 of the safety analysis report for the Dresden 2 & 3 EPU addresses the radiological consequences of design basis accidents. While this section identifies the magnitude of change in the results, the application does not adequately identify the methodology, assumptions, and inputs used by ComEd in arriving at these conclusions. This information is necessary for the staff to determine whether the ComEd analyses are acceptable and meet the provisions of the ELTR1 and the staff SER on the ELTR1. Please provide the following additional information, or provide a cross-reference to where the information can be found in docketed material.*

1. *For any conclusion provided in this section that was derived in total or in part from generic analyses, please describe the analysis or provide a citation to that description. Please explain how the results were determined to be applicable to the Dresden 2 & 3 design basis as modified by this uprate.*

   Response

   The accident dose consequences and conclusions presented in Section 9.3, "Design Basis Accidents," of the power uprate safety analysis report (PUSAR) contained in Reference 1 are not derived from generic analyses.

2. *If any of the accident dose results were obtained by plant-specific re-analysis, as opposed to scaling previous FSAR results, please provide a tabulation of analysis inputs and assumptions that will enable the staff to evaluate the acceptability of these assumptions, and as necessary, perform confirming calculations. Please identify any changes to prior design basis analysis inputs, assumptions, and methodologies, including offsite and control room atmospheric dispersion coefficients, incorporated in these re-analyses.*

   Response

   The extended power uprate (EPU) accident dose consequences presented were obtained by applying scaling techniques to previous results to reflect the increased core inventory and increased mass releases, as applicable. No plant specific re-analyses were performed. There were no changes to prior design basis analysis inputs, assumptions, or methodologies, including offsite and control room atmospheric dispersion coefficients, except with regard to the gland steam release path for the control rod drop accident (CRDA) at Dresden Nuclear Power Station (DNPS). As discussed in Section 9.3 of the PUSAR, the dose contribution due to melted fuel via the gland steam release path for the CRDA was previously unaccounted for in the Updated Final Safety Analysis Report (UFSAR), since it was determined that the assumption of melted fuel was excessively conservative. For EPU, the contribution due to melted fuel was included by scaling the activities released from the melted fuel.

3. *The application reports that the LOCA, CRDA, and FHA offsite thyroid and whole body doses increased by 26 % and 17% respectively. This suggests that these results were obtained by multiplying the previous doses by a factor based on the increase in core inventory. This methodology is generally acceptable. However, the requested power uprate is only 17 percent. The application implies that the lack of proportionality might be due to the difference in U-235 and Pu-239 fission yields. However, ORIGEN data available to the staff (NUREG/CR-6703) indicates that the inventory of I-131 (Ci/MWt) increases by less than 2% from 22 to 75 GWD/MTU. Please explain the derivation of the 26% and 17% factors providing sufficient information for the staff to confirm the acceptability of these factors. Similarly, please explain why the control room factors for the LOCA differ from the factors used for the other accidents. If analyses were performed to derive the 26% and 17% factors, please describe the inputs, assumptions, and methodologies used.*

<u>Response</u>
As discussed in Section 9.3 of the PUSAR, the pre-EPU thyroid dose analyses for the loss of coolant accident (LOCA), CRDA, and fuel handling accident (FHA) used the iodine inventories, in curies/megawatt (Ci/MW), from Table I of Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactor Sites," (Reference 2), which was published on March 23, 1962. These core inventories were based on the fission product yields of U-235 only, which was the typical practice for low burnup fuel at that time. This data represented the state of the art for determining fission product generation at that time.

As discussed in Section 9.3 and Table 1-3, "NSSS Computer Codes Used for EPU," of the PUSAR, the EPU core inventory for GE14 fuel and a 24 month fuel cycle was calculated using the ORIGEN2 code which uses current methodology and data for fission product generation. This methodology includes up to date fission product yields and decay chains, and also includes the fission product contributions from transuranic nuclides (e.g., Pu-239 and Pu-241), which become important with the present day higher burnup fuel cycles.

The scaling factor for thyroid dose is the ratio of the ORIGEN2 iodine inventories, weighted for dose factor, at the EPU power level to the TID-14844 iodine inventories at the pre-EPU power level. This factor was found to be 1.26 (i.e., the 26% increase discussed in Section 9.3 of the PUSAR) and includes the effects of the increase in power level and the increases in iodine nuclides due to the 24 month cycle, reflecting the change in fission product generation described above.

A similar scaling factor was determined for the whole body dose. The whole body dose is primarily affected by the noble gas fission product inventories. This factor was found to be less than the increase in power level alone (i.e., less than 17%) because the ORIGEN2 code has shown decreased noble gas production compared to previous methods. However, the EPU scaling factor for the whole body dose is conservatively maintained at the percentage of uprate (i.e., the 17% increase discussed in Section 9.3 of the PUSAR).

As noted in Section 9.3 of the PUSAR, the EPU LOCA control room thyroid and whole body dose scaling factors are 1.29 and 1.19, respectively, as a result of the addition of an additional 2% margin. This 2% margin was previously included in all of the other pre-uprate accident analyses, except the control room LOCA dose analysis.

As requested by the NRC, the significant EPU core inventories, calculated using the ORIGEN2 computer code, are provided in Table 1 of this attachment. The sixteen nuclides listed represent the significant contributors to the core inventory. Other nuclides were not shown as their contribution is small relative to those listed.

4. *Have any UFSAR or CURRENT results in Tables 9-7 through 9-8 been revised as a result of any analysis changes since this application was docketed?*

<u>Response</u>
There have been no new analyses performed that change the results in Tables 9-7 or 9-8 since the EPU license amendment request was submitted. However, subsequent review of Table 9-9, "FHA Radiological Consequences," has determined that revisions are required to some of the reported dose values as a

result of an oversight in updating this table. Table 2 of this attachment provides a revised Table 9-9, to reflect the correct dose values.

5. *In Table 9-7, please explain the difference between the "UFSAR" and "CURRENT" values tabulated for the control room thyroid dose. Please explain why only the control room thyroid dose changed.*

   Response
   At the time the EPU license amendment request was submitted, a change was pending to the UFSAR regarding the control room dose resulting from a LOCA. The calculation of control room dose following a LOCA had been revised and a change package to update the UFSAR had been developed and received site approval. The DNPS UFSAR containing the revision was not scheduled to be submitted until June 2001. Thus, the control room dose shown in the UFSAR was reported in the "UFSAR" column of Table 9-7 and the result from the revised calculation was reported in the "Current" column of Table 9-7.

6. *Section 15.6.5.5.2 of the FSAR discusses the control room infiltration rates for Dresden. An earlier submittal dated May 19, 1997, subsequently withdrawn, indicated that the in-leakage measured with tracer gas testing shows that the observed leakage was less than the calculated leakage. Please confirm that this conclusion is still valid.*

   Response
   The conclusion that the actual unfiltered control room in-leakage is less than the calculated unfiltered in-leakage given in the UFSAR is still valid. The basis for this conclusion is that the site completed the baseline tracer gas testing in January 1997 and has the following ongoing programs and surveillance tests to assure that any degradation in unfiltered control room in-leakage is identified and corrected.

   - A plant barrier control program maintains the integrity of the control room envelope boundary, including ductwork.
   - A periodic test assesses the control room in-leakage. It involves a walk down of the control room envelope, including visual inspections and a smoke test of the negative pressure ductwork located outside of the control room envelope, isolation dampers, and door seals.
   - A periodic surveillance required by the DNPS Technical Specifications (TS) that measures filtered outside supply air quantity and control room pressure with the control room emergency ventilation system operating in emergency mode provides an indication of any changes to the in-leakage. An increase in the control room pressure without an increase in filtered outside air supply

indicates an increase in in-leakage. Although this test does not provide quantitative results of actual in-leakage, it is used as a tool to provide warning of any changes in in-leakage and identify the need for further investigation. This test has not shown any indication of increased in-leakage.

7. *Section 4.7 identifies that the time to reach the oxygen limit decreases from 25 hours for pre-EPU to 19 hours post-EPU. Does this observation affect any analysis assumption regarding the dose impacts of the operation of the CAD system post-LOCA? If so, please describe how this was considered in determining the LOCA dose.*

   Response
   The dose impacts due to operation of the nitrogen containment atmosphere dilution (NCAD) system following a LOCA are unchanged due to EPU. No venting due to operation of the NCAD system is necessary within a 30-day period following a LOCA.

   No venting is required for 30 days the following reason. The decrease in time from 25 hours to 19 hours to reach the primary containment oxygen limit results in earlier operation of the NCAD system. Even with earlier injection of nitrogen, however, primary containment pressure remains below the repressurization limit of 31 psig (i.e., 50% of design pressure) stated in 10 CFR 50.44, "Standards for combustible gas control system in light-water-cooled power reactors," until 32 days following a LOCA. Pre-EPU analyses indicated the pressurization limit was reached at 48 days following a LOCA. Upon reaching the pressurization limit, any necessary containment venting would be controlled by the plant emergency operating procedures.

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

2. U. S. Atomic Energy Commission Technical Information Document (TID) – 14844, "Calculation of Distance Factors for Power and Test Reactor Sites," March 23, 1962

## Attachment A
## Additional Offsite Dose Information Supporting the License Amendment Request to Permit Uprated Power Operation at Dresden Nuclear Power Station, Units 2 and 3

**Table 1**
**Extended Power Uprate (EPU) Equilibrium Core Inventory of Noble Gases and Iodines (Curies/Megawatt -Thermal) Utilized for Site Boundary and Control Room Dose Analyses Dresden and Quad Cities Nuclear Power Stations**

| Isotope | EPU Core |
|---|---|
| I 131 | 2.710E+04 |
| I 132 | 3.914E+04 |
| I 133 | 5.501E+04 |
| I 134 | 6.035E+04 |
| I 135 | 5.157E+04 |
| KR 83M | 3.249E+03 |
| KR 85 | 4.364E+02 |
| KR 85M | 6.772E+03 |
| KR 87 | 1.291E+04 |
| KR 88 | 1.815E+04 |
| XE 131M | 3.040E+02 |
| XE 133 | 5.282E+04 |
| XE 133M | 1.726E+03 |
| XE 135 | 2.144E+04 |
| XE 135M | 1.089E+04 |
| XE 138 | 4.500E+04 |

## Attachment A
## Additional Offsite Dose Information Supporting the License Amendment Request to Permit Uprated Power Operation at Dresden Nuclear Power Station, Units 2 and 3

### Table 2
### Revised Power Uprate Safety Analysis Report (PUSAR) Table 9-9

Table 9-9

**FHA Radiological Consequences**

| Location | UFSAR | Current | EPU | Limit |
|---|---|---|---|---|
| Fuel Handling Accident (Single fuel bundle and handling equipment dropped) | | | | |
| Offsite: | | | | |
| Whole Body Dose, rem | 3.74E-3(1) | (2) | (2) | ≤ 25 |
| Thyroid Dose, rem | 1.33E-3(1) | (2) | (2) | ≤ 300 |
| Exclusion Area: | | | | |
| Whole Body Dose, rem | (1) | 0.156(3) | 0.183 | ≤ 6.25 |
| Thyroid Dose, rem | (1) | 3.05(3) | 3.84 | ≤ 75 |
| Low Population Zone: | | | | |
| Whole Body Dose, rem | (1) | 2.03E-2(3) | 2.38E-2 | ≤ 6.25 |
| Thyroid Dose, rem | (1) | 0.362(3) | 0.456 | ≤ 75 |
| Control Room: | | | | |
| Whole Body Dose, rem | Not reported in UFSAR | 1.32E-2(3) | 1.54E-2 | ≤ 5 |
| Thyroid Dose, rem | | 8.09(3) | 10.2 | ≤ 30 |
| Beta Dose, rem | | 0.491(3) | 0.575 | ≤ 30 |

Notes:

(1) UFSAR Table 15.7-8 lists doses as a function of distance and meteorological condition. The values are at 1/2 mile under unstable 2 mph wind speed meteorological condition and represent the worst case values reported.

(2) Not evaluated as it is considered historical information.

(3) Doses developed to support proposed conversion to Improved Technical Specifications (ITS) as described in a letter from R. M. Krich (ComEd) to U. S. NRC, "Request for Technical Specifications Changes for Dresden, Units 2 and 3, LaSalle County Station, Units 1 and 2, and Quad Cities Nuclear Power Station, Units 1 and 2, to Implement Improved Standard Technical Specifications," dated March 3, 2000.

## Attachment B
## Additional Offsite Dose Information Supporting the License Amendment Request to Permit Uprated Power Operation at Quad Cities Nuclear Power Station, Units 1 and 2

Question

*The ComEd application cites Section 5.4 of the ELTR1, which states that the uprate application will provide justification that current radiological consequences are still bounding and within applicable criteria, or provide re-analysis of any areas adversely affected by power uprate. Appendix H of the ELTRI describes the methodology and assumptions for these re-analyses. Appendix H does not provide specific assumptions, noting only that the analyses will be based on the methodology, assumptions, and analytical techniques described in the regulatory guides, the SRP, and previous safety evaluations. The staff's SER on ELTR1 dated September 14, 1998, notes that radiological consequences will be assessed on a plant-specific basis using NRC-approved methods.*

*Section 9.3 of the safety analysis report for the Quad Cities 1 & 2 EPU addresses the radiological consequences of design basis accidents. While this section identifies the magnitude of change in the results, the application does not adequately identify the methodology, assumptions, and inputs used by ComEd in arriving at these conclusions. This information is necessary for the staff to determine whether the ComEd analyses are acceptable and meet the provisions of the ELTR1 and the staff SER on the ELTR1. Please provide the following additional information, or provide a cross-reference to where the information can be found in docketed material.*

1. *For any conclusion provided in this section that was derived in total or in part from generic analyses, please describe the analysis or provide a citation to that description. Please explain how the results were determined to be applicable to the Quad Cities 1 & 2 design basis as modified by this uprate.*

   Response

   The accident dose consequences and conclusions presented in Section 9.3, "Design Basis Accidents," of the power uprate safety analysis report (PUSAR) contained in Reference 1 are not derived from generic analyses.

2. *If any of the accident dose results were obtained by plant-specific re-analysis, as opposed to scaling previous FSAR results, please provide a tabulation of analysis inputs and assumptions that will enable the staff to evaluate the acceptability of these assumptions, and as necessary, perform confirming calculations. Please identify any changes to prior design basis analysis inputs, assumptions, and methodologies, including offsite and control room atmospheric dispersion coefficients, incorporated in these re-analyses.*

   Response

   The extended power uprate (EPU) accident dose consequences presented in Section 9.3 of the PUSAR were obtained by applying scaling techniques to previous results to reflect increased core inventory and increased mass releases, as applicable. No plant specific re-analyses were performed. There were no changes to prior design basis analysis inputs, assumptions, or methodologies, including offsite and control room atmospheric dispersion coefficients.

3. *The application reports that the LOCA, CRDA, and FHA offsite thyroid and whole body doses increased by 27% and 18% respectively. This suggests that these results were obtained by multiplying the previous doses by a factor based on the increase in core inventory. This methodology is generally acceptable. However, the requested power uprate is only 17 percent. The application implies that the lack of proportionality might be due to the difference in U-235 and Pu-239 fission yields. However, ORIGEN data available to the staff (NUREG/CR-6703) indicates that the inventory of I-131 (Ci/MWt) increases by less than 2% from 22 to 75 GWD/MTU. Please explain the derivation of the 27% and 18% factors providing sufficient information for the staff to confirm the acceptability of these factors. Similarly, please explain why the control room factors for the LOCA differ from the factors used for the other accidents. If analyses were performed to derive the 27% and 18% factors, please describe the inputs, assumptions, and methodologies used.*

   Response
   As discussed in Section 9.3 of the PUSAR, the pre-EPU thyroid dose analyses for the loss of coolant accident (LOCA), CRDA, and fuel handling accident (FHA) used the iodine inventories, in curies/megawatt (Ci/MW), from Table I of Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactor Sites," (Reference 2), which was published on March 23, 1962. These core inventories were based on the fission product yields of U-235 only, which was the typical practice for low burnup fuel at that time. This data represented the state of the art for determining fission product generation at that time.

   As discussed in Section 9.3 and Table 1-3, "NSSS Computer Codes Used for EPU," of the PUSAR, the EPU core inventory for GE14 fuel and a 24-month fuel cycle was calculated using the ORIGEN2 code, which uses current methodology and data for fission product generation. The present methodology includes up to date fission product yields and decay chains, but also includes the fission product contributions from transuranic nuclides (e.g., Pu-239 and Pu-241), which become important with the present day higher burnup fuel cycles.

   The scaling factor for thyroid dose is the ratio of the ORIGEN2 iodine inventories, weighted for dose factor, at the EPU power level to the TID-14844 iodine inventories at the pre-EPU power level. This factor was found to be 1.27 (i.e., the 27% increase discussed in Section 9.3 of the PUSAR) and includes the effects of the increase in power level and the increases in iodine nuclides due to the 24 month cycle, reflecting the change in fission product generation described above.

A similar scaling factor was determined for the whole body dose. The whole body dose is primarily affected by the noble gas fission product inventories. This factor was found to be less than the increase in power level alone (i.e., less than 18%) because the ORIGEN2 code has shown decreased noble gas production compared to previous methods. However, the EPU scaling factor for the whole body dose is conservatively maintained at the percentage of uprate (i.e., the 18% increase discussed in Section 9.3 of the PUSAR).

As noted in Section 9.3 of the PUSAR, the EPU LOCA control room thyroid and whole body dose scaling factors are 1.30 and 1.20, respectively, as a result of the addition of an additional 2% margin. This margin was previously included in all of the other pre-uprate accident analyses, except the control room LOCA dose analysis.

As requested by the NRC, the significant EPU core inventories, calculated using the ORIGEN2 computer code, are provided in Table 1 of this attachment. The sixteen nuclides listed represent the significant contributors to the core inventory. Other nuclides were not shown, as their contribution is small relative to those listed.

4. *Have any UFSAR or CURRENT results in Tables 9-7 through 9-8 been revised as a result of any analysis changes since this application was docketed?*

   <u>Response</u>
   There have been no new analyses performed that change the results in Tables 9-7 through 9-8 since the EPU license amendment request was submitted. However, subsequent review of Table 9-7, "LOCA Radiological Consequences," and Table 9-9, "FHA Radiological Consequences," has determined that revisions are required to some of the reported dose values as a result of an oversight in updating these tables. Table 2 of this attachment provides a revised Table 9-9, to reflect the correct dose values. The changes to the table are noted with revision bars. The changes to Table 9-7 are discussed in the response to question 5 below.

5. *In Table 9-7, please explain the difference between the "UFSAR" and "CURRENT" values tabulated for the control room thyroid dose. Please explain why only the control room thyroid dose changed.*

Response
At the time the EPU license amendment request was submitted, a change was pending to the Updated Final Safety Analysis Report (UFSAR) regarding the control room dose resulting from a LOCA. The calculation of control room dose following a LOCA had been revised and a change package to update the UFSAR had been developed and was awaiting site approval. Thus, the control room dose shown in the UFSAR was reported in the "UFSAR" column of Table 9-7 and the result from the updated calculation was reported in the "Current" column of Table 9-7. The control room whole body and beta doses from the current calculation were not updated in Table 9-7 due to an oversight. Table 3 of this attachment provides an updated Table 9-7. The changes have been indicated with revision bars.

6. *Section 15.6.5.5.3.3 of the FSAR discusses the control room infiltration rates for Quad Cities. An earlier submittal dated May 19, 1997, subsequently withdrawn, indicated that the in-leakage measured with tracer gas testing shows that the observed leakage was less than the calculated leakage. Please confirm that this conclusion is still valid.*

Response
The conclusion that the actual unfiltered control room in-leakage is less than the calculated unfiltered in-leakage given in the UFSAR is still valid. The basis for this conclusion is that the site completed the baseline tracer gas testing in April 1997 and has the following ongoing programs and surveillance tests to assure that any degradation in unfiltered control room in-leakage is identified and corrected.

- A plant barrier control program maintains the integrity of the control room envelope boundary, including ductwork.
- A periodic test assesses the control room in-leakage. It involves a walk down of the control room envelope, including visual inspections and a smoke test of the negative pressure ductwork located outside of the control room envelope, isolation dampers, and door seals.
- A periodic surveillance required by the QCNPS Technical Specifications (TS) that measures filtered outside supply air quantity and control room pressure with the control room emergency ventilation system operating in emergency mode provides an indication of any changes to the in-leakage. An increase in the control room pressure without an increase in filtered outside air supply indicates an increase in in-leakage. Although this test does not provide quantitative results of actual in-leakage, it is used as a tool to provide warning of any changes in in-leakage and identify the need for further investigation.

This test has not shown any indication of increased in-leakage.

7. *Section 4.7 identifies that the time to reach the oxygen limit decreases from 25 hours for pre-EPU to 19 hours post-EPU. Does this observation affect any analysis assumption regarding the dose impacts of the operation of the CAD system post-LOCA? If so, please describe how this was considered in determining the LOCA dose.*

   Response
   The dose impacts due to operation of the nitrogen containment atmosphere dilution (NCAD) system following a LOCA are unchanged due to EPU. No venting due to operation of the NCAD system is necessary within a 30-day period following a LOCA.

   No venting is required for 30 days the following reason. The decrease in time from 25 hours to 19 hours to reach the primary containment oxygen limit results in earlier operation of the NCAD system. Even with earlier nitrogen injection, however, primary containment pressure remains below the repressurization limit of 31 psig (i.e., 50% of design pressure) stated in 10 CFR 50.44, "Standards for combustible gas control system in light-water-cooled power reactors," until 32 days following a LOCA. Pre-EPU analyses indicated the pressurization limit was reached at 48 days following a LOCA. Upon reaching the pressurization limit, any necessary containment venting would be controlled by the plant emergency operating procedures.

References

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

2. U. S. Atomic Energy Commission Technical Information Document (TID) – 14844, "Calculation of Distance Factors for Power and Test Reactor Sites," March 23, 1962

## Attachment B
## Additional Offsite Dose Information Supporting the License Amendment Request to Permit Uprated Power Operation at Quad Cities Nuclear Power Station, Units 1 and 2

**Table 1**
**Extended Power Uprate (EPU) Equilibrium Core Inventory of Noble Gases and Iodines (Curies/Megawatt -Thermal) Utilized for Site Boundary and Control Room Dose Analyses Dresden and Quad Cities Nuclear Power Stations**

| Isotope | EPU Core |
|---|---|
| I 131 | 2.710E+04 |
| I 132 | 3.914E+04 |
| I 133 | 5.501E+04 |
| I 134 | 6.035E+04 |
| I 135 | 5.157E+04 |
| KR 83M | 3.249E+03 |
| KR 85 | 4.364E+02 |
| KR 85M | 6.772E+03 |
| KR 87 | 1.291E+04 |
| KR 88 | 1.815E+04 |
| XE 131M | 3.040E+02 |
| XE 133 | 5.282E+04 |
| XE 133M | 1.726E+03 |
| XE 135 | 2.144E+04 |
| XE 135M | 1.089E+04 |
| XE 138 | 4.500E+04 |

## Attachment B
## Additional Offsite Dose Information Supporting the License Amendment Request to Permit Uprated Power Operation at Quad Cities Nuclear Power Station, Units 1 and 2

### Table 2
### Revised Power Uprate Safety Analysis Report (PUSAR) Table 9-9

Table 9-9
**FHA Radiological Consequences**

| Location | UFSAR | Current | EPU | Limit |
|---|---|---|---|---|
| Fuel Handling Accident (Single fuel bundle and handling equipment dropped) | | | | |
| Offsite: | | | | |
| Whole Body Dose, rem | 5.9E-3(1) | (2) | (2) | ≤ 25 |
| Thyroid Dose, rem | 4.1E-3(1) | (2) | (2) | ≤ 300 |
| Exclusion Area: | | | | |
| Whole Body Dose, rem | (1) | 0.358(3) | 0.422 | ≤ 6.25 |
| Thyroid Dose, rem | (1) | 9.92(3) | 12.6 | ≤ 75 |
| Low Population Zone: | | | | |
| Whole Body Dose, rem | (1) | 3.8E-2(3) | 4.48E-2 | ≤ 6.25 |
| Thyroid Dose, rem | (1) | 0.687(3) | 0.873 | ≤ 75 |
| Control Room: | | | | |
| Whole Body Dose, rem | Not | 1.20E-2(3) | 1.40E-2 | ≤ 5 |
| Thyroid Dose, rem | reported in | 7.66(3) | 9.73 | ≤ 30 |
| Beta Dose, rem | UFSAR | 0.462(3) | 0.545 | ≤ 30 |

Notes:

(1) UFSAR Table 15.7-3 lists doses as a function of distance and meteorological condition. The doses reported above are at ¼ mile under unstable 2 mph wind speed meteorological condition and represent the worst case values reported.

(2) Not evaluated as it is considered historical information.

(3) Doses developed to support proposed conversion to Improved Technical Specifications (ITS) as described in a letter from R.M. Krich (ComEd) to U.S. NRC, "Request for Technical Specifications Changes for Dresden Nuclear Power Station, Units 2 and 3, LaSalle County Station, Units 1 and 2, and Quad Cities Nuclear Power Station, Units 1 and 2, to Implement Improved Standard Technical Specifications," dated March 3, 2000

**Attachment B**
**Additional Offsite Dose Information Supporting the License Amendment Request to Permit Uprated Power Operation at Quad Cities Nuclear Power Station, Units 1 and 2**

**Table 3**
**Revised Power Uprate Safety Analysis Report (PUSAR) Table 9-7**

Table 9-7

**LOCA Radiological Consequences**

| Location | UFSAR | Current | EPU | Limit |
|---|---|---|---|---|
| Offsite: | | | | |
| Whole Body Dose, rem | 5.3E-4 (1) | (2) | (2) | ≤ 25 |
| Thyroid Dose, rem | 1.3E-4 (1) | (2) | (2) | ≤ 300 |
| Exclusion Area: | | | | |
| Whole Body Dose, rem | 5 (3) | 5 | 6 | ≤ 25 |
| Thyroid Dose, rem | 120 (3) | 120 | 152 | ≤ 300 |
| Low Population Zone: | | | | |
| Whole Body Dose, rem | < 5 (3) | <5 | < 6 | ≤ 25 |
| Thyroid Dose, rem | < 120 (3) | < 120 | < 152 | ≤ 300 |
| Control Room: | | | | |
| Whole Body Dose, rem | 0.118 (4) | 0.314 | 0.377 | ≤ 5 |
| Thyroid Dose, rem | 21.88 (4) | 22.75 | 29.6 | ≤ 30 |
| Beta Dose, rem | 1.23 (4) | 8.71 | 10.5 | ≤ 30 |

Notes:
(1) UFSAR Sect. 15.6.5.5.1, Table 15.6-6 (original analysis). This table lists doses as a function of distance and meteorological condition. The doses listed above are at ¼ mile under unstable 2 mph wind speed meteorological condition and represent the worst case values reported.
(2) Not evaluated as it is considered historical information.
(3) UFSAR Sect. 15.6.5.5.1, AEC analysis, 1% per day primary containment leak rate.
(4) UFSAR Sect. 15.6.5.5.3, Table 15.6-8.