Commonwealth Edison
1400 Opus Place
Downers Grove, Illinois 60515

October 29, 1993

Dr. Thomas E. Murley, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attn: Document Control Desk

Subject: Dresden Nuclear Power Station Units 2 and 3
Quad Cities Nuclear Power Station Units 1 and 2
Request for EXIGENT LICENSE AMENDMENT,
to Facility Operating Licenses DPR-19, DPR-25, DPR-29, and DPR-30
NRC Docket Nos. 50-237/249 and 50-254/265

References: (a) L. DelGeorge letter to T. Murley, dated September 15, 1993.

(b) P. Piet letter to T. Murley, dated October 8, 1993.

(c) Meeting between Commonwealth Edison and the NRC Staff, dated October 21, 1993.

Dear Dr. Murley:

Pursuant to 10 CFR 50.91(a)(6), Commonwealth Edison Company (CECo) proposes to amend Facility Operating Licenses DPR-19, DPR-25, DPR-29, and DPR-30 and requests that the Nuclear Regulatory Commission (NRC) grant these amendments on an Exigent basis. This exigent License Amendment request supersedes the Reference (b) submittal in its entirety. The amendment is needed by November 22, 1993 at 11:59 PM based upon the CECo modification schedule outlined in the Reference (a) letter.

The proposed License Amendment dispositions Unreviewed Safety Questions (USQ) related to proposed plant modifications associated with Reactor Water Level Instrumentation. These modifications have been initiated to mitigate the circumstances outlined in NRC Bulletin (IEB) 93-03, "Resolution of Issues related to Reactor Vessel Water Level Instrumentation in BWRs."

The attached safety analysis shows that the License Amendment will have minimal impact on safety. The need for this exigent change could not be avoided because the design for the backfill instrumentation was modified on a schedule to ensure the Reactor Vessel Water Level Instrumentation System (RVLIS) concerns were addressed during the planned maintenance outage for Quad Cities Unit 1 scheduled to be completed before November 22, 1993. The technical resolution of the final modification design, however,

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exceeded a time frame that would allow normal License Amendment processing by the NRC Staff and still allow startup from the planned maintenance outage for Quad Cities Unit 1 (scheduled for completion before November 22, 1993).

Therefore, this condition was not created by the failure to make a timely application for a License Amendment. The need for this exigent License Amendment was previously discussed with members of the NRC Staff during the Reference (c) meeting.

In support of this request, the following information is provided:

- Attachment A provides a description and safety analysis of the proposed change.

- Attachment B describes CECo's evaluation performed in accordance with 10 CFR 50.92(c), which confirms that no significant hazards consideration is involved.

- Attachment C provides an Environmental Assessment for the proposed change.

This request for an Exigent Technical Specification Amendment has been reviewed and approved by CECo Station Management as well as On-Site Review and Off-Site Review in accordance with CECo procedures.

To the best of my knowledge and belief, the statements contained above are true and correct. In some respect these statements are not based on my personal knowledge, but obtained information furnished by other Commonwealth Edison employees, contractor employees, and consultants. Such information has been reviewed in accordance with company practice, and I believe it to be reliable.

Pursuant to 10 CFR 50.91(b)(1), a copy of this request has been forwarded to the Illinois Department of Nuclear Safety.

Please direct any questions you may have concerning this submittal to this office.

Sincerely,

Peter L. Piet
Nuclear Licensing Administrator

Attachments:

A. Description and Safety Analysis of the Proposed Changes
B. Significant Hazards Consideration
C. Environmental Assessment Statement Applicability Review

cc: J. B. Martin, Regional Administrator - RIII
T. Taylor, Senior Resident Inspector - Quad Cities
M. N. Leach, Senior Resident Inspector - Dresden
J. L. Kennedy, Project Manager - NRR
C. P. Patel, Project Manager - NRR
J. F. Stang, Project Manager - NRR
Office of Nuclear Facility Safety - IDNS

Signed before me on this 29th day
of October, 1993,
by ______________________.
Notary Public

" OFFICIAL SEAL "
SANDRA C. LARA
NOTARY PUBLIC, STATE OF ILLINOIS
MY COMMISSION EXPIRES 6/25/94

ATTACHMENT A

DESCRIPTION AND SAFETY ANALYSIS OF THE PROPOSED CHANGES

Schedule for Modification Installation and Review of License Amendment

CECo proposes to install Reactor Pressure Vessel Backfill Modifications to address the concerns discussed in NRC Bulletin (IEB) 93-03. The current schedule outlines the complete installation and implementation of the modifications during planned maintenance outages for Quad Cities Unit 1 (November 1993) and Quad Cities Unit 2 (December 1993). For Dresden Unit 2, CECo plans to implement the modifications during the first Cold Shutdown after December 1, 1993. For Dresden Unit 3, CECo plans to implement the modifications during the upcoming Refueling Outage (D3R13) currently scheduled to begin during March 1994.

The submittal of this proposed License Amendment is based upon CECo's determination that the proposed modifications involve an Unreviewed Safety Question (USQ). The proposed modifications increase the probability of a previously analyzed accident due to the potential for inadvertent closure of the Reference Leg root valve and subsequent pressurization of the Reactor Pressure Vessel (RPV) Level and Containment Pressure instrumentation. This potential event was first identified by the industry and communicated to CECo in August 1993. CECo incorporated this information into the design and evaluation of the proposed Backfill modification at Dresden and Quad Cities Stations. The non-safety-related and non-seismic CRD system will be actively connected to each of the safety-related divisions of RPV instrumentation. The proposed modification is considered to introduce the potential for a "malfunction of a different type than any evaluated previously in the safety analysis report..." The installation of the modification with this configuration represents an Unreviewed Safety Question. Therefore, CECo requests review and approval of the proposed amendment to Facility Operating Licenses DPR-19, DPR-25, DPR-29, and DPR-30 prior to November 22, 1993 in order to support the current implementation schedule.

Description of Proposed License Amendment

The Reactor Vessel Level Indication System (RVLIS) backfill modification is being installed at Dresden Units 2 and 3 and Quad Cities Units 1 and 2 in response to NRC Bulletin 93-03. The actual physical routing of the designs at both stations is similar to the design that has previously been installed within the industry and for the indication only reference legs at LaSalle Station. The proposed backfill subsystem will resolve the concern of inaccurate Reactor Pressure Vessel (RPV) level indication due to the presence of noncondensible gases in the RVLIS reference legs after a depressurization of the vessel.

Implementation of the modification creates the potential for pressurizing one of the RVLIS reference legs to CRD drive water header pressure in the event of a root valve closure with the backfill subsystem in operation (see attached drawing). The condition could activate RPV low level safety features as well as RPV high pressure safety features depending on which valve is closed.

The likelihood of this event occurring is small due to the physical location of the valves and the administrative controls that govern the operation of the valves. For these reasons, the potential increase in the probability of a previously analyzed accident is not significant.

The proposed configuration connects the non-safety-related CRD system to each division of RPV instrumentation. The failure of the CRD piping may result in instrument line leakage. However, the isolation action of the redundant safety-related reference leg backfill instrument check valves limits the consequences associated with this malfunction to those consequences associated with previously analyzed accidents. This design is per GDC 14 and IEEE-279.

Description of the Proposed Modifications

As previously stated, a modification to RVLIS is currently planned for installation at Dresden and Quad Cities Stations to meet the requirements of NRC Bulletin 93-03. This modification includes the connection of a low flow, high pressure water supply to 4 reference legs to provide a continuous backfill through the reference leg, condensate pot and the reactor vessel. The source of the water supply is the CRD drivewater header, which currently has pressure control at approximately 300 psi above reactor pressure. The new backfill subsystem provides deaerated water to the reference leg to prevent the accumulation of dissolved gases that can later come out of solution during reactor vessel depressurization.

The new subsystem prevents degraded level indications commonly appearing as "notches" by 1) forming a barrier of degassed water that will prevent gases from dissolving in the condensate pot and being transported down the reference leg, 2) purging the reference leg with deaerated water to "sweep" dissolved gases from the reference leg, and 3) providing a continuous fill of the reference leg in case non-condensible gases prevent adequate condensation in the condensate pot to otherwise keep the reference leg full of water.

ATTACHMENT A

The new backfill subsystem utilizes water from the CRD drivewater header. This water supply flows to each of 2 instrument racks, 2201(2)-5 and 2201(2)-6 at Quad Cities and 2202(3)-5 and 2202(3)-6 at Dresden. Near these racks is a new panel with 2 flow stations and a water filter which acts as a pressure snubber. Each flow station consists of: needle valves for system startup and shutdown; metering valves for flow regulation; local flow indicators for setting flow rates through the backfill line for each reference leg; multiple check valves for SR-to-NSR and/or system separation; instrument taps for testing components; a vent connection for purging air from the lines; and isolation valves to isolate components for maintenance. A detailed drawing of the preliminary design is attached.

During power operation, there exists a small potential for the inadvertent valve manipulation error of a reference leg root valve. These valves currently exist and are located near the drywell penetration for the reference leg instrument lines. As previously stated, these valves are only occasionally closed to perform maintenance during outages.

If a reference leg root valve were to be inadvertently closed in the current unmodified configuration, the affected instruments using the reference leg would be inoperable. The pressure instruments would indicate the existent pressure at the time of isolation or show a declining pressure if there is leakage from the reference leg. Level instruments would also lose accuracy but would not immediately cause an ESF actuation. The following event is possible, though very unlikely, in the modified RVLIS system, if 1) the Unit is at power, 2) the reference leg root valve is inadvertently closed, and 3) the backfill system has not been first isolated and continues injecting CRD drivewater into the reference leg to pressurize it to approximately 1300 psig at normal vessel pressure. This event causes the affected pressure instruments to indicate a false high reactor pressure and the level instruments to indicate or trip on a false low level indication.

The speculated consequences of the inadvertent closing of each reference leg root valve without first isolating the backfill subsystem are estimated below.

<u>Condensate Pot [Quad Cities 1(2)-0263-12A Dresden 2(3)-0263-12A]</u>

* All reactor vessel relief valves open immediately due to high indicated pressure (greater than 1135 psig)

* High Pressure reactor SCRAM

ATTACHMENT A

* Isolation Condenser initiation due to high pressure [Dresden]

* Low water level reactor SCRAM

* Low-low water level Group 1 isolation (MSIV's close)

* HPCI starts and injects on the (false) low-low water level

* RCIC starts and injects on (false) low-low water level [Quad Cities]

* Low pressure ECCS systems start on low pressure permissive signal or after 8.5 minutes, whichever occurs first

* Some false low water level indications in the Control Room, but most are unaffected

  NOTE: Normal feedwater level control, but feedwater pump and HPCI [RCIC and main turbine also for Quad Cities] trips on high water level are inoperable. Turbine trip on high water level is still available from unaffected reference leg.

Condensate Pot [Quad Cities 1(2)-0263-12B Dresden 2(3)-0263-12B]

* Same as 1(2)-0263-12A [Quad Cities] or 2(3)-0263-12A [Dresden], except that reactor vessel relief valves will not open immediately on high pressure but may open after 8.5 minutes to perform the function of ADS

Condensate Pot [Quad Cities 1(2)-0263-13A Dresden 2(3)-0263-13A]

* If LT 1(2)-0646-A [Quad Cities] or 2(3)-0646A [Dresden] is selected for normal feedwater level control, feedwater flow will increase to makeup for the (false) low water level indication. Pumps will trip on high water level trips, if they are not secured manually or level control is not switched to other level channel.

  If LT 1(2)-0646-B [Quad Cities] or 2(3)-0646B [Dresden] is selected for normal water level control, feedwater level control is unaffected.

* Some false low water level indications in the Control Room, but most indicators are unaffected

* Some false pressure indication in the Control Room, but most indicators are unaffected

Condensate Pot [Quad Cities 1(2)-0263-13B Dresden 2(3)-0263-13B]

* If LT 1(2)-0646-B [Quad Cities] or 2(3)-0646B [Dresden] is selected for normal feedwater level control, feedwater flow will increase to makeup for the (false) low water level indication. Pumps will trip on high water level trips, if they are not secured manually or level control is not switched to other level channel.

  If LT 1(2)-0646-A [Quad Cities] or 2(3)-0646A [Dresden] is selected for normal water level control, feedwater level control is unaffected.

* Some false low water level indications in the Control Room, but most indicators are unaffected

* Some false pressure indication in the Control Room, but most indicators are unaffected

The increased significance of the inadvertent valve closure described above causes these events to be similar to previously analyzed accidents and transients. For example, the event on the reference leg from the 12A condensate pot causes the Target Rock and Electromatic Relief Valves to immediately open causing a loss of coolant. Other ECCS systems and RCIC (for Quad Cities) start and inject into the vessel in a manner similar to a Loss of Coolant Accident (LOCA). If the reference leg from the 13A or 13B condensate pot is isolated and that loop is being used for feedwater level control, the response of the plant is similar to a failure open of a feedwater regulating valve. The feedwater pump runout trip would not occur, but the high water level trip of the feedwater pumps and turbine would still be operable. The potential valve manipulation errors, therefore, can be considered an "increase in the probability of occurrence or consequences of an accident or malfunction of equipment important to safety." The installation of the modification with this configuration represents an Unreviewed Safety Question under paragraph "a(2)i" of 10CFR50.59. NRC review and concurrence with the No Significant Hazards Analysis is required.

The non-safety-related and non-seismic CRD system will be actively connected to each division of RPV instrumentation. The proposed modification, by the introduction of the non-safety-related/non-seismic CRD system may introduce the potential for a "malfunction of a different type than any evaluated previously in the safety analysis report..." The connection of the non-safety-related backfill piping to the safety-related vessel instrumentation line requires that a safety-related isolation boundary be established. The isolation boundary will ensure that the vessel reference leg piping remains filled in the event of challenges to the piping integrity or depressurization of the CRD system piping. This boundary is provided by two (2) safety-related check valves in series that mitigates this condition. The check valves allow flow to the vessel instrumentation reference leg piping and prevent flow out of the reference leg piping. The installation of the modification with this configuration represents an Unreviewed Safety Question under paragraph "a(2)ii" of 10CFR50.59. NRC review and concurrence with the No Significant Hazards Analysis is required.

Justification of Proposed Modifications and License Amendment

The modifications described above are being installed in response to an issue identified in NRC Bulletin (IEB) 93-03, "Resolution of Issues Related to Reactor Water Level Instrumentation in BWRs." The installation of these modifications will enhance plant safety by assuring that the degassing phenomenon described in Bulletin 93-03 will not be encountered at Dresden and Quad Cities Stations.

The modifications are similar in design to modifications that have been installed at other plants. To date, Commonwealth Edison is unaware of any difficulties or operational concerns that other licensees have encountered as a result of implementing similar types of backfill modifications to address the degassing phenomenon described in IEB 93-03.

The potential safety issues (described in the Description of Proposed Modifications section above) associated with installation of these modifications are not significant for the following reasons:

> The root valve(s) referred to above are installed so that the instrument lines can be isolated. Isolation of the instrument lines is required when the excess flow check valves in those lines are repaired, tested, or when the instrument lines are taken out of service for other reasons. These administrative measures will be procedurally controlled at both Dresden and Quad Cities Stations. The testing and/or repair of the instrument lines almost exclusively occur when the reactor is not in an operational mode to which the phenomenon described in IEB 93-03 is applicable or a concern. The station procedural controls governing the out-of-service process lessens the possibility of a valve manipulation error. The out-of-service process also ensures that the valves are properly returned to service. This administrative control is performed in conjunction with the usage of valve checklists that

are performed prior to a unit startup. In addition, the status of the safety lock that will be installed on the valves is checked at the end of each refuel outage prior to a unit startup.

The primary concern is a mismanipulation of the valves while the unit is at power. This scenario is unlikely because these valves are located at the penetration where there are no normally operated valves. To further minimize the possibility of a valve error, the valves will be locked in the open position with a lock, the keys for which are administratively controlled. In addition, labels that clearly identify the valves will be provided at the valve location indicating that operation of the valves will result in a plant transient and that they are not to be operated without permission of shift supervision.

To further guard against mismanipulation, the operators and Instrument Maintenance Technicians will be trained on the location and purpose of the valves and on the consequences of closing the valves without first taking the backfill system out of service. Also, the process of taking the backfill system out of service and returning it to service will be administratively controlled by station procedures.

Training will be provided to the control room operators as part of the modification. This training will include directions concerning how to recognize the indications that the root valves have been mispositioned, and what actions to take to control a possible resultant transient.

The backfill lines are connected in such a manner that they do not have an adverse effect on the capability of the connected instruments to perform their function. The backfill lines have no effect on the response time and an insignificant impact on instrument accuracy. The design of the backfill system satisfies the redundancy, independence and testability requirements of the reactor protection system. The safety-related portions of the backfill lines are designed to the same level of quality as the existing instrument lines; the check valves will not close accidentally during normal operation but will close if instrument line integrity is challenged during normal or accident conditions.

The failure of the CRD piping integrity could result in challenges to the RPV instrumentation due to reference leg leakage. The isolation action of the two (2) backfill instrument line check valves mitigates this condition. The check valves allow flow to the vessel instrumentation reference leg piping and prevent flow out of the reference leg piping.

Leakage criteria will be established to provide assurance that vessel level instrumentation integrity is adequately maintained in the event of CRD system depressurization. The basis for the check valve leakage shall be calculated to verify that the leakage from the reference leg piping over an acceptable time period is limited to an appropriate value. A loss of inventory greater than this value to the reference leg would excessively degrade indicated vessel level. Small deviations are acceptable for ensuring level instrumentation actuations are within analytical limits and ensures that vessel level indication is adequately provided to the Operator for assessing plant conditions. A time period will be established to ensure that recognition of the condition is made and isolation of the piping occurs.

Schedule

CECo requests review and approval of the proposed amendment to Facility Operating Licenses DPR-19, DPR-25, DPR-29, and DPR-30 prior to November 22, 1993 in order to support the current implementation schedule.

ATTACHMENT B

SIGNIFICANT HAZARDS CONSIDERATION

Commonwealth Edison has evaluated the proposed License Amendment and determined that it does not represent a significant hazards consideration. Based on the criteria for defining a significant hazards consideration established in 10 CFR 50.92, operation of Dresden and Quad Cities Stations in accordance with the proposed amendment will not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated because:

   The addition of the backfill instrumentation piping does not significantly increase the probability of an accident previously evaluated due to the low probability of the inadvertent closure of the root valve(s). CECo has evaluated the estimated frequency of the inadvertent closure of the root valve(s) at approximately 1E-08 per reactor year given the implementation of administrative controls. The resulting condition (valve mismanipulation) cycles the Reactor Pressure Vessel in a similar manner as a plant LOCA (i.e., simulates LOCA conditions). The current (pre-modification) LOCA initiation frequency is predicted to be approximately 1E-04 per reactor year. Therefore, the proposed modifications do not significantly increase the probability of any previously evaluated accident.

   The consequences of any previously evaluated accident are not increased by the proposed modifications. For example, the consequence of closing the root valve for the reference leg from condensing chamber 12A, without first isolating the backfill injection, is the in advertent pressurization of the reference leg resulting in the opening of the SRV and all Electromatic reliefs. This is equivalent to an inadvertent actuation of the automatic depressurization system (ADS) - an event that is not analyzed in the safety analysis as an initiating event. Regardless, the event is bounded by the recirculation line break analysis in terms of the RPV response. Because this event would release reactor inventory to the suppression pool, it has less significant consequence than other events previously analyzed for Dresden and Quad Cities Stations.

2) Create the possibility of a new or different kind of accident from any accident previously evaluated because:

   For Dresden and Quad Cities Station, a spectrum of Loss-of-Coolant Accidents have previously been evaluated. The accident in question associated with the proposed modifications can be categorized as a LOCA due to the resultant plant response following the initiating conditions. The previously analyzed LOCA analyses bound the conditions introduced by the proposed modifications. As such, the proposed amendment request for Dresden and Quad Cities Stations do not introduce any new or different kinds of accidents.

The proposed modification connects the non-safety-related CRD system to each division of RPV instrumentation. The failure of the CRD piping may result in instrument line leakage. However, this event is mitigated by the isolation action of the reference leg backfill instrument check valves. Although the proposed modifications may introduce the potential for a malfunction of equipment of a different type than previously evaluated in the safety analysis report, the proposed amendment request for Dresden and Quad Cities Stations does not introduce any new or different kinds of accidents.

3) Involve a significant reduction in the margin of safety because:

The previously analyzed LOCA consequences bound the consequences introduced by the inadvertent closure of the root valve(s) and subsequent LOCA conditions. As such, the previously approved safety margin remains unchanged. Therefore, the proposed modifications do not significantly reduce the margin of safety for both Dresden and Quad Cities Stations.

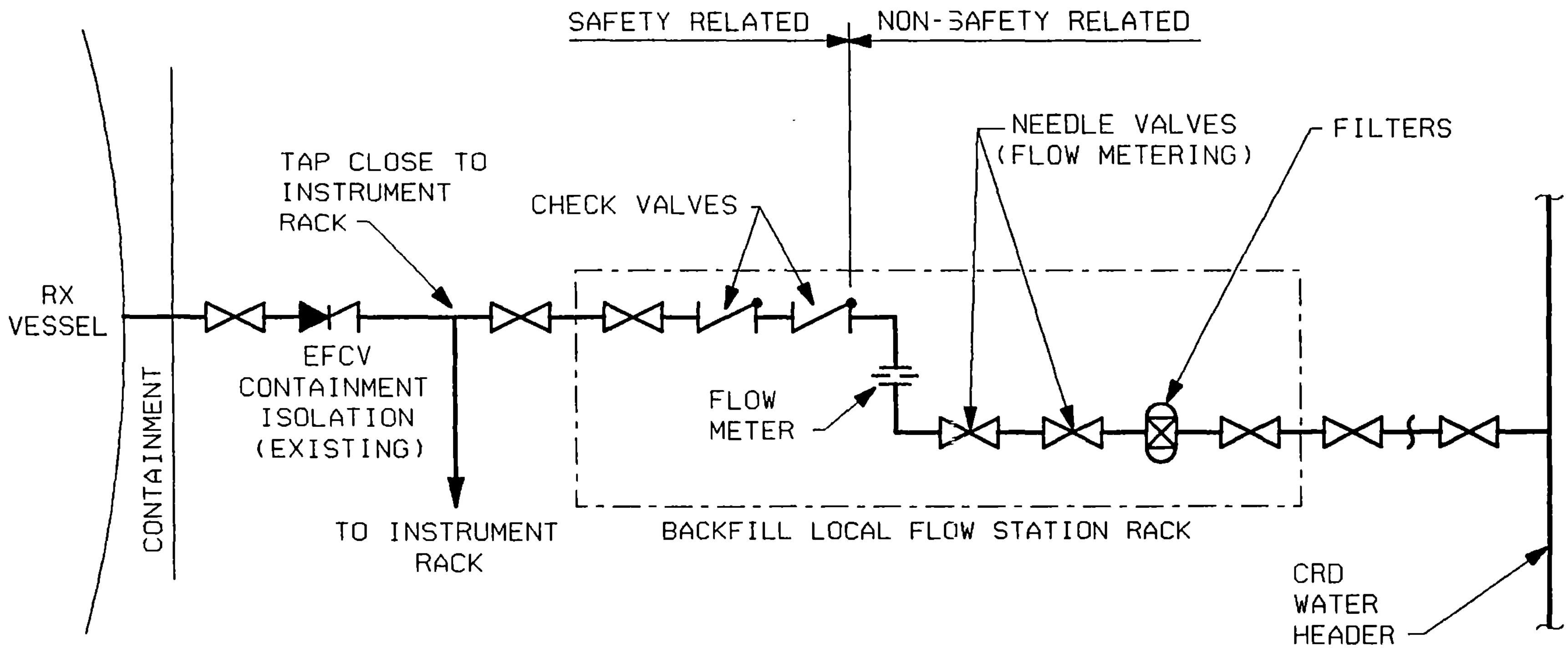
The proposed amendment request does not involve a significant relaxation of the criteria used to establish safety limits, a significant relaxation of the bases for the limiting safety system settings, or a significant relaxation of the bases for the limiting conditions for operations. Therefore based on the guidance provided in the Federal Register and criteria established in 10 CFR 50.92(c), the proposed change does not constitute a significant hazards consideration.

ATTACHMENT C

ENVIRONMENTAL ASSESSMENT STATEMENT APPLICABILITY REVIEW

Commonwealth Edison has evaluated the proposed amendment against the criteria for identification of licensing and regulatory action requiring environmental assessment in accordance with 10 CFR 51.20. It has been determined that the proposed changes meet the criteria for a categorical exclusion as provided under 10 CFR 51.22(c)(9). This conclusion has been determined because the changes requested do not pose significant hazards considerations or do not involve a significant increase in the amounts, and no significant changes in the types, of any effluents that may be released off-site. Additionally, this request does not involve a significant increase in individual or cumulative occupational radiation exposure.

BACKFILL MODIFICATION
DRESDEN/QUAD CITIES



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