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RS-01-162

August 13, 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 Mechanical Information Supporting the License Amendment Request to Permit Uprated Power Operation at Dresden Nuclear Power Station and Quad Cities Nuclear Power Station

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) Letter from K. A. Ainger (Exelon Generation Company, LLC) to U. S. NRC, "Additional Mechanical Information Supporting the License Amendment Request to Permit Uprated Power Operation at Dresden Nuclear Power Station and Quad Cities Nuclear Power Station," dated August 8, 2000

In Reference 1, 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 (DNPS), Units 2 and 3, and Quad Cities Nuclear Power Station (QCNPS), Units 1 and 2, to allow operation with an extended power uprate (EPU). In a July 23, 2001, teleconference between members of the NRC and representatives of EGC, the NRC requested additional information regarding these proposed changes. The first portion of this information was provided in Reference 2. Attachment A to this letter provides the remainder of the requested information.

A001

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 B 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 C 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.

Respectfully,

K. A. Ainger

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

Attachments:

Affidavit
Attachment A: Additional Mechanical Information Supporting the License Amendment Request to Permit Uprated Power Operation, Dresden Nuclear Power Station, Units 2 and 3, Quad Cities Nuclear Power Station, Units 1 and 2 (Proprietary version)
Attachment B: Affidavit for Withholding Portions of Attachment A from Public Disclosure
Attachment C: Additional Mechanical Information Supporting the License Amendment Request to Permit Uprated Power Operation, 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 Mechanical Information Supporting the License Amendment Request to Permit Uprated Power Operation, 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.

K. A. Ainger

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

Subscribed and sworn to before me, a Notary Public in and for the State above named, this 13th 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 Mechanical Information Supporting the License Amendment Request to Permit 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-01-466, *Mechanical RAIs*, (GE Proprietary Information), dated August 7, 2001. The proprietary information is delineated by bars marked in the margin adjacent to the specific material in the Attachment 1, *GE Response to NRC Mechanical 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 7th day of August 2001.

George B. Stramback
General Electric Company

Subscribed and sworn before me this 7th day of August 2001.

TERRY J. MORGAN
Commission # 1304914
Notary Public - California
Santa Clara County
My Comm. Expires May 18, 2005

Notary Public, State of California

Attachment C
Additional Mechanical Information Supporting the License Amendment Request to Permit Uprated Power Operation (Non-Proprietary)
Dresden Nuclear Power Station, Units 2 and 3
Quad Cities Nuclear Power Station, Units 1 and 2

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

Attachment C
Additional Mechanical Information Supporting the License Amendment Request to Permit Uprated Power Operation (Non-Proprietary)
Dresden Nuclear Power Station, Units 2 and 3
Quad Cities Nuclear Power Station, Units 1 and 2

This attachment contains responses to NRC Questions 1, 2, 3, 7, 10, 11D, and 12D. Responses to NRC Questions 4, 5, 6, 8, 9, 11, (Parts A, B, C, and E), 12 (Parts A, B, and C), 13, and 14 were provided in a previous transmittal (Reference 1).

Question

2. In reference to Section 3.3.4 for the reactor internal structural evaluation, you stated that the structural assessment used guidelines and procedures similar to those in the design basis analyses. All applicable service levels, namely normal, upset, emergency, and faulted are considered consistent with the current design basis analyses. The loads considered in the evaluation include the reactor internal pressure differences, seismic loads, flow induced and acoustic loads due to the postulated recirculation line break (RLB-LOCA), thermal load effects, dead weight, and flow loads.

1A. Confirm whether the loads considered for the evaluation of the reactor internal components include the fuel lift loads, the safety relief valve discharge loads, annulus asymmetric pressurization and jet reaction loads during a main steam or a feedwater line break.

1B. Discuss the effects of the proposed extended power uprate (EPU) on the RLB-LOCA load and other design basis loads mentioned above.

Response

1B. The effects of EPU on the loads considered for the core support structure and non-core support structure components are discussed in detail for each component in Section 3.3.4 (a) through (o) of Reference 2, Attachment E, Power Uprate Safety Analysis Report (PUSAR). Governing loads and stresses of reactor internals components are provided in the response to Question 2B.

Question

2A. In Section 3.3.2, you indicated that the reduction in some fatigue usage factors (CUFs) in Table 3-3a is a result of reduction in the conservatism and/or number of thermal cycles from the original analysis. Describe how you arrived at an accurate representation of the fatigue cycles which resulted in a reduction of CUF from 0.94 to 0.862 for the shroud support as provided in Table 3-3a.

2B. In regard to Section 3.3.4, provide the maximum calculated stress and CUFs for the reactor internal components evaluated for both the current design condition and the uprate power condition, the allowable code limits, and the code and code edition used in the evaluation for the power uprate. If different from the code of record, provide your justification.

Response

2A. The EPU CUF of 0.862 for the support skirt was reduced by removing conservatism from a previous analysis (CUF = 0.945). The 1989 analysis used 278 startup/shutdown cycles and 361 scram cycles. The support skirt was re-analyzed in 2000 to account for the then-current thermal cycle information; using 250 startup/shutdown cycles and 232 scram cycles (CUF = 0.838). The power uprate calculation used this information to scale up the CUF to 0.862. Consequently, the power uprate fatigue usage was an increase in CUF from the 2000 CUF of 0.838 to 0.862.

Scaling Technique
General Electric has developed a technique to conservatively scale the original stress report stresses to account for changes in the original pressures, temperatures, and nozzle flows as a result of EPU.

Many pressure vessel calculations select the three stress directions of the orthogonal coordinate system such that the shear stress components are zero; the normal stress components are the principal stresses. With this orientation, the pressure stresses are directly proportional to the increase in coolant pressure, and the magnitude of the principal stress resulting from thermal cycling is proportional to the temperature change during a thermal transient. When there are no changes in mechanical loads as a result of the EPU, the new magnitude of the principal stress is:

$$\sigma_{new} = \sigma_p * (P_{new}/P_{old}) + \sigma_t * (\Delta t_{new}/\Delta t_{old}) + \sigma_m$$

where:

σ_p = Original pressure stress
σ_t = Original thermal stress
σ_m = Original mechanical stress
P_{new} = EPU pressure
P_{old} = Original pressure
Δt_{new} = EPU temperature range
Δt_{old} = Original temperature range

or:

$$\sigma_{new} = \sigma_p * SCF_p + \sigma_t * SCF_t + \sigma_m$$

where:

SCF_p = Pressure stress scaling factor
SCF_t = Thermal stress scaling factor

Components that experience a change in internal coolant flow during operation have a flow scaling factor, SCF_f. The magnitude of the internal flow changes the convective heat transfer coefficient. The Biot Modulus is used to determine the effect of increased nozzle flows on the nozzle thermal skin stresses. It can be shown that:

$B_i \;\alpha\; h \;\alpha\; V^{0.8}$

where:

B_i = Biot Modulus, hL/k
h = Film convection coefficient
V = Flow velocity through nozzle

This relationship allows the flow scaling factor to be determined by the following:

$$SCF_f = (V_{new}/V_{old})^{0.8}$$

When the flow scaling factor is applied, a new thermal SCF is calculated using both the SCF_t and the SCF_f. The new thermal scaling factor is calculated using the following formula:

$$SCF_T = SCF_t * SCF_f$$

Most stress reports do not separately report the pressure, thermal, and mechanical stresses; therefore, it is not practical to calculate the scaled pressure or scaled thermal stresses. A conservative scaling technique, using the larger of the pressure and temperature scaling factors, is used to scale the entire stress magnitude. If a calculated SCF is less than unity, a SCF = 1.0 is used instead. This method is a conservative alternative to scaling the individual stress components because:

- The largest scaling factor is used for both the pressure and temperature SCF.
- The mechanical stresses are increased by the SCF even though the design mechanical loads did not increase.
- Conditions which generate a stress reduction (a SCF less than 1.0) are ignored.

The stress scaling technique may be further simplified by applying the SCF to the stress intensity alone, rather than applying the SCF to the principal stress components. A stress intensity, or stress difference, used to compare with the American Society of Mechanical Engineers (ASME) Code allowable values is determined by selecting the absolute value of the maximum difference between any pair of principal stresses. Consider the following example:

$$\begin{aligned} S_{12,new} &= \sigma_{1,new} - \sigma_{2,new} \\ &= \sigma_{1,old} * SCF - \sigma_{2,old} * SCF \\ &= (\sigma_{1,old} - \sigma_{2,old}) * SCF \\ &= S_{12,old} * SCF \end{aligned}$$

Scaling Factors for the Support Skirt
The support skirt is located in Region B of the reactor and only experienced a change in temperature due to EPU conditions. Based on the scale factor equations discussed above, the highest normal startup SCF is 1.002 and the highest SCRAM SCF is 1.083.

Scaling Factors for Region B

		Pre- Power Uprate Condtions		Power Uprate Conditions		Scaling Factor
Zones	Description	Initial Temperature, F	Final Temperature, F	Initial Temperature, F	Final Temperature, F	SCF_1
3 – 4	Normal Startup	100	546	100	547	1.002
4 – 5	Normal Startup	546	538	547	539	1.000
4 – 5	Normal Startup	538	520	539	530	0.500
10 – 11	Scram	400	520	400	530	1.083

Support Skirt
The support skirt was re-analyzed in 1989 and accounted for the latest thermal cycle information at that time. Since the QCNPS and DNPS RPVs have the same usage factors, the QCNPS results apply to DNPS. The limiting transients for the support skirt are heatup and cooldown. The maximum primary plus secondary stress range (P + Q) is 82.88 Ksi which exceeds the Code allowable limit of $3S_m$. The P + Q stress intensity with thermal bending removed is scaled up by using the appropriate SCF and compared to the Code allowable.

$P + Q = (P + Q - \text{Thermal Bending})_{old} * SCF$
54.41 Ksi = 53.31 Ksi * 1.002
54.41 Ksi < $3S_m$ = 69.9 Ksi

Since the calculated value of the maximum primary plus secondary stress is greater than the $3S_m$ limit, an elastic-plastic analysis, as described in the Code is performed.

$$S_{alt,new} = S_{P+Q+F} / 2 = [(S_n * SCF*(K_t - 1) + S_{surf} * SCF)/2]*K_e *(E_c/E_a)$$
$$= [(82.88*1.002*(2.12-1)+117.77*1.002)/2]*1.15*(30/28)$$
$$= 130.0 \text{ Ksi.}$$

where: $K_e = 1.0$ for $S_n \leq 3S_m$
$= 1.0 + [(1/n - 1)/(m - 1)]*(S_n / 3S_m - 1) = 1.15$
for $3S_m < S_n < 3mS_m$
$= 1/n$, for $S_n > 3mS_m$

where:
n = 0.2 for Low Alloy Steel
m = 2.0 for Low Alloy Steel
$3S_m$ = 80.1 Ksi
$S_n = P + Q$ = 82.88 Ksi

S_{surf} = 117.77 Ksi
E_c = 30E6 psi
E_a = 28E6 psi
K_t = 2.12
SCF = 1.002 (Region B, Startup)

A thermal stress ratcheting check was performed and shakedown will occur. Therefore, thermal stress ratcheting is not a concern. For an alternating stress of 130.0 Ksi, the allowable number of cycles is 304. For a total number of startup/shutdown cases of 250, the fatigue usage factor is 0.822. Similarly, for SCRAM, the alternating stress was calculated using the above method with a SCF of 1.083 and is 45.5 Ksi. This stress allows 5860 SCRAM cycles. There are 232 SCRAM cases predicted for a 40 year life which gives a fatigue usage factor of .040. The total combined fatigue usage factor is:

$U_{Total} = U_{SU/SD} + U_{SCRAM}$
$U = n_1/N_1 + n_2/N_2 = 250/304 + 232/5860 = .822 + .040 = .862 < 1.0$

Question
3. In Section 3.3.5, you evaluated the effects of the EPU on the potential for flow-induced vibration of the reactor internal components due to the increase in steam produced (>20%) in the core, the increase in the core pressure drop, and the increase in the recirculation pump speed. You indicated that the evaluation was based on the vibration data for the reactor internal components recorded during the startup testing of DNPS and QCNPS plants and on operating experience from similar plants. The expected vibration levels under EPU conditions were estimated by extrapolating the vibration data recorded during startup testing at the DNPS and QCNPS units.

3A. Discuss whether and how the recorded vibration data can be applicable for your calculation of the flow induced vibration stress level after the steam separators and dryers hardware modifications that are required for the EPU.

3B. Provide a sample evaluation for the most critical components (i.e., steam dryers and steam separators) and the basis for using the operating experience of similar plants.

3C. Discuss the potential for flow-induced vibration of the reactor internal components due to various mechanisms, including, in particular, the fluid-elastic instability in the steam separators and dryers at the proposed power level. If the details of the analysis and the results are documented in a report, submit the report for staff review.

3D. Provide a discussion on the potential for excessive vibrations, high noise levels, and the instrument lines leakage that might be caused by the increased recirculation pump speed or flow for the proposed power uprate, as described in the NRC Information Notice 95-16.

Response
3A. There is no recorded vibration data for the steam dryer. It is a non-safety related component and it was not instrumented during startup.

There were no modifications to the steam separator due to EPU.

3B. The steam dryer has no safety function. The sole function of the steam dryer is to remove moisture from the steam in order to minimize erosion of the piping and turbine and to improve the turbine efficiency. The Boiling Water Reactor Vessel Internals Project (BWRVIP) document BWRVIP-06 (Reference 3), which was endorsed by the NRC, also states that the dryer is non-safety related and failure of a dryer component may cause an operability concern but has no safety impact. Hence the dryer was not instrumented during startup testing and no measured vibration data is available for the prototype plant.

The design criteria for the steam dryer is that the structural integrity of the dryer is maintained when subjected to a steam line break occurring beyond the main steam isolation valves. Since the dome pressure is not changed under EPU conditions, steam dryer structural integrity evaluations performed for a steam line break for the current rated thermal power is applicable to EPU conditions.

The operational history of steam dryers in similar plants was also studied to see if there were any flow induced vibration related problems in the dryer. Only drain channel cracks at steady state conditions and outer bank hood damage due to turbine stop valve (TSV) closure were found due to vibration effects. Drain channel cracking has occurred even during normal operation and is usually repaired after detection. The outer bank hoods adjacent to the steam outlet nozzles at DNPS and QCNPS are four times thicker than at the plant where the damage occurred, while the TSV closure time is identical. Hence it is expected that the outer bank hood can withstand the transient. While instances of drain channel cracking and hood cracking have occurred at operating plants, it is an operational issue only, relating to proper drying of the steam before it leaves the dryer. No structural integrity problems have been observed with these cracks. The dryers are visually inspected during removal in each refueling outage and any observed cracking can be repaired.

The steam separator is also not a safety-related component. However, the steam separator loads act on the shroud through the shroud head. Since the shroud is a safety related component, the separator/shroud structure was tested at various power conditions up to the rated power during plant startup.

3C.

3D. The vibration issue associated with increased containment noise and vibration levels due to increased recirculation pump speed was investigated and reported in GE SIL No. 600. The conclusion of this investigation was that the increased noise and vibration levels associated with higher recirculation pump speeds were a direct result of a residual heat removal (RHR) testable check valve not being properly seated. Testing demonstrated that the containment noise and RHR vibration levels were greatly attenuated when the RHR testable check valve was properly seated.

The containment noise and vibration associated with the RHR testable check valve, at increased pump speeds and flow rates, was determined by testing to have no detrimental effect on plant equipment, including the reactor recirculation system (RRS) piping, RHR piping, the recirculation pumps and motors, and the containment structure.

<u>Question</u>
7. In Section 3.3.6, you stated that EPU conditions result in an increase in saturated steam generated in the reactor core. For constant core flow, this in turn results in an increase in the separator inlet quality and dryer face velocity and a decrease in the water level inside the dryer skirt, all of which affect the steam separator-dryer performance. The results of the evaluation demonstrate that the steam separator-dryer performance remains acceptable up to some portion of extended power prior to any substantive hardware modification. To reduce the moisture content, hardware modifications are required. These modifications will be completed before EPU implementation.
Confirm whether and how your evaluation in Section 3.3.4 for the structural integrity of steam separators and dryers will be affected by the required hardware modifications due to the proposed EPU at DNPS and QCNPS.

<u>Response</u>

<u>Question</u>
10.A. In Section 3.5.5, you indicated that the main steam (MS) and feedwater (FW) piping will experience increased vibration levels, approximately proportional to the square of the flow velocities. For the proposed power uprate, the flow rates and flow velocities will increase by more than 20 percent of the flow rate at the original rated thermal power for the MS and FW piping systems.
Provide an evaluation of the cumulative fatigue usage factor (in addition to the startup and shutdown cycles), and the potential for flow-induced vibration in the MS and FW piping (during the normal and upset operations) and in heat exchangers following the power uprate.

10B. In Section 10.4.3, you indicated that the vibration level may even be higher if other flow induced vibration mechanisms occur.
Provide a discussion on the potential for flow-induced vibration of the main steam and feedwater piping due to various mechanisms, including, in particular, the fluid-elastic instability at the proposed power level.

Response
10A. The steady state flow induced vibration (FIV) maximum stress levels of the main steam (MS) and feedwater (FW) piping must remain below the endurance limit of the piping material. This is because many, many cycles of vibration will be encountered over the remaining design life of the plant. For austenitic (stainless) steel piping material, the mean value of endurance limit stress, at which high cycle fatigue failures can occur, is in the vicinity of 30,000 psi. The actual design fatigue endurance limit is set well below this value. The design fatigue endurance limit for steady state alternating stresses from vibration is 13,600 psi (zero to peak) for austenitic (stainless) steel piping materials. The design fatigue endurance limit for steady state alternating stresses from vibration is 7,690 psi (zero to peak) for carbon steel piping materials. These fatigue design endurance limits were taken from ASME Section III Pressure Vessel and Piping code and the American National Standard, OM S/G 1997.

If the steady state vibration levels of the MS and FW piping are measured and found to be below these design limits, which are well below the actual material fatigue endurance limits, then no fatigue usage can ever occur from FIV at the new and 20% higher flow rates. These 20% higher MS and FW flow rates are the flow rates required for EPU conditions.

The potential for flow-induced vibration of the main steam and feedwater piping due to various FIV mechanisms, such as a fluid-elastic instability, is possible. However, it is not possible to analytically predict which FIV mechanism, if any, may occur within the MS or FW piping at the new and higher MS and FW flow rates associated with the new EPU flow conditions. For this reason, Exelon Generation Company, LLC (EGC) will performing a startup piping vibration test program during initial plant operations during power ascension to the new EPU conditions. These new startup tests will show that the steady state MS and FW piping FIV levels at the new and higher EPU flow conditions are well below the fatigue endurance limit of the piping material.

Startup, shutdown, normal, and upset conditions or transient vibration cycles associated with the MS and FW piping are assessed in the piping evaluation report prepared for the planned EPU at the planned EPU flow conditions. MS and FW piping system are analyzed to the following codes.

- B31.1 Power Piping Code, 1967 edition
- B31.1 Power Piping Code, 1967 edition and 1973 through 1976 Summer Addenda.
- ASME Code Section III, Sub-section NC (Class 2), 1977 through 1978 Winter Addenda.
- ASME Code Section III, Sub-section ND (Class 3), 1974 through 1976 Summer Addenda.

These industry codes do not require fatigue analysis.

Heat exchangers in the main steam and feedwater systems, such as the main condenser and feedwater heaters, were evaluated for EPU operational conditions. Refer to the response to Questions 12C (Reference 1) and 12D (below) for discussions of these evaluations.

10B. In Section 10.4.3, it was stated that the vibration level may even be higher if other flow induced vibration mechanisms occur. The startup piping vibration test program planned for the MS and FW piping during initial plant operation at the new, higher EPU flow conditions, will be expected to show that the FIV levels are acceptable and well below the fatigue endurance limit of the piping material, independent of the FIV mechanism occurring.

<u>Question</u>
11D. Discuss the effects of the proposed power uprate on the pressure locking and thermal binding of safety-related power-operated gate valves for Generic Letter (GL) 95-07.

<u>Response</u>
11D. The results of GL 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," were determined to be unaffected by EPU as reported in Reference 4.

<u>Question</u>
12D. Provide a discussion on the potential for flow-induced vibration of the main condenser tubes, and heat exchangers due to increased temperature and flow in the main steam and feedwater systems.

<u>Response</u>
12D. For EPU, the range of circulating water flow rates through the main condenser tubes is unchanged. Flow-induced vibration for the main condenser tubes from steam is addressed in Question 12C as provided in Reference 1.

The feedwater heaters were analyzed and verified to be acceptable for the higher feedwater heater flows for EPU. The feedwater heaters maximum shell-side velocities were determined to be in compliance with the design guidelines of Heat Exchanger Institute, "Standard for Closed Feedwater Heaters," except for one heater group where the shell drain outlet velocity exceeded the allowable by less than 0.7 ft/sec. In addition, to assess the tube-side mechanical effect of EPU operation on the feedwater heaters, flow velocities were evaluated based on the heat exchange industry guidelines for tube side flow velocity to minimize tube end erosion. The maximum existing tube plugging in the heaters was considered. The tube-side flow velocities in all but one group of heaters are predicted to slightly exceed the guidelines, the highest by less than 3 ft/sec. These heaters have been identified for erosion monitoring.

Attachment C
Additional Mechanical Information Supporting the License Amendment Request to Permit Uprated Power Operation (Non-Proprietary)
Dresden Nuclear Power Station, Units 2 and 3
Quad Cities Nuclear Power Station, Units 1 and 2

References

1. Letter from K. A. Ainger (Commonwealth Edison Company) to U. S. NRC, "Additional Mechanical Information Supporting the License Amendment Request to Permit Uprated Power Operation at Dresden Nuclear Power Station and Quad Cities Nuclear Power Station," dated August 8, 2000

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

3. EPRI TR-105707, BWR Vessel and Internals Project, "Safety Assessment of BWR Reactor internals (BWRVIP-06)", dated October 1995

4. General Electric Company Licensing Topical Report, "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate," NEDC-32523P-A, Class III, February 2000, and Supplement 1, Volumes I and II