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RS-01-151

August 7, 2001

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

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 Plant Systems Information Supporting the License Amendment Request to Permit Uprated Power Operation at Dresden Nuclear Power Station and Quad Cities Nuclear Power Station

References 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 (TS) for Dresden Nuclear Power Station, Units 2 and 3, and Quad Cities Nuclear Power Station, Units 1 and 2, to allow operation at uprated power levels. In a telephone conference call between representatives of EGC and Mr. L. W. Rossbach and other members of the NRC staff, the NRC requested additional information regarding these proposed changes. Attachments A and B to this letter provide a portion of the requested information. The remainder of the requested information will be provided in a separate letter.

Some of the information in Attachment A is proprietary information to the General Electric Company, and EGC requests that it be withheld from public disclosure in accordance with 10 CFR 2.790(a)(4), "Public Inspections, Exemptions, Requests for Withholding." This information is indicated with sidebars. Attachment C provides the affidavit supporting the request for withholding the proprietary information in Attachment A from public disclosure, as required by 10 CFR 2.790(b)(1). Attachment D contains a non-proprietary version of Attachment A.

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

APO1

Respectfully,

for Terrence W. Simpkins
K. A. Ainger
Director – Licensing
Mid-West Regional Operating Group

Attachments:

Affidavit
Attachment A: Additional Plant Systems Information Supporting the License Amendment Request to Permit Uprated Power Operation at Dresden Nuclear Power Station, Units 2 and 3, Quad Cities Nuclear Power Station, Units 1 and 2 (proprietary version)
Attachment B: Draft Revision to Updated Final Safety Analysis Report Section 6.2.1.3
Attachment C: Affidavit for Withholding Portions of Attachment A from Public Disclosure
Attachment D: Additional Plant Systems Information Supporting the License Amendment Request to Permit Uprated Power Operation at Dresden Nuclear Power Station, Units 2 and 3, Quad Cities Nuclear Power Station, Units 1 and 2 (non-proprietary version)

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 Plant Systems Information Supporting the License Amendment Request to Permit Uprated Power Operation at Dresden Nuclear Power Station and Quad Cities 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.

Jeffrey A Benjamin

J. A. Benjamin
Vice President – Licensing and Regulatory Affairs

Subscribed and sworn to before me, a Notary Public in and for the State above named, this 7th day of August, 2001.

V. Lynn Bernhagen

Notary Public

"OFFICIAL SEAL"
V. LYNN BERNHAGEN
Notary Public, State of Illinois
My Commission Exp. 05/30/2004

Attachment B
Additional Plant Systems Information Supporting the License Amendment Request for Uprated Power Operation
Dresden Nuclear Power Station, Units 2 and 3
Quad Cities Nuclear Power Station, Units 1 and 2

Draft Revision to Updated Final Safety Analysis Report Section 6.2.1.3

<u>Revised Pages for Dresden Nuclear Power Station</u>
6.2-24
Insert page for 6.2-24

<u>Revised Pages for Quad Cities Nuclear Power Station</u>
6.2-17
Insert page for 6.2-17

Statement B may appear to contradict existing test data which shows as much as an 11-psi increase in peak drywell pressure due to prepurging. This apparent disparity is attributable to the effects of two phenomena discussed below.

A. Condensation on drywell walls: Due to the high ratio of drywell wall surface area to blowdown flow area, the effects of condensation reduced the peak drywell pressure in tests with cold drywell walls. Prepurging eliminated any significant surface condensation, and higher peak drywell pressures resulted. The calculation of peak drywell pressure did not take credit for surface condensation with or without prepurging.

B. Liquid carryover into drywell vents: The calculation of peak drywell pressure assumes complete carryover of all liquid in the drywell into the drywell vents which increases the peak drywell pressure. However, test data from the Humboldt Bay series of pressure suppression tests[12] reveal that carryover is more likely to be complete if the drywell is initially hot. Hence, the increased carryover would increase the measured pressure compared to a test with less carryover; i.e., one with no purge. Hence, prepurging of the drywell does not significantly affect the peak drywell pressure so long as condensation is neglected and complete liquid carryover is assumed for both the prepurged and nonprepurged cases.

CW Stark 400 7/24/01

~~The pressure and temperature responses of the containment, as originally calculated using Moody's model, are shown in Figures 6.2-19 and 6.2-20. As can be seen in Figure 6.2-19, the calculated peak drywell pressure is 47 psig, which is well below the design allowable pressure of 62 psig.~~

INSERT B

Additional analyses of the containment pressure and temperature response to small break accidents (SBA), intermediate break accidents (IBA), and the DBA were conducted as part of the Mark I Program. Refer to Section 6.2.1.3.6.4 for a description of these additional analyses.

On June 5, 1970, Dresden Unit 2 experienced a transient which caused a safety valve to open and fail to reseat. As a result, the containment atmosphere is postulated to have reached 320°F after approximately 1 hour. A general case in which the containment wall is postulated to be 340°F has been analyzed to demonstrate the adequacy of the containment. It was found that as a result of thermal expansion of the drywell shell against the concrete walls of the containment structure, the thermally induced loads for 340°F at 0.5 psig are the same as for the design condition of 281°F at zero psig. At 340°F and zero psig the loads are slightly greater and result in a slight decrease in safety factor from 2.2 to 1.9. Therefore, it was concluded that the containment structure (design temperature of 281°F) provides adequate safety margin for the maximum steam superheat temperature of 340°F.

Insert A to Section 6.2.1.3.2.1 (Page 6.2-24)

Based on the methodology described in Section 6.2.1.3.2.1, the containment pressure and temperature responses are evaluated at the core thermal power of 3016 MWt (102% of the rated thermal power of 2957 MWt). The short-term results are presented in Figures 6.2-31a and 6.2-32a, and the long-term results in Figures 6.2-19a and 6.2-20a. It is noted that the short-term response calculations are based on input assumptions that maximize the pressure response, whereas the suppression pool temperature response is of primary concern in the long-term response calculations. As shown in Figure 6.2-31a, the peak drywell pressure is calculated to be 43.9 psig. (For historical purposes, the containment pressure and temperature responses (short-term and long-term combined), as originally calculated for the original rated thermal power of 2527 MWt, are shown in Figures 6.2-19 and 6.2-20.)

As the size of the vessel orifice increases, the vessel blowdown rate is overpredicted and the overprediction of peak drywell pressure increases. This trend is illustrated in Figure 6.2-15, where calculated and measured peak drywell pressures are compared. In no case did the model underpredict the test data.

CWB 7/11/01

INSERT A

~~The calculated containment pressure and temperature responses are shown in Figures 6.2-16 and 6.2-17. As shown in Figure 6.2-16, the calculated peak drywell pressure is 47 psig, which is well below the design pressure of 56 psig.~~

6.2-29 Revised analysis of the pressure and temperature response of a similar primary containment (Dresden Unit 2) following an actual LOCA was performed in which peak drywell temperature was calculated to be 320°F. This concern was addressed in Dresden Unit 2 reports entitled "Special Report of Incident of June 5, 1970" and "Supplement to the Special Report of June 5, 1970". The LOCA which caused this peak drywell temperature was a special case small break LOCA (actually a steam leak) which did not have any effect on the design temperature and pressure of the containment (281°F, 56 psig) because the pressure associated with the higher temperature was not a saturation pressure. The resulting combination of slightly higher temperature and significantly lower pressure was less severe than design conditions.

6.2.1.3.2.2 T400 CWB 1/5/01

~~6.2.1.3.3~~ Containment Long Term Response to A Design Basis Accident

6.2-30 After the blowdown immediately following a postulated recirculation line break, the temperature of the suppression chamber water would approach 130°F and the primary containment system pressure equalizes at about 25 psig. Most of the noncondensible gases would be transported to the suppression chamber during blowdown. As condensation in the drywell began, the drywell pressure would decrease and the gases would redistribute between the drywell and the suppression chamber via the vacuum-breaker system.

6.2-31 The core spray system would remove decay heat and stored heat from the core, thereby minimizing core heatup and limiting metal-water reaction to less than 0.1%. The core spray system would transport core heat out of the reactor vessel through the broken recirculation line in the form of hot water. This hot water would flow from the drywell into the suppression chamber via the connecting vent pipes. Steam flow would be negligible. The energy transported to the suppression chamber water would ultimately be removed from the primary containment system by the residual heat removal (RHR) system heat exchangers.

Prior to activation of the containment cooling mode of RHR (arbitrarily assumed to occur at 600 seconds after accident initiation) at least three RHR pumps in the low pressure coolant injection (LPCI) mode would add liquid to the reactor vessel along with core spray. After the reactor vessel was flooded, the excess flow would discharge through the break into the drywell. This flow, in addition to heat losses to the walls, would offer considerable cooling to the drywell and would cause a depressurization of the containment as the steam in the drywell condensed. At 600 seconds, the RHR system may be transferred from the LPCI mode to the containment cooling mode. The containment spray would not be necessary at all and the transfer to containment cooling mode would not be necessary for several hours.

Insert A to Section 6.2.1.3.2.1 (Page 6.2-17)

Based on the methodology described in Section 6.2.1.3.2, the containment pressure and temperature responses are evaluated at the core thermal power of 3016 MWt (102% of the rated thermal power of 2957 MWt). The short-term results are presented in Figures 6.2-22a and 6.2-25a, and the long-term results in Figures 6.2-16a and 6.2-18a. It is noted that the short-term response calculations are based on input assumptions that maximize the pressure response, whereas the suppression pool temperature response is of primary concern in the long-term response calculations. As shown in Figure 6.2-22a, the peak drywell pressure is calculated to be 43.9 psig, which is well below the design pressure of 56 psig. (For historical purposes, the containment pressure and temperature responses (short-term and long-term combined), as originally calculated for the original rated thermal power of 2511 MWt, are shown in Figures 6.2-16 and 6.2-17.)

Attachment C
Additional Plant Systems Information Supporting the License Amendment Request for Uprated Power Operation
Dresden Nuclear Power Station, Units 2 and 3
Quad Cities Nuclear Power Station, Units 1 and 2

Affidavit for Withholding Portions of Attachment A from Public Disclosure

General Electric Company

AFFIDAVIT

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

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

(2) The information sought to be withheld is contained in Attachment 1 to letter GE-DQC–EPU-01-464, *Plant Systems RAIs*, (GE Proprietary Information), dated August 6, 2001. The proprietary information is delineated by bars marked in the margin adjacent to the specific material in the Attachment 1 to Letter GE-DQC–EPU-01-464, *GE Response to NRC Plant Systems RAIs.*

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

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

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

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

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

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

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

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

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

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

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

(8) The information identified in paragraph (2), above, is classified as proprietary because it contains further details regarding the GE proprietary report NEDC-32961P, *Safety Analysis Report for Quad Cities 1 & 2 Extended Power Uprate,* Class III (GE Proprietary Information), dated December 2000, and NEDC-32962P, *Safety Analysis Report for Dresden 2 & 3 Extended Power Uprate,* Class III (GE Proprietary Information), dated December 2000, which contain detailed results of analytical models, methods and processes, including computer codes, which GE has

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

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

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

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

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

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

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

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

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

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

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

Executed at San Jose, California, this 6th day of August 2001.

George B. Stramback
General Electric Company

Subscribed and sworn before me this 6 TH day of AUGUST 2001.

Notary Public, State of California

ANNA HANLIN
Commission # 1184507
Notary Public - California
Santa Clara County
My Comm. Expires Jun 19, 2002

Attachment D
Additional Plant Systems Information Supporting the License Amendment Request for Uprated Power Operation
Dresden Nuclear Power Station, Units 2 and 3
Quad Cities Nuclear Power Station, Units 1 and 2

Additional Plant Systems Information Supporting the License Amendment Request to Permit Uprated Power Operation (non-proprietary version)

Attachment D
Additional Plant Systems Information Supporting the License Amendment Request for Uprated Power Operation
Dresden Nuclear Power Station, Units 2 and 3
Quad Cities Nuclear Power Station, Units 1 and 2
(Non-Proprietary Version)

This attachment contains responses to NRC Questions 1 through 8, 12, 15, and 17 through 30. Responses to NRC Questions 9, 10, 11, 13, 14, and 16 will be provided separately.

<u>Question</u>
1. *During a telephone call on April 30, 2001, your staff noted that changes were being planned in the feedwater and condensate systems to improve the trip avoidance capability of the plant from transients initiated in these systems at the extended power uprate (EPU) full power conditions. These changes were not described in your application. For both Dresden and Quad Cities, describe the various existing features and planned changes (e.g., delayed tripping of a main feedwater pump on low suction pressure; reactor recirculation pump runbacks) which will minimize plant trips from these conditions. Describe plant startup testing and/or post modification testing which will examine these modifications.*

<u>Response</u>
The EPU feedwater and condensate modifications being implemented to avoid spurious reactor scrams are the addition of a reactor recirculation pump runback feature, changes to the reactor feedwater pump low suction pressure trip logic, and changes to the scaling of feedwater control and indication loops.

<u>Reactor Recirculation Pump Runback</u>
A reactor recirculation pump runback is being added as a trip avoidance feature to reduce the potential for a reactor low water level scram on the loss of either a feedwater or condensate pump at extended power uprate (EPU) conditions. In addition, the reactor low water level scram and isolation setpoint is being changed as discussed in Reference 1, Attachment E, "Power Uprate Safety Analysis Report," (PUSAR), Section 5.3.8. A dynamic analysis of a single feedwater pump trip at EPU conditions indicates that an automatic reactor recirculation runback can reduce core flow and thermal power to within the capability of the running feedwater pumps and avoid a reduction in reactor water level to the scram setpoint. The runback on loss of a condensate pump is initiated in anticipation of the reduced feedwater pump suction pressure when only three of the four condensate pumps remain in operation.

The runback logic is enabled when reactor power exceeds the capability of two feedwater pumps, as measured by total steam flow. A runback is initiated when less than three feedwater pumps are running and reactor water level drops below the low level alarm setpoint, or when less than four condensate pumps are running and total feedwater flow exceeds the capacity of two feedwater pumps. The runback will rapidly reduce core flow to approximately 70% of rated core flow, which is equivalent to 82% of uprated thermal power on the highest rod line. This feature will not alter the response characteristics of the reactor recirculation speed control system under normal operating conditions.

Proper operation of the runback logic will be verified in a post modification functional test. The feedwater control system response will be verified at various power levels. These tests will be used to confirm the runback dynamic analysis.

Reactor Feedwater Pump Low Suction Pressure Trip Logic
Since EPU conditions require the operation of three feedwater and four condensate pumps, there is an increased potential for low feedwater pump suction pressure in the event of a condensate pump trip. Both stations currently have feedwater pump protection logic for low suction pressure conditions. For EPU, this logic will be modified to stagger the feedwater pump trips consistent with three pump operation. The revised logic will trip one feedwater pump if suction pressure decreases to the low suction pressure trip setpoint for 3 to 5 seconds, then trip a second pump if suction pressure remains below this setpoint for 12 to 15 seconds. The remaining feedwater pump will not trip until suction pressure decreases to the low-low suction pressure trip setpoint. All pumps will continue to trip immediately if suction pressure decreases below the low-low suction pressure trip setpoint. Proper operation of the feedwater pump low suction pressure trip logic will be verified in a post modification functional test.

Scaling of Feedwater Control and Indication Loops
To accommodate the increased flow rates, the scaling of steam and feedwater flow loops will be increased to 3.5 million pounds mass per hour (Mlb/hr) for each steam line, 7 Mlb/hr for each feedwater pump, and 14 Mlb/hr each for total steam and feedwater flow. The feedwater pump runout (i.e., maximum feedwater flow) logic will also be revised to accommodate three pump operation. In addition to normal loop calibration and functional testing, the dynamic response of the control system will be verified by incremental step changes at various power levels. Feedwater flow indication will be verified at 90 % and 100 % rated thermal power (RTP) using installed ultrasonic flow devices. Feedwater pump performance will be monitored at various power levels to confirm runout protection requirements. Steam flow will be verified against feewater flow.

Question
2. Provide additional discussion of the effect of the EPU on the feedwater system, including your plans for handling additional flow in the system including heater drains. Are the line and valve sizing and system characteristics adequate for EPU conditions or are changes required? The regulatory concern is challenges to operators and safety systems caused by loss of feedwater heater strings and challenges to fuel integrity caused by the transients associated with loss of feedwater heating.

Response
The results of the EPU performance assessments of the balance of plant main thermal-hydraulic power cycle systems (i.e., feedwater, condensate, and heater drain valves), including consideration of the effect of the actual material conditions, indicate that the design capacity of these systems is sufficient to permit operation at EPU conditions. The thermal-hydraulic power cycle systems do not present a significant risk of additional feedwater transients as a result of EPU.

Evaluation of the feedwater heater drain system piping, valves, and instruments was performed at the pressures and temperatures expected at EPU conditions, assuming the turbine control valves were fully open. Reviews of the feedwater heater level control valve (LCV) and drain valve flows required for a range of uprated power levels were performed and compared to their flow passing capabilities. These reviews determined that the EPU operating conditions do not significantly challenge the inherent flow passing capabilities of the LCVs, with the exception of the DNPS Units 2 and 3 feedwater heater "B" normal LCVs. Further evaluations confirmed that trim replacement for these DNPS LCVs is required. These trim replacements will be implemented during each of the DNPS EPU refueling outages. QCNPS previously made similar changes to these valves and thus does not require this modification. Therefore, the system design will be adequate to prevent additional loss of feedwater transients.

The feedwater heater vessels were evaluated at EPU flows, pressures, and temperatures. With the planned shell modifications to re-rate the "C" and "D" feedwater heaters for increased pressures described in the Attachment G of the PUSAR, these vessels will be adequate to support operation at EPU conditions.

Therefore, based upon the above evaluations, additional challenges to operators, safety systems, and fuel integrity are not expected as a result of operation at EPU conditions.

<u>Question</u>
3. State the re-rated conditions for the feedwater heaters.

<u>Response</u>
The re-rated conditions for the feedwater heaters are provided in Figure 1.

<u>Question</u>
4. With the proposed modifications to the steam dryers, will the moisture carryover remain within the original design bases following EPU? If not, what reviews have been conducted to evaluate the increased moisture carryover?

<u>Response</u>
Review and analysis of current moisture carryover data, and the potential impact of EPU on moisture carryover, determined the need for a modification to the present steam dryer assemblies. The design criteria for the modification was to maintain carryover $\leq$ 0.2 wt% under most normal operating conditions, which is equivalent to the original startup test acceptance criteria. This design criteria was established based upon actual moisture carryover data collected from both the Dresden and Quad Cities Stations. Physical testing of the modified steam dryer assemblies confirmed the carryover fraction to be consistent with the modification design criteria.

<u>Question</u>
5. You have requested a significant increase in the magnitude of a main steam line break that will not be isolable automatically by the main steam isolation signal. You requested to raise the main steam isolation flow from 120% pre-EPU to 125% post-EPU for Dresden Unit 2; 120% pre-EPU to 140% post-EPU for Dresden Unit 3; and 138% pre-EPU to 254.3 psid for Quad Cities. The stated basis in NEDC-32424P-A for the increased magnitude of a main steam line break is to keep the same basis (expressed as a percentage of steam flow) to assure that reactor trip avoidance is maintained. For Dresden and Quad Cities, with a 17% power uprate, this

corresponds to an increase of 20% flow if the same percentage of steam line flow were maintained as addressed in the topical report.

What analyses have been performed for the safety impact (e.g., on core damage frequency or on high energy line break (HELB) analyses) of this additional range of steam line breaks (beyond the increase addressed in the EPU topical report), that is no longer automatically isolable? Provide the basis for the additional requested steam line break flow.

Response

The ELTR (Reference 2), Section F.4.2.5, "MSIV Closure on High Steam Flow Setpoint," states, "The setpoint for initiation of MSIV closure on high steam flow shall be raised to be equivalent to ≤ 140% of the uprated steam flow in each steamline." The proposed DNPS and QCNPS setpoints are all within the topical report setpoint.

The HELB analyses performed for EPU for main steam line breaks identified no additional impact due to EPU. The bounding steam line break is a complete rupture that results in choked flow through the flow restrictor. Since system pressure is not changed under EPU conditions, mass releases from such breaks are the same as before EPU. Since the mass releases are unchanged, there is no additional impact on the reactor core or on structures, systems, or components due to EPU.

For main steam line breaks resulting in less flow than the high flow setpoint, two diverse isolation signals will isolate postulated breaks. In RUN mode, low steam line pressure will result in isolation for breaks large enough to depressurize the steam line. Breaks that pass from 120%-140% flow will result in the low pressure isolation signal. These breaks are therefore still automatically isolable following EPU, regardless of location. In addition, postulated breaks in the main steam tunnel would actuate the area high temperature switches and result in isolation. Operability of both of these isolation signals is governed by the DNPS and QCNPS Technical Specifications (TS).

Question

6. Provide short term and long term results (curves or tables of calculated values as a function of time) of calculations for

- *drywell short term pressure and temperature*
- *suppression pool short term temperature*
- *wetwell atmosphere short term pressure and temperature*
- *suppression pool long term temperature*
- *wetwell atmosphere long term pressure and temperature*

If the long term calculation results are different from those used for calculating NPSH, provide the suppression pool long term temperature and wetwell atmosphere long term pressure and temperature used for the NPSH calculation.

Response
Drywell short term pressure and temperature are provided in Figures 2 and 3. Suppression pool short term temperature is provided in Figure 3. Wetwell atmosphere short term pressure and temperature are provided in Figures 2 and 3. Suppression pool long term temperature is provided in Figures 5 and 9. Wetwell atmosphere long term pressure and temperature are provided in Figures 4,5,8, and 9. Notes for the figures are provided in Table 1.

Figures 6,7,10, and 11 provide the curves showing the wetwell atmosphere pressure and temperature and the suppression pool temperature used for the net positive suction head (NPSH) calculation. As noted in Table 1, appropriate assumptions were used to minimize the containment pressure available and maximize the required NPSH.

Question
7. For Quad Cities, provide additional detail of the confirmatory calculations validating the SHEX computer code (ELTR1 SER Section 2.6(a)).

Response
Case E of the QCNPS UFSAR Table 6.2-3 was selected as a benchmark case to validate the SHEX computer code. This benchmark case was analyzed with SHEX, using input assumptions, which best represent the case. Case E assumed an instantaneous double-ended break of a reactor recirculation suction line (DBA-LOCA). It was assumed that one RHR loop equipped with one RHR pump and one service water pump is available. The input assumptions used in the benchmark analysis were not necessarily the same as those used for the EPU analysis. For instance, feedwater addition, which would result in higher peak pool temperature, was included in the EPU analysis, whereas the benchmark analysis ignored its effect. The following table provides key input assumptions used in the benchmark analysis.

Parameter	Value	Remarks
Decay heat	May-Witt	In the Long Term Program (LTP) for Mark I containment, the May-Witt decay heat values were used. It was assumed that the same decay heat values had been used around 1969 in the absence of information on the decay heat values used in the UFSAR analysis.
Feedwater addition	None	It is believed that feedwater addition was ignored in the original UFSAR analysis, since there is no mention about that in the UFSAR. Feedwater addition would result in a higher peak pool temperature.
Initial pool temperature	90°F	Based on plots.
RHR heat	276.1	The K-value is defined as total heat removal

Parameter	Value	Remarks
exchanger K-value	Btu/sec-°F	rate (Btu/sec) divided by the heat exchanger inlet temperature difference. The K-value was obtained from process diagram.
Service water temperature	95°F	Obtained from process diagram. For typical analyses, the service water temperature is assumed to be the same as or lower than pool temperature.

Thus, the benchmark case (Case E of UFSAR Table 6.2-3) was analyzed with SHEX, and the peak pool temperature from the SHEX run was 181°F, which is 4°F higher than the 177°F reported in the UFSAR. This benchmark calculation, though based on limited information on the UFSAR analysis assumptions, shows that the SHEX prediction is representative, compared with the UFSAR analysis.

Question

8. Dresden proposed Technical Specification bases section B 3.6.1.4 is changed to reflect a reduced calculated peak drywell pressure of 43.9 psig for the limiting event. Additionally, the listed reference is changed to Updated Final Safety Analysis Report (UFSAR) Section 6.2.1.3, which was not provided in the application. Provide the referenced UFSAR Section or a draft of the section if it has not been revised for the EPU uprate.

Response

The proposed UFSAR Section 6.2.1.3, "Design Evaluation," is currently in draft form and is provided as Attachment B for both DNPS and QCNPS, for information only.

Question

12. Section 4.7 on post-LOCA combustible gas control notes margin changes in various parameters associated with the EPU and additional impact of GE14 fuel introduction on metal-water hydrogen production. Provide long term results (curves or tables of calculated values as a function of time) of calculations for

- *hydrogen and oxygen production*
- *hydrogen and oxygen concentrations*
- *nitrogen containment atmosphere dilution system nitrogen cumulative usage and capacity*
- *containment pressure buildup demonstrating meeting the 30-day acceptance limit.*

Response

Hydrogen production is provided in Figure 12. Oxygen production is not specifically presented, but equals one half of the hydrogen production.

Hydrogen and oxygen concentrations in primary containment without the use of the nitrogen containment atmosphere dilution (NCAD) system are provided in Figure 13.

Hydrogen and oxygen concentrations with the use of the NCAD system are provided in Figure 14.

NCAD system nitrogen cumulative usage is provided in Figure 15. The NCAD nitrogen storage system has a minimum volume of 200,000 scf as discussed in PUSAR Section 4.7, "Post-LOCA Combustible Gas Control."

Containment pressure buildup demonstrating meeting the 30-day acceptance limit of 50% of design pressure is provided in Figure 16.

Question

15. What effect, if any, does the EPU have on the service water system heat loads for the HPCI and LPCI room coolers?

Response

DNPS ECCS room cooler equipment consists of coolers for the LPCI/CS rooms and the HPCI room. QCNPS ECCS room cooler equipment consists of coolers for the RHR rooms and the HPCI room.

Under normal operating conditions, EPU has no effect on HPCI and LPCI room coolers.

After a design basis LOCA, the EPU suppression pool temperature will be higher than the pre-EPU pool temperature. Thus, EPU affects the RHR and LPCI/CS pump rooms since the pumps and the heat exchangers in these rooms process the higher temperature water from the suppression pool during emergency operation. This will increase the piping and heat exchanger heat loads to the rooms. The electrical heat load in ECCS rooms is not affected by EPU.

The QCNPS RHR corner room service water heat load will increase from 319,798 BTU/hr to 335,800 BTU/hr due to the higher EPU suppression pool temperature. The RHR corner room cooler capacity is 570,000 BTU/hr, which is greater than the EPU heat load. Therefore, the design LOCA room temperature of 150°F is not affected by EPU.

The DNPS LPCI/CS room service water heat load also increases due to EPU. However, no credit is given to the DNPS LPCI/CS corner room coolers for removal of the heat load. Under EPU post accident conditions, the peak room temperature is conservatively calculated to be 189° F without the use of the room coolers. The safety related components in these rooms have either been environmentally qualified to the higher temperatures or are being replaced with instruments that are environmentally qualified to the higher EPU post accident temperatures.

HPCI operation involving the suppression pool is not changed by EPU. Following a design basis LOCA, the reactor quickly depressurizes below the limit for HPCI operation. Therefore, HPCI is not credited and its components are not required to be environmentally qualified to the higher room temperatures resulting from a design basis LOCA.

HPCI room coolers are utilized to maintain the HPCI rooms as a mild environment during testing and for operation of the HPCI system during a small break LOCA. The maximum HPCI process temperature is unchanged for EPU, and is accommodated by the existing room coolers.

Question

17. Section 6.4.5 addresses the adequacy of the ultimate heat sink (UHS). In the event of downstream dam losses, the water trapped in the intake and discharge bay becomes the UHS for Quad Cities 1&2 and the water trapped in the intake canal becomes the UHS for Dresden 2&3. Considering the increased decay heat associated with the EPU, provide details of the analyses of the available water supply trapped in these UHSs for safe shutdown for all units; addressing conformance with Regulatory Guide 1.27. Include any revised timing of required operator actions to maintain the UHS; if any.

Response

The design basis for the DNPS and QCNPS Cities UHS was established prior to the issuance of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants." The design bases for the DNPS and QCNPS UHS are provided in UFSAR Section 9.2.5, "Ultimate Heat Sink," for each plant.

The capability of the UHS for operation at EPU conditions was evaluated within the context of the UHS design bases as stated above. The results are provided as follows.

QCNPS UHS Evaluation

At EPU conditions, with the use of the main condenser for 24 hours after shutdown, and the use of three portable pumps delivering 5100 gpm to the Residual Heat Exchanger Service Water (RHRSW) intake, the water in the suppression pool remains below the acceptance value of 177° F. The temperatures reached are 156° F prior to EPU and 166° F for EPU. The maximum cribhouse intake temperature remains below the acceptance value of 109° F. The temperatures reached are 106.5° F prior to EPU and 108° F for EPU for operation of one RHR pump and one RHRSW pump per unit.

Manual actions for placing and operating the portable diesel pumps in the event of a postulated failure of Lock and Dam No. 14 do not change as a result of EPU. The time available to position and operate the portable pumps to provide the makeup water from the river to the UHS is dependent only on the time to reach separation between the UHS and river (i.e., approximately two days) and is not affected by EPU operation.

DNPS UHS Evaluation

The response for the DNPS UHS evaluation will be provided separately.

Question

18. Section 7.1 Considering reactor power may now be limited by main generator capability, discuss implications of potentially load cycling the reactor due to environmental changes – such as diurnal heating and cooling effects changing cycle efficiency. Will this mode result in additional radioactive wastes being generated?

Response
The main generator will be limited to 912 megawatts-electric (MWe). Because of this limitation, and the change in plant efficiency over the course of an operating cycle, the thermal power of the reactor will generally be less than the rated thermal power of 2957 megawatts-thermal (MWth). It is expected that to maintain 912 Mwe during the coldest winter days, the reactor thermal power would be on the order of 2850 Mwth (i.e., approximately 96% of EPU RTP), while on the warmest summer days the reactor power would be expected to be near 2957 MWth (100% of EPU RTP). This 4% yearly variation in reactor power is easily achieved via a combination of changes in the operating control rod pattern and reactor core flow. These changes are very small over the time interval in which they occur. On a daily basis these changes due to plant efficiency parameters do not approach the magnitude of reactor power changes required for surveillance testing and rod pattern adjustments.

Radioactive waste generated is primarily affected by an increase in conductivity and increase in the amount of feedwater flow as a result of operating at higher power level. The effect on the generation of radioactive wastes due to load following is negligible. The conductivity and feedwater flow and the radioactive waste generated will not increase beyond that determined for the operation of the reactors at the maximum EPU power level for an entire operating cycle.

Question
19. PUSAR Section 4.1.1.1.(b), Local Pool Temperature with RV plus SRV Discharge, notes that because these plants have quenchers no evaluation nor limit is necessary as long as steam ingestion into the ECCS suction is not a concern. The NRC approved elimination of the local temperature limit provided quenchers were at an elevation above the ECCS suction. Since Dresden and Quad Cities have quenchers and suction strainers located in the same bays; an evaluation of the behavior of the steam plumes from the quenchers, relative to the entrainment flow path to the ECCS strainers was performed. Provide the details of this evaluation demonstrating that steam ingestion is not a concern. Include a description of the units' ECCS suction elevation relative to the suction strainers.

Response
The NRC previously approved elimination of the local pool temperature limit at DNPS as noted in Reference 4. As part of EPU, it was decided to include elimination of the local pool temperature limit for QCNPS as well.

An evaluation of the likelihood of steam ingestion into the ECCS suction strainers during safety relief valve (SRV) actuation was performed for DNPS and QCNPS. The evaluation was performed at EPU conditions for the most limiting geometry from the two plant designs, which was a case where the suction strainer and t-quencher are in the same torus bay with the least physical separation. The evaluation used the conservative assumption that the suppression pool is locally saturated in the region around the SRV quenchers and ECCS suction strainers. The evaluation also conservatively assumed operation of all ECCS pumps simultaneously with full SRV discharge flow.

The conservative premise was that steam ingestion would be predicted if the quencher steam plume intersects any part of the ECCS suction strainer or the entrainment envelope surrounding the suction strainer. The size of the steam plume generated from an SRV quencher and the envelope of flow drawn into the suction strainers were quantified and evaluated for overlap, which could result in steam ingestion. The results of these evaluations show that the steam plume from the SRV quencher located closest to a suction strainer will not intersect either the suction strainer or the envelope of flow (i.e., the entrainment envelope) drawn into the strainer. Therefore, steam ingestion is not predicted.

Since steam ingestion is not predicted at EPU conditions for the most limiting geometry, it is concluded that steam ingestion will not occur at DNPS or QCNES.

The ECCS pump suctions are located approximately six feet below the torus penetration connecting the ECCS ring header to the suction strainers (centerline to centerline). The strainer itself extends approximately five feet vertically into the torus from the penetration.

Question
20. Section 7.1 Provide the results of the evaluation of low pressure turbine missile analyses. Did these reanalyses confirm the potential need to change turbine overspeed protection settings?

Response
A missile analysis was previously performed for the DNPS and QCNPS turbines in 1986. Based upon review of this missile analysis, the predominant stresses that could cause a LP rotor failure were attributed to the centrifugal loading with only a small thermal stress contribution. Since the geometry of the LP rotors and blading is not changing as a result of EPU, the centrifugal stresses also do not change, and the existing analysis remains valid.

The overspeed to limit of 120% of rated speed is the limit used during original design and is not changing for EPU. Because EPU increases steam flow, turbine overspeed protection settings were reviewed by the original equipment manufacturer (OEM). As a result of this review, the current trip settings will be reduced, as applicable, to preclude rotor train speeds in excess of 120% of rated speed in the unlikely event of a simultaneous full load rejection and failure of both control and intermediate valves. A change to the backup overspeed trip (BUOT) setpoint in accordance with the OEM's recommendation is required at QCNPS Units 1 and 2. The current DNPS BUOT setpoint is within the range of the OEM recommendations and does not require revision.

Question
21. Section 7.1 notes that for the turbine-generator; valves, control systems and other support systems were evaluated for the effects of EPU. The results of the evaluation show that modifications to the high pressure turbine and some non-safety-related equipment should ensure satisfactory turbine-generator performance. Describe these modifications.

Response

Station	Modification Description
DNPS Units 2 and 3 QCNPS Units 1 and 2	Cross around relief valve (CARV) setpoint change – provide new relief pressure settings for all CARVs
DNPS Units 2 and 3 QCNPS Units 1 and 2	Replacement of HP turbine rotors and diaphragms – new HP boreless rotor and HP nozzle diaphragms for increased volumetric flows
DNPS Units 2 and 3 QCNPS Units 1 and 2	Stator water cooling alarm and runback setpoint changes - adjusted for revised flow conditions
DNPS Units 2 and 3 QCNPS Units 1 and 2	Electrohydraulic control / turbine supervisory instrumentation changes • Steam line resonance compensator – addition of 3rd harmonic filter • Diode function generator calibration – adjustment of control valve characteristic with increased steam flow • Turbine 1st stage pressure transmitter change – adjustment for new rated condition • Power load unbalance input span changes – adjustment of turbine intermediate pressure and generator current for new rated conditions • Differential expansion detector – change in detector calibration and alarm
DNPS Unit 2 and 3	Stator water cooling service water restriction orifices – increase heat removal capacity

Question

22. Section 8.2.1 addresses the impact of the EPU on the condenser off-gas system; noting an increase of (radiolytic) hydrogen flow from 26.3 to 30.9 lb_m per hour under hydrogen water chemistry conditions. Additionally, the radioactive releases to be handled (held-up) by the off-gas system are estimated to increase proportionately to the power increase of 17%. Address how the combination of these proposed changes impact the design hold up times for the off-gas system; including the ability of the system to hold up a minimum of 30 minutes under conditions associated with 100 μCi/sec/Mwt release rates for noble gases; and (2) the operational impacts associated with the increase radiation shine effects caused by the increased feedwater hydrogen injection rates/main steam flow rates. As noted in Section 8.4.1.1, the impacts of hydrogen water chemistry on source terms are considered without credit for use of the effects of the NobleChem process, which considerably lowers the hydrogen feedwater injection requirements. Alternately, state if the use of NobleChem process to limit these effects is considered as part of the EPU basis.

Response

Holdup time in the offgas system, both in the delay line downstream of the recombiner and on the charcoal adsorbers, is affected only by main condenser air inleakage, and not by radiolytic hydrogen flow, which is removed in the offgas recombiner at the entrance to the system. Because EPU does not affect main condenser air inleakage, offgas system holdup times for noble gases are not affected by EPU. Therefore, an estimated increase in noble gas source term of 17% will result in a like increase in system release rate during periods of operation with significant fuel cladding leaks. However, because the current and expected fuel defect rate is extremely small, the actual offgas release rate is not expected to increase. Further, the maximum allowed release rate in the TS is not being changed for EPU.

As discussed in Reference 5, the calculated offsite dose due to turbine shine will increase in proportion to the uprate (i.e., 17%). This includes the effect of the increased hydrogen injection rate. However, the actual increase in turbine shine is expected to be less than the calculated increase. For a given feedwater hydrogen concentration, the increase in N-16 over baseline normal water chemistry conditions (i.e., no feedwater hydrogen) does not change for EPU. Therefore, the estimated 17% increase in N-16 due to EPU is offset by the approximately 19% increase in steaming rate (see PUSAR Sections 8.4.1 and 8.6), regardless of whether the plant is operating with or without hydrogen water chemistry (HWC) or NobleChem™. Under NobleChem™ operation, reduced feedwater hydrogen injection rates will significantly reduce the N-16 multiples resulting from HWC operation. Thus, for normal water chemistry operation and for HWC operation, offsite dose from turbine shine under EPU conditions will remain the same as current levels or will increase a small amount due to the small decrease in delay time in the steam lines from the increased steaming rate. For NobleChem™ application, turbine shine will decrease relative to levels prior to NobleChem™ application. Dose effects as presented in the EPU basis do not take credit for reduction due to NobleChem™, and therefore are bounding.

Question

23. Section 8.4.3 Clarify the statement in section 8.4.3 that the EPU does not change the design noble gas release rate from the fuel, specifically with respect to SRP 11.3 which provides guidance that the source term for noble gasses is a linear function of the power level and with respect to the stated original design bases of 0.2 Ci/sec after a thirty minute delay. Does the 0.2 Ci/sec original design basis bound the effect of a linear increase in power on the instantaneous off-gas limit noted in SRP 11.3?

Response

For the DNPS and QCNPS plants, the design basis is 0.2 Ci/s referenced to a 30 minute decay time. This design value was based upon past fuel performance at the time of original design (i.e., 1970 to 1972) and provided a margin of approximately a factor of two over the range in which these plants were expected to operate. Since that time period, improvements in fuel and operations have continually reduced the expected operating offgas values to small fractions of the original design basis. The expected offgas releases based upon current plant performance were estimated based on ANS Standard ANS/ANSI 18.1-1999 "Radiological Source for Normal Operation for Light Water Reactor," for the EPU condition. The results of this analysis show the offgas rate as evaluated by that standard to be a fraction of the original design basis. Therefore, the 0.2 Ci/sec design basis bounds the effect of the increase in power on the off-gas release rate.

Question

24. Section 8.4.3 Explain the stated expectation of no increase in fission product releases from the fuel as a result of EPU. Why won't the expected release rate increase in proportion to the reactor power level increase of 17%?

Response

To correct the statements in PUSAR Section 8.4.3, it is expected that some increase in fission product activity in reactor coolant will be seen. Using the formula in ANSI/ANS 18.1-1999, "Radiological Source Term for Normal Operation for Light Water Reactors," the increase would result in a calculated 12% increase in concentration. Even with this increase, the reactor coolant activity levels will be fractional parts of the design basis coolant concentrations.

Question

25. Section 10.1.1.1 addresses the main steam high energy line break and notes that the critical parameter affecting the HELB analyses is reactor dome pressure which is not being changed by the EPU. Do any of the HELB analyses credit isolation of the main steam lines to limit mass-energy released? If so, address the effects.

Response

The bounding main steam line break, a circumferential rupture that results in choked flow through the flow restrictor, credits isolation to limit the mass release. Since the steam pressure does not change due to EPU, the mass release from the limiting break is also unchanged. The DNPS and QCNPS UFSARs define the specific design basis break locations analyzed for HELB. All such postulated breaks in the main steam lines are located in the pipe tunnel. These breaks are isolated by any of three signals: the high steam line flow isolation, low steam line pressure isolation (in RUN mode), or high steam tunnel temperature isolation. Operability of all of these isolation signals are governed by the DNPS and QCNPS TS.

Question

26. Section 10.1.1.2 notes that for the EPU, the feedwater system line break results in a 6% increase in feedwater mass and energy release. The safety analysis further notes that design margins within the high energy line break analyses are conservative and remain bounding. Provide details of the main steam tunnel HELB analysis that addresses these margins, including major assumptions and results.

Response

The feedwater line break was used with a concurrent main steam line break to establish the peak pressure and the long term temperature environment in the main steam tunnel for DNPS and QCNPS.

(proprietary information removed)

The licensing design basis pressure for the main steam tunnel for DNPS and QCNPS is 27.5 psia. The main steam tunnel peak calculated pressure is 27.2 psia at 0.1 seconds.

(proprietary information reemoved)

Therefore, it can be concluded that the pre-EPU design basis main steam tunnel environmental parameters bound the EPU values for the feedwater HELB concurrent with a main steam line break.

Question

27. Section 10.2 notes that moderate energy line break protection features are based on system parameters unchanged by the EPU. Are portions of the condensate and feedwater system considered within the scope of this analyses? If so, has the additional flow associated with operation of four condensate pumps been evaluated? Are any changes in flow or system operation being proposed for the condenser circulating water system to accommodate increased heat load of EPU, or will the EPU otherwise impact the potential for flooding from a line break in this system?

Response

The condensate and feedwater systems are considered high energy systems. Postulated flooding from these systems is not within the scope of the moderate energy line break analysis, but was covered under flooding from high energy breaks.

Safety related equipment in the turbine building that could be subjected to the effects of flooding are the containment cooling service water (CCSW) pumps at DNPS and the RHR service water pumps at QCNPS. At DNPS, two of these pumps are located in watertight vaults. At QCNPS, all of the RHR service water pumps are located in watertight vaults. These passive protection features are designed to protect against the limiting event of a circulating water system rupture, which could flood the building to the level of the river. This flooding event bounds the consequences of a postulated condensate system rupture, regardless of the number of condensate pumps operating.

Feedwater system ruptures could affect safety-related equipment in the main steam tunnel due to flooding. The pre-EPU analysis of such ruptures assumed the tunnel would flood completely. This analysis therefore bounds the EPU case.

The circulating water system can accommodate the EPU heat load at the current system flow rate. The existing protective features for a circulating water system rupture include a trip of the pumps on high level in the condenser pit area. The ultimate consequences of such a rupture are due to subsequent gravity feed resulting in the flood water level reaching river level. Existing flood protection features are not affected by EPU.

<u>Question</u>
28. Section 11.3 notes that the quantity of spent fuel will not be affected by the uprate; although the short-term radioactivity will be higher but within limits. Please clarify this statement. Is there not an expectation that additional spent fuel assemblies will be required to support the 17% power increase; or is the entire power uprate accommodated in increased burn-up of fuel assemblies?

<u>Response</u>
This statement is incorrect. The statement on this issue in the Environmental Report (Attachment D to Reference 1), Section 3.3, "Radiological Environmental Impacts," is correct. The quantity of spent fuel discharged at the end of each uprated cycle will be larger than that discharged from the pre-EPU cycles.

<u>Question</u>
29. Is the capacity of the hardened vent sufficient to accommodate the power uprate?

<u>Response</u>
The design basis for the containment hardened vent is to mitigate loss of decay heat removal sequences, and to prevent further pressurization with the containment at its pressure limit. The vent was reanalyzed for EPU conditions to ensure this basis was still met.

For DNPS, the hardened vent will have a capacity of 1% of RTP after EPU. The QCNPS hardened vent will have a capacity of 0.85% of RTP after EPU. The EPU decay heat curve reaches 1% at 11,000 seconds, or 3.1 hours, and reaches 0.85% at 20,000 seconds, or 5.6 hours. Under EPU conditions, the containment will not reach the pressure limit until 20 hours after a loss of decay heat removal. The DNPS and QCNPS hardened vents are thus capable of relieving EPU decay heat with ample margin to the time when venting is required. Therefore, the existing DNPS and QCNPS hardened vent capacity is sufficient to accommodate the power uprate.

Question
30. The environmental qualification of non-metallic components, (i.e. seals, gaskets, lubricants, diaphragms, etc.) has not been addressed. Please demonstrate that plant operations at the proposed EPU level will have no impact on the environmental qualification of mechanical equipment located both inside and outside containment.

Response
Operating and environmental conditions are included in procurement specifications for such material.

Changes in operating conditions as well as normal and accident environmental conditions have been determined for EPU. These changes, as indicated in PUSAR Tables 4-1, "DBA-LOCA Containment Performance Results," 10-1, "High Energy Line Break," and 10-2, "Environmental Changes for Equipment Qualification and Affected Equipment Types," are very minor relative to the range of conditions normally allowed for such materials. The most severe change in conditions is due to the post LOCA increase in the torus water temperature. Due to the potentially higher fluid operating temperature of the Core Spray and LPCI pumps at DNPS only, which use process water for bearing cooling, the bearing lube oil is being changed for EPU. No other changes in materials of this type were identified for operation at EPU conditions.

Figure 1
Re-Rated Conditions for Feedwater Heaters

Feedwater Heater	**Design Conditions Pre-EPU**	**Design Conditions Post-EPU**
LP Drain Cooler A-2 Design Pressure (PSIG) Design Temperature (°F)	 50 300	 50 300
LP Heater #A1 Design Pressure (PSIG) Design Temperature (°F)	 50 298	 50 298
LP Heater B Design Pressure (PSIG) Design Temperature (°F)	 50 350	 50 350
LP Heater C Design Pressure (PSIG) Design Temperature (°F)	 75 (DNPS), 83 (QCNPS) 350	 100 350
HP Heater D Design Pressure (PSIG) Design Temperature (°F)	 150 450	 178 450

Table 1
Remarks on Figures 2 - 11

Figure	Parameter	Remarks
2 & 3	Pressure and temperature response from short term DBA-LOCA analysis	Break flow rate and enthalpy are calculated with LAMB (with Moody's slip critical flow model), using the model representing both Dresden and Quad Cities, and these break flow values are used as input to the M3CPT code. The initial wetwell and suppression pool temperature was conservatively assumed to be 98°F, as compared with 95°F assumed in the long-term SHEX analysis. The initial drywell and wetwell pressure are assumed to be their maximum expected normal operating values. The peak drywell pressure for the current power, based on the current method, was predicted to be lower than the UFSAR value, but higher than the peak value obtained during the Long Term Program (LTP) for Mark I containment, as shown below. UFSAR / Current Analysis / LTP Dresden: 47 psig / 42.8 psig / 41.2 psig Quad Cities: 47 psig / 42.8 psig / 40.6 psig The difference between the original UFSAR analysis and the current analysis may be mainly due to differences in the blowdown flow rates, although the same critical flow model (Moody's slip model) was used for both analyses. Depending upon the vessel modeling that provides input conditions for the critical flow model, the blowdown values could be different. The UFSAR indicated that the DBA-LOCA blowdown values used in that analysis might be over-predicted, which would cause the over-prediction of peak drywell pressure. It is noted that the current method resulted in higher peak drywell pressure, compared to the LTP analysis that was reviewed and approved by the NRC. The LTP analysis used the vessel blowdown model built into the M3CPT code, as compared with the current analysis based on the LAMB blowdown model.
4 & 5	Dresden pressure and temperature response from long term DBA-LOCA analysis with direct pool cooling	The wetwell and suppression pool temperature was assumed to be 95°F. The initial drywell and wetwell pressure were assumed to be their maximum expected normal operating values. It was conservatively assumed that the suppression pool surface stays unperturbed. This assumption results in an unrealistically high wetwell temperature early in the event, because of compression effects, while allowing no mixing between the airspace and pool even during the blowdown phase. Only one RHR pump was assumed to be available to maximize the pool temperature response
6 & 7	Dresden pressure and	The wetwell and suppression pool temperature was assumed to be 95°F. The initial drywell

Figure	Parameters Plotted	Remarks
	temperature response from DBA-LOCA analysis for NPSH	and wetwell pressure were assumed to be their minimum expected values. Two cases, short-term and long-term, were analyzed using different assumptions regarding operation of LPCI/containment cooling pumps. The short-term case, which applies to the time period before 600 seconds, assumed that water from four LPCI pumps flows into the drywell through the break to minimize the pressure response. The long-term case (applying to the time period after 600 seconds) assumed that before 600 seconds two LPCI pumps are operating without dumping the water into the drywell, and at 600 seconds one of the two LPCI pumps is switched to containment spray, while turning off the other. This case will maximize the pool temperature response. For NPSH evaluations, the results for the short-term case are used for the time period before 600 seconds, and the long-term results are used for the time period after 600 seconds. Figures 5 and 6 show the combined results (short-term results before 600 seconds and long-term results after 600 seconds). Because of different assumptions between the two cases (before and after 600 seconds), Figures 5 and 6 exhibits sudden changes in the pressure and temperature response at 600 seconds. For instance, the wetwell pressure is approximately 20 psia (the short-term result) at 600 seconds as a result of low drywell pressure (due to high (4 pumps) LPCI flow into the drywell), which is followed by opening of wetwell-drywell vacuum breakers. Right after 600 seconds, the wetwell pressure is approximately 30 psia (the long-term results), because of the difference in the event scenario between the short-term and long-term cases.
8 & 9	Quad Cities pressure and temperature response for long term DBA-LOCA with direct pool cooling	Same assumptions as for the Dresden analysis, using Quad Cities RHR heat exchanger K-value of 262 Btu/sec-°F compared with 281.7 Btu/sec-°F for Dresden.
10 & 11	Quad Cities pressure and temperature response for DBA-LOCA for NPSH	Same assumptions as for the Dresden analysis, using Quad Cities RHR heat exchanger K-value of 262 Btu/sec-°F compared with 281.7 Btu/sec-°F for Dresden. As mentioned above for the Dresden analysis, two cases, short-term and long-term, were analyzed using different assumptions regarding operation of LPCI/RHR (containment cooling) pumps. The combined results (short-term results before 600 seconds and long-term results after 600 seconds) are plotted in Figures 9 and 10. Because of different event scenarios between the two cases, Figures 9 and 10 show sudden changes in the pressure and temperature response at 600 seconds.

Attachment D
Additional Plant Systems Information Supporting the License Amendment Request for Uprated Power Operation
Dresden Nuclear Power Station, Units 2 and 3
Quad Cities Nuclear Power Station, Units 1 and 2
(Non-Proprietary Version)

Figure 2
Dresden and Quad Cities Short-Term DBA-LOCA Containment Pressure Response

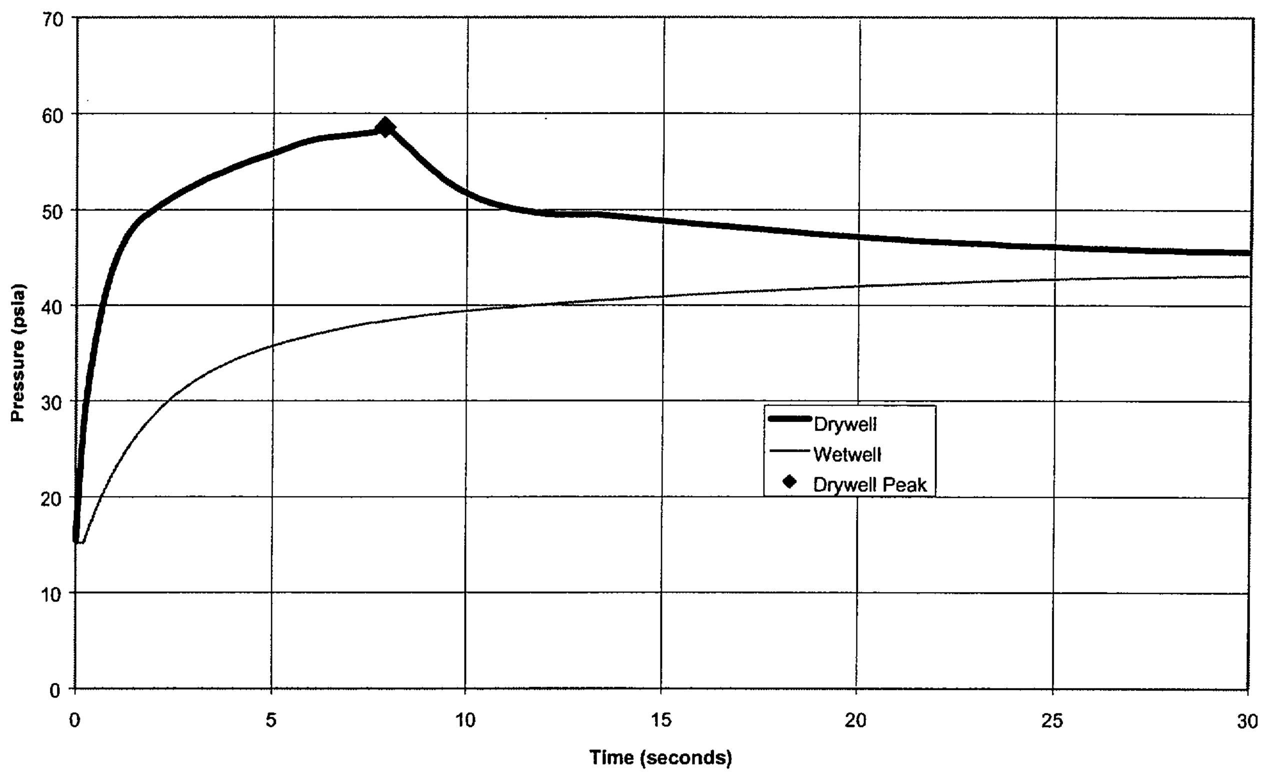


Figure 3
Dresden and Quad Cities Short-Term DBA-LOCA Containment Temperature Response

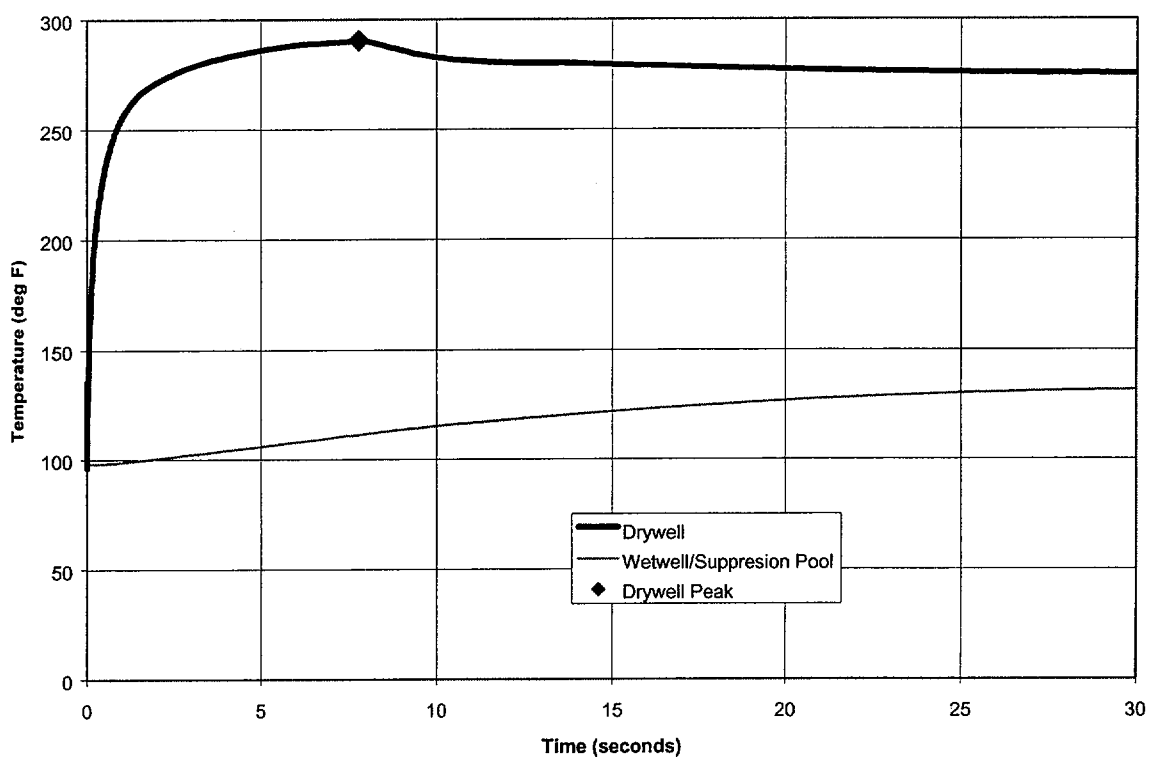
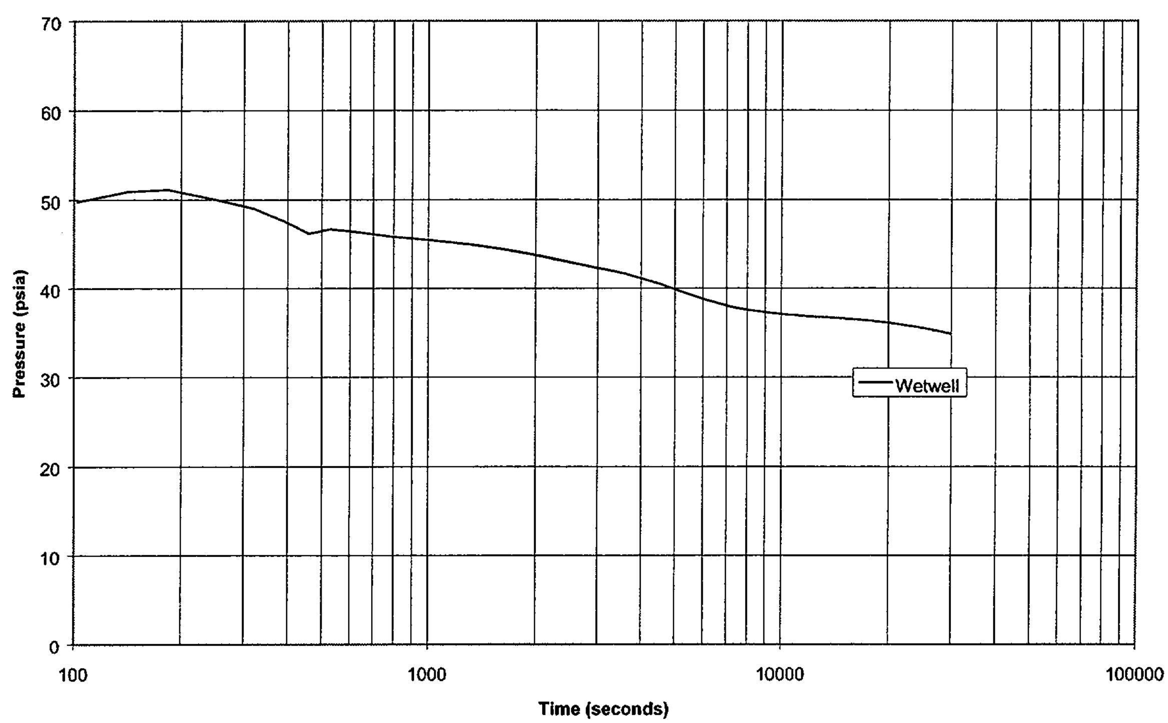
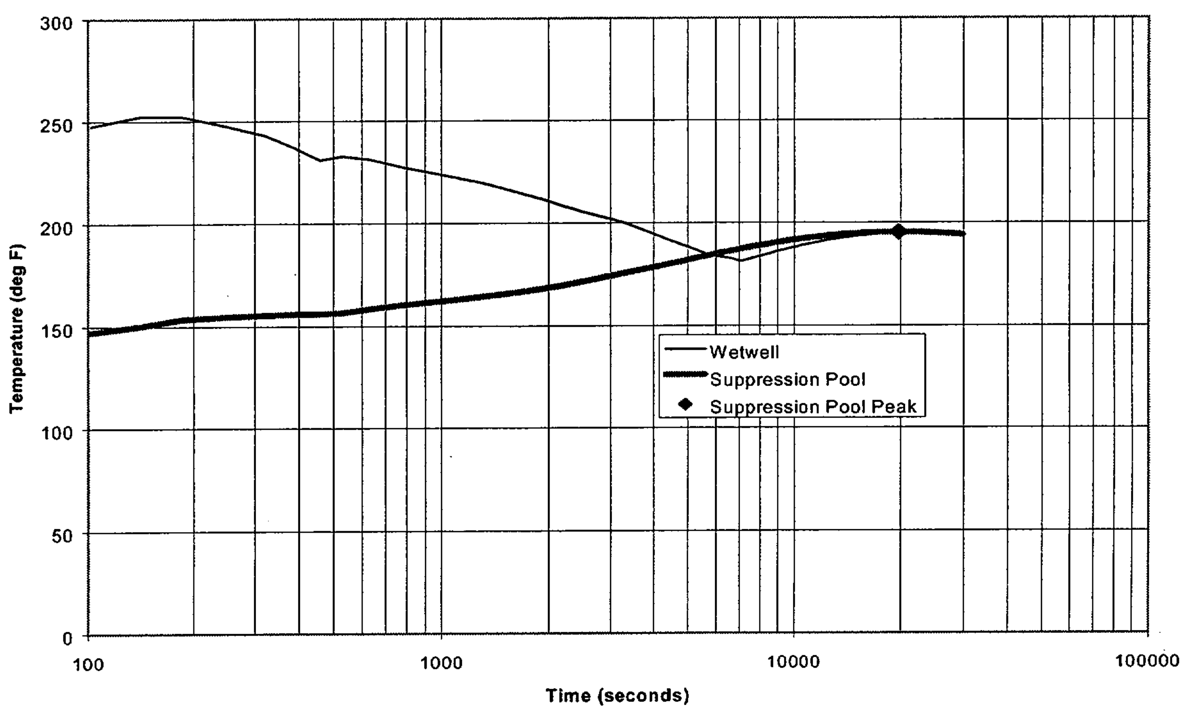


Figure 4
Dresden DBA-LOCA with Direct Pool Cooling
Containment Pressure Response

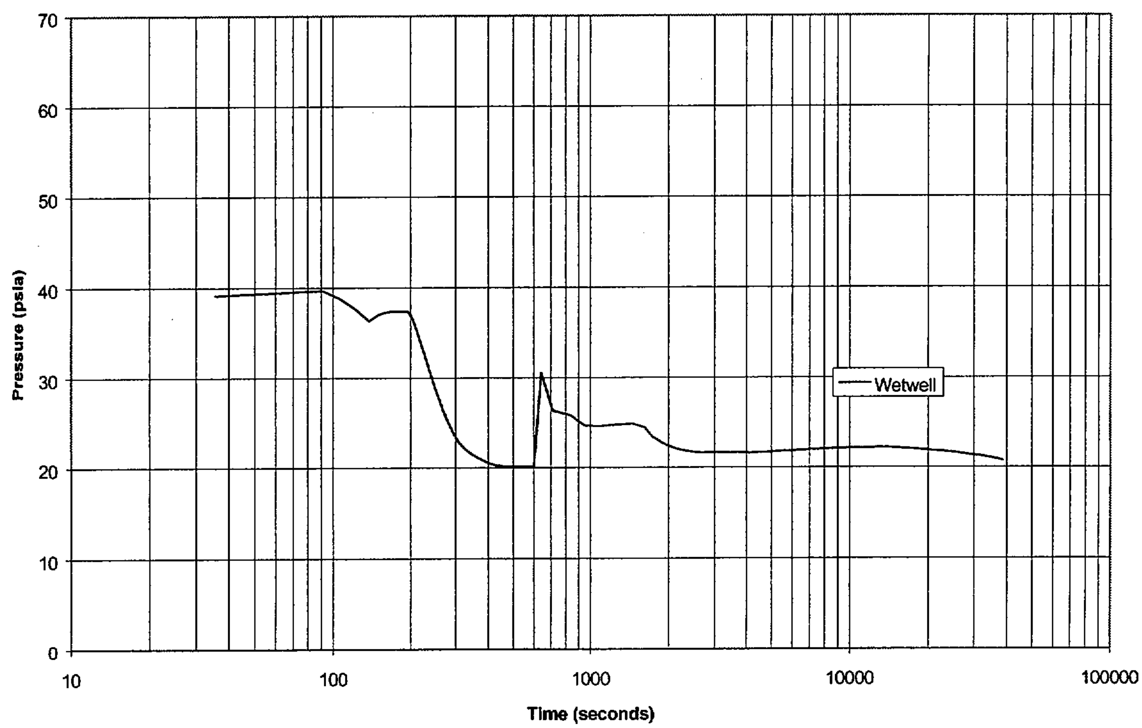


Attachment D
Additional Plant Systems Information Supporting the License Amendment Request for Uprated Power Operation
Dresden Nuclear Power Station, Units 2 and 3
Quad Cities Nuclear Power Station, Units 1 and 2
(Non-Proprietary Version)

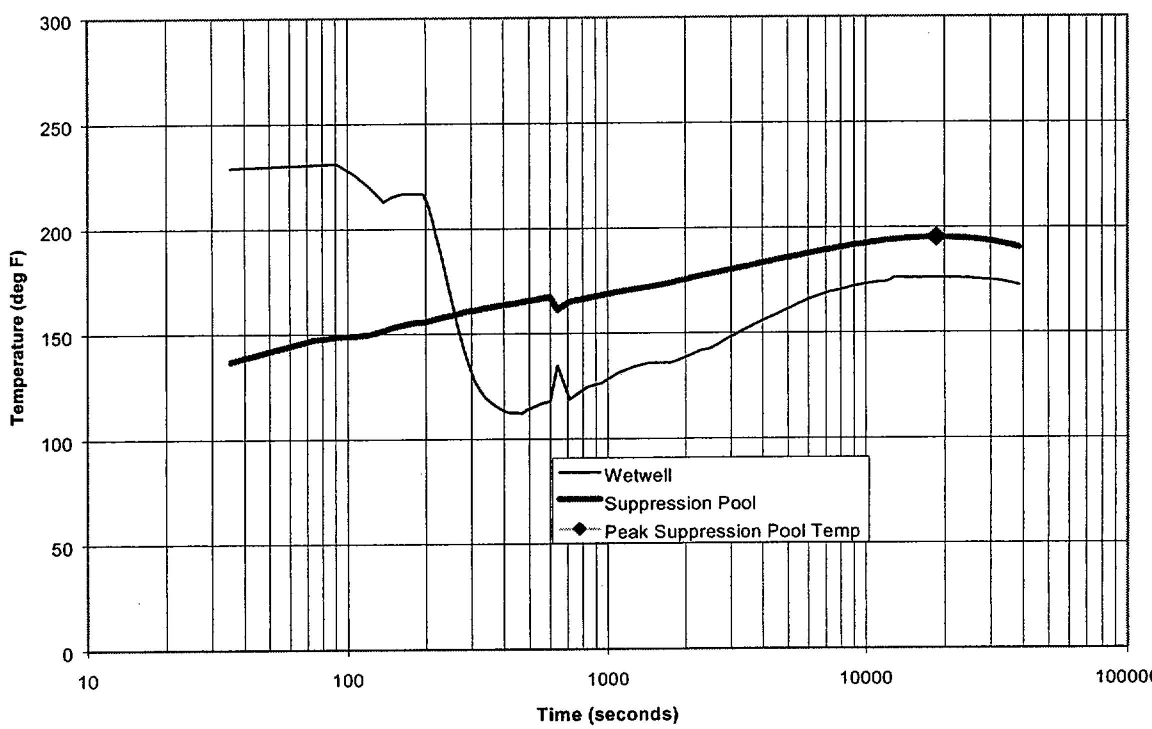
Figure 5
Dresden DBA-LOCA with Direct Pool Cooling
Containment Temperature Response



Attachment D
Additional Plant Systems Information Supporting the License Amendment Request for Uprated Power Operation
Dresden Nuclear Power Station, Units 2 and 3
Quad Cities Nuclear Power Station, Units 1 and 2
(Non-Proprietary Version)
Figure 6
Dresden DBA-LOCA for NPSH
Containment Pressure Response



Attachment D
Additional Plant Systems Information Supporting the License Amendment Request for Uprated Power Operation
Dresden Nuclear Power Station, Units 2 and 3
Quad Cities Nuclear Power Station, Units 1 and 2
(Non-Proprietary Version)
Figure 7
Dresden DBA-LOCA for NPSH
Containment Temperature Response



Attachment D
Additional Plant Systems Information Supporting the License Amendment Request for Uprated Power Operation
Dresden Nuclear Power Station, Units 2 and 3
Quad Cities Nuclear Power Station, Units 1 and 2
(Non-Proprietary Version)
Figure 8
Quad Cities DBA-LOCA with Direct Pool Cooling
Containment Pressure Response

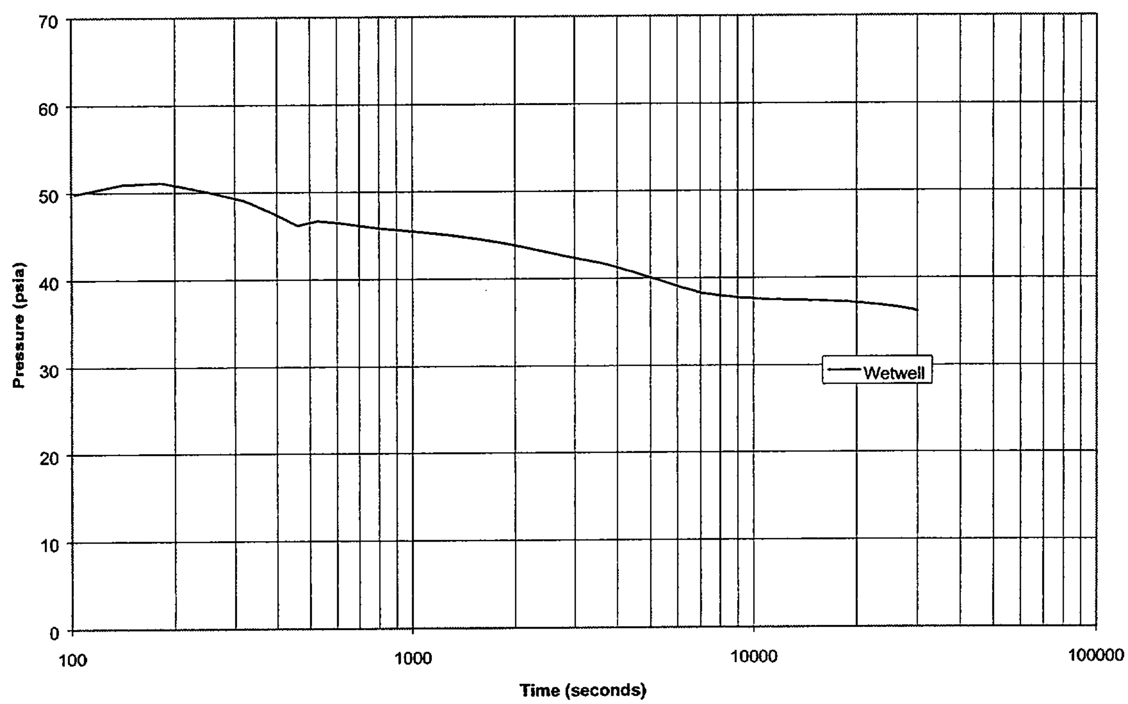


Figure 9
Quad Cities DBA-LOCA with Direct Pool Cooling
Containment Temperature Response

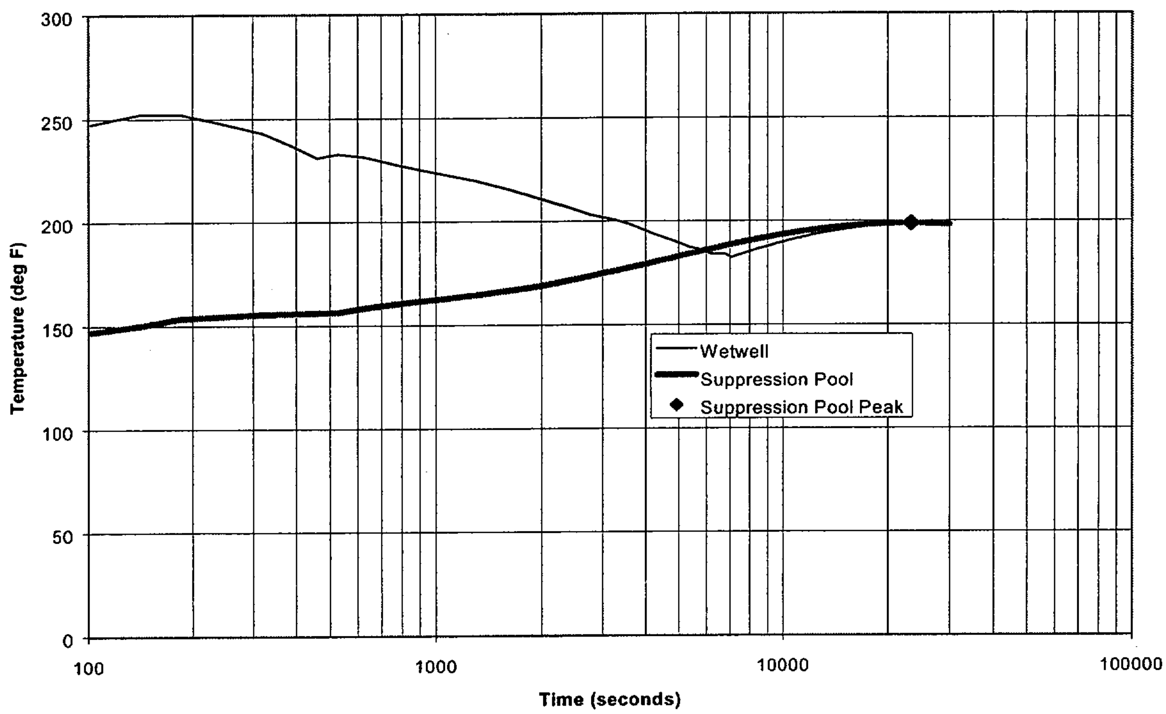
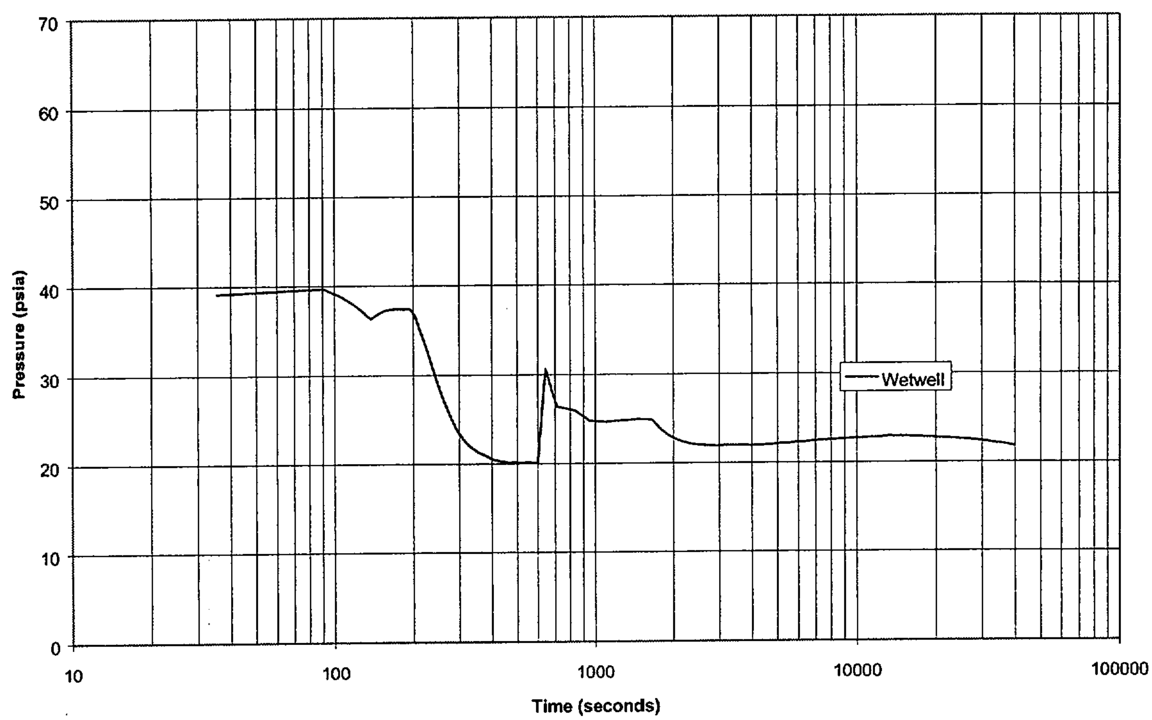


Figure 10
Quad Cities DBA-LOCA for NPSH
Containment Pressure Response



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Dresden Nuclear Power Station, Units 2 and 3
Quad Cities Nuclear Power Station, Units 1 and 2
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Figure 11
Quad Cities DBA-LOCA for NPSH
Containment Temperature Response

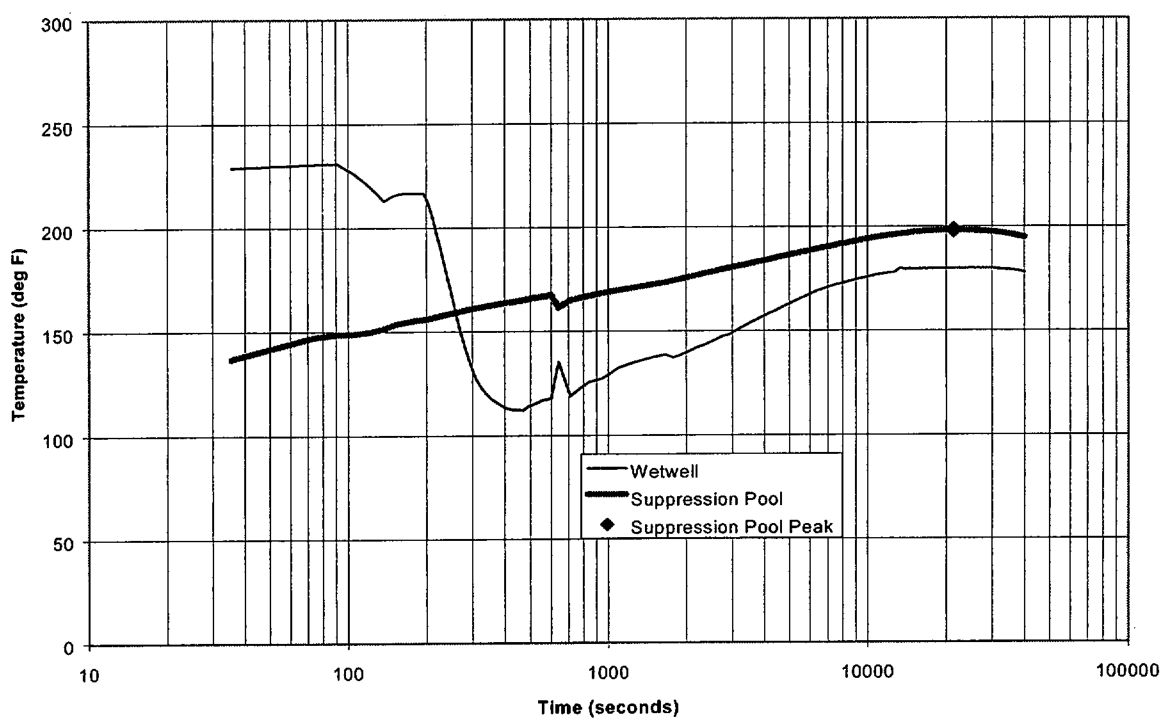
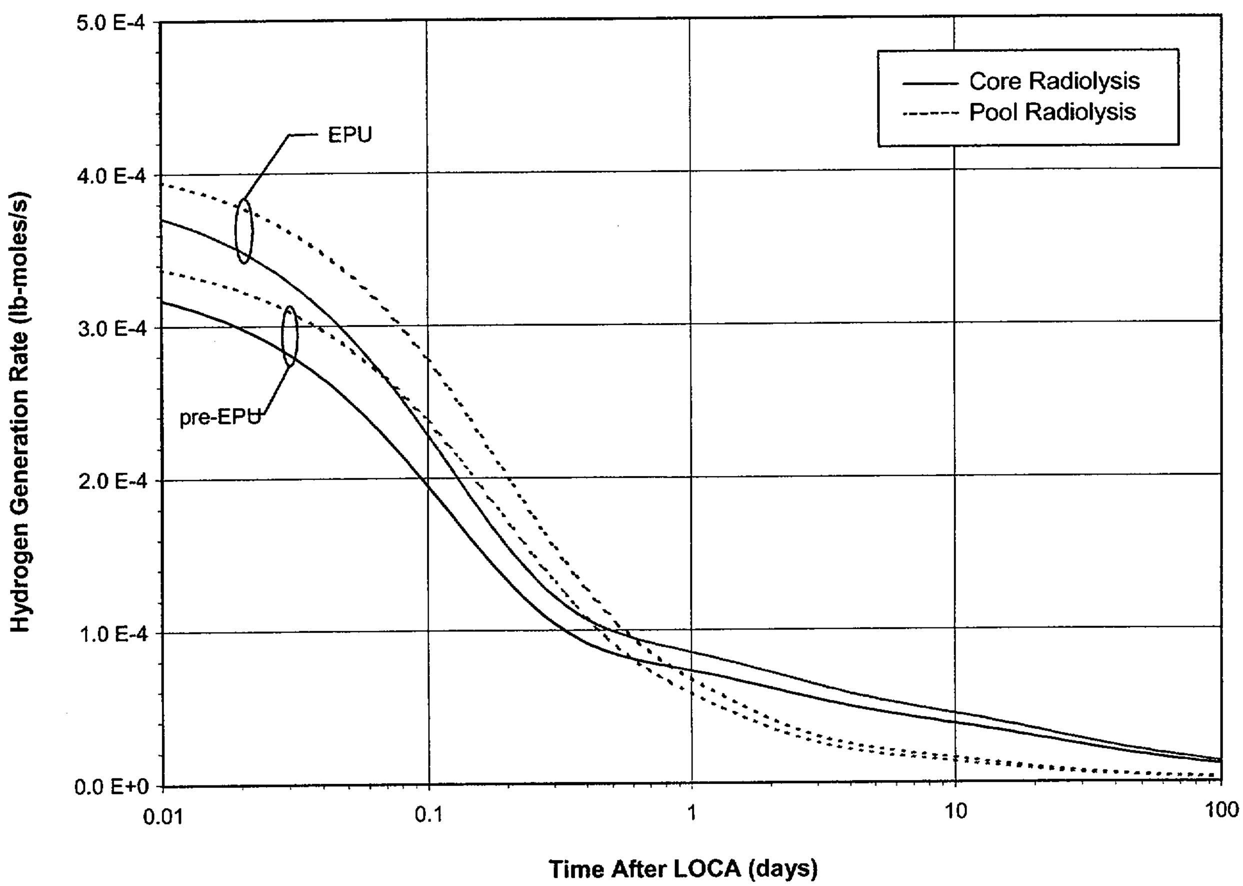


Figure 12
Hydrogen Generation Rate in Containment Following Loss of Coolant Accident



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Additional Plant Systems Information Supporting the License Amendment Request for Uprated Power Operation
Dresden Nuclear Power Station, Units 2 and 3
Quad Cities Nuclear Power Station, Units 1 and 2
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Figure 13
Containment Hydrogen and Oxygen Concentrations Without Nitrogen Containment Atmosphere Dilution System

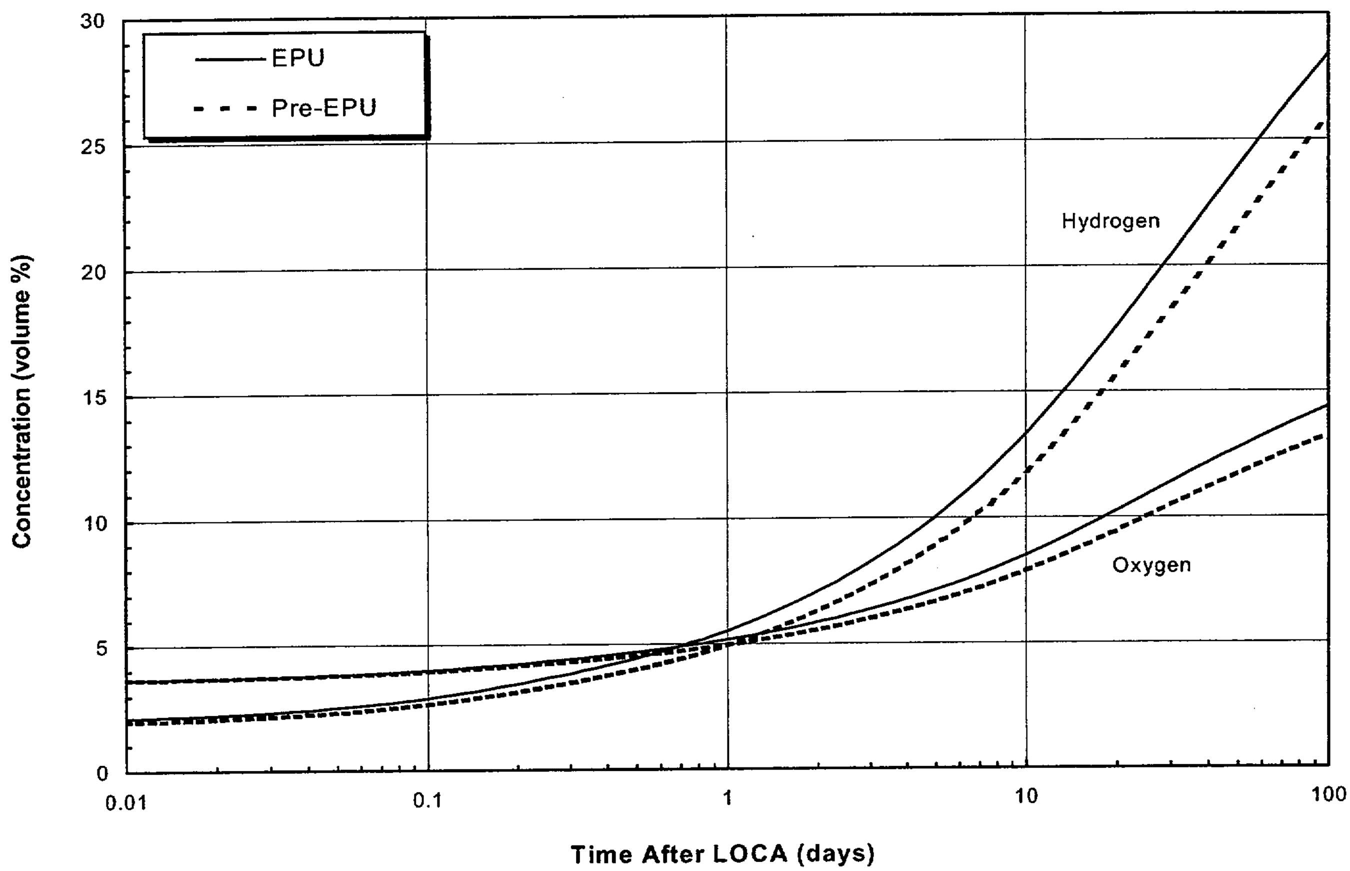


Figure 14
Containment Hydrogen and Oxygen Concentrations
With Nitrogen Containment Atmosphere Dilution System Operation

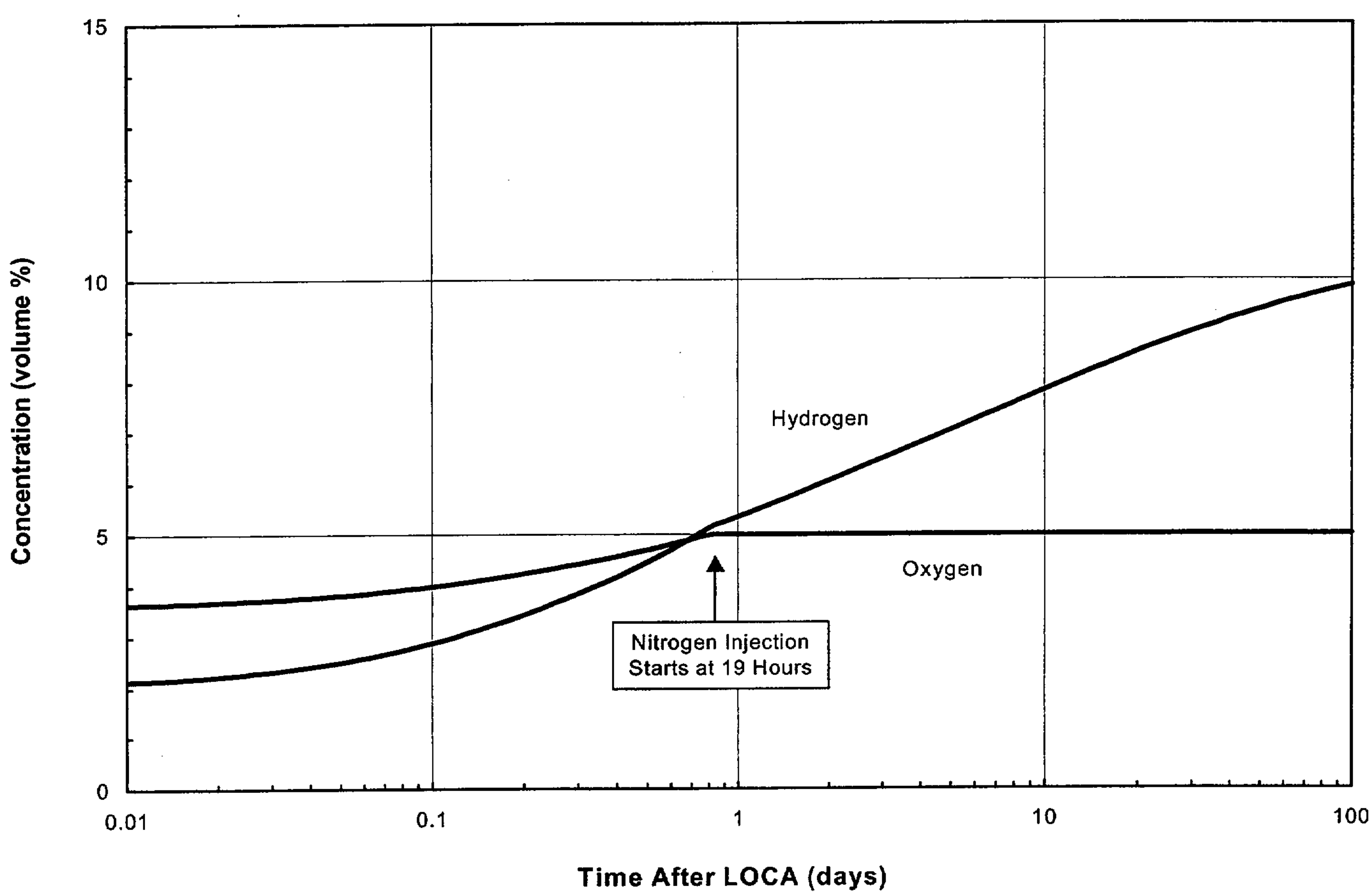


Figure 15
NCAD System Nitrogen Cumulative Usage

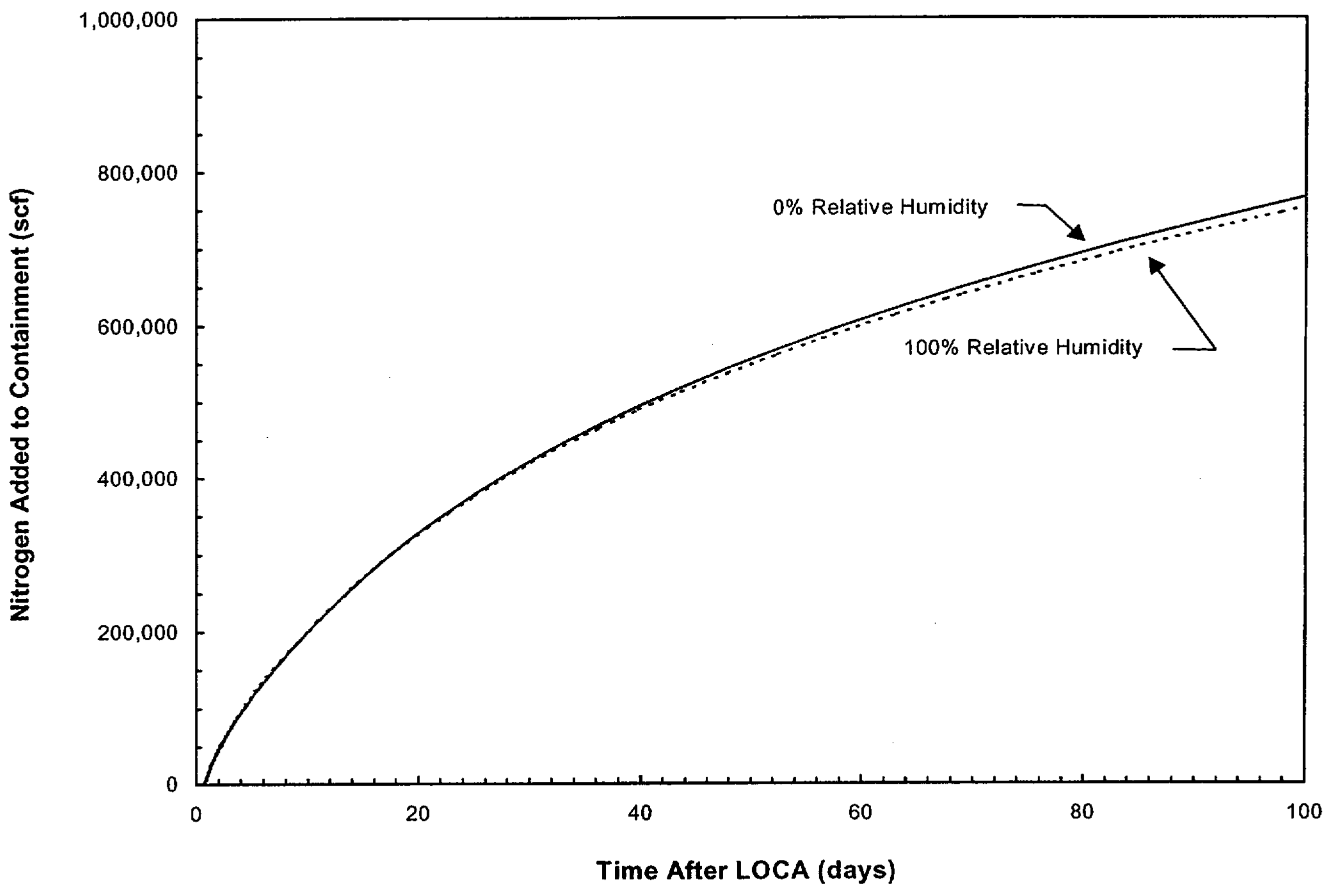
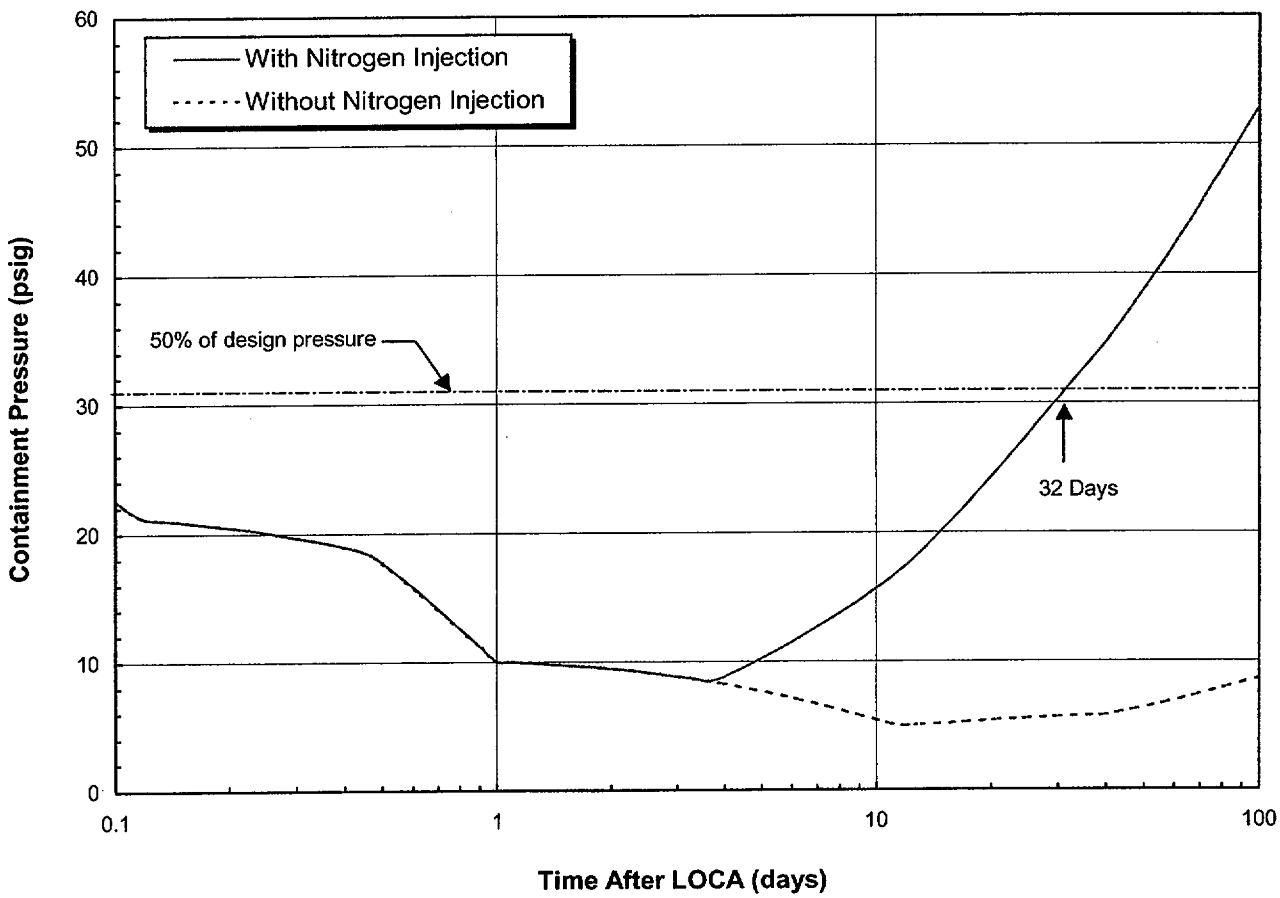


Figure 16
Containment Pressure Response to Nitrogen Containment Atmosphere Dilution System Operation



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Dresden Nuclear Power Station, Units 2 and 3
Quad Cities Nuclear Power Station, Units 1 and 2
(Non-Proprietary Version)

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

3. Letter from U. S. NRC to G.L. Sozzi (General Electric), "Staff Position Concerning General Electric Boiling-Water Reactor Extended Power Uprate Program," dated February 8, 1996

4. Letter from U. S. NRC to I. Johnson (Commonwealth Edison Company), "Issuance of Amendments," dated April 30, 1997

5. Letter from R. M. Krich (Exelon Generation Company, LLC) to U. S. NRC, "Additional Health Physics Information Supporting the License Amendment Request to Permit Uprated Power Operation at Dresden Nuclear Power Station and Quad Cities Nuclear Power Station," dated May 29, 2000