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RS-01-157

August 8, 2001

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Dresden Nuclear Power Station, Units 2 and 3
Facility Operating License Nos. DPR-19 and DPR-25
NRC Docket Nos. 50-237 and 50-249

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

Subject: Additional Mechanical Information Supporting the License Amendment Request to Permit Uprated Power Operation at Dresden Nuclear Power Station and Quad Cities Nuclear Power Station

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

In the referenced letter, Commonwealth Edison (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. Attachment A to this letter provides the requested information. This letter provides the first 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 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.

AP01

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

Respectfully,

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

Attachments:

Affidavit
Attachment A: Additional Mechanical Systems 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 Systems 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

Attachment B
Additional Mechanical Systems 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

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 Systems 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.

Terrence W. Simpkin

T. W. Simpkin
Manager – Licensing

Subscribed and sworn to before me, a Notary Public in and for the State above named, this 8th day of August, 2001.

Timothy Byam

Notary Public

" OFFICIAL SEAL "
Timothy A. Byam
Notary Public, State of Illinois
My Commission Expires 11/24/2001

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
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

Terry J. Morgan
Notary Public, State of California

Attachment C
Additional Mechanical Systems 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

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

Attachment C
Additional Mechanical Systems 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

This attachment contains 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. Responses to NRC Questions 1, 2, 3, 7, 10, 11D, and 12D will be provided separately.

<u>Question</u>
4. A. In reference to Sections 3.3.2 and 3.3.4, provide a discussion of the methodology, assumptions and loading combinations used for evaluating the reactor vessel and internal components with regard to the stresses and fatigue usage for the power uprate.

B. Were the analytical computer codes used in the evaluation different from those used in the original design-basis analysis? If so, identify the new codes used and provide your justification for their use by specifying how were these codes benchmarked for such applications.

<u>Response</u>
A. The methodology, assumptions and loading combinations used for evaluating the reactor vessel and internal components are described in Reference 1, Appendix I, "Methods and Assumptions for Vessel and Components Evaluations."

B.

<u>Question</u>
5. In Section 4.1.2.3 regarding the subcompartment pressurization, you stated that the increase in actual asymmetrical loads on the vessel, attached piping and biological shield wall, due to the postulated main steam and feedwater pipe breaks in the annulus between the reactor vessel and biological shield wall is minor. You also indicated that the biological shield wall and component designs remain adequate, because there is sufficient pressure margin available.

Discuss quantitatively how will the biological shield wall and the reactor vessel and internals be affected by the proposed power uprate as a result of increase in the applied asymmetrical pressurization and jet loads.

<u>Response</u>
PUSAR Section 4.1.2.3, "Subcompartment Pressurization," discusses asymmetrical loads without specifically referring to a main steam or feedwater line break. A postulated rupture of a recirculation suction line was previously evaluated for both Dresden Nuclear Power Station (DNPS) and Quad Cities Nuclear Power Station (QCNPS) to assess the structural capability of the biological shield wall.

For both DNPS and QCNPS, the largest line which has the safe end located in the annulus region between the reactor vessel and the biological shield wall is a 4 inch jet pump instrument line. The maximum calculated wall differential pressure (i.e., 1 psid) for this postulated break is well below the structural capability of the wall.

These previous evaluations were used as a basis to quantify the changes expected due to EPU. A simplified subcompartment pressurization model of the DNPS and QCNPS annulus region was developed and expected mass and energy releases at pre-EPU and EPU conditions were determined.

Recirculation suction line break mass and energy releases at pre-EPU and EPU conditions were calculated using the standard General Electric (GE) methods, using inputs from the reactor heat balances at both pre-EPU and EPU conditions.

The following assumptions were used to determine the pre-EPU and EPU mass and energy releases.

- Initial mass release rates (i.e., inventory period) are based on Moody saturated critical flow, with a flow multiplier of 1, through the break area from both the pipe side and reactor side of the break.
- Energy release rates are based on the core inlet enthalpy.
- After the initial blowdown (i.e., inventory period) the flow is conservatively based on the Henry-Fauske subcooled critical flow, rather than the Moody subcooled critical flow, from the nozzle area on the reactor side of the break. The flow from the pipe side of the break is based on the total area of 10 jet pump nozzles plus the reactor water clean up (RWCU) line area.
- The safe end weld is within the biological shield wall penetration. This penetration is included in the evaluation to account for a flow split between the annulus and the drywell.

The resulting maximum incremental increase in mass release due to EPU was determined to be 6% for DNPS and 6.2% for QCNPS. The maximum incremental increase in energy release due to EPU was determined to be 5.5% for DNPS and 5.8% QCNPS.

Benchmark subcompartment pressurization analyses of the DNPS and QCNPS annulus region were performed using the COMPARE computer code and pre-EPU mass and energy releases for a recirculation suction line break. The same model was rerun using mass and energy releases calculated at EPU conditions.

The biological shield wall pressurization has been evaluated for the effects of these small increases in mass and energy. An analysis was performed to determine the effect on annulus pressure expected for the above changes in mass and energy releases. This resulted in a minor reduction in pressure margin. The study resulted in an increase of 0.9 psi for DNPS and 1.2 psi for QCNPS in the maximum calculated biological shield wall differential pressure. The incremental increase in annulus pressure was applied to results of the previous evaluations dicussed above. The pressure margins are provided below.

PARAMETER	DNPS	QCNPS
Annulus differential pressure at which biological shield wall failure would begin (psid)	41	46
Maximum annulus pressure from a recirculation line break (psid)	36	38
Pre-uprate margin (psid)	5	8
Incremental change due to EPU (psi)	0.9	1.2
EPU margin (psid)	4.1	6.8

The jet loads are evaluated in PUSAR Section 10.1.2, "Pipe Whip and Jet Impingement." The review shows that there is no change in the operating pressure of high energy main steam piping. Thus, the jet impingement load evaluation results remain unchanged for the main steam piping system due to EPU. For the feedwater piping, the internal pressure increase is less than 10 psi. The less than 10 psi change in the internal pressure represents an approximately 1% change that was judged to be insignificant for jet impingement load evaluation.

<u>Question</u>
6. In the evaluation of the reactor jet pumps in Section 3.3.4, you stated that additional engineering evaluations will be performed to determine if the jet pump riser brace will be susceptible to vibration from the recirculation pump vane passing frequency (VPF). The evaluations will determine if modifications are required to alter the natural frequency of the jet pump braces.

A. Provide your evaluation associated with the possible VPF vibrations due to the EPU.

B. Confirm whether and how your evaluation for the structural integrity of jet pumps will be affected by the VPF vibrations due to EPU at DNPS and QCNPS.

<u>Response</u>
A. An extensive test program was conducted at the GE test facilities in San Jose from February to July 2001 to determine the natural frequencies of the DNPS Unit 2 and Unit 3 riser braces. The DNPS Unit 3 riser braces are representative of the QCNPS Units 1 and 2 riser braces. A full scale mockup of the jet pump riser pipe and riser brace was constructed and set up to determine the residual loads and natural frequencies of the riser brace leaves in air and also while submerged under water. A total of 26 strain gages and 6 accelerometers were installed and the natural frequencies of these jet pump components were computed from the dynamic response to impacts from an instrumented hammer. The results of the test program showed that the reactor recirculation system VPF during EPU operation is well removed from the riser brace natural frequencies and no modifications are required to alter the natural frequency of the riser braces.

B. The VPF vibrations at non-resonant conditions were considered in the evaluation of the jet pumps. The above described test was conducted to see if there is any potential for resonance of the riser brace leaves due to VPF at EPU conditions. Since the VPF is well removed from the riser brace leaf natural frequency, the response due to VPF is small and the existing evaluation is not affected.

Question
8. A. In reference to Section 3.5, provide a discussion of the methodology and assumptions used for evaluating the reactor coolant pressure boundary piping systems for the proposed power uprate.

B. Provide the calculated maximum stresses and fatigue usage factors at the current design basis and the proposed power uprate conditions, corresponding critical locations and piping systems, allowable stress limits, and the code and code edition used in the evaluation for the power uprate. If different from the Code of record, justify and reconcile the differences.

Response
A. The reactor coolant pressure boundary (RCPB) piping evaluated includes the following piping systems.

- Reactor recirculation (RR) system
- Main steam (MS) piping inside containment
- Branch piping from RR and MS systems, including safety and relief valve discharge lines, shutdown cooling system (residual heat removal (RHR) for QCNPS), RWCU, low pressure coolant injection (LPCI), and others
- Reactor pressure vessel (RPV) head vent, RPV bottom drain line, and/or isolation condenser (IC) (Reactor Core Isolation Cooling (RCIC) for QCNPS)
- MS drain lines
- Small bore piping attached to these systems

Existing design and licensing basis documents, such as design specifications and piping stress reports, were reviewed to determine the design and analytical basis for these piping systems. The proposed uprate parameters of the RCPB piping systems were compared with the existing analytical bases to determine any increases in temperature, pressure, and flow due to the uprate conditions. During the evaluation process, the original code of record, code allowables, and the same analytical techniques were used. No new assumptions or computer codes were used except for in the evaluation of the MS lines as described in the response to Question 13A.

For the majority of these systems, it was determined that there are no changes in the analysis parameters. The RR system was determined to be subject to a slight increase in temperature, but less than the acceptance criteria outlined in the response to Question 9A. The MS piping will not experience an increase in temperature. However, a significant increase in flow will be seen, which will have an impact on the turbine stop valve (TSV) closure transient. A detailed description of the methodology and assumptions used in the evaluation of the MS system is provided in the response to Question 13A. Some of the branches off the RCPB piping (i.e., core spray (CS), LPCI, etc.) were also found to experience temperature increases due to long term post-LOCA conditions in which water is being drawn from the suppression pool (i.e., torus). These systems were evaluated with the large bore torus water piping systems and the methodology and assumptions used in those evaluations are described in the response to Question 9A. All other RCPB piping systems are either not impacted by EPU, or the changes are within acceptance criteria.

B. The majority of the RCPB piping systems are designed to American National Standards Institute (ANSI) B31.1.0, 1967 requirements, which are not subject to fatigue requirements. In addition, the RCPB piping is under the jurisdiction of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section I, 1965 Edition, through Summer 1966 Addenda including Code Cases N-1 thru N-3 and N-7 thru N-11. In accordance with these codes and code cases, fatigue is not part of the design or licensing basis for these systems. For DNPS only, the one exception is the RR system piping for DNPS Unit 3, which was replaced in the mid 1980s. The stress analysis for Class I piping covered by the scope of the RR pipe replacement project was performed in accordance with ASME Code, Section III, Subsection NB, 1980 Edition, including the Summer 1982 Addenda, which includes fatigue requirements. The RR system piping was determined to have a only minor increase in the temperature, which was considered negligible. Any small increase in stresses due to the slight temperature increase is bounded by inherent conservatisms in the existing analysis. Therefore, the calculated maximum stresses and fatigue usage factors are unchanged as a result of the proposed uprate. The critical locations and piping systems, allowable stress limits, and the code and code edition used are also unchanged.

Question

9.A. Provide a summary of your evaluation of the pipe supports, nozzles, penetrations, guides, valves, pumps, heat exchangers and anchors at the power uprate condition. The evaluation should include the methodology, assumptions, and the results of evaluation for the critical piping systems affected by the proposed power uprate.

B. Were the analytical computer codes used in the evaluation different from those used in the original design-basis analysis? If so, identify the new codes and provide your justification for their use by specifying how these codes were benchmarked for such applications.

Response

A. Operation at EPU conditions may increase piping stresses caused by higher operating temperatures, pressures and flow rates. Additionally, piping components (i.e., pipe supports, equipment nozzles, etc.) may be potentially subjected to increased loadings due to the EPU.

The piping system evaluations for power uprate were performed by determining "change factors" for the changes in thermal, pressure, flow rate, and total design load conditions. This method is based on determining a "change factor" by conservatively comparing the ratio of power uprate temperature, pressure and flow conditions to the corresponding pre-EPU conditions. The method (described below) used to evaluate DNPS and QCNPS is the same method used on several other power uprates - most recently for the Turkey Point, Byron and Braidwood power uprates. The recent Byron and Braidwood NRC Safety Evaluation for power uprate (Reference 3) concluded that, "The staff finds the methodology to be acceptable considering the conservatism in the calculation of the scaling factors for the power uprate stress and loads."

This method is based on determining a "change factor" by conservatively comparing the ratio of power uprate temperature, pressure and flow conditions to the corresponding pre-uprate conditions.

Where the "change factor" is less than or equal to 1.0, the pre-EPU (i.e., existing) conditions envelop or equal the power uprate conditions and no further review is performed.

For minor changes resulting in a "change factor" between 1.0 and 1.05 (i.e., 5%), the increase was considered acceptable since the small increase is offset by conservatism inherent in the analytical methods used to calculate the existing stresses and loads. The conservatism include, but are not limited to, the industry practice of enveloping multiple operating conditions and modeling pipe supports without consideration of gaps between piping and supports. Pressure effects are considered in conjunction with other loading conditions which are unchanged by the EPU (e.g., weight, seismic) thus the overall effect of the pressure change factor is reduced. Therefore for "change factors" between 1.0 and 1.05, the existing stress and load values were considered to be acceptable and remain within allowable limits.

For "change factors" greater than 1.05, simple and conservative evaluations were performed to address the specific increase in stress and load values. Where the simple evaluation yielded a resultant stress ratio (i.e., calculated / allowable) that was less than or equal to 1.0, the resultant stress remains acceptable. For those conditions where the resultant stress ratio is greater than 1.0, the calculations were revised and/or piping support modifications were performed to bring the stress at EPU conditions within allowable limits.

The thermal "change factor" was based on the ratio of the thermal power uprate to pre-thermal power uprate operating temperature. That is, the thermal change factor is $(T_{uprate}-70°F)/(T_{pre-uprate}-70°F)$. Using this method for the thermal change factor, evaluations resulted in a bounding evaluation of the thermal impact on piping stresses and loads.

Similarly, the pressure "change factor" was determined by the $P_{uprate}/P_{pre\text{-}uprate}$ ratio and the flow rate "change factor" was determined by the $Flow_{uprate}/Flow_{pre\text{-}uprate}$ ratio. The total design load change factor is the total combined load associated with EPU conditions divided by the allowable design load, and was determined by the following formula:

[Dead Weight (DW) + $\text{Pressure}_{\text{uprate}}$ + $\text{Thermal}_{\text{uprate}}$ + $\text{TransientLoad}_{\text{uprate}}$ + Seismic] / Design $\text{Load}_{\text{analyzed}}$.

Thermal changes were found to be the most significant, primarily for systems using the suppression pool as a water suction source during long term post-LOCA conditions. No changes to the suppression pool loads (i.e., pool swell, condensation oscillation, chugging and SRV discharge) will result from the EPU because previous load definitions were determined to be bounding. Pressure changes were typically found to be negligible and were unchanged for most systems. There is a slight increase in predicted design basis accident (DBA) pressures inside the torus. However, most torus attached piping systems and components were previously analyzed for the maximum intermediate break analysis pressures, which bound even the new DBA pressures. Flow changes were found to be significant only for the MS and feedwater/condensate systems. A detailed evaluation of the MS system was performed for the increased flow rate and is discussed in more detail in the response to Question 13A.

All piping systems subject to changes in temperature, pressure or flow were screened to determine the impact on the piping and piping components (i.e. supports, penetrations, equipment nozzles, etc.). Piping systems subjected to minor operating condition increases due to EPU were excluded from a detailed evaluation, as follows.

Thermal load increases of up to 5% (i.e., change factors between 1.00 and 1.05), were considered acceptable since these increases are offset by conservatism in analytical methods used to calculate the existing stresses and loads. Conservatisms include the enveloping of multiple thermal operating conditions and not considering pipe support gaps in the thermal analyses.

Furthermore, in accordance with industry practice, piping systems that have operating temperatures less than 150°F did not require evaluation for thermal change effects.

Pressure load increases up to 5% were considered acceptable due to margins in piping wall thickness.

Transient load increases up to 5% resulting from EPU related fluid flow rate changes were considered acceptable due to conservatism in load combinations (i.e., transient loads are combined with other conservative loads such as thermal and seismic).

Total design load increases of 5% were considered minor and acceptable by engineering judgment due to inherent conservatism in piping analysis methodology, as previously described.

The total design load criteria was not used for drywell steel, corner room steel, and/or flued head anchors without reviewing their qualification documentation to ensure that similar reasoning to this criteria had not been previously invoked for other load increases.

If the increases described above exceeded 5%, the analyzed margin between design load and the allowable load prior to uprate was used to justify the increases for uprate conditions (e.g., if the load increased by 15%, but the piping component analysis showed a 20% margin to allowable, the component was considered acceptable).

If the load increase on a piping component was greater than the calculated available margin, then a detailed evaluation of the component was performed to evaluate the adequacy of the component for EPU conditions. If the detailed evaluation could not justify the increased EPU loads in accordance with the previously defined acceptance criteria, a modification was designed for that component such that the modified component would meet that acceptance criteria. A description of the modifications required to qualify the piping and piping components for EPU conditions is provided in the response to Question 13B.

All piping systems and piping components with changes in temperature, pressure or flow rate were screened for impact by EPU. If the change factor for the piping system was less than 1.05, the whole system, including the piping components (i.e., supports, penetrations, equipment nozzles, etc.), was considered acceptable. If any of the change ratios exceeded 5%, each piping component was reviewed independently.

The evaluation methodology used to assess impact of the long term post-LOCA temperature increase on torus water piping system components (piping components in systems pumping or exposed to the torus water) is provided in more detail below, by component type:

Pipe Stress
The basic approach for the pipe stress evaluation was to scale up the existing Level A ASME Equation 10 pipe stresses by the thermal change ratio. The revised stress was then compared to the allowable pipe stress associated with the post-LOCA thermal condition. The application of ASME and B31.1 for the EPU pipe stress evaluations is consistent with the existing design and licensing basis.

The allowable pipe stress for post-LOCA conditions was based on the code of record for each piping system for one time secondary loads (e.g., single non-repeated anchor movement). For ASME piping, the allowable stress was taken as 3 S_h (equal to 45,000 psi for A-106 Gr. B piping). For B31.1 piping, the allowable was taken as 1.8 S_h (equal to 27,000 psi for A-106 Gr. B piping). For B31.1 piping, as an alternate, an allowable of 3 S_h minus the actual deadweight (DW) and pressure stresses is allowed by Section 102.3.2d of B31.1.

Rigid Pipe Supports
Rigid supports were categorized as those supports that rigidly support both static and dynamic loads and include rod hangers where applicable, struts, guides, and piping anchors, etc. The basic approach was to calculate a revised post-LOCA load combination of DW plus EPU thermal (T) (i.e., thermal expansion plus thermal anchor movement) plus safe shutdown earthquake (SSE) plus EPU torus displacement (TD). This load combination was classified as a

Level D or faulted load combination. Therefore, a revised interaction coefficient (IC) (i.e., actual stress divided by allowable stress) was calculated by multiplying the maximum IC in the existing calculation by the total design load change factor defined as the new post-LOCA load combination (DW+T+SSE+TD) divided by the largest peak qualified load. In addition, for supports subjected to frictions loads (i.e., guide supports), or supports with integral welded attachments, additional evaluations were performed.

Snubbers
Since snubbers do not resist thermal loads, the new EPU thermal conditions will not affect the snubber loads. The thermal displacement will increase however, so there is a potential for a top out or bottom out condition associated with the increased thermal displacements from EPU. In the late 1980s, allowable cold setting ranges were determined for each snubber to ensure that sufficient travel was available such that the snubbers would not bottom or top out on their range during thermal expansion. Included in this range calculation was a minimum of a ½ inch travel margin provided on each end of the range. Therefore, a minimum of ½ inch of travel is available to handle additional thermal expansion above and beyond the current design displacements. A generic evaluation was performed, which concluded that the increase in thermal displacements due to the EPU would not exceed the ½ inch available travel.

In addition, the increased displacement will cause an increase in the swing angle for snubbers and other pinned supports. A generic evaluation was performed, which concluded that the increase in swing angles due to EPU conditions is minor and will not impair the functionality of the pinned type supports.

Spring Hanger Supports
For each affected spring hanger, the increased vertical thermal displacement was compared to the available displacement to top/bottom-out conditions. If the additional displacement exceeded the available displacement by more than 5%, then a modification was issued to reset or replace the existing spring can. The increase/decrease in the spring hanger load due to movement change is considered to be negligible.

Displacements at Interferences
Some piping models have displacement checks at certain locations where there may be interferences with nearby structures (i.e., slab or wall penetrations, nearby plant equipment, etc.). The locations that were impacted were evaluated to make sure the revised thermal displacements did not result in damaging contact with these interferences.

Flanges
Some of the piping models have in-line flanges that have been evaluated for piping moments. These moments in the piping system are affected by the increase in temperature for these lines. For the affected flanges, revised thermal moments were calculated for the flanged joints and compared to the previously calculated allowables.

Valves
The stresses in valve bodies were already enveloped by the stresses reported for the piping, so these valves were covered in the piping stress evaluation. For valves with extended operators (i.e., motor operated valves (MOVs)), the stresses are a function of the valve acceleration and are not affected by increased thermal loads.

Containment Penetrations
Some of the piping systems penetrate the primary containment boundary (i.e., the torus or the drywell). At these penetrations, the containment shell is evaluated for the local stresses in the vicinity of the penetration due to the reactions at the penetration. The total stress in the containment shell is a combination of the local stresses due to the reaction loads from the piping, combined with the global shell stresses due to conditions inside containment. The revised post-LOCA forces and moments were calculated for all six degrees of freedom and compared to the previously qualified loads. In some cases, revised combined stresses in the containment were calculated and compared to the allowable stresses.

Equipment Nozzles
The existing design basis for piping loads on equipment is that the nozzles and casings are considered acceptable if the attached piping stress at the nozzles meets the code requirements for the piping. For certain equipment, a seismic qualification utility group (SQUG) type evaluation had previously been performed, where the equipment anchorage was evaluated considering the piping reaction loads. This approach was extended to cover non-SQUG equipment such as the core spray (CS) pumps. The affected equipment included the LPCI and CS pumps and the LPCI heat exchangers at DNPS and the RHR and CS pumps and the RHR heat exchangers at QCNPS. If the loads on this equipment increased by more than 5%, the equipment anchorage was re-evaluated. In some cases, it was concluded that certain equipment is bounded by other similar equipment that had been previously evaluated and accepted (i.e., identical equipment with higher nozzle loads).

Reactor Nozzles
Some of the piping systems tie directly into reactor nozzles. At these nozzles, an evaluation was performed to determine the impact of the nozzle reaction loads on the RPV. The revised stresses in the RPV nozzles were calculated for EPU conditions and compared to the previously calculated allowable stresses. The nozzles were also previously evaluated for fatigue considerations. Since the EPU post-LOCA thermal condition is a one-time event, its impact on the fatigue analysis of the nozzle was determined to be negligible.

Results
The results of the piping evaluations are provided in Tables 9A-1, 9A-2, 9A-1QC, 9A-2QC, 9A-3, 9A-4, 9A-3QC, and 9A-4QC. All large bore (i.e., > 4" normal pipe size (NPS)) torus water piping systems were evaluated for the effect of increased operating temperatures and pressures. The resulting pipe stress for each piping system and the corresponding allowable stresses are shown in Tables 9A-1, 9A-2, 9A-1QC, and 9A-2QC. The scope of the small bore torus water

piping systems that were evaluated for EPU conditions included small bore piping directly attached to the torus and small bore piping connected to large bore piping that is directly attached to the torus. Also, small bore lines attached to large bore lines that are not torus attached but transmit torus water during the long term post-LOCA mode were evaluated. The current and resulting EPU pipe stress for each small bore piping system and the corresponding allowable stresses are shown in Tables 9A-3, 9A-4, 9A-3QC, and 9A-4QC.

Piping components (i.e., pipe supports, etc.) were evaluated as described above. In some cases modifications were required to ensure the components could handle the increased thermal loads due to the EPU. If modifications were required, the stresses shown in the tables reflect the post-modification calculated stresses. A summary of all the piping component modifications is provided in the response to Question 13B.

B. In some instances different software codes were used in the evaluation of various piping systems and piping components (i.e., pipe supports) when detailed analysis was required to evaluate a system or component. The following software codes were used, along with a description of how they were benchmarked.

Piping Analysis Software
PIPSYS was used for piping analysis for certain torus water piping systems when a more detailed analysis was required. These piping systems were previously analyzed using the proprietary software PISTAR. In these cases PIPSYS was only used to analyze non-Mark I load cases (i.e., deadweight, seismic, and thermal). PIPSYS is a widely used piping analysis software which was procured from Sargent & Lundy (S&L) and has been verified and validated for use on nuclear projects in accordance with the S&L Quality Assurance Program.

NUPIPE-SWPC was used for piping analysis for certain torus water and main steam piping systems when a more detailed analysis was required. NUPIPE-SWPC is suitable for use in nuclear safety related applications and has been benchmarked to industry standards and codes. It is documented, reviewed, approved and controlled in accordance with the Stone & Webster Quality Assurance Program.

Frame Analysis Software
GT-STRUDL and PC-PREPS were used for frame analysis for certain torus water and main steam piping supports when a more detailed analysis was required. Some of these supports were previously analyzed using GENSAP or using manual calculations. GT-STRUDL and PC-PREPS are suitable for use in nuclear safety related applications and have been benchmarked to industry standards and codes. They are documented, reviewed, approved and controlled in accordance with the Stone & Webster Quality Assurance Program.

STAAD-III was used in the frame analysis of certain MS pipe supports inside the drywell. These supports were previously analyzed manually. STAAD-III is a widely used analysis software which has been verified and validated for use on nuclear projects in accordance with the S&L Quality Assurance Program.

Baseplate Analysis Software
NPLATE was used for baseplate analysis for certain torus water pipe supports. Some of these supports were previously analyzed using SDAL or BASEPLATE II software or by hand calculations. NPLATE is a widely used baseplate analysis software which was procured from Duke Engineering and was verified and validated for use on nuclear projects as part of the Duke Engineering Quality Assurance Program.

Fluid Transient Forcing Function Development Software
STEHAM was used for fluid transient forcing function development for main steam piping when a more detailed analysis was required. STEHAM is suitable for use in nuclear safety related applications and has been benchmarked to industry standards and codes. It is documented, reviewed, approved and controlled in accordance with the Stone and Webster Quality Assurance Program.

Integral Welded Attachment Analysis Software
ANSYS, PILUG, PITRUST and PITRIFE were used for integral welded attachment analysis for certain torus water and main steam piping supports when a more detailed analysis was required. ANSYS, PILUG, PITRUST and PITRIFE are suitable for use in nuclear safety related applications and have been benchmarked to industry standards and codes. They are documented, reviewed, approved and controlled in accordance with the Stone and Webster Quality Assurance Program.

Question
11. A. Discuss the functionality of safety-related mechanical components (i.e., all safety-related valves and pumps, including air-operated valves (AOV) and safety and relief valves) affected by the proposed power uprate to ensure that the performance specifications and technical specification requirements (e.g., flow rate, close and open times) will be met for the proposed power uprate.

B. Confirm that safety-related air operated valves (AOVs) and motor-operated valves (MOVs) will be capable of performing their intended function(s) following the proposed power uprate including such affected parameters as fluid flow, temperature, pressure and differential pressure, and ambient temperature conditions.

C. Identify the mechanical components that were not evaluated at the uprated power level.

E. Provide an evaluation of the effect of increased temperature due to power uprate on thermally-induced pressurization of piping runs penetrating the containment that were evaluated in response to Generic Letter 96-06, "Assurance of Equipment Operability and Containment Integrity During Design Basis Accident Conditions."

<u>Response</u>
A. Plant mechanical systems, including safety-related mechanical components, were evaluated to assess operating condition changes at EPU. As described in Reference 1, some plant systems were determined to be not impacted or only slightly impacted by EPU. For the remaining plant systems, further evaluations were performed to ensure the adequacy of the system components to operate as required at EPU conditions. This review included all safety-related mechanical components (e.g., pumps and valves) within the system. Safety-related pumps, safety relief valves and other components were determined to be adequately designed for operation at EPU conditions.

Refer to the response to Question 11B for further discussion on AOVs and MOVs.

B. In addition to the mechanical component review discussed in the response to Question 11A, AOVs and MOVs were reviewed in more detail. All MOVs in the Generic Letter (GL) 89-10, "Safety-Related Motor-Operated Valve Testing And Surveillance," program have been evaluated for EPU process and ambient conditions changes, including parameters such as fluid flow, temperature, pressure, differential pressure and ambient temperature. These evaluations confirmed that the existing analysis for each MOV bounds the EPU conditions.

Safety-related AOVs have been categorized into an AOV Program and evaluated utilizing the Joint Owners' Group (JOG) methodology. All AOVs included in this program have been evaluated for EPU process and ambient conditions changes, including parameters such as temperature, pressure, flow and differential pressure, similar to that previously described for MOVs to confirm the AOVs operate as required after EPU implementation.

C. There is no listing of the mechanical components that were not specifically evaluated or determined not to be impacted by EPU. However, PUSAR Section 6.8, "Systems Not Impacted by EPU," identifies those systems that were generically dispositioned as unaffected by EPU in Reference 1, Section J, "Methods and Assumptions for System Equipment Evaluation."

For systems that are impacted by EPU, the components affected are discussed on a system by system basis throughout the PUSAR.

E.
<u>*DNPS*</u>
Piping runs penetrating the containment that were evaluated in the response to GL 96-06 were confirmed adequate for uprate conditions by one of the following methods.

- Penetration piping with relief valves. Relief valves set pressures are not affected by uprate conditions. Existing relief capacities are much greater than required, enveloping any slight increase in relief capacity required from heat transfer to the isolated section due to EPU.

- Penetration piping with a bypass. Piping runs containing a bypass line with a spring check valve are not affected by uprate conditions for thermal overpressurization.
- Other water-filled penetration piping. In some cases, EPU conditions slightly increase the heat transfer to the isolated water-filled piping. Adequate conservatism exists in the original design to accommodate the resulting slight increase in internal pressure.

QCNPS
Piping runs penetrating the containment that were evaluated in the response to GL 96-06 were confirmed adequate for uprate conditions by one of the following methods.

- Penetration piping with relief valves. Relief valves set pressures are not affected by uprate conditions. Existing relief capacities are much greater than required, enveloping any slight increase in relief capacity required from heat transfer to the isolated section due to EPU.
- Other water-filled penetration piping. In some cases, EPU conditions slightly increase the heat transfer to the isolated water-filled piping. Adequate conservatism exists in the original design to accommodate the resulting slight increase in internal pressure.

Question
12. A. In reference to Section 3.11, provide a summary addressing your evaluation of the effects of the proposed power uprate on the balance-of-plant (BOP) piping, components, and pipe supports, nozzles, penetrations, guides, valves, pumps, heat exchangers and anchorages.

B. Provide the calculated maximum stresses and fatigue usage factors for the most critical BOP piping systems, the allowable limits, the code of record and code edition used for the power uprate conditions. If different from the code of record, justify and reconcile the differences.

C. In Appendix G of the submittal, you indicated that some feedwater heater relief valves will be adjusted or replaced and the heaters will be rerated to compensate for the increased feedwater flow and the associated pressure change. You also indicated that condenser tube staking is planned for the main condensers to provide adequate protection against tube vibration damage at uprated power conditions. Provide a summary of your evaluation of the main condenser tubes at the uprated condition.

Response
A. The BOP piping systems include all other affected piping systems not included in the piping systems addressed in the response to Questions 8, 9, and 13. These systems were evaluated using the same methodology and criteria discussed in the response to Question 9A. With the exception of MS, which is described in Question 13A, most of these BOP systems will not experience significant changes in operating conditions due to EPU. A description of the piping systems examined, and the results of these evaluations are provided in Table 12A-1.

B. The calculated maximum stresses and fatigue usage, the allowable limits, the code of record and code edition used for the EPU conditions factors for the most critical piping systems are provided in the response to Questions 8, 9, and 13. The remaining BOP affected systems passed the screening criteria discussed in the response to Question 9A, and no new analyses were required.

C.
DNPS
The main condenser tubes were evaluated at EPU conditions to determine which areas of the condenser tube bundle would be subject to potentially damaging tube vibration and to determine the extent and length of the stakes required to prevent such damage. Heat transfer relations were used to determine the overall performance of the condenser at the uprated condition. Steam flow velocities within the condenser were then determined based on the calculated heat transfer performance of the condenser. These velocities were used to evaluate the vibration criteria established from H. J. Conners, "Fluid-Elastic Vibration of Heat Exchanger Tube Arrays."

The plots of the Conners vibration parameters analyzed at winter conditions (i.e., worst case) indicate areas susceptible to fluid-elastic vibration. From this, the location and length of required stakes were determined.

QCNPS
The main condenser tubes were evaluated at the uprated conditions to determine which areas of the condenser tube bundle would be subject to potentially damaging tube vibration and to determine the extent and length of the stakes required to prevent such damage. Heat transfer relations were used to determine the overall performance of the condenser at the uprated condition. Steam flow velocities within the condenser were then determined based on the calculated heat transfer performance of the condenser. These velocities were used to evaluate the vibration criteria established from H. J. Conners, "Fluid-Elastic Vibration of Heat Exchanger Tube Arrays."

The plots of the Conners vibration parameters analyzed at winter conditions (i.e., worst case) indicate areas susceptible to fluid-elastic vibration. From this, the location and length of required stakes were determined. The currently installed staking was then compared to the stake locations and lengths determined in the analysis and was found to be adequate. No additional staking will be installed.

Question
13. A. In reference to Sections 3.5 and 4.1.2, provide a discussion of the evaluation of piping systems attached to the torus shell, vent penetrations, pumps, and valves, that are affected by increased torus temperature and changes in LOCA dynamic loads (pool swell, condensation oscillation, and chugging) and increased temperature and flow in the main steam and feedwater systems due to the proposed power uprate.

B. Identify supports and piping systems that require modifications as a result of the proposed extended power uprate.

Response

B. For piping systems other than MS, refer to the responses to Questions 8, 9, and 12.

The EPU does not affect design basis loads for the MS system. However, the MS system flow will increase by approximately 20% for EPU. A review of the increase in flow related loads associated with EPU indicates that piping loads due to the dynamic effects of the TSV fast closure, which is not included in the design basis loads, results in significant loads for the MS piping and supports.

DNPS and QCNPS are pre-General Design Criteria Plant (GDC) plants and were designed to USAS B31.1 – 1967, which required consideration of the most severe condition of coincident pressure, temperature, and loading. B31.1 – 1967 required that the plant transient dynamic load for safety valve opening be included in the design requirements. The Standard Review Plan (SRP), Section 10.3, "Main Steam Supply System," Revision 3, stated that main steam systems must be designed to withstand the effects of rapid valve closure. However Subsection V, "Implementation," of SRP Section 10.3 states that currently licensed plants (i.e., prior to 1984) do not need to adhere to this requirement. Thus, neither the GDC nor SRP requirements regarding consideration of transient dynamic loads due to TSV closure have been applied to DNPS or QCNPS.

Even though consideration of TSV loads was determined to be beyond the design basis, it is prudent to address these loads. The EPU evaluation approach for the TSV loads is based on an acceptance criteria for the TSV loads which are less restrictive than the current application of the ASME and American Institute for Steel Construction (AISC) codes, but which ensure that no permanent deformation of the piping, piping supports or supporting structural steel will occur as a result of the event.

Under EPU conditions the TSV closure loads were analyzed and modifications were implemented to ensure that the TSV closure does not result in MS piping failure. Since there is no current licensing basis for the acceptance criteria for the TSV loads, load combinations and acceptance criteria for the TSV loads were developed for the EPU evaluations. The MS piping, pipe supports, and supporting structures were evaluated for the TSV fluid transient loads in combination with pressure, deadweight, thermal, safety relief valve (SRV), and pipe break loads, as appropriate. Since a seismic event may cause a unit trip and a TSV closure, the TSV transient loads were also considered concurrent with applicable seismic loads. Since the TSV closure event is considered beyond the current licensing basis, a TSV event was considered to occur concurrently with the SSE only. The evaluation method is to demonstrate pressure boundary integrity of the piping and associated member/component evaluated to ensure that no gross deformation or integrity failure occurs. Also, due to the time relationships between the significant loads resulting from TSV, SRV discharge, and pipe break events (i.e., LOCA), no combination of these loads is required.

To demonstrate piping pressure boundary integrity subsequent to a TSV closure event, the piping, pipe supports and supporting structures were evaluated for the following additional loading combinations (LC).

Piping:
LC 1 Dead Load + Pressure + TSV Loads
LC 2 Dead Load + Pressure + $[(\text{TSV Loads})^2 + (\text{SSE Loads})^2]^{1/2}$

Pipe Supports and Pipe Support Structures:
LC 3 Dead Load + Operating Thermal Loads + TSV Loads
LC 4 Dead Load + Operating Thermal Loads + $[(\text{TSV Loads})^2 + (\text{SSE Loads})^2]^{1/2}$

The TSV fluid transient loads were generated utilizing the representative and bounding effective closing time for the TSV. For dynamic load combinations, oscillator (i.e., piping system) damping were considered to be 2% when considering TSV alone (i.e., LC 1) and 3% when combined with seismic (i.e., LC 2), in accordance with guidance contained in Reg. Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants." Seismic damping values are based on the values stipulated in the Updated Final Safety Analysis Report (UFSAR).

For evaluation of the supporting drywell steel, where supports from different main steam lines are attached to the same drywell steel, the TSV loads were combined by the square root of the sum of the squares (SRSS) method. This is due to the variation in actuation time, which results in the pressure wave for different MS lines being out-of-phase with the peak loads occurring at different times.

Design Criteria for Structural Steel and Pipe Support Evaluations
LC 3 – Dead load + Operating Thermal Loads + TSV Loads
Acceptance criteria: The allowable stresses shall be limited to 1.33 x Normal AISC Allowable stresses.

The following table summarizes the acceptance criteria for the load combinations listed above.

Attachment C
Additional Mechanical Systems 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

APPLICABLE TSV LOAD COMBINATIONS	**ACCEPTANCE CRITERIA**
STRUCTURAL & AUXILIARY STEEL	
DW + TH* + TR**	NORMAL 1.33 x AISC Allowable
DW + TH + $(SSE^2 + TR^2)^{1/2}$	FAULTED 1.60 x AISC Allowable < 0.95 x Fy***
EXPANSION ANCHOR BOLTS	
DW + TH + TR	SAFETY FACTOR = 4
DW + TH + $(SSE^2 + TR^2)^{1/2}$	SAFETY FACTOR = 2
PIPE SUPPORT COMPONENTS	
DW + TH + TR	ASME LEVEL C
DW + TH + $(SSE^2 + TR^2)^{1/2}$	ASME LEVEL D
PIPING	
DW + P + TR	ASME Level C
DW + P + $(SSE^2 + TR^2)^{1/2}$	ASME Level D

*TH = thermal loads
*TR = transient Loads such as TSV
*** Plastic section modulus can be used to determine the section stresses but must meet ductility criteria.

LC 4 – Dead Load + Operating Thermal Loads + SSE Loads + TSV Loads

Structural Steel Members Acceptance Criteria

Stress	Design Limit
Bending	1.6 x AISC allowable based on plastic section modulus with stresses not to exceed 0.95 x Fy. For this to be used, the section should satisfy the compact section criteria and lateral bracing requirements of the AISC Code. AISC LRFD Specification may be consulted to obtain further clarifications.
Axial	1.6 x AISC allowable not < 0.95 x Fy
Shear	$0.95 \times Fy / (3)^{1/2} = 0.548 \times Fy$

Plate Materials Acceptance Criteria

Stress	Design Limit
Bending about Weak Axis	0.95 x Fy based on plastic section modulus
Bending about Strong Axis	0.95 x Fy based on plastic section modulus or 1.0 x Fcr based on elastic section modulus, whichever is smaller.
Shear	0.95 x Fy / $(3)^{1/2}$ = 0.548 x Fy

Bolts Acceptance Criteria
1.60 x AISC Allowables.

Welds Acceptance Criteria
1.60 x AISC Allowables. The base metal shear for welds other than fillets shall not exceed 0.548 x Fy of the base metal. Base metal stress shall not govern for fillet welds.

Where the MS pipe supports combined loads as defined in combinations LC3 and LC4 do not exceed the original design basis loads (i.e., LC3 compared to operating basis earthquake (OBE) loads, and LC4 compared to SSE loads), the supporting structure was not reevaluated for the beyond design basis combinations.

The maximum stress ratios for each of the MS piping subsystems impacted by the TSV loads are provided in Table 13-1. The resultant pipe supports and drywell steel modifications are summarized in the response to Question 13B. With the modifications, the MS piping, pipe supports, and supporting drywell steel meet the above acceptance criteria. In addition, the current design and license basis criteria are met for the EPU conditions.

B. Table 13-2 identifies supports and piping systems that require modifications as a result of the extended power uprate.

Question
14. In Appendix G of the submittal, you indicated that restriction orifices to the stator water cooling system will be resized to accommodate the increased heat load. Additional cooling towers will be installed to ensure that the temperature of the water released to the environment remains within existing limits.

Confirm whether the proposed power uprate will increase the accident temperature, pressure and sub-compartment pressurization that affect the design basis analyses for steel and concrete in the containment, steam tunnel and the spent fuel pool. If the structural steel and concrete will *be affected, provide the design basis margin and margins after considering increased accident loading due to the proposed power uprate.*

Response
The EPU accident temperatures and pressures are bounded by the original structural design temperatures and pressures of the containment and containment sub-compartments, including the pressure suppression system and torus. Refer to PUSAR Sections 4.1.1, "Containment Pressure and Temperature Response," and 4.1.2, "Containment Dynamic Loads."

Temperatures and pressures due to feedwater and RWCU HELBs at EPU conditions increased slightly in some sub-compartments outside the containment, including the main steam tunnel (refer to PUSAR Table 10-1). The subcompartment structures were evaluated and are adequate as designed for the slightly increased pressures and temperatures.

Maximum Structural Margin Changes

Structure	**Interaction Ratio (IC)***	
	Pre-EPU	**EPU**
Concrete Sub-Compartments	0.946	0.995
Corner Room Structural Steel	0.62	0.83

* Maximum Allowable Interaction Ratio is 1.0.

The maximum EPU temperatures and pressures for the fuel pool structure and fuel racks are unchanged from the pre-EPU conditions (refer to PUSAR Table 6-2).

Attachment C
Additional Mechanical Systems 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

Table 9A-1 Large Bore Torus Water Piping Stress Results Dresden Unit 2

Piping Model	Description	Code	Pre-EPU Stress (psi)	(1)EPU Stress (psi)	Allowable Stress (psi)	Stress Ratio
D2.02	ECCS Ring Header	EQ. 10a, ASME Class II	37132	42126	45000	0.94
D2.03/D2.04	LPCI/CS Suction	102.3.2d, ANSI B31.1	33906	37007	37888	0.98
D2.08	LPCI Discharge	EQ. 10a, ASME Class II	33844	14700	45000	0.33
D2.05	HPCI Suction	EQ. 10a, ASME Class II	32241	32241	45000	0.72
D2.09.1	LPCI/CS Discharge	EQ. 10a, ASME Class II	25502	44159	45000	0.98
D2.09.2	CS Discharge	102.3.2c, ANSI B31.1	5384	7458	27000	0.28
D2.10	Vacuum Relief	EQ. 10a, ASME Class II	8049	9131	45000	0.20
D2.11	Pressure Suppression	EQ. 10a, ASME Class II	28247	28247	45000	0.63
D2.12	HPCI Turbine Exhaust	EQ. 10a, ASME Class II	13666	18931	45000	0.42
D2.13.1 (Internal)	LPCI Discharge	EQ. 10a, ASME Class II	29619	35435	45000	0.79
D2.13.1 (External)	LPCI Discharge	EQ. 10a, ASME Class II	25205	34916	45000	0.78
D2.13.2/D2.14.2	LPCI Discharge	EQ. 10a, ASME Class II	26010	42786	45000	0.95
D2.14.1 (Internal)	LPCI Discharge	EQ. 10a, ASME Class II	24283	29051	45000	0.65
D2.14.1 (External)	LPCI Discharge	EQ. 10a, ASME Class II	28969	40130	45000	0.89
D2-LPCI-09C	LPCI Discharge	102.3.2c, ANSI B31.1	23802	11601	27000	0.43
D2-LPCI-10C	LPCI Discharge	102.3.2c, ANSI B31.1	23871	11635	27000	0.43
D2-LPCI-12C (2)	Drywell Spray Header	102.3.2c, ANSI B31.1	0	0	27000	0.00
D2-LPCI-13C (2)	Drywell Spray Header	102.3.2c, ANSI B31.1	0	0	27000	0.00
D2-COSP-02B(C)	CS Discharge, Inside Drywell	102.3.2c, ANSI B31.1	7305	10119	27000	0.37
D2COSP-04C	CS Discharge	102.3.2d, ANSI B31.1	39173	32090	37500	0.86
D2-COSP-01B(C)	CS Discharge, Inside Drywell	102.3.2c, ANSI B31.1	15026	20815	27000	0.77

(1) Calculated Stress is for TE2 + THAM2 + TD4, where TE2 is thermal expansion, THAM2 is thermal anchor movements, and TD4 is torus displacement. All loads are based on the long term post-LOCA conditions associated with the EPU.

(2) Thermal stress is considered negligible for the torus spray header since the spray Header and the torus expand uniformly.

Attachment C
Additional Mechanical Systems 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

Table 9A-2 Large Bore Torus Water Piping Stress Results Dresden Unit 3

Piping Model	Description	Code	Pre-EPU Stress (psi)	(1)EPU Stress (psi)	Allowable Stress (psi)	Stress Ratio
D3.02	ECCS Ring Header	EQ. 10a ASME CL II	30074	35979	45000	0.80
D3.03/D3.06	LPCI / CS Suction	EQ. 10a ASME CL II	30158	41600	45000	0.92
D3.04/D3.07	LPCI / CS Suction	EQ. 10a ASME CL II	27654	44308	45000	0.98
D3.08.1/08.3	LPCI Discharge	EQ. 10a ASME CL II	29299	34284	45000	0.76
D3.08.2	LPCI Discharge	EQ. 10a ASME CL II	7324	10146	45000	0.23
D3.05	HPCI Suction	EQ. 10a ASME CL II	10503	10503	45000	0.23
D3.09.1	LPCI/CS Discharge	EQ. 10a ASME CL II	18605	32216	45000	0.72
D3.09.2	CS Discharge	EQ. 10a ASME CL II	12080	16734	45000	0.37
D3.09.3	CS Discharge	EQ. 10a ASME CL II	8706	12060	45000	0.27
D3.10	Vacuum Relief	EQ. 10a ASME CL II	18021	24964	45000	0.55
D3.11	Pressure Suppression	EQ. 10a ASME CL II	25427	14001	45000	0.31
D3.12 (Internal)	HPCI Turbine Exhaust	EQ. 10a ASME CL II	19916	27589	45000	0.61
D3.12 (External)	HPCI Turbine Exhaust	EQ. 10a ASME CL II	19916	27589	45000	0.61
D3.13.1 (Internal)	LPCI Discharge	EQ. 10a ASME CL II	26648	31881	45000	0.71
D3.13.1 (External)	LPCI Discharge	EQ. 10a ASME CL II	24088	33368	45000	0.74
D3.13.3	LPCI Discharge	EQ. 10a ASME CL II	14055	18493	45000	0.41
D3.13.2/D3.14.2	LPCI Discharge	EQ. 10a ASME CL II	14079	23160	45000	0.51
D.3.14.1 (Internal)	LPCI Discharge	EQ. 10a ASME CL II	31549	37744	45000	0.84
D.3.14.1 (External)	LPCI Discharge	EQ. 10a ASME CL II	31359	43440	45000	0.96
D3.14.3	LPCI Discharge	EQ. 10a ASME CL II	20662	25828	45000	0.57
D3-LPCI-11C (2)	Drywell Spray Header	102.3.2c, ANSI B31.1	0	0	27000	0.00
D3-LPCI-12C (2)	Drywell Spray Header	102.3.2c, ANSI B31.1	0	0	27000	0.00
D3-COSP-RP01	CS Discharge, Inside Drywell	EQ. 12 ASME CL I	N/A	26156	60000	0.44
D3-COSP-RP02	CS Discharge, Inside Drywell	EQ. 12 ASME CL I	N/A	5020	52620	0.10
D3-RRCI-RP01	Recirc	EQ. 10a ASME CL II	27772	13053	45000	0.29
D3-RRCI-RP02	Recirc	EQ. 10a ASME CL II	15026	7062	45000	0.16

(1) Calculated Stress is for TE2 + THAM2 + TD4, where TE2 is thermal expansion, THAM2 is thermal anchor movements, and TD4 is torus displacement. All loads are based on the long term post-LOCA conditions associated with the EPU.

(2) Thermal stress is considered negligible for the torus spray header since the spray header and the torus expand uniformly.

Attachment C
Additional Mechanical Systems 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

Table 9A-1QC Large Bore Torus Water Piping Stress Results Quad Cities Unit 1

Piping Model	Description	Code	Pre-EPU Stress (psi)	(1)EPU Stress (psi)	Allowable Stress (psi)	Stress Ratio
Q1.02	ECCS Ringheader	Eq 10a, ASME Class II	15301	16780	45000	0.37
Q1.03	RCIC Suction	Eq 10a, ASME Class II	22721	24917	45000	0.55
Q1.04	HPCI Suction	Eq 10a, ASME Class II	11953	16558	45000	0.37
Q1.05	RHR A/B Suction	Eq 10a, ASME Class II	50190	44660	52500	0.85
Q1.06	RHR C/D Suction	Eq 10a, ASME Class II	32627	35781	45000	0.80
Q1.07	Core Spray Suction	Eq 10a, ASME Class II	27998	30704	45000	0.68
Q1.08	Vacuum Relief	Eq 10a, ASME Class II	36037	43509	45000	0.97
Q1.09.1	RHR A/B Discharge	Eq 10a, ASME Class II	37168	40761	45000	0.91
Q1.09.2	RHR A/B Discharge	Eq 10a, ASME Class II	15316	18324	45000	0.41
Q1.09.3	RHR A/B Discharge	Eq 10a, ASME Class II	15316	18324	45000	0.41
Q1.10.1	CS Discharge	Eq 10a, ASME Class II	13727	15054	45000	0.33
Q1.10.2	CS Discharge	Eq 10a, ASME Class II	34021	37310	45000	0.83
Q1.11.1	RHR C/D Discharge	Eq 10a, ASME Class II	29089	31901	45000	0.71
Q1.11.2	RHR C/D Discharge	Eq 10a, ASME Class II	29300	35375	45000	0.79
Q1.11.3	RHR C/D Discharge	Eq 10a, ASME Class II	19350	20372	45000	0.45
Q1.13	HPCI Turbine Exhst	Eq 10a, ASME Class II	20253	22211	45000	0.49
Q1.14	RCIC Turbine Exhst	Eq 10a, ASME Class II	16244	22502	45000	0.50
Q1.15	Pressure Suppression	Eq 10a, ASME Class II	18288	10070	45000	0.22
Q1-RHRS-14B(C)	RHR Fuel Pool Cooling	102.3.2d, ANSI B31.1	21923	26228	27000	0.97
Q1-RHRS-09C	RHR Spray Header	102.3.2d, ANSI B31.1	16381	15796	27000	0.59
EMD-066699	RHR to Recirc	See Note 2				
Q1-COSP-01C	CS Disch Inside drywell	See Note 2				
Q1-COSP-02C	CS Disch Inside drywell	See Note 2				

(1) Calculated Stress is for TE2 + THAM2 + TD4, where TE2 is thermal expansion, THAM2 is thermal anchor movements, and TD4 is torus displacement. All loads are based on the long term post-LOCA conditions associated with the EPU.

(2) EPU condition does not control since analyzed at a temperature greater than 201.6 °F.

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Attachment C
Additional Mechanical Systems 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

Table 9A-2QC Large Bore Torus Water Piping Stress Results Quad Cities Unit 2

Piping Model	Description	Code	Pre-EPU Stress (psi)	(1)EPU Stress (psi)	Allowable Stress (psi)	Stress Ratio
Q2.02	ECCS Ringheader	Eq 10a, ASME Class II	29687	32557	45000	0.72
Q2.03	RCIC Suction	Eq 10a, ASME Class II	8234	9030	45000	0.20
Q2.04	HPCI Suction	Eq 10a, ASME Class II	26154	28682	45000	0.64
Q2.05	RHR A/B Suction	Eq 10a, ASME Class II	18020	19762	45000	0.44
Q2.06	RHR C/D Suction	Eq 10a, ASME Class II	22705	24975	45000	0.56
Q2.07	Core Spray Suction	Eq 10a, ASME Class II	27808	38521	45000	0.86
Q2.08	Vacuum Relief	Eq 10a, ASME Class II	25128	30338	45000	0.67
Q2.09.1	RHR A/B Discharge	Eq 10a, ASME Class II	23098	37996	45000	0.84
Q2.09.2	RHR A/B Discharge	Eq 10a, ASME Class II	22752	27220	45000	0.60
Q2.09.3	RHR A/B Discharge	Eq 10a, ASME Class II	22752	27220	45000	0.60
Q2.10.1	CS Discharge	Eq 10a, ASME Class II	18442	20225	45000	0.45
Q2.10.2	CS Discharge	Eq 10a, ASME Class II	5975	6553	45000	0.15
Q2.10.3	CS Discharge	Eq 10a, ASME Class II	8300	9102	45000	0.20
Q2.11.1	RHR C/D Discharge	Eq 10a, ASME Class II	35941	39415	45000	0.88
Q2.11.2	RHR C/D Discharge	Eq 10a, ASME Class II	29749	35591	45000	0.79
Q2.11.3	RHR C/D Discharge	Eq 10a, ASME Class II	23230	24457	45000	0.54
Q2.13	HPCI Turbine Exhst	Eq 10a, ASME Class II	16819	23299	45000	0.52
Q2.14	RCIC Turbine Exhst	Eq 10a, ASME Class II	7500	10500	45000	0.23
Q2.15	Pressure Supp.	Eq 10a, ASME Class II	18168	10004	45000	0.22
Q2-RHRS-09B(C)	RHR Fuel Pool Cooling	102.3.2d, ANSI B31.1	13997	14855	27000	0.55
Q2-RHRS-09C	RHR Spray Header	See Note 2				
EMD-066794	RHR to Recirc	See Note 2				
EMD-067695	CS Disch inside drywell	102.3.2d, ANSI B31.1	19600	19600	26400	0.74

(1) Calculated Stress is for TE2 + THAM2 + TD4, where TE2 is thermal expansion, THAM2 is thermal anchor movements, and TD4 is torus displacement. All loads are based on the long term post-LOCA conditions associated with the EPU.

(2) EPU condition does not control since analyzed at a temperature greater than 201.6 °F.

Table 9A-3 Small Bore Torus Water Piping Stress Results Dresden Unit 2

Calculation Number	System Identification***	Pre-EPU Stress (psi)	EPU Stress (psi)	Allowable Stress (psi)	Stress Ratio (EPU/Allowable)
27.0200.2053.007	PS	20524	28431	45000	0.63
27.0200.2053.009	PS	24658	34158	45000	0.76
27.0200.2053.010	DAP	27712	38388	45000	0.85
27.0200.2053.013	PS	31280	43331	45000	0.96
27.0200.2053.014	PS	24243	33583	45000	0.75
27.0200.2053.015	PS	24243	33583	45000	0.75
27.0200.2053.016	PS	35205	19385	45000	0.43
27.0200.2053.028	N	35284	35284	45000	0.78
27.0200.2053.030	Core Spray	3514	4868	45000	0.11
27.0200.2053.040	Core Spray	16638	27370	45000	0.61
27.0200.2053.041	Core Spray	16527	27187	45000	0.60
27.0200.2053.043	LPCI	22329	30932	45000	0.69
27.0200.2053.051	LPCI	25552	35396	45000	0.79
27.0200.2053.059	LPCI	23592	38809	45000	0.86
27.0200.2053.061	LPCI	1651	5743	45000	0.13
27.0200.2053.062	LPCI	21879	35113	45000	0.78
27.0200.2053.063	LPCI	30095	30614	45000	0.68
27.0200.2053.074	LPCI	17934	36398	45000	0.81
27.0200.2053.077	LPCI	20924	33580	45000	0.75
27.0200.2053.078	LPCI	26073	41844	45000	0.93
27.0200.2053.079	LPCI	36901	34547	45000	0.77
27.0200.2053.089	HPCI	23780	41177	45000	0.92
27.0200.2053.090	HPCI	15108	24853	45000	0.55
27.0200.2053.102	CAM	38584	53449	56400	0.95
27.0200.2053.103	CAM	40117	55573	56400	0.99
27.0200.2053.104	ACAD	33094	43185	45000	0.96
27.0200.2053.105	ACAD	33118	44904	45000	1.00
D2-LPCI-02B(C)/Analysis	LPCI	34910	41853	45000	0.93

*** PS = Pressure Suppression
DAP = Drywell Air Particulate Sampling
LPCI = Low Pressure Coolant Injection
HPCI = High Pressure Coolant Injection
CAM = Containment Atmosphere Monitoring
ACAD = Atmosphere Containment Atmosphere Dilution
N = Nitrogen Inerting and Drywell Oxygen Sampling

Table 9A-4 Small Bore Torus Water Piping Stress Results Dresden Unit 3

Calculation Number	**System** Identification***	**Pre-EPU Stress (psi)**	**EPU Stress (psi)**	**Allowable Stress (psi)**	**Stress Ratio (EPU/Allowable)**
27.0200.2058.007	PS	30656	42467	45000	0.94
27.0200.2058.008	PS	27414	37976	45000	0.84
27.0200.2058.009	DAP	34792	48196	56400	0.85
27.0200.2058.013	PS	29963	36795	45000	0.82
27.0200.2058.014	PS	15562	20354	45000	0.45
27.0200.2058.015	PS	11961	12220	45000	0.27
27.0200.2058.016	PS	33689	18550	45000	0.41
27.0200.2058.049	Core Spray	29989	44889	45000	1.00
27.0200.2058.050	Core Spray	20314	35175	45000	0.78
27.0200.2058.051	LPCI	2047	2836	45000	0.06
27.0200.2058.052	LPCI	14702	20366	45000	0.45
27.0200.2058.061	LPCI	6963	9646	45000	0.21
27.0200.2058.062	LPCI	26056	36094	45000	0.80
27.0200.2058.075	LPCI	22376	38746	45000	0.86
27.0200.2058.089	LPCI	20364	35262	45000	0.78
27.0200.2058.095	LPCI	26166	41993	45000	0.93
27.0200.2058.113	HPCI	25906	37122	45000	0.82
27.0200.2058.114	HPCI	15108	24853	45000	0.55
27.0200.2058.120	CAM	28674	37884	56400	0.67
27.0200.2058.121	CAM	24308	32009	56400	0.57
27.0200.2058.122	ACAD	24684	32738	45000	0.73
27.0200.2058.123	ACAD	32547	43121	45000	0.96
D3-LPCI-02B(C)/Analysis	LPCI	11813	14766	45000	0.33

*** PS = Pressure Suppression
DAP = Drywell Air Particulate Sampling
LPCI = Low Pressure Coolant Injection
HPCI = High Pressure Coolant Injection
CAM = Containment Atmosphere Monitoring
ACAD = Atmosphere Containment Atmosphere Dilution

Attachment C
Additional Mechanical Systems 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

Table 9A-3QC Small Bore Torus Water Piping Stress Results Quad Cities Unit 1

Calculation Number	System Identification***	Pre-EPU Stress (psi)	EPU Stress (psi)	Allowable Stress (psi)	Stress Ratio (EPU/Allowable)
27.0200.1053.001	PS	15000	24675	45000	0.55
27.0200.1053.002	PS	19000	31255	45000	0.69
27.0200.1053.006	PS	26680	36959	45000	0.82
27.0200.1053.007	PS	14341	19866	45000	0.44
27.0200.1053.008	PS	25019	13776	45000	0.31
27.0200.1053.010	DAP	30454	42187	56400	0.75
27.0200.1053.011	PS	12658	17535	45000	0.39
27.0200.1053.012	PS	7524	10423	45000	0.23
27.0200.1053.019	Core Spray	24390	33787	45000	0.75
27.0200.1053.020	Core Spray	21771	30159	45000	0.67
QDC-1000-S-0456	RH	28749	31528	45000	0.70
27.0200.1053.043	RH	32295	42722	45000	0.95
27.0200.1053.047	RH	17654	24455	45000	0.54
27.0200.1053.059	HPCI	18205	25219	45000	0.56
Q1-HPCI-04B(C)	HPCI	13915	13915	45000	0.31
27.0200.1053.069	HPCI	15000	24675	45000	0.55
27.0200.1053.074	RCIC	7702	10669	45000	0.24
27.0200.1053.077	RCIC	41052	43639	45000	0.97
27.0200.1053.088	HPCI	15356	25261	45000	0.56
27,0200.1053.089	RCIC	16681	27440	45000	0.61
27.0200.1053.117	HPCI	28787	34756	45000	0.77

*** PS = Pressure Suppression
DAP = Drywell Air Particulate Sampling
RH = Residual Heat Removal
HPCI = High Pressure Coolant Injection
RCIC = Reactor Core Isolation Cooling

Table 9A-4QC Small Bore Torus Water Piping Stress Results Quad Cities Unit 2

Calculation Number	System Identification***	Pre-EPU Stress (psi)	EPU Stress (psi)	Allowable Stress (psi)	Stress Ratio (EPU/Allowable)
27.0200.1058.001	Instrument Air	25501	35326	56400	0.63
27.0200.1058.004	PS	26400	36571	45000	0.81
27.0200.1058.005	PS	28612	39635	45000	0.88
27.0200.1058.010	PS	27903	38653	45000	0.86
27.0200.1058.011	PS	11118	15401	45000	0.34
27.0200.1058.012	PS	25116	34792	45000	0.77
27.0200.1058.013	DAP	15686	21729	49800	0.44
27.0200.1058.017	PS/NO	4798	5793	45000	0.13
27.0200.1058.018	PS	30072	36057	45000	0.80
27.0200.1058.032	RH	29342	32178	45000	0.72
27.0200.1058.051	RH	21361	29591	45000	0.66
Q2-RHRS-08B(C)	RH	29147	18179	45000	0.40
27.0200.1058.059	HPCI	29547	40930	45000	0.91
Q2-HPCI-02B(C)	HPCI	12372	12372	45000	0.27
27.0200.1058.066	HPCI	31514	43655	45000	0.97
27.0200.1058.079	HPCI	26405	36578	45000	0.81
27.0200.1058.080	HPCI	31675	43878	45000	0.98
27.0200.1058.081	HPCI	32352	44816	45000	1.00
27.0200.1058.085	RCIC	5077	7033	45000	0.16
27.0200.1058.095	RCIC	25965	37967	45000	0.84
27.0200.1058.096	CAM	44552	34060	56400	0.60
27.0200.1058.097	CAM	19787	27410	56400	0.49
27.0200.1058.102	HPCI	27228	37718	45000	0.84
27.0200.1058.103	Core Spray	13011	14269	45000	0.32
27.0200.1058.104	Core Spray	17922	19654	45000	0.44
QDC-1400-M-033	Core Spray	6640	3121	27000	0.12
Q2-RHRS-06B(C)	RH	36459	36459	45000	0.81
QDC-1000-M-185	RH	21800	21800	27000	0.81

*** PS = Pressure Suppression
DAP = Drywell Air Particulate Sampling
RH = Residual Heat Removal
HPCI = High Pressure Coolant Injection
RCIC = Reactor Core Isolation Cooling
CAM = Containment Atmosphere Monitoring
NO = Drywell Nitrogen and Oxygen Analyzer

Table 12A-1
Balance of Plant Piping System Evaluation Method and Results
DNPS Units 2 & 3

Piping System	Evaluation Method	Evaluation Results
Main Steam (outside RCPB)	See the response to Question 13A	See the response to Question 13A
Feedwater (outside RCPB)	Increases < 5%	Pass*
Reactor Recirculation	Increases < 5%	Pass
Control Rod Drive	Increases < 5%	Pass
RPV Bottom Head Drain	Increases < 5%	Pass
RPV Head Vent	Increases < 5%	Pass
Isolation Condenser	Increases < 5%	Pass
Shutdown Cooling	Increases < 5%	Pass
SRV Discharge	Increases < 5%	Pass
Reactor Water Clean Up	Increases < 5%	Pass
CCSW	Increases < 5%	Pass
Fuel Pool Cooling	Increases < 5%	Pass
Main Steam Drain Lines	Increases < 5%	Pass
Neutron Monitoring	Increases < 5%	Pass
MS Turbine By-Pass	Increases < 5%	Pass
Standby Liquid Control	Increases < 5%	Pass
Off Gas	Increases < 5%	Pass
Standby Gas	Increases < 5%	Pass
High Radiation Sampling	Increases < 5%	Pass
MS Cross Around Piping	Increases < 5%	Pass
Turbine Cross Around Piping	Increases < 5%	Pass
Condensate & Heater Drain	Increases < 5%	Pass

* FW flow increase factor 1.20, however system contains no fast acting valves and increase in flow is acceptable

Table 12A-1
Balance of Plant Piping System Evaluation Method and Results
QCNPS Units 1 & 2

Piping System	Evaluation Method	Evaluation Results
Main Steam (outside RCPB)	See the response to Question 13A	See the response to Question 13A
Feedwater (outside RCPB)	Increases < 5%	Pass*
Reactor Recirculation	Increases < 5%	Pass
Control Rod Drive	Increases < 5%	Pass
RPV Bottom Head Drain	Increases < 5%	Pass
RPV Head Vent	Increases < 5%	Pass
RCIC	Increases < 5%	Pass
SRV Discharge	Increases < 5%	Pass
Reactor Water Clean Up	Increases < 5%	Pass
CCSW	Increases < 5%	Pass
Fuel Pool Cooling	Increases < 5%	Pass
Main Steam Drain Lines	Increases < 5%	Pass
Neutron Monitoring	Increases < 5%	Pass
MS Turbine By-Pass	Increases < 5%	Pass
Standby Liquid Control	Increases < 5%	Pass
Off Gas	Increases < 5%	Pass
Standby Gas	Increases < 5%	Pass
High Radiation Sampling	Increases < 5%	Pass
MS Cross Around Piping	Increases < 5%	Pass
Turbine Cross Around Piping	Increases < 5%	Pass
Condensate & Heater Drain	Increases < 5%	Pass

* FW flow increase factor 1.20, however system contains no fast acting valves and increase in flow is acceptable

TABLE 13-1 DNPS

Unit	Subsystem	Code	Calculated Stress (psi)	Allowable Stress (psi)
DNPS Unit 2	MS Line A - RPV to drywell Pen	ASME Level C	24,991	27,000
		ASME Level D	26,766	36,000
	MS Line B - RPV to drywell Pen	ASME Level C	22,532	27,000
		ASME Level D	33,247	36,000
	MS Line C - RPV to drywell Pen	ASME Level C	14,256	27,000
		ASME Level D	25,368	36,000
	MS Line D - RPV to drywell Pen	ASME Level C	22,633	27,000
		ASME Level D	33,504	36,000
DNPS Unit 3	MS Line A - RPV to drywell Pen	ASME Level C	23,487	27,000
		ASME Level D	35,260	36,000
	MS Line B - RPV to drywell Pen	ASME Level C	21,856	27,000
		ASME Level D	34,102	36,000
	MS Line C - RPV to drywell Pen	ASME Level C	17,864	27,000
		ASME Level D	29,610	36,000
	MS Line D - RPV to drywell Pen	ASME Level C	23,607	27,000
		ASME Level D	33,385	36,000
DNPS Unit 2	MS Lines A, B, C & D Outside Drywell	ASME Level C	14,972	27,000
		ASME Level D	13,989	36,000
DNPS Unit 3	MS Lines A, B, C & D Outside Drywell	ASME Level C	14,972	27,000
		ASME Level D	13,989	36,000

ASME Level C = DW + PR +TSV
ASME Level D = DW + PR + SRSS(SSE + TSV)

DW = deadload stress (psi)
PR = pressure stress (psi)
TSV = turbine stop valve stress (psi)
SSE = safe shutdown earthquake stress (psi)

TABLE 13-1 QCNPS

Unit	Subsystem	Code	Calculated Stress (psi)	Allowable Stress (psi)
Quad Cities Unit 1	MS Line A - RPV to drywell Pen	ASME Level C	24,119	27,000
		ASME Level D	33,922	36,000
	MS Line B - RPV to drywell Pen	ASME Level C	20,139	27,000
		ASME Level D	33,733	36,000
	MS Line C - RPV to drywell Pen	ASME Level C	26,025	27,000
		ASME Level D	35,770	36,000
	MS Line D - RPV to drywell Pen	ASME Level C	21,000	27,000
		ASME Level D	35,306	36,000
Quad Cities Unit 2	MS Line A - RPV to drywell Pen	ASME Level C	25,291	27,000
		ASME Level D	35,336	36,000
	MS Line B - RPV to drywell Pen	ASME Level C	26,638	27,000
		ASME Level D	34,459	36,000
	MS Line C - RPV to drywell Pen	ASME Level C	22,441	27,000
		ASME Level D	34,546	36,000
	MS Line D - RPV to drywell Pen	ASME Level C	16,484	27,000
		ASME Level D	29,127	36,000
QCNPS Unit 1	MS Lines A, B, C & D Outside Drywell	ASME Level C	21,673	27,000
		ASME Level D	27,260	36,000
QCNPS Unit 2	MS Lines A, B, C & D Outside Drywell	ASME Level C	21,673	27,000
		ASME Level D	27,260	36,000

ASME Level C = DW + PR +TSV
ASME Level D = DW + PR + SRSS(SSE + TSV)

DW = deadload stress (psi)
PR = pressure stress (psi)
TSV = turbine stop valve stress (psi)
SSE = safe shutdown earthquake stress (psi)

Table 13-2 DNPS

SUPPORT NUMBER	MODIFICATION DESCRIPTION
DNPS Unit 2 - TAP SUPPORT MODIFICATIONS	
SUPPORT NUMBER	MODIFICATION DESCRIPTION
2-15-M321315	Change spring hanger loads
2-15-M321423	Revise baseplate mounting
2-15-M3381	Revise U-Bolt
2-14-M320924	Revise baseplate design and add new brace
2-14-M320808	Replace rigid strut with snubber
DNPS Unit 2 - MS PIPE SUPPORT AND DRYWELL STEEL MODIFICATIONS	
2-3001A-49	Replace snubber assembly and add stiffener angle and welds
2-02-2870SH1 2-02-2870SH2	Add two box frame supports at MS bypass loop in Turbine Building
2-02-2870SH3 2-02-2870SH4	Add lateral guides inside 2 G-line wall sleeves
DRYWELL STEEL	Strengthen various beam end connections using packing, bumper and stiffener plates
2-3001-H86 2-3001-H89	Remove existing pipe supports
DNPS Unit 3 - TAP SUPPORT MODIFICATIONS	
3-14-M340919	Increase the size of existing welds on support cleats
3-14-M340921	Install additional stiffener plates and associated welds
3-15-M340819	Add additional welds to existing support
3-15-M340827	Install additional stiffener plates and add additional welds to existing support
3-15-M340906	Add new brace with associated baseplate and anchor bolts
DNPS Unit 3 - MS PIPE SUPPORT AND DRYWELL STEEL MODIFICATIONS	
3-3001A-S2	Add new welds and stiffener plates to existing members
3-3001C-S2	Add new support member and welds and reduce length of snubber extension piece
DRYWELL STEEL	Strengthen various beam end connections using packing, bumper and stiffener plates
3-02-3870SH1 3-02-3870SH2	Add two box frame supports at MS bypass loop in Turbine Building
3-02-3870SH3 3-02-3870SH4	Add lateral guides inside 2 G-line wall sleeves
3-02-M778ASH26 3-02-M778ASH27	Add new supports for rerouting of MS drain line
3-3001-H86 3-3001-H89	Remove existing pipe supports

Table 13-2 QCNPS

SUPPORT NUMBER	MODIFICATION DESCRIPTION
QCNPS Unit 1 - MS PIPE SUPPORT AND DRYWELL STEEL MODIFICATIONS	
1-3001B-20-S1	Replace snubber assembly, replace support structure by tube steel members
1-3001B-20-S2	Relocate pipe clamp to accommodate new clamp for 1-3001B-20-S1
1-3001C-S2	Add new welds, replace a snubber
1-3001D-R1	Add new welds, replace support member
1-3001-988D-8-1 1-3001-988D-8-2 1-3001-988D-8-3 1-3001-988D-8-4	Add special LISEGA Clamps and horizontal and vertical struts to main steam lines
1-3059-988D-8-5 1-3059-988D-8-6	Add new supports for rerouting of MS equalizing line
DRYWELL STEEL	Strengthen various beam end connections using packing, bumper and stiffener plates
QCNPS Unit 2 - TAP SUPPORT MODIFICATIONS	
2-1810-07	Reset spring can displacements
2-1810-35	Add stiffener plate
QCNPS Unit 2 - MS PIPE SUPPORT AND DRYWELL STEEL MODIFICATIONS	
2-3001A-R4	Add stiffeners to existing steel beam
2-3001B-S2	Add new welds, strengthening structural beam
2-3001B-R1	Replace existing strut
2-3001C-R1	Replace existing strut
2-3001C-S2	Replace entire support structure by tube steel members and add stiffeners to steel beam
2-3001-1020D-6-1 2-3001-1020D-6-2 2-3001-1020D-6-3 2-3001-1020D-6-4	Add special LISEGA Clamps and horizontal and vertical struts to main steam lines
DRYWELL STEEL	Strengthen various beam end connections using packing, bumper and stiffener plates, replace bolting at 5 connections (EL. 593)

Attachment C
Additional Mechanical Systems 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

References:

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

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. Letter from U. S. NRC to O. D. Kingsley (Exelon Generation Company, LLC), "Issuance of Amendments; Increase in Reactor Power, Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2," dated May 4, 2001