From: "Daflucas, Ronda" <rdafluc@entergy.com>
To: "Rick Ennis" <RXE@nrc.gov>
Date: 9/18/05 7:31PM
Subject: BVY 05-086, EPU Suppl 34

Rick,

Entergy letter, BVY 05-086, EPU Supplement 34 was signed today.

Attached is the cover letter, Attachments 1, 2, 3, 5, 6, 7, 8.

I will follow with an additional E-mail providing Attachment 4, proprietary information.

Attachment 5 is the non-proprietary version of Attachment 4.

Ronda Daflucas
Vermont Yankee Project Manager, NRR
Entergy Nuclear Operations, Inc.
802-258-4232

CC: "Gucwa, Len" <LGUCW90@prod.entergy.com>, "Hobbs, Brian" <bhobbs@prod.entergy.com>, "Nichols, Craig" <cnicho1@prod.entergy.com>, "Dreyfuss, John" <jdreyfu@prod.entergy.com>

Mail Envelope Properties (432DF8D5.48B : 0 : 29835)

Subject: BVY 05-086, EPU Suppl 34
Creation Date: 9/18/05 7:29PM
From: "Daflucas, Ronda" <rdafluc@entergy.com>

Created By: rdafluc@entergy.com

Recipients
nrc.gov
owf4_po.OWFN_DO
RXE (Rick Ennis)

prod.entergy.com
jdreyfu CC (John Dreyfuss)
cnichol CC (Craig Nichols)
bhobbs CC (Brian Hobbs)
LGUCW90 CC (Len Gucwa)

Post Office	**Route**
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	prod.entergy.com

Files	**Size**	**Date & Time**
MESSAGE	409	09/18/05 07:29PM
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BVY 05-086 Attachment 7.pdf	197555	
BVY 05-086 Attachment 8.pdf	738147	
BVY 05-086 Attachment 5.pdf	865490	
BVY 05-086 letter_Attachments1-2-3 and 6.pdf		1209828
Mime.822	4127094	

Options

Expiration Date:	None
Priority:	Standard
Reply Requested:	No
Return Notification:	None
Concealed Subject:	No
Security:	Standard

Entergy

Entergy Nuclear Northeast
Entergy Nuclear Operations, Inc.
Vermont Yankee
P.O. Box 0500
185 Old Ferry Road
Brattleboro, VT 05302-0500
Tel 802 257 5271

September 18, 2005

Docket No. 50-271
BVY 05-086
TAC No. MC0761

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

Subject: **Vermont Yankee Nuclear Power Station**
Technical Specification Proposed Change No. 263 – Supplement No. 34
Extended Power Uprate – Additional Information

References:

1) Entergy letter to U.S. Nuclear Regulatory Commission, "Vermont Yankee Nuclear Power Station, License No. DPR-28 (Docket No. 50-271), Technical Specification Proposed Change No. 263, Extended Power Uprate," BVY 03-80, September 10, 2003

2) U.S. Nuclear Regulatory Commission (Richard B. Ennis) letter to Entergy Nuclear Operations, Inc. (Michael Kansler), "Request for Additional Information – Extended Power Uprate, Vermont Yankee Nuclear Power Station (TAC No. MC0761)," September 7, 2005

3) Entergy letter to U.S. Nuclear Regulatory Commission, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263 – Supplement No. 32, Extended Power Uprate – Additional Information," BVY 05-083, September 10, 2005

4) Entergy letter to U.S. Nuclear Regulatory Commission, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263 – Supplement No. 33, Extended Power Uprate – Response to Request for Additional Information," BVY 05-084, September 14, 2005

This letter provides additional information regarding the application by Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. (Entergy) for a license amendment (Reference 1) to increase the maximum authorized power level of the Vermont Yankee Nuclear Power Station (VYNPS) from 1593 megawatts thermal (MWt) to 1912 MWt.

The attachments to this letter provide supplemental information in response to requests for additional information from the NRC staff (Reference 2) and other supplemental information to update the application for a license amendment. As a result of recent discussions with the NRC staff and its recent audit of analytical methodologies of General Electric (GE) that are used for the design and evaluation of VYNPS' fuel, the NRC staff identified the need for additional information reflected in several of the requests for additional information (RAIs) contained in Reference 2. Because of the recency of the requests, one (Reference 2) RAI remains to be addressed (i.e., NRC RAI SRXB-A-68); the remaining RAI will be addressed in a submittal that will be made by September 23, 2005.

Attachment 1 to this letter is a revision to Exhibit EMEB-B-18-1, Rev. 1, Attachment 4 (regarding the steam dryer acoustic load uncertainty evaluation) that was provided to the NRC staff in Reference 4. Inadvertently, several figures were not included in the original submittal. The omitted figures include comparisons of power spectral densities for certain transmitter locations. Attachment 1 consists of thirty figures (EMEB-B-18-1-4-1 through EMEB-B-18-1-4-30) and supersedes, in its entirety, Exhibit EMEB-B-18-1, Rev. 1, Attachment 4 provided in Reference 4, Attachment 1 (Proprietary Information) and Attachment 8 (Non-Proprietary Version). Attachment 1 to this letter does not contain proprietary information.

In the response to RAI SRXB-A-66 (Reference 3), Entergy stated that certain tabulated data supporting the response to the RAI would be submitted to the NRC staff as Microsoft Excel spreadsheets. That information is included herein as Attachment 2 on a compact disk. The data contained on the compact disk is considered Proprietary Information to General Electric and is covered by the affidavit accompanying the response to SRXB-A-66 in Reference 3. An explanatory "Read Me" file (non-proprietary) contained on the CD is included in hardcopy as part of Attachment 2.

As a result of discussions with the NRC staff, Entergy is providing in Attachment 3 a more extensive response to RAI SRXB-A-64. This response supplements the response that was originally provided in Reference 3.

Attachment 4 contains responses to NRC Reactor Systems Branch RAIs SRXB-A-65 and SRXB-A-67 that were posed in Reference 2. These RAIs and the responses thereto contain Proprietary Information as defined by 10CFR2.390 and should be handled in accordance with the provisions of that regulation. Attachment 4 is considered to be Proprietary Information in its entirety. Attachment 5 is a non-proprietary version of Attachment 4. An affidavit provided by General Electric Company, supporting the proprietary nature of the document, is provided as Attachment 7.

Attachment 6 provides a response to RAI SRXB-A-71 that was asked in Reference 2.

Attachment 8 of this letter provides a copy of the demonstrated shutdown margin (SDM) calculation for the current operating cycle (i.e., cycle 24). This SDM calculation is referenced in the response to RAI SRXB-A-67, part (b).

There are no new regulatory commitments contained in this submittal.

This supplement to the license amendment request provides additional information to clarify Entergy's application for a license amendment and does not change the scope or conclusions in the original application, nor does it change Entergy's determination of no significant hazards consideration.

The following attachments are included in this submittal:

Attachment	Title
1	Revised Exhibit EMEB-B-18-1, Rev. 1, Attachment 4
2	RAI SRXB-A-66 Data (Compact Disk) (PROPRIETARY INFORMATION)
3	Supplemental Response to SRXB-A-64
4	Responses to RAIs SRXB-A-65 and SRXB-A-67 (Proprietary Information)
5	Responses to RAIs SRXB-A-65 and SRXB-A-67 (Non-Proprietary Version)
6	Response to RAI SRXB-A-71
7	General Electric Affidavit
8	Demonstrated Shutdown Margin

Entergy stands ready to support the NRC staff's review of this submittal and suggests meetings at your earliest convenience to resolve any remaining issues. If you have any questions or require additional information, please contact Mr. James DeVincentis at (802) 258-4236.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on September 18, 2005.

Sincerely,

Norman L. Rademacher
Director, Nuclear Safety Assurance
Vermont Yankee Nuclear Power Station

Attachments (8)

cc: (see next page)

cc: Mr. Richard B. Ennis, Project Manager
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Mail Stop O 8 B1
Washington, DC 20555

Mr. Samuel J. Collins (w/o attachments)
Regional Administrator, Region 1
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406-1415

USNRC Resident Inspector (w/o attachments)
Entergy Nuclear Vermont Yankee, LLC
P.O. Box 157
Vernon, Vermont 05354

Mr. David O'Brien, Commissioner (w/o proprietary information)
VT Department of Public Service
112 State Street – Drawer 20
Montpelier, Vermont 05620-2601

Attachment 1

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 263 – Supplement No. 34

Extended Power Uprate – Additional Information

Revised Exhibit EMEB-B-18-1, Rev. 1, Attachment 4

Total number of pages in Attachment 1 (excluding this cover sheet) is 30.

Exhibit EMEB-B-18-1, Rev. 1, VYNPS Steam Dryer Load Uncertainty
Attachment 4 (Rev 1)

PSD Comparison, QC2 Data vs. ACA Predictions plus Uncertainty

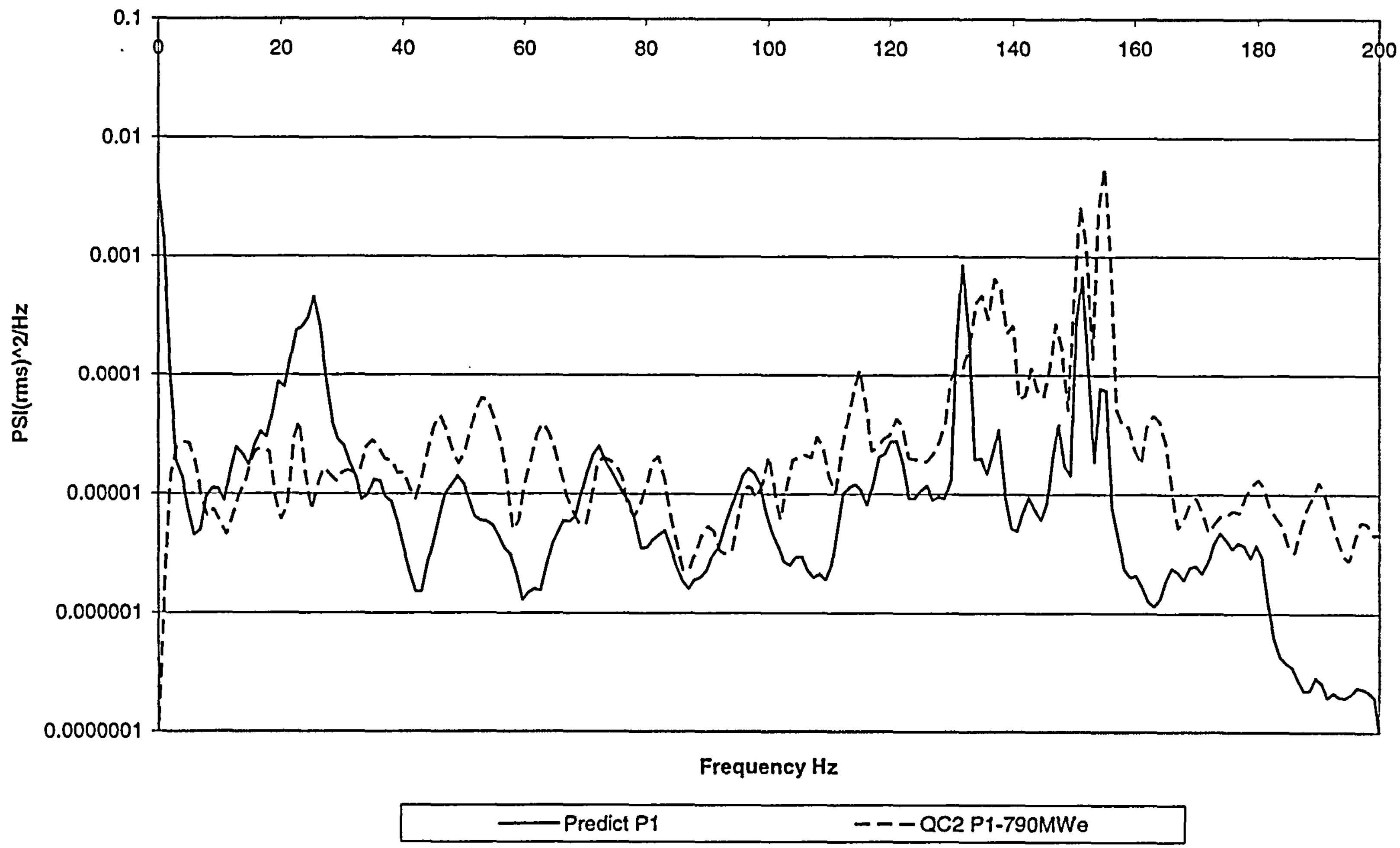

Figure EMEB-B-18-1-4-1

Exhibit EMEB-B-18-1, Rev. 1, VYNPS Steam Dryer Load Uncertainty
Attachment 4 (Rev 1)

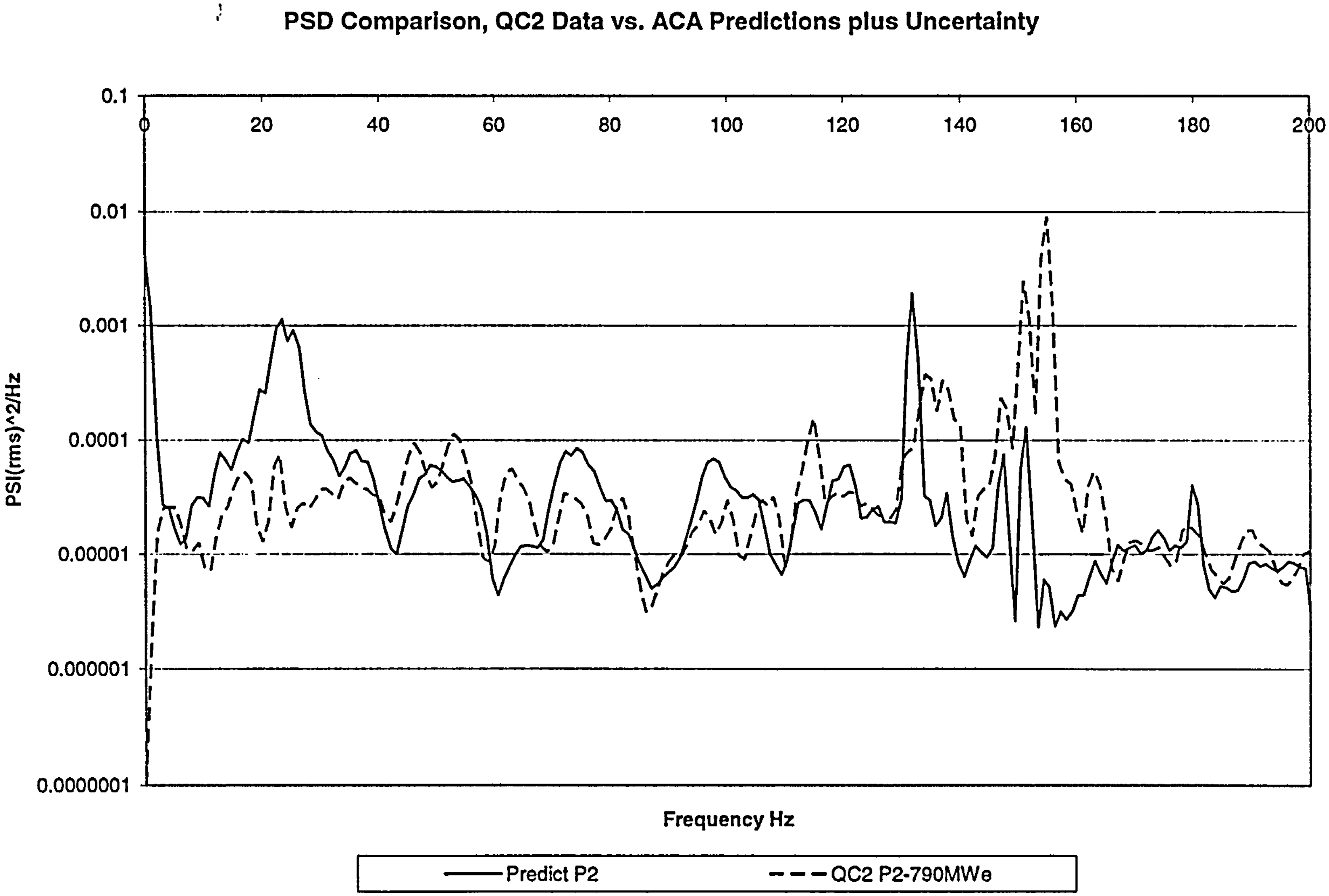

Figure EMEB-B-18-1-4-2

Exhibit EMEB-B-18-1, Rev. 1, VYNPS Steam Dryer Load Uncertainty
Attachment 4 (Rev 1)

PSD Comparison, QC2 Data vs. ACA Predictions plus Uncertainty

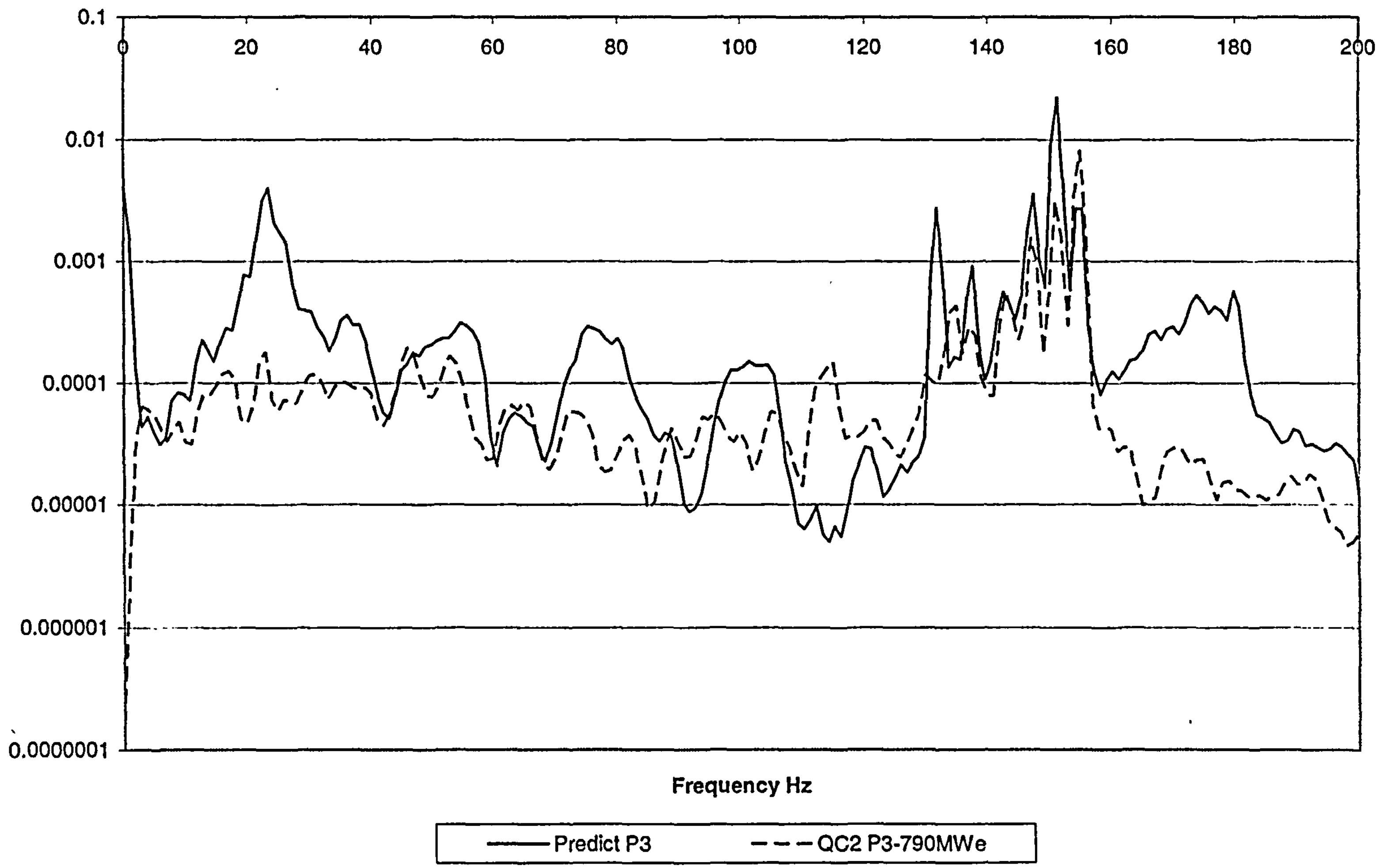

Figure EMEB-B-18-1-4-3

Exhibit EMEB-B-18-1, Rev. 1, VYNPS Steam Dryer Load Uncertainty
Attachment 4 (Rev 1)

PSD Comparison, QC2 Data vs. ACA Predictions plus Uncertainty

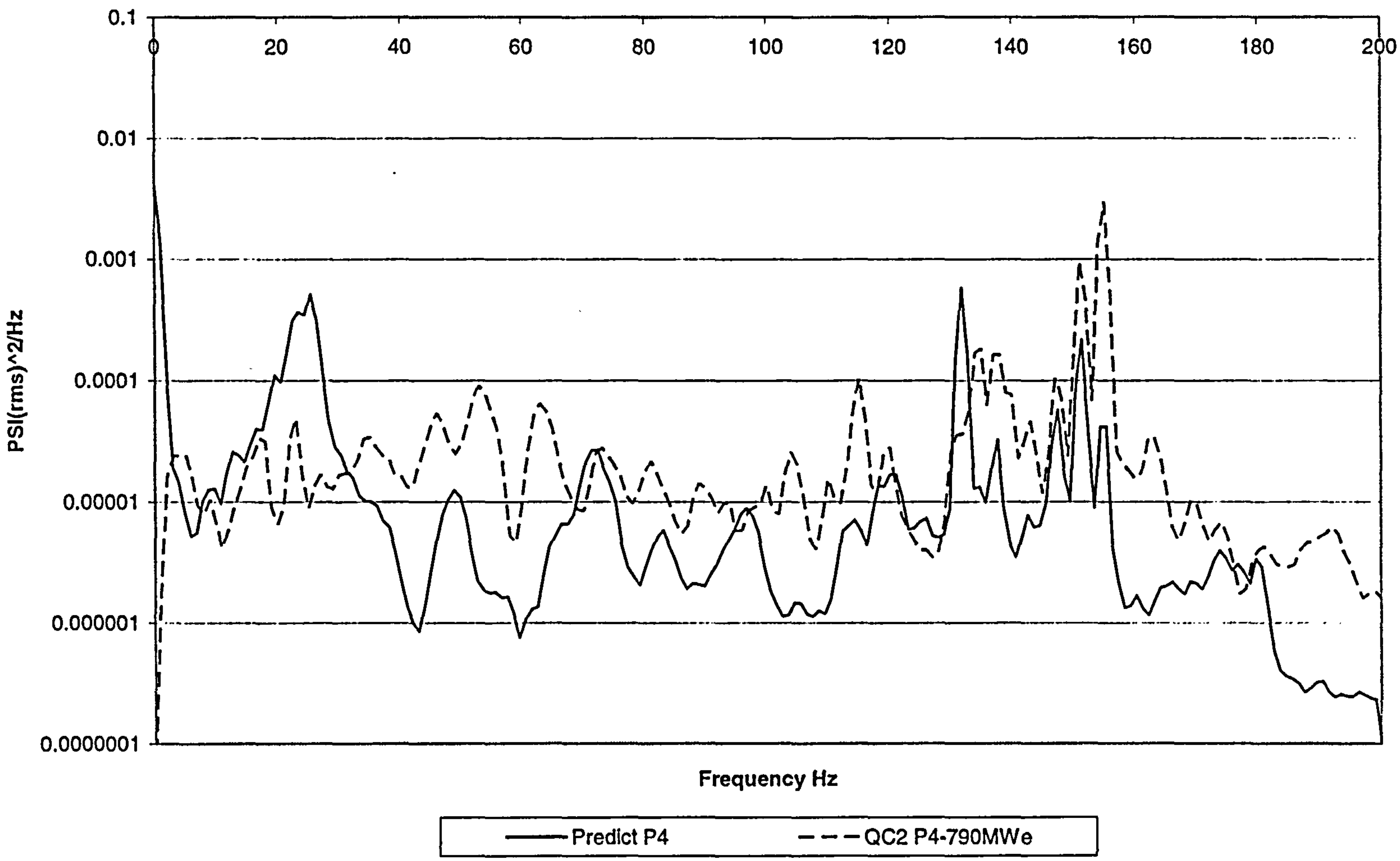

Figure EMEB-B-18-1-4-4

Exhibit EMEB-B-18-1, Rev. 1, VYNPS Steam Dryer Load Uncertainty
Attachment 4 (Rev 1)

PSD Comparison, QC2 Data vs. ACA Predictions plus Uncertainty

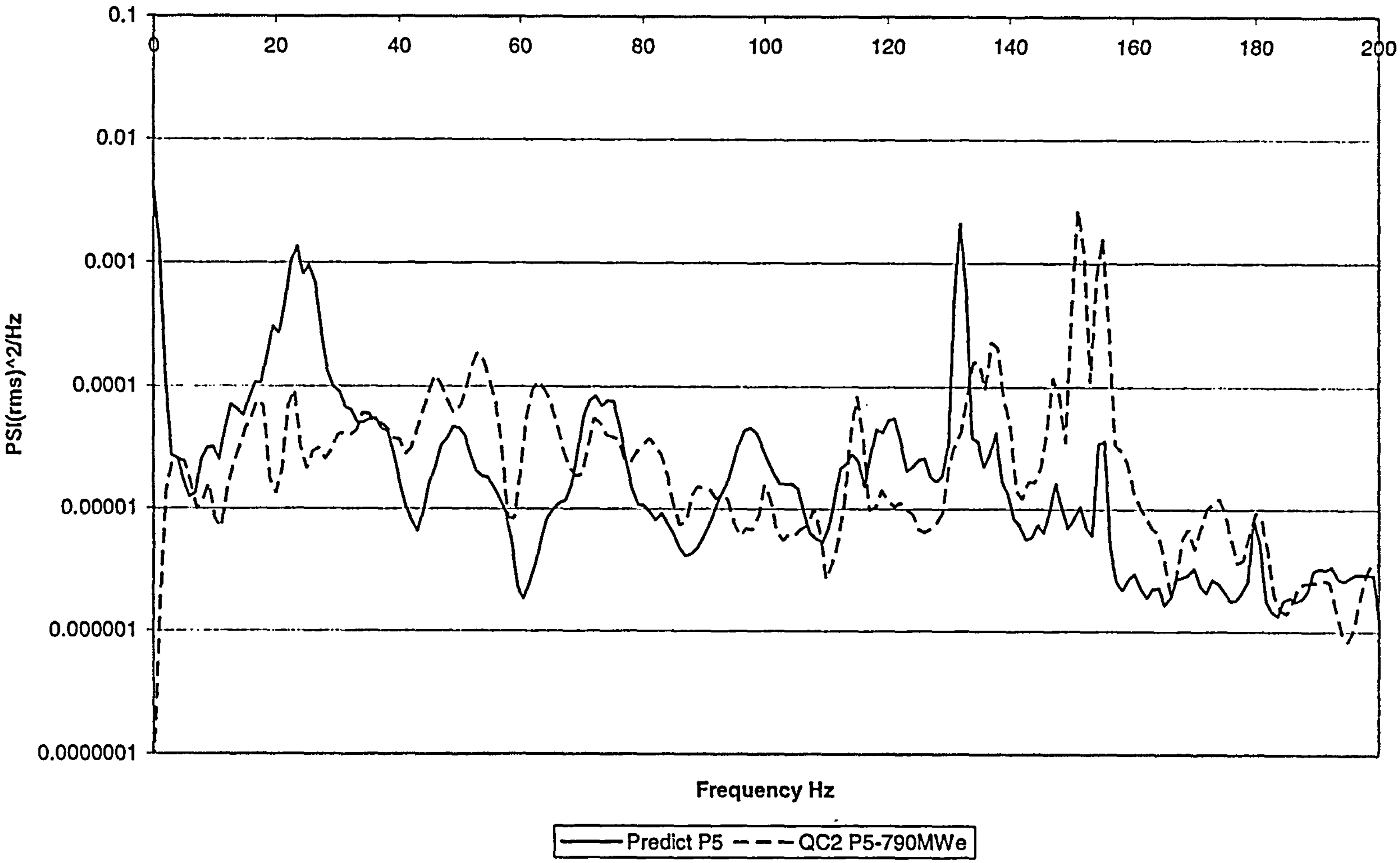

Figure EMEB-B-18-1-4-5

Exhibit EMEB-B-18-1, Rev. 1, VYNPS Steam Dryer Load Uncertainty
Attachment 4 (Rev 1)

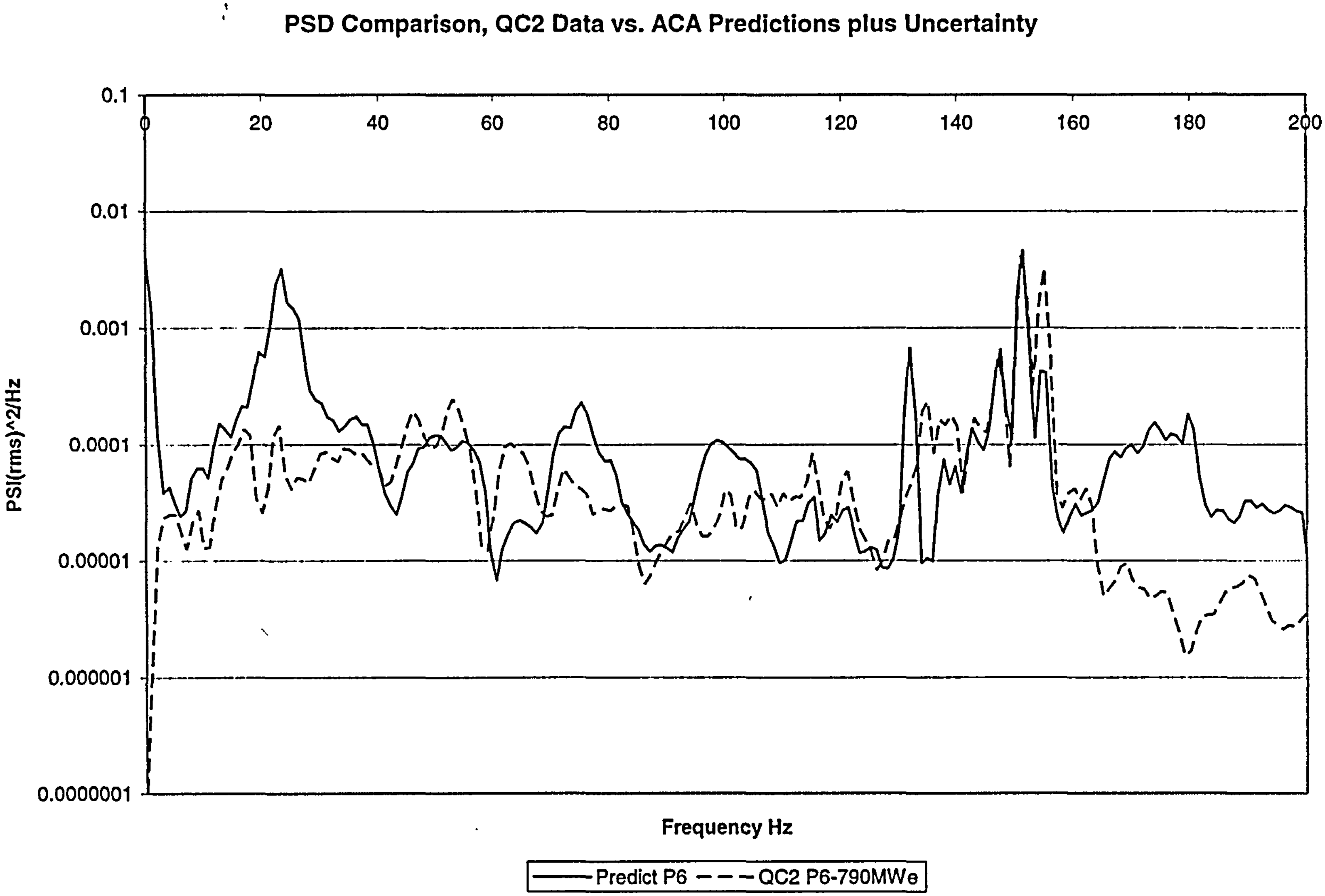

Figure EMEB-B-18-1-4-6

Exhibit EMEB-B-18-1, Rev. 1, VYNPS Steam Dryer Load Uncertainty
Attachment 4 (Rev 1)

PSD Comparison, QC2 Data vs. ACA Predictions plus Uncertainty

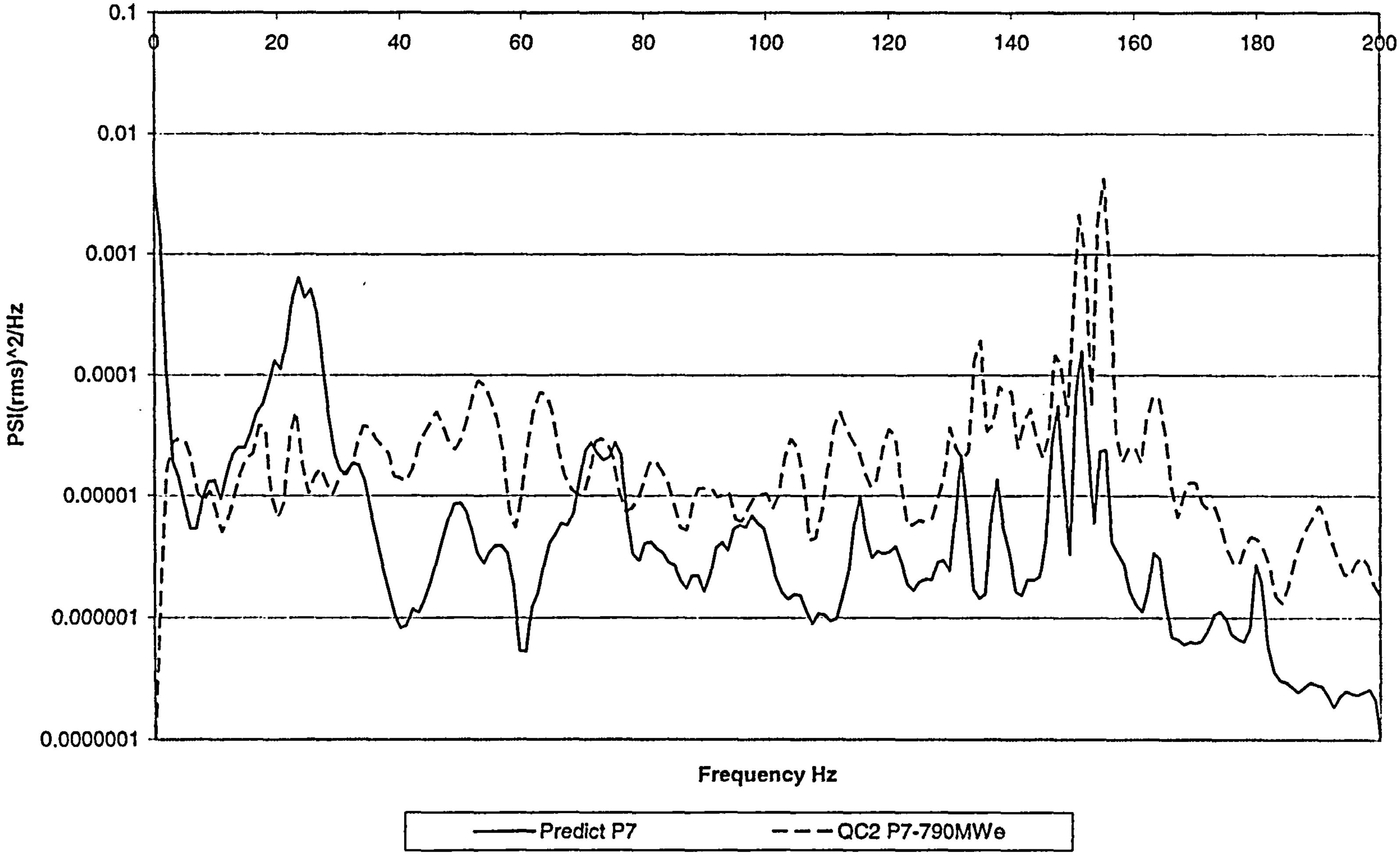

Figure EMEB-B-18-1-4-7

Exhibit EMEB-B-18-1, Rev. 1, VYNPS Steam Dryer Load Uncertainty
Attachment 4 (Rev 1)

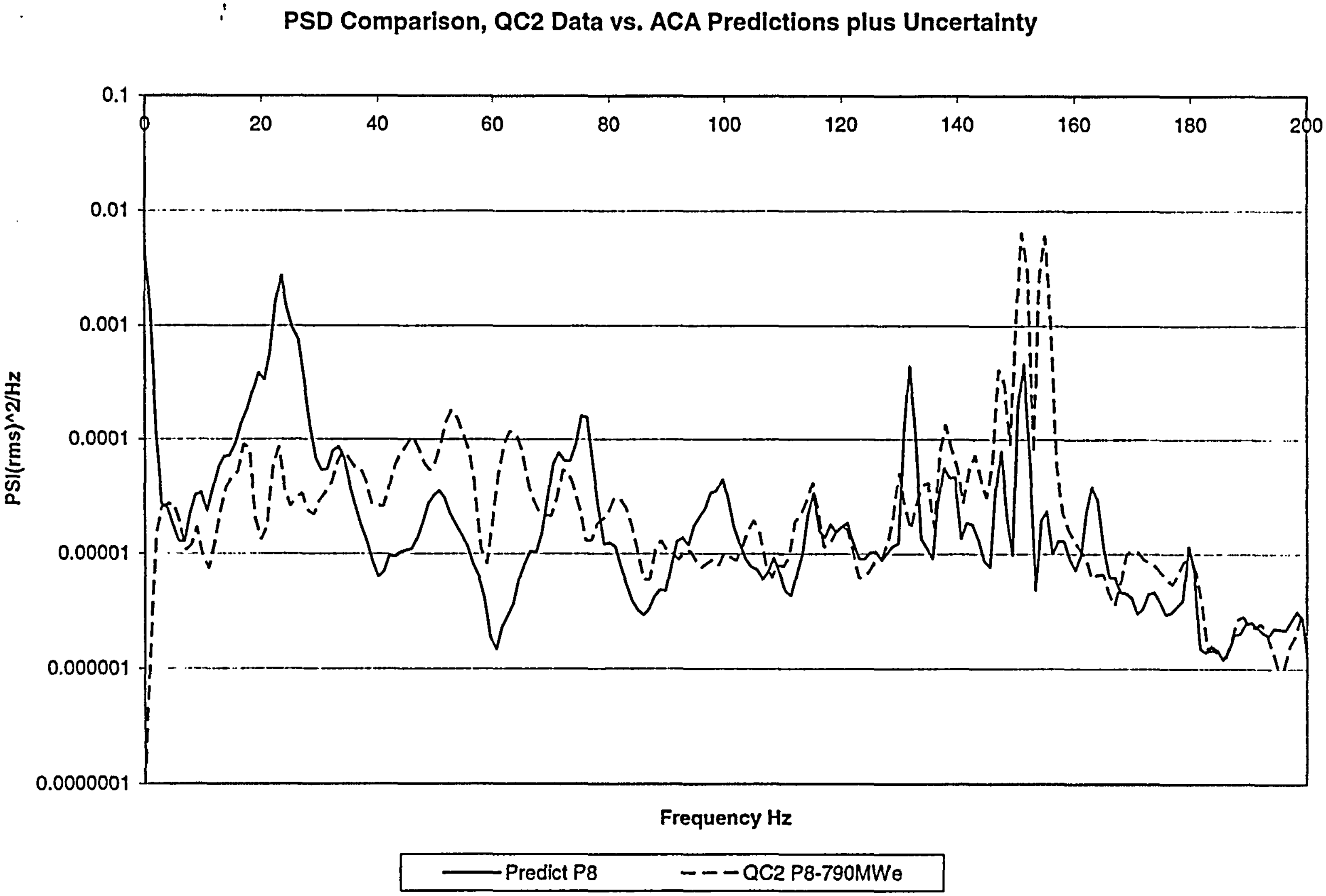

Figure EMEB-B-18-1-4-8

Exhibit EMEB-B-18-1, Rev. 1, VYNPS Steam Dryer Load Uncertainty
Attachment 4 (Rev 1)

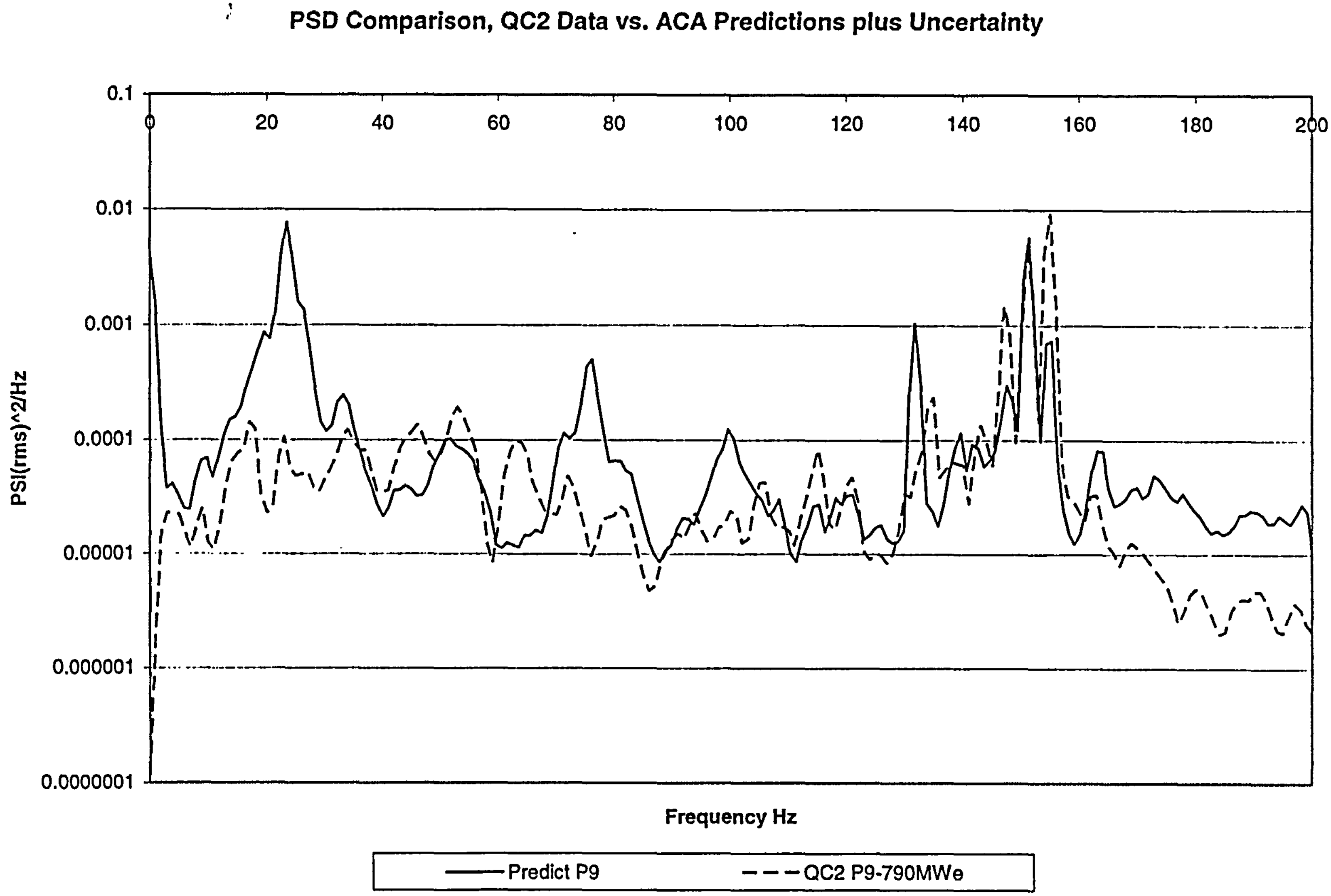

Figure EMEB-B-18-1-4-9

Exhibit EMEB-B-18-1, Rev. 1, VYNPS Steam Dryer Load Uncertainty
Attachment 4 (Rev 1)

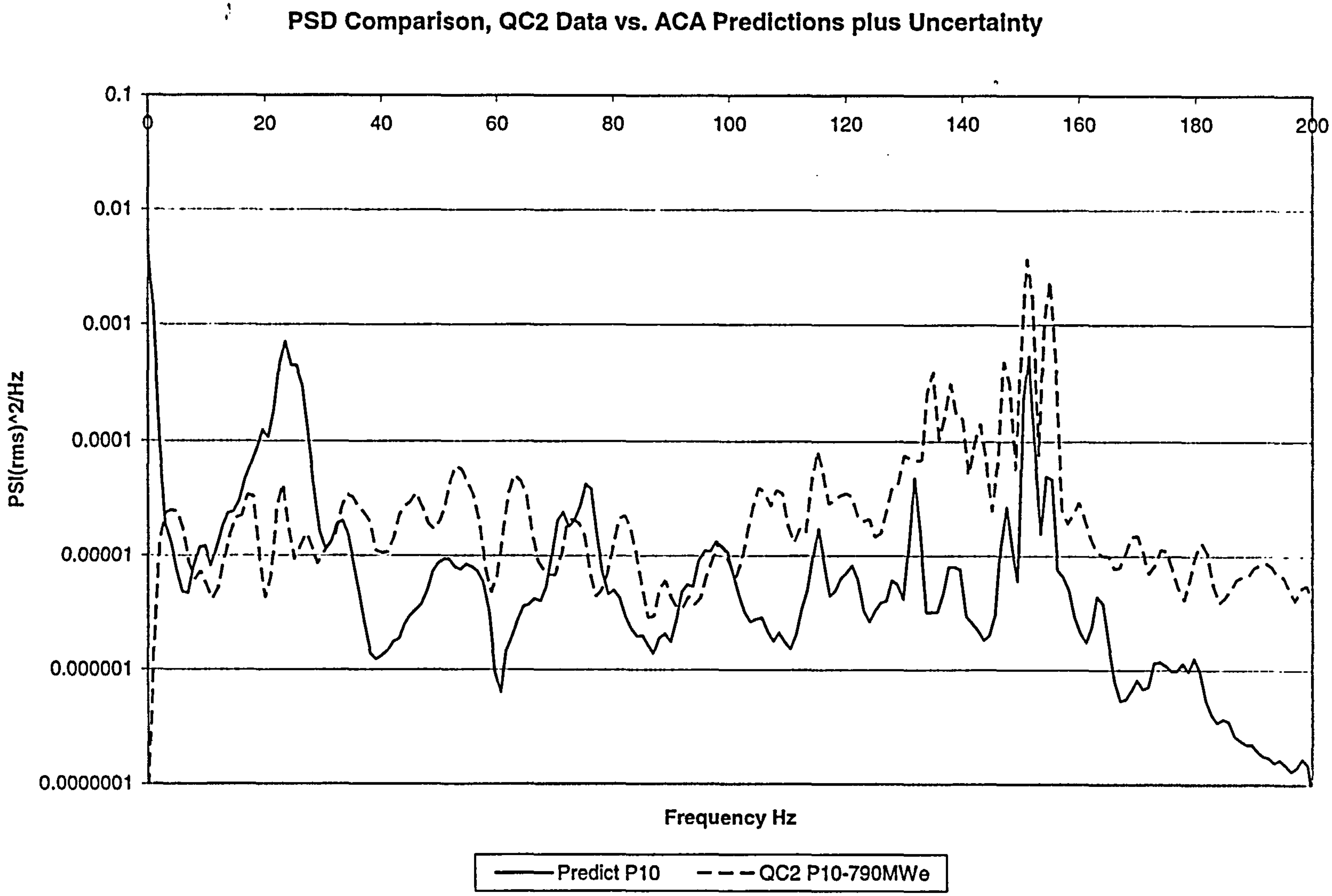

Figure EMEB-B-18-1-4-10

Exhibit EMEB-B-18-1, Rev. 1, VYNPS Steam Dryer Load Uncertainty
Attachment 4 (Rev 1)

PSD Comparison, QC2 Data vs. ACA Predictions plus Uncertainty

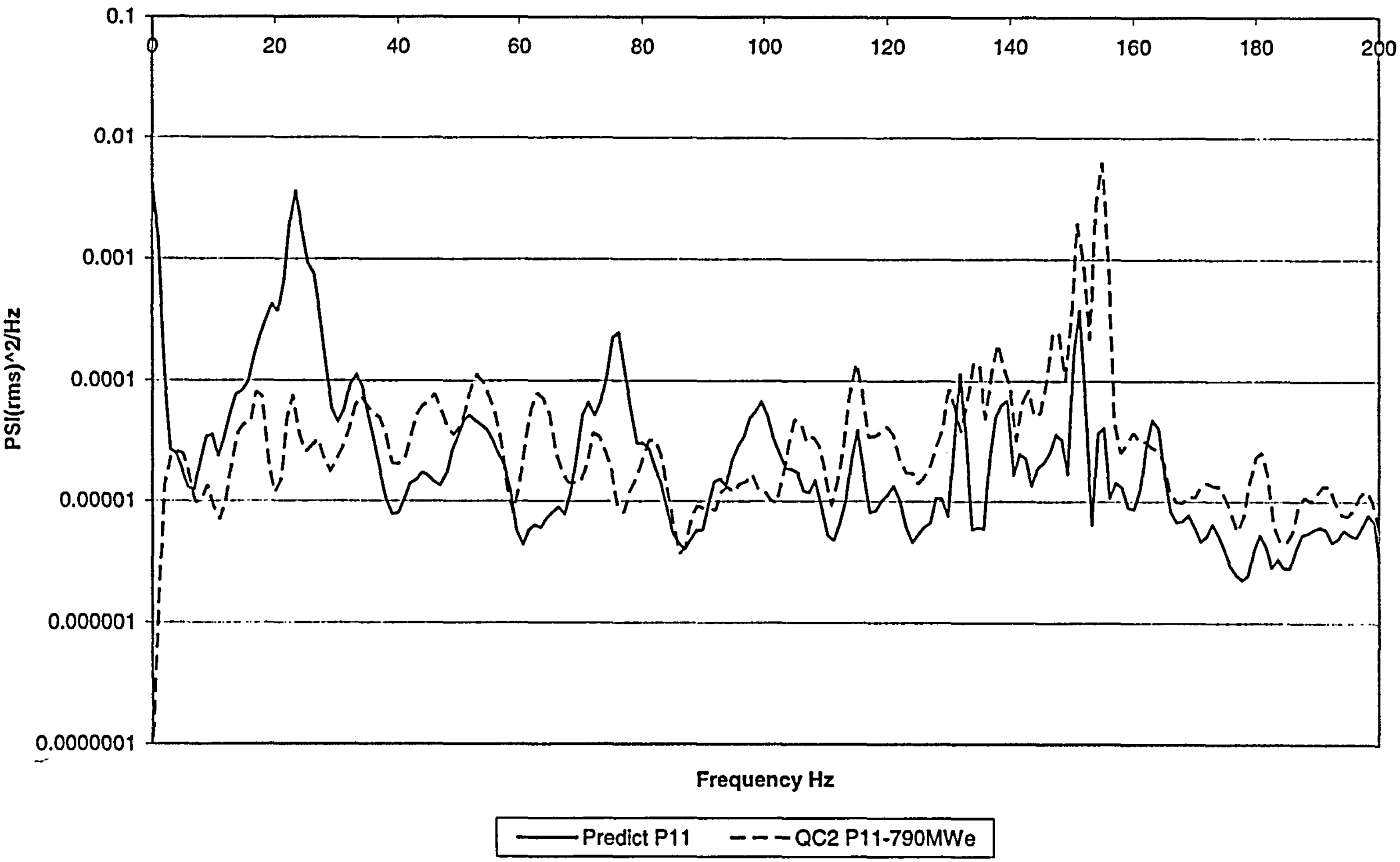

Figure EMEB-B-18-1-4-11

Exhibit EMEB-B-18-1, Rev. 1, VYNPS Steam Dryer Load Uncertainty
Attachment 4 (Rev 1)

PSD Comparison, QC2 Data vs. ACA Predictions plus Uncertainty

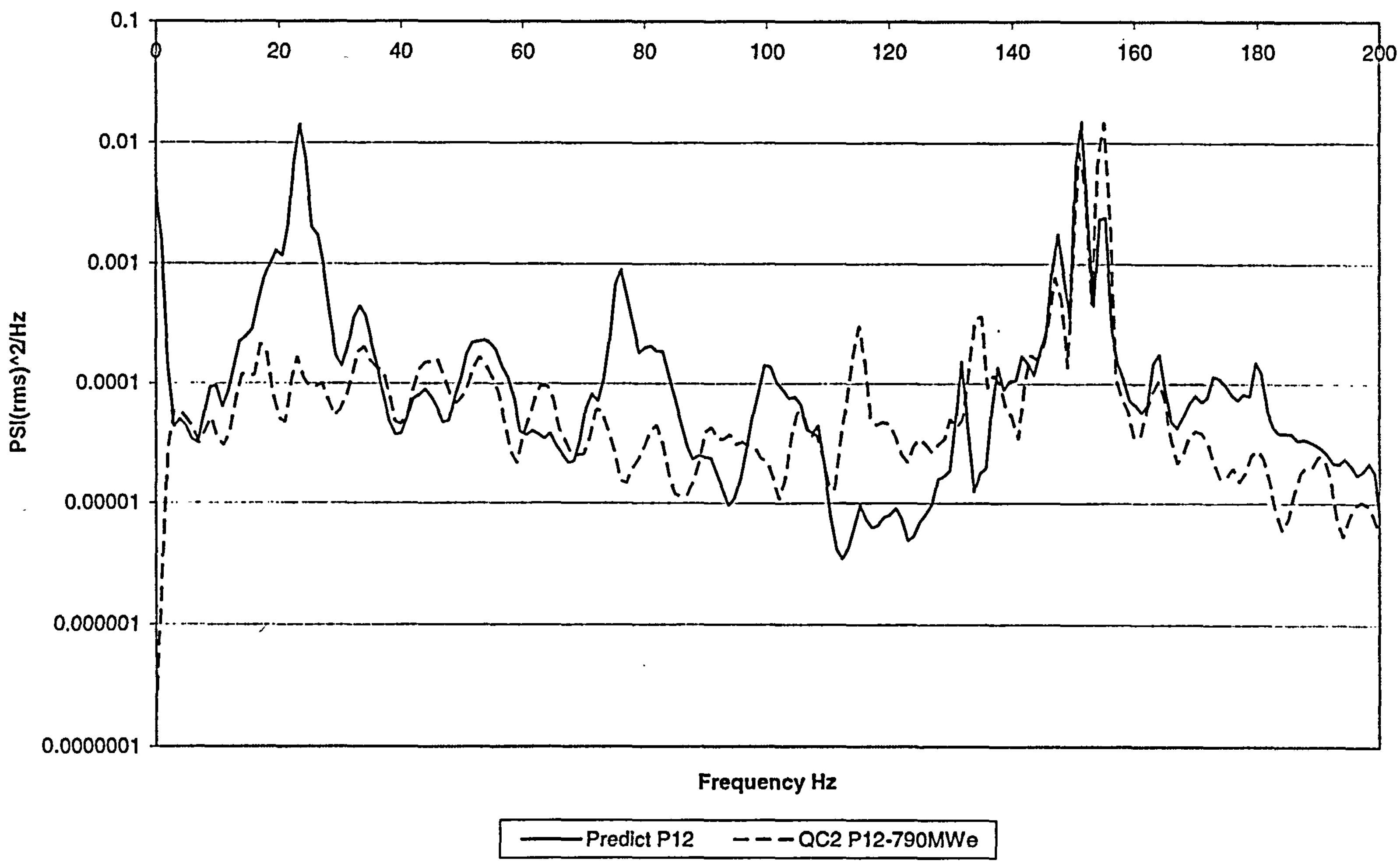

Figure EMEB-B-18-1-4-12

Exhibit EMEB-B-18-1, Rev. 1, VYNPS Steam Dryer Load Uncertainty
Attachment 4 (Rev 1)

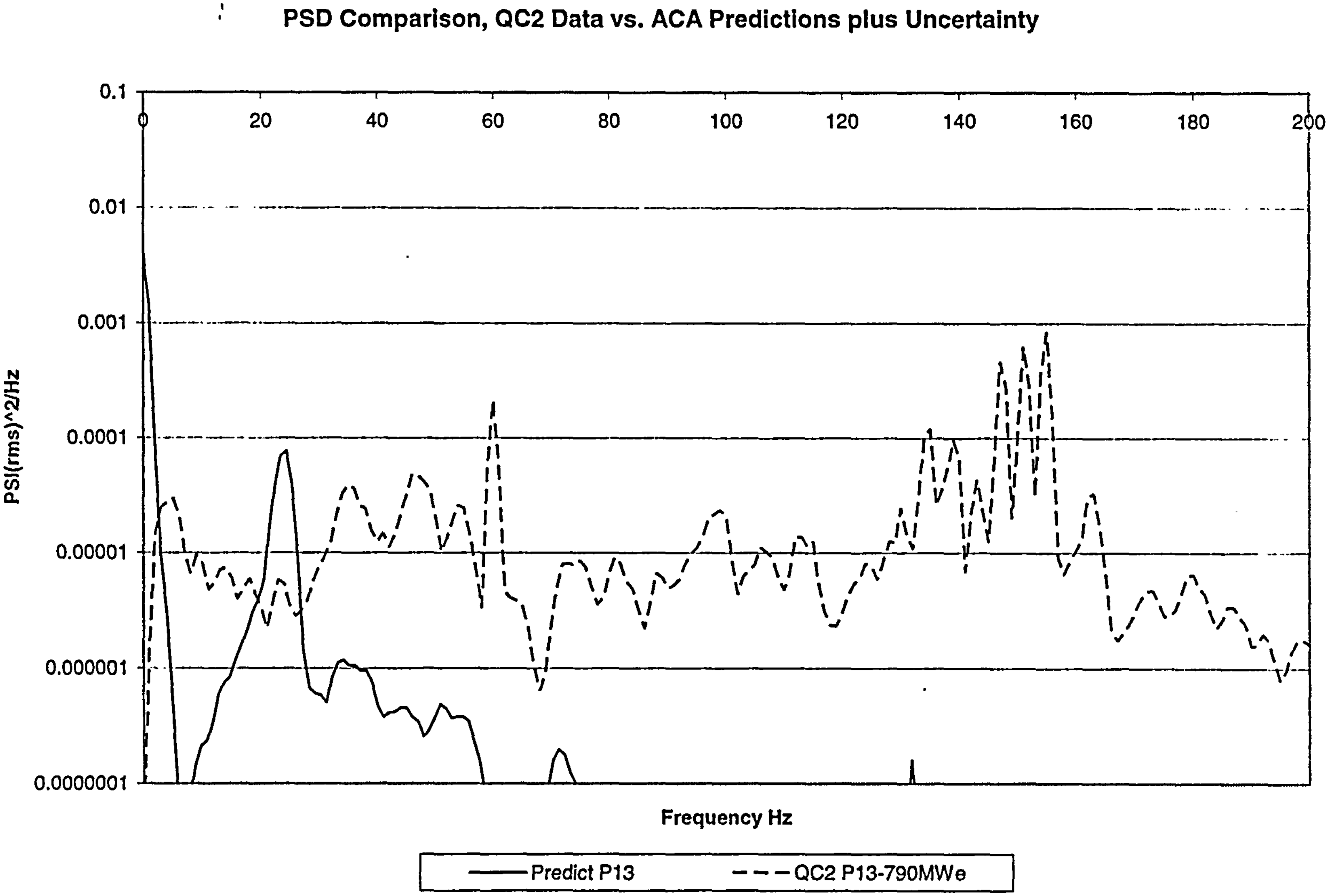

Figure EMEB-B-18-1-4-13

Exhibit EMEB-B-18-1, Rev. 1, VYNPS Steam Dryer Load Uncertainty
Attachment 4 (Rev 1)

PSD Comparison, QC2 Data vs. ACA Predictions plus Uncertainty

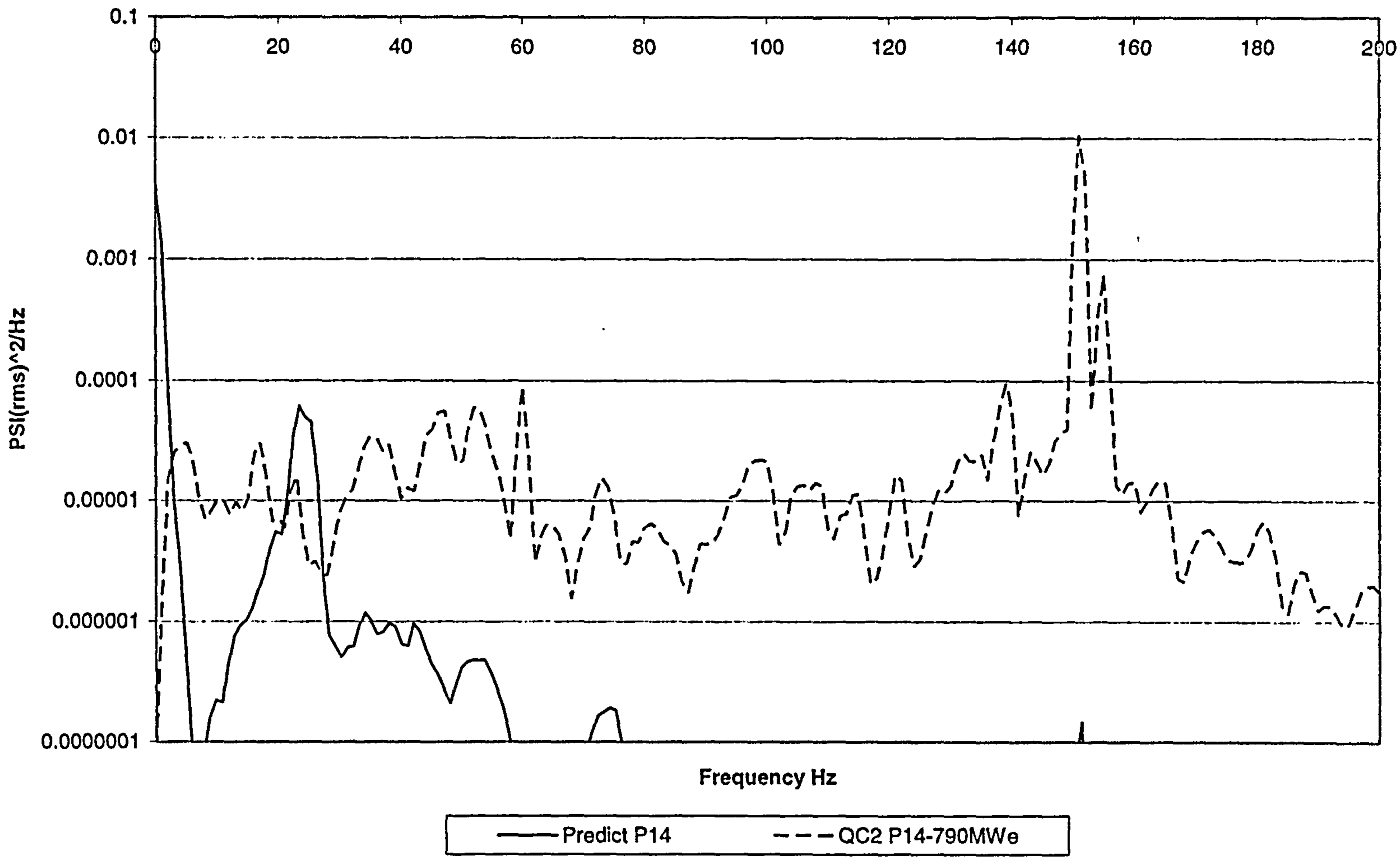

Figure EMEB-B-18-1-4-14

Exhibit EMEB-B-18-1, Rev. 1, VYNPS Steam Dryer Load Uncertainty
Attachment 4 (Rev 1)

PSD Comparison, QC2 Data vs. ACA Predictions plus Uncertainty

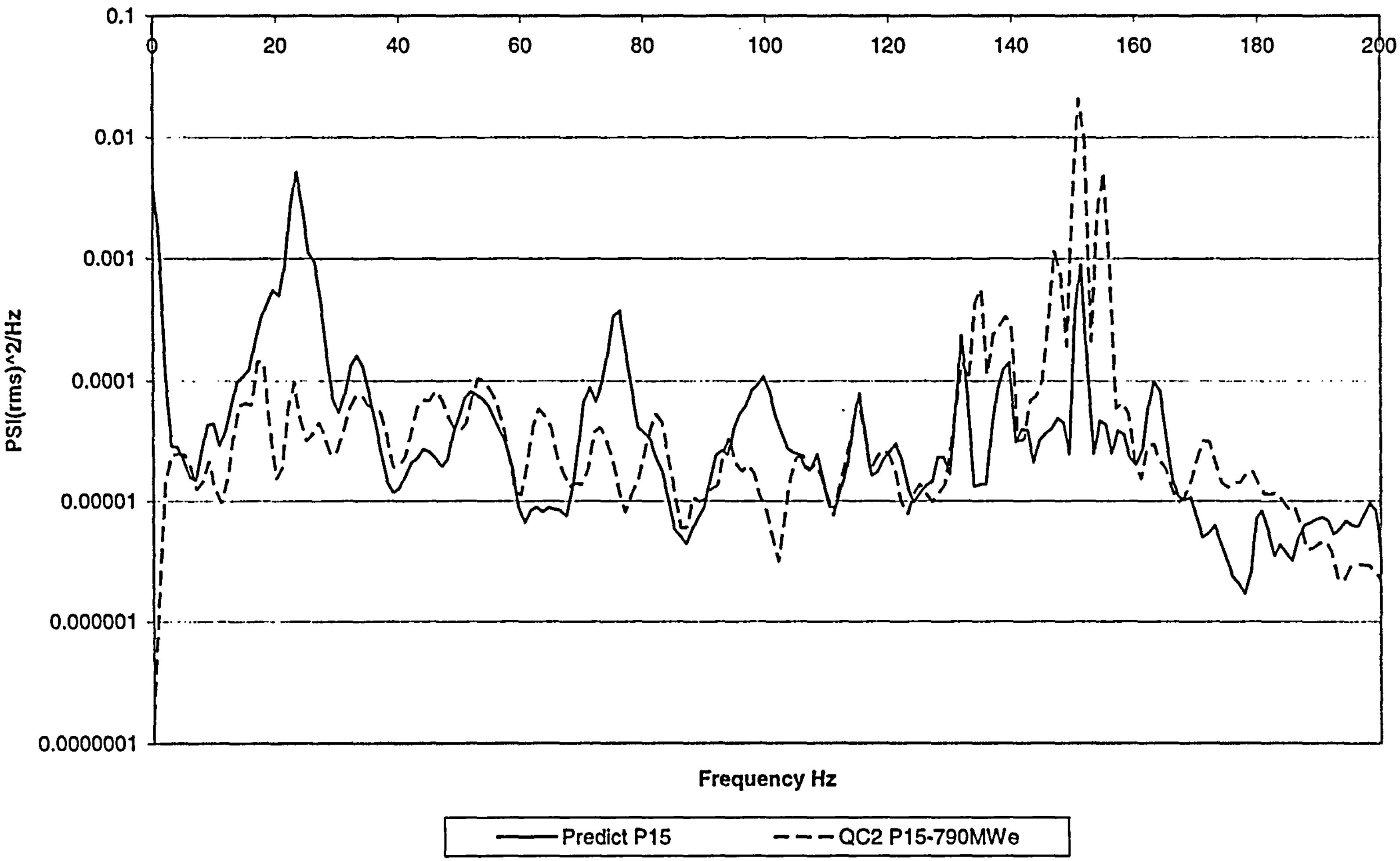

Figure EMEB-B-18-1-4-15

Exhibit EMEB-B-18-1, Rev. 1, VYNPS Steam Dryer Load Uncertainty
Attachment 4 (Rev 1)

PSD Comparison, QC2 Data vs. ACA Predictions plus Uncertainty

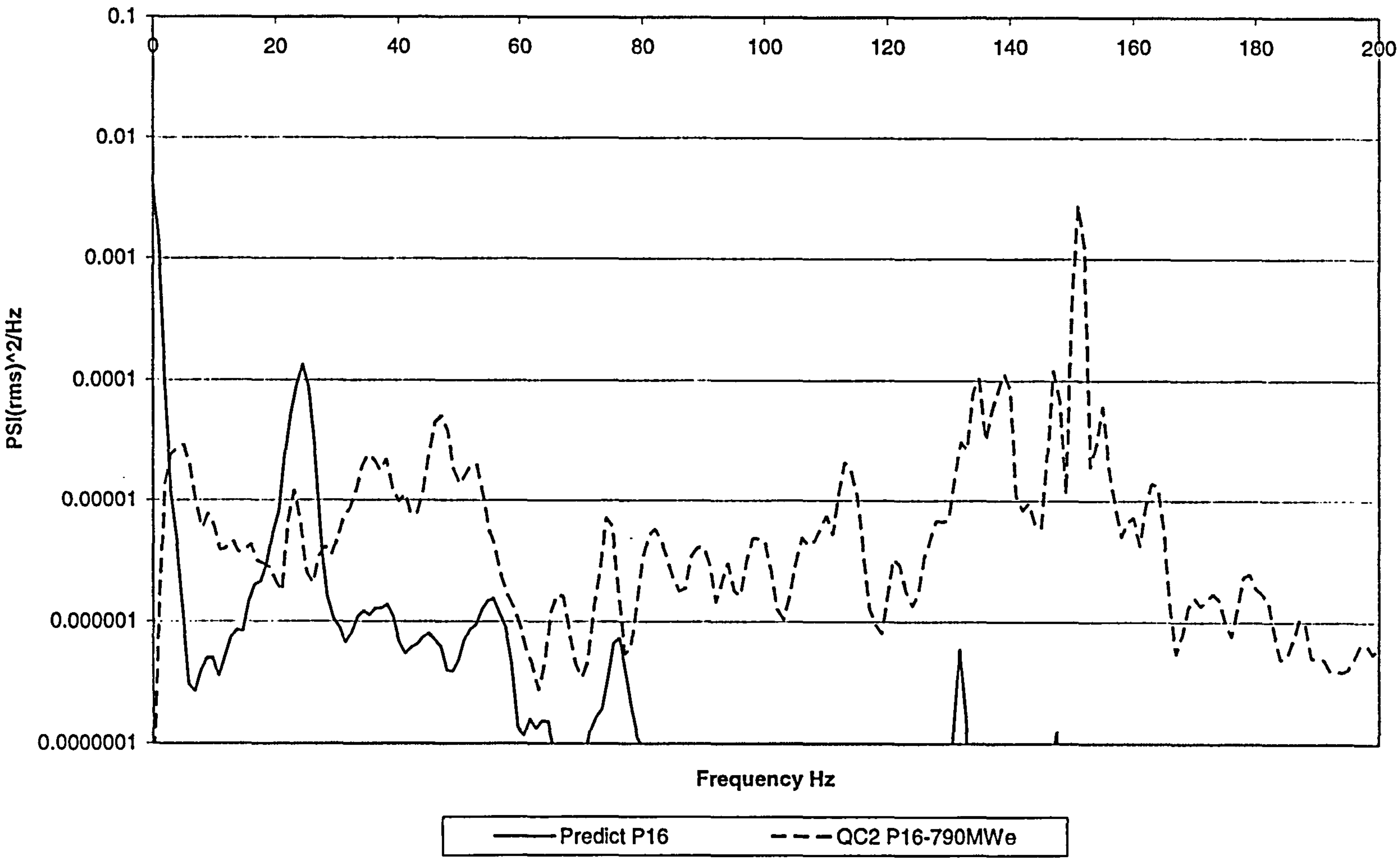

Figure EMEB-B-18-1-4-16

Exhibit EMEB-B-18-1, Rev. 1, VYNPS Steam Dryer Load Uncertainty
Attachment 4 (Rev 1)

PSD Comparison, QC2 Data vs. ACA Predictions plus Uncertainty

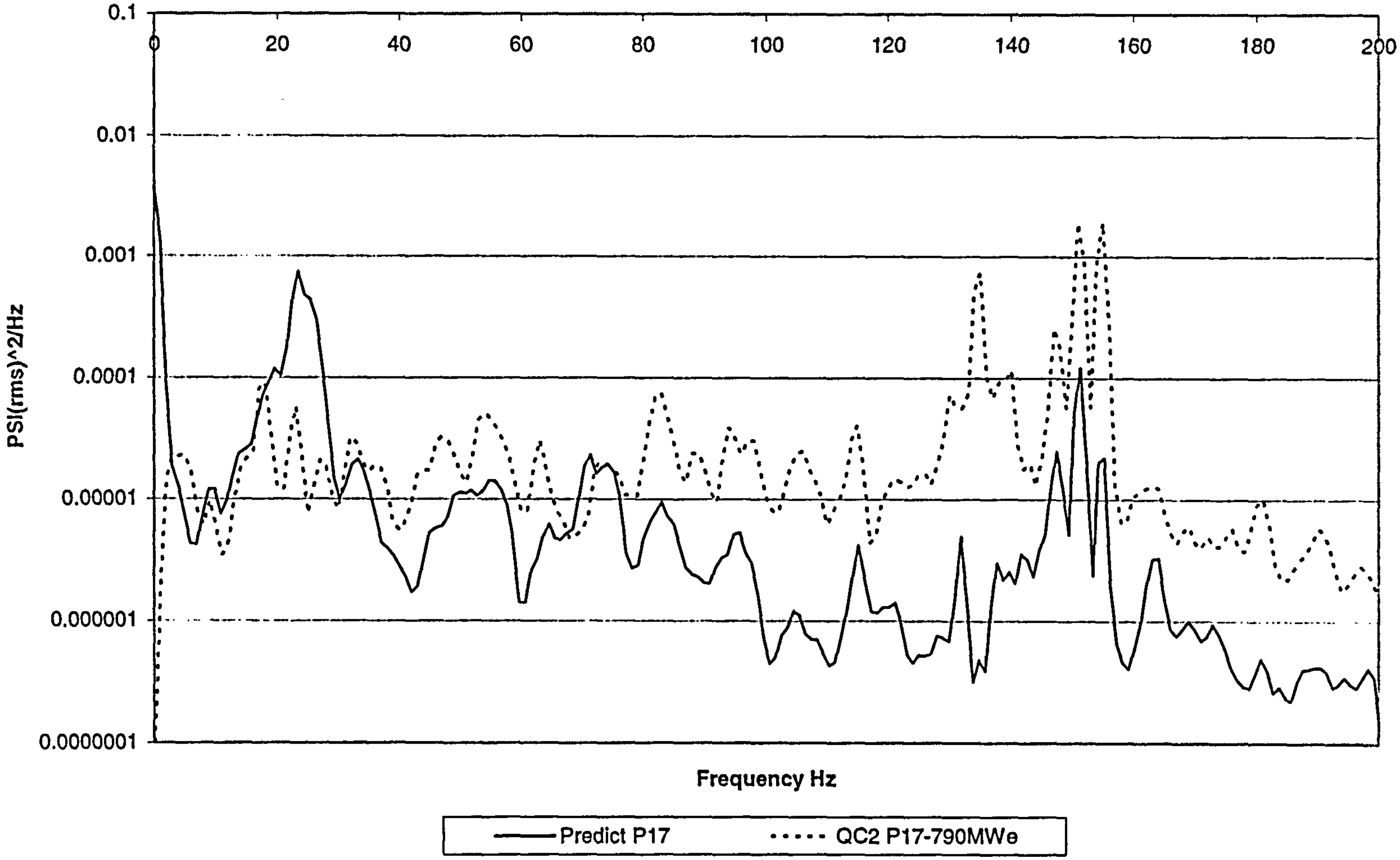

Figure EMEB-B-18-1-4-17

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Attachment 4 (Rev 1)

PSD Comparison, QC2 Data vs. ACA Predictions plus Uncertainty

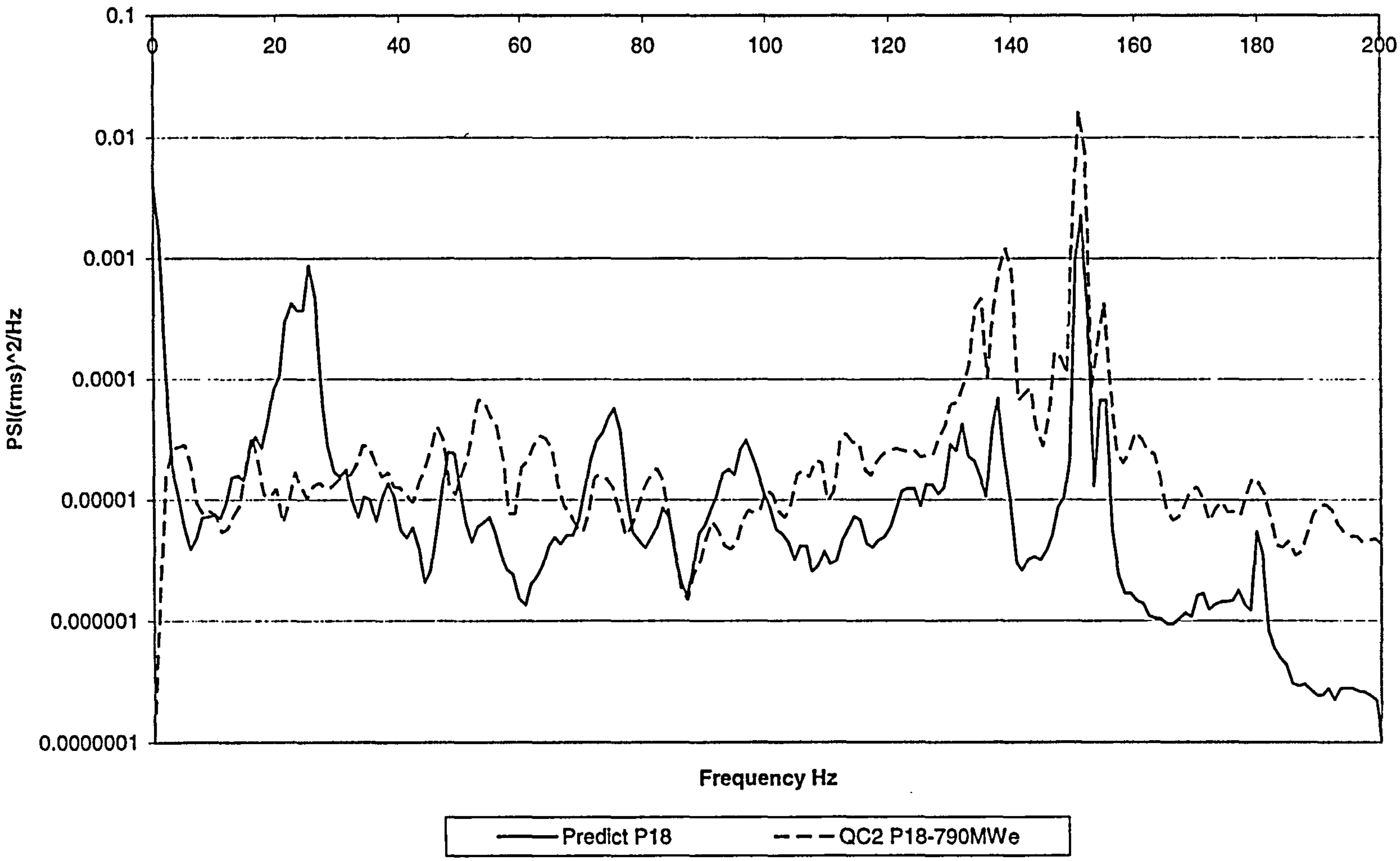

Figure EMEB-B-18-1-4-18

Exhibit EMEB-B-18-1, Rev. 1, VYNPS Steam Dryer Load Uncertainty
Attachment 4 (Rev 1)

PSD Comparison, QC2 Data vs. ACA Predictions plus Uncertainty

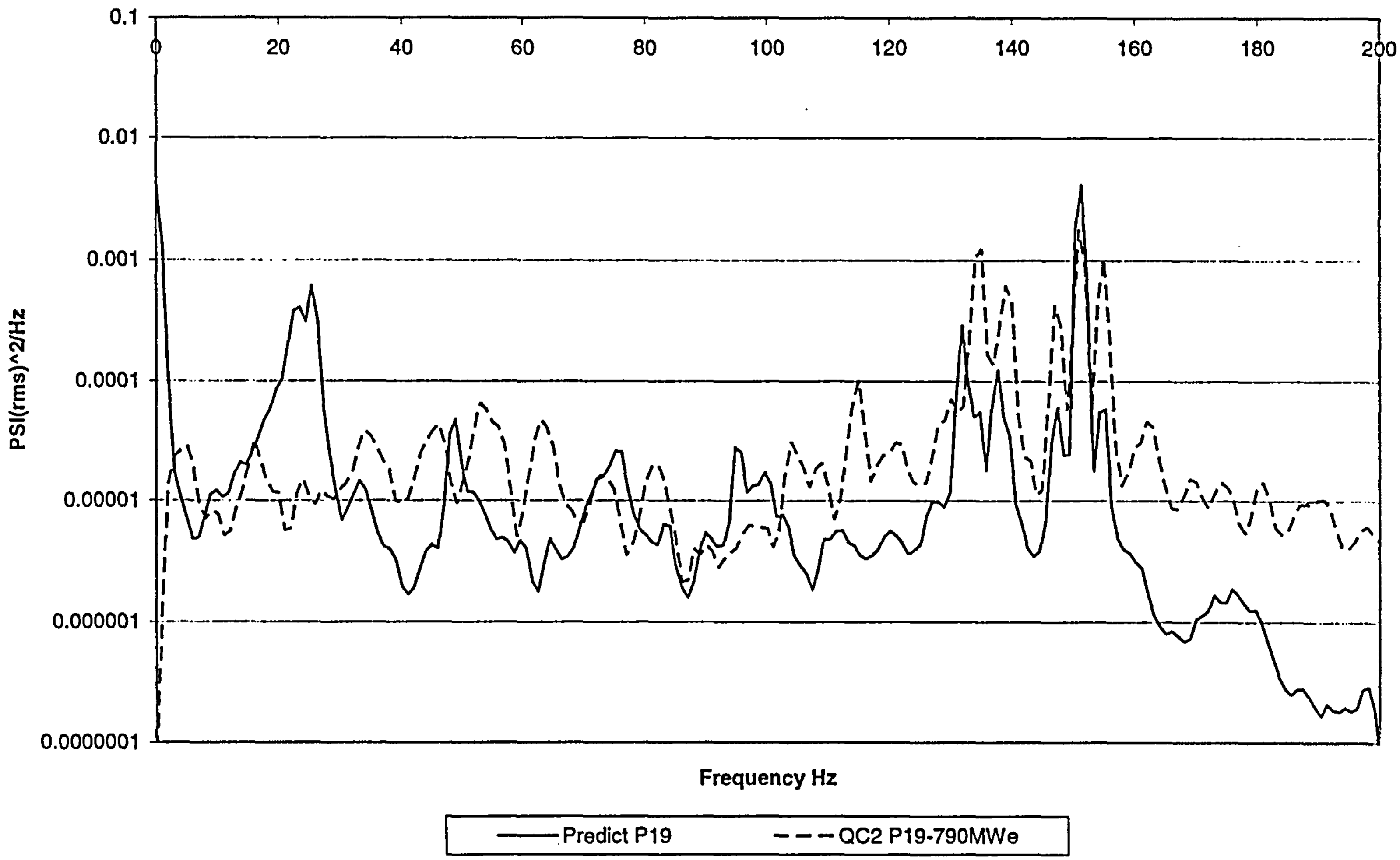

Figure EMEB-B-18-1-4-19

Exhibit EMEB-B-18-1, Rev. 1, VYNPS Steam Dryer Load Uncertainty
Attachment 4 (Rev 1)

PSD Comparison, QC2 Data vs. ACA Predictions plus Uncertainty

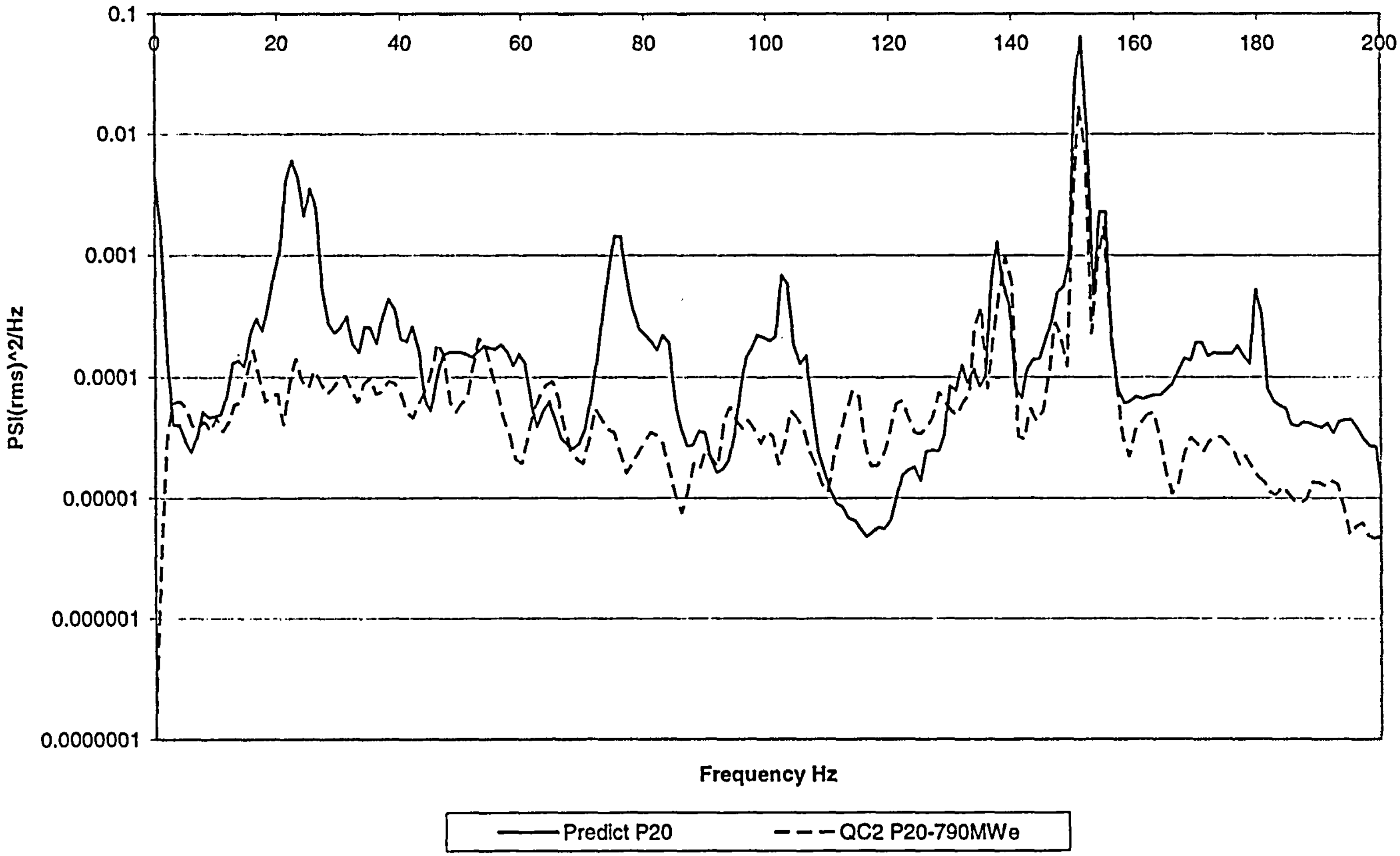

Figure EMEB-B-18-1-4-20

Exhibit EMEB-B-18-1, Rev. 1, VYNPS Steam Dryer Load Uncertainty
Attachment 4 (Rev 1)

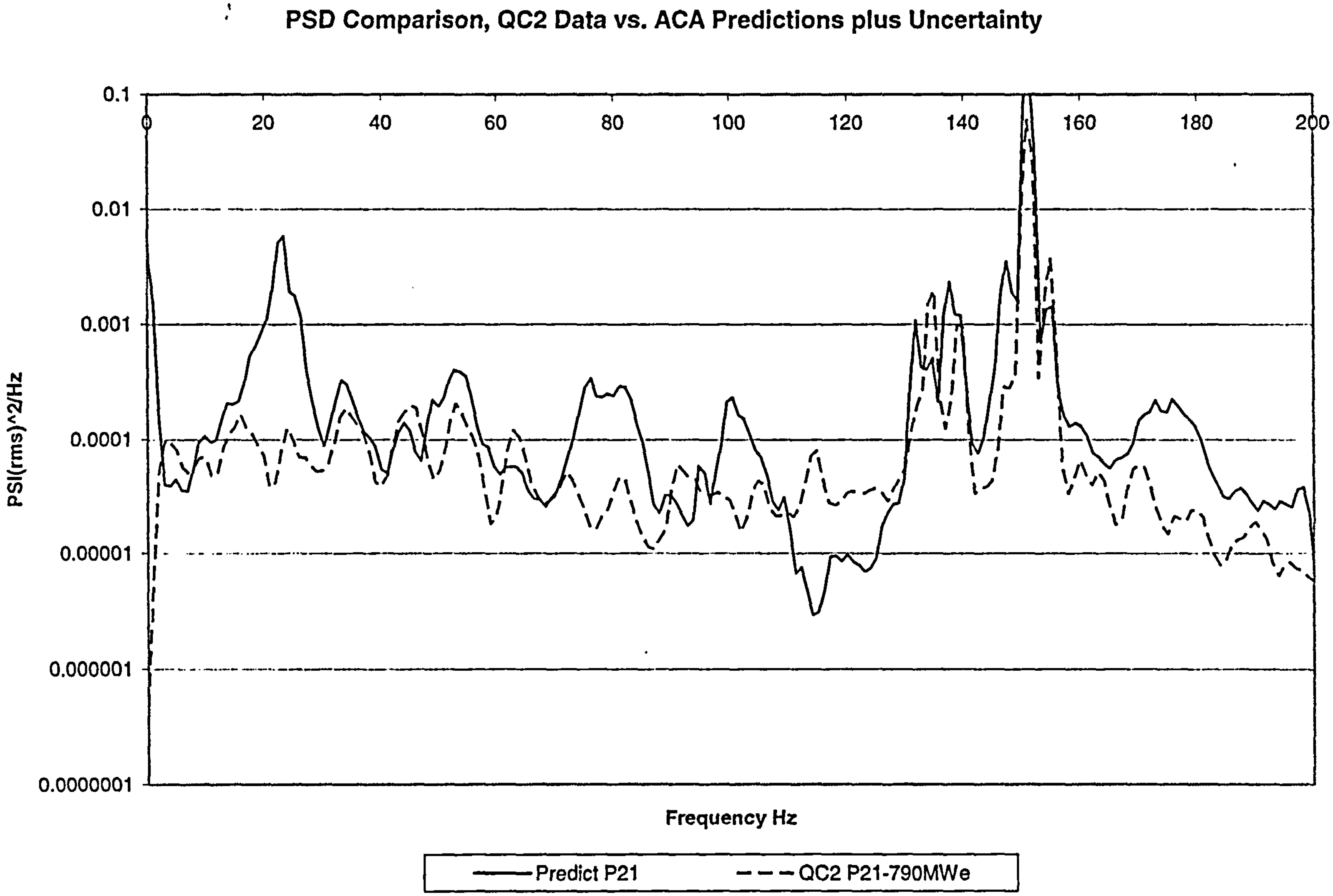

Figure EMEB-B-18-1-4-21

Exhibit EMEB-B-18-1, Rev. 1, VYNPS Steam Dryer Load Uncertainty
Attachment 4 (Rev 1)

PSD Comparison, QC2 Data vs. ACA Predictions plus Uncertainty

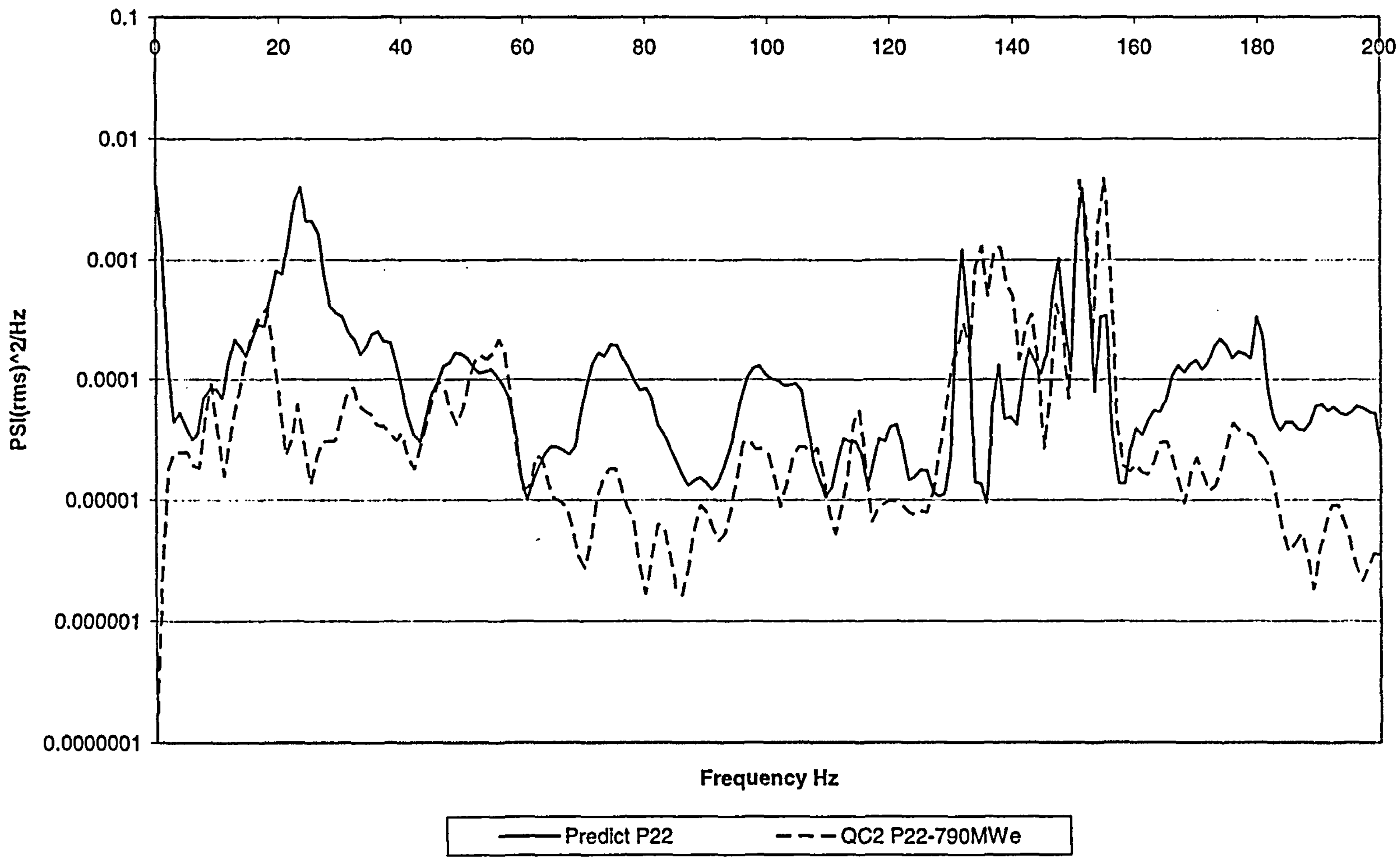

Figure EMEB-B-18-1-4-22

Exhibit EMEB-B-18-1, Rev. 1, VYNPS Steam Dryer Load Uncertainty
Attachment 4 (Rev 1)

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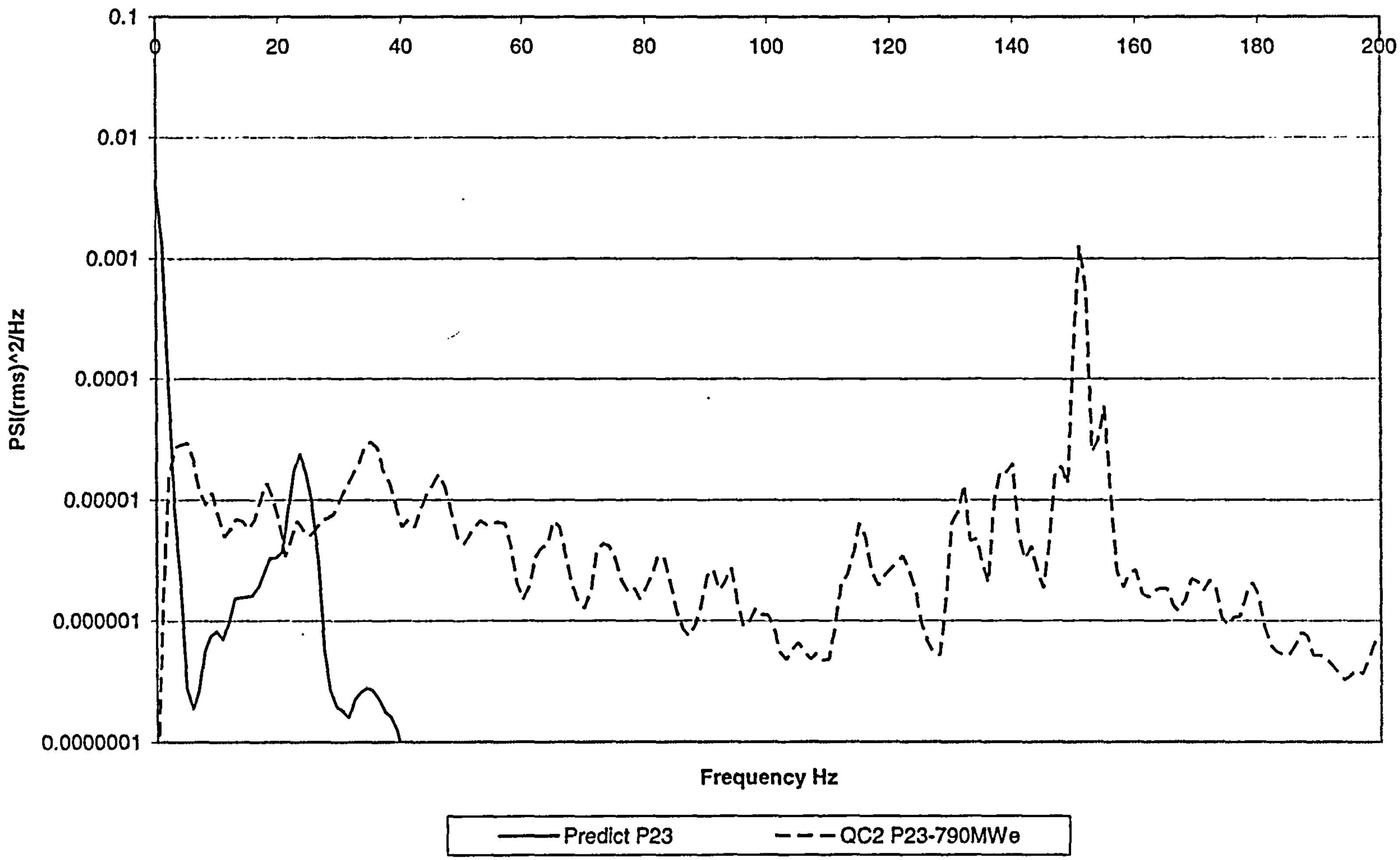

Figure EMEB-B-18-1-4-23

Exhibit EMEB-B-18-1, Rev. 1, VYNPS Steam Dryer Load Uncertainty
Attachment 4 (Rev 1)

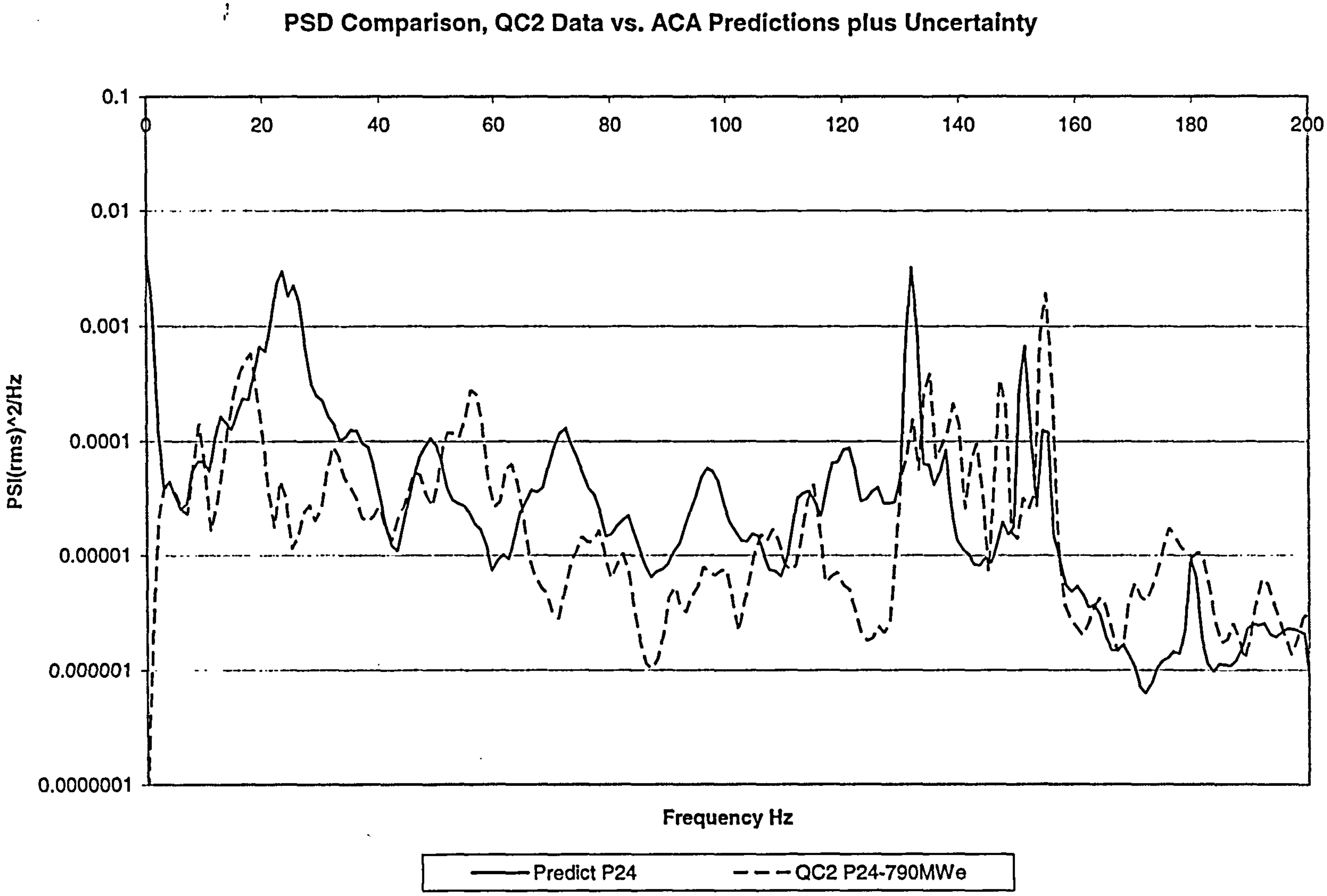

Figure EMEB-B-18-1-4-24

Exhibit EMEB-B-18-1, Rev. 1, VYNPS Steam Dryer Load Uncertainty
Attachment 4 (Rev 1)

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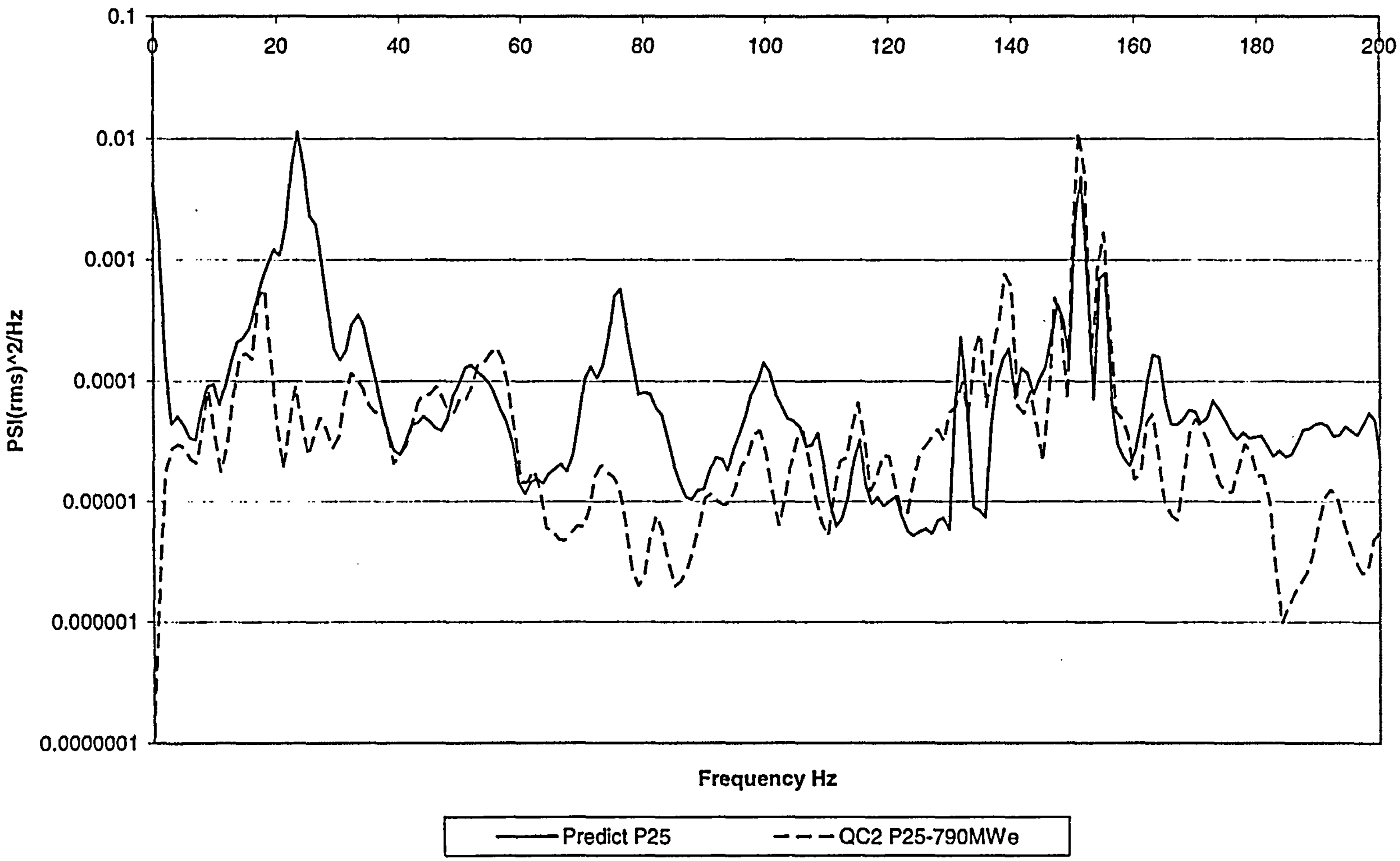

Figure EMEB-B-18-1-4-25

Exhibit EMEB-B-18-1, Rev. 1, VYNPS Steam Dryer Load Uncertainty
Attachment 4 (Rev 1)

PSD Comparison, QC2 Data vs. ACA Predictions plus Uncertainty

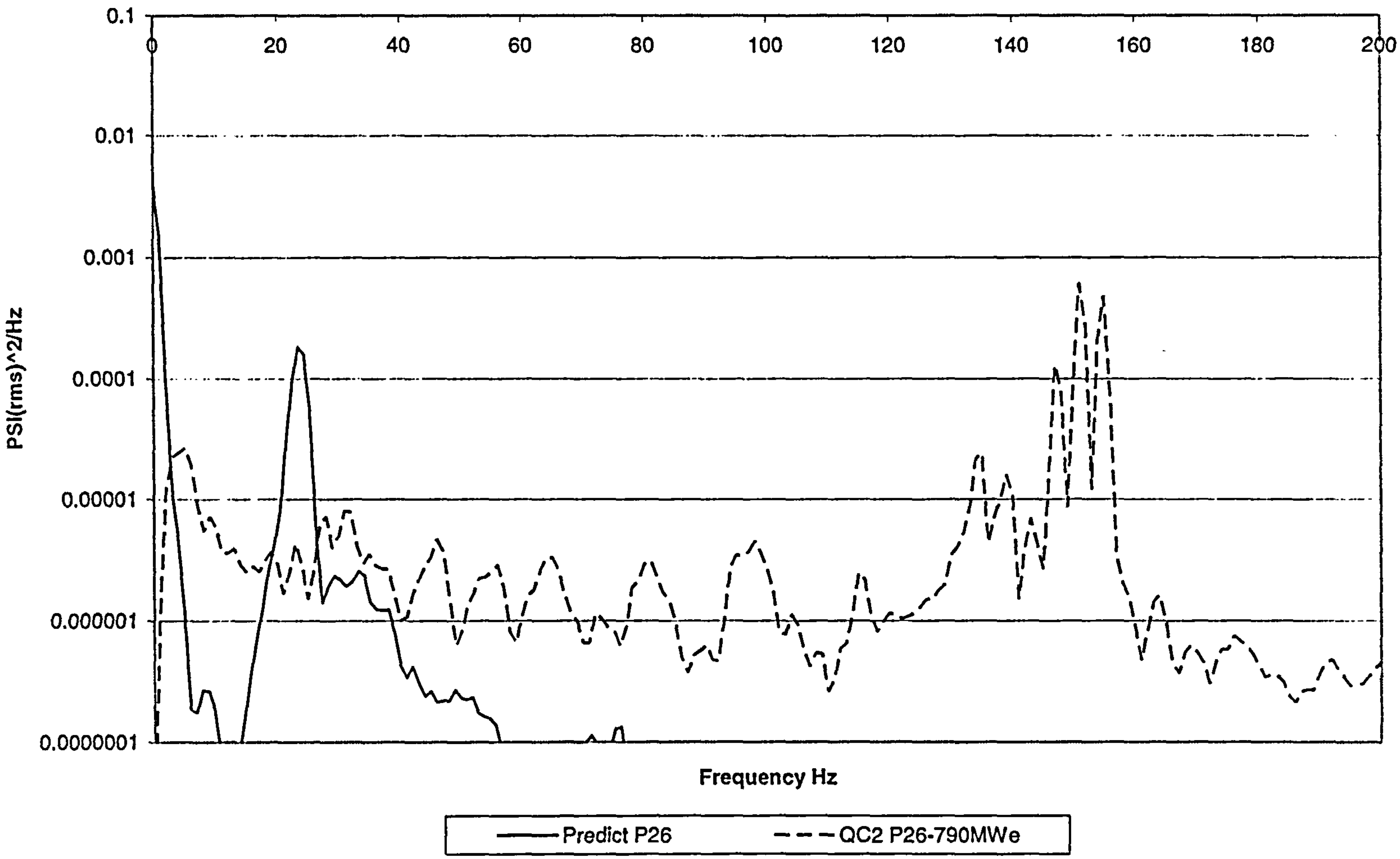

Figure EMEB-B-18-1-4-26

Exhibit EMEB-B-18-1, Rev. 1, VYNPS Steam Dryer Load Uncertainty
Attachment 4 (Rev 1)

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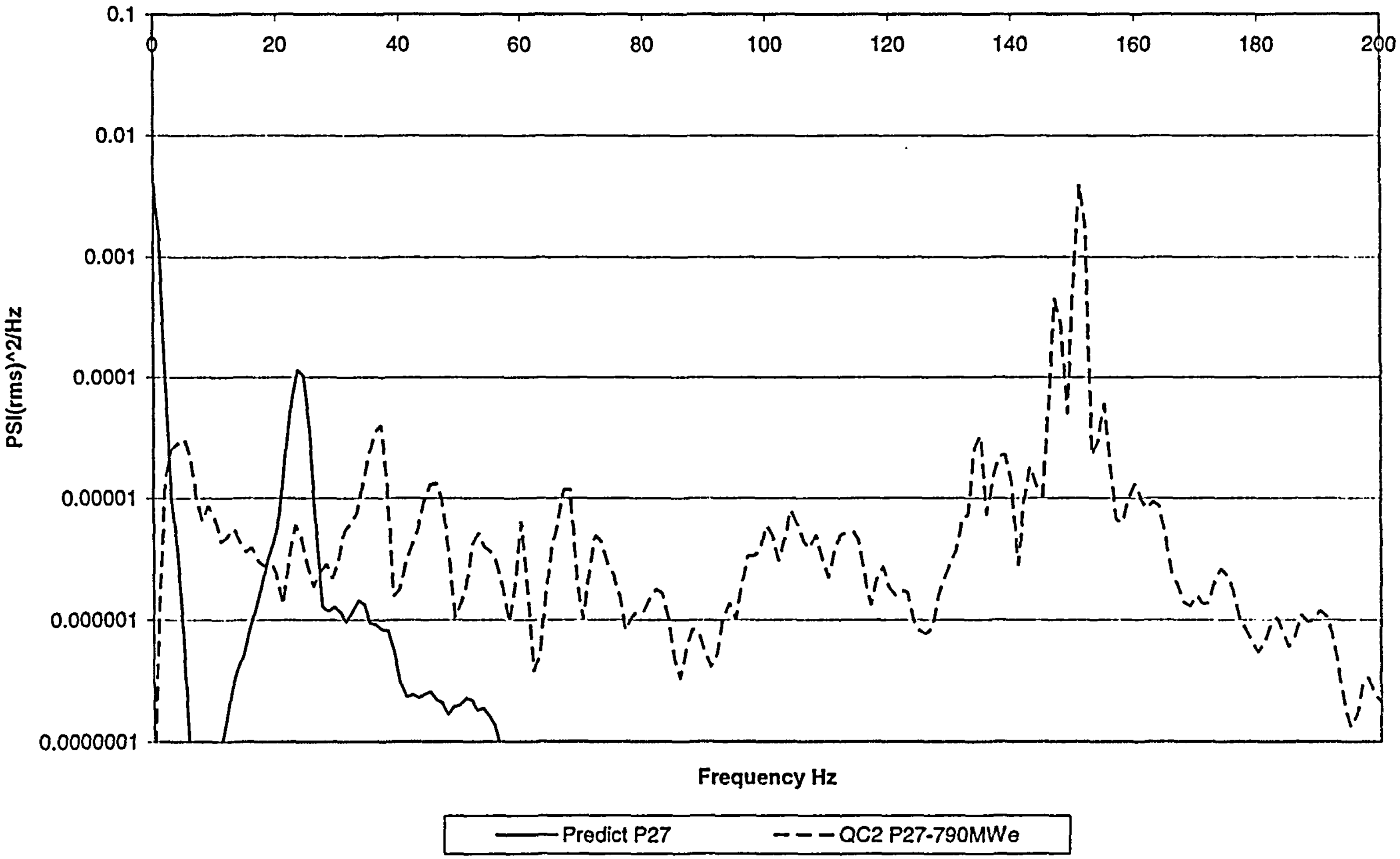

Figure EMEB-B-18-1-4-27

Exhibit EMEB-B-18-1, Rev. 1, VYNPS Steam Dryer Load Uncertainty
Attachment 4 (Rev 1)

PSD Comparison, QC2 Data vs. ACA Predictions plus Uncertainty

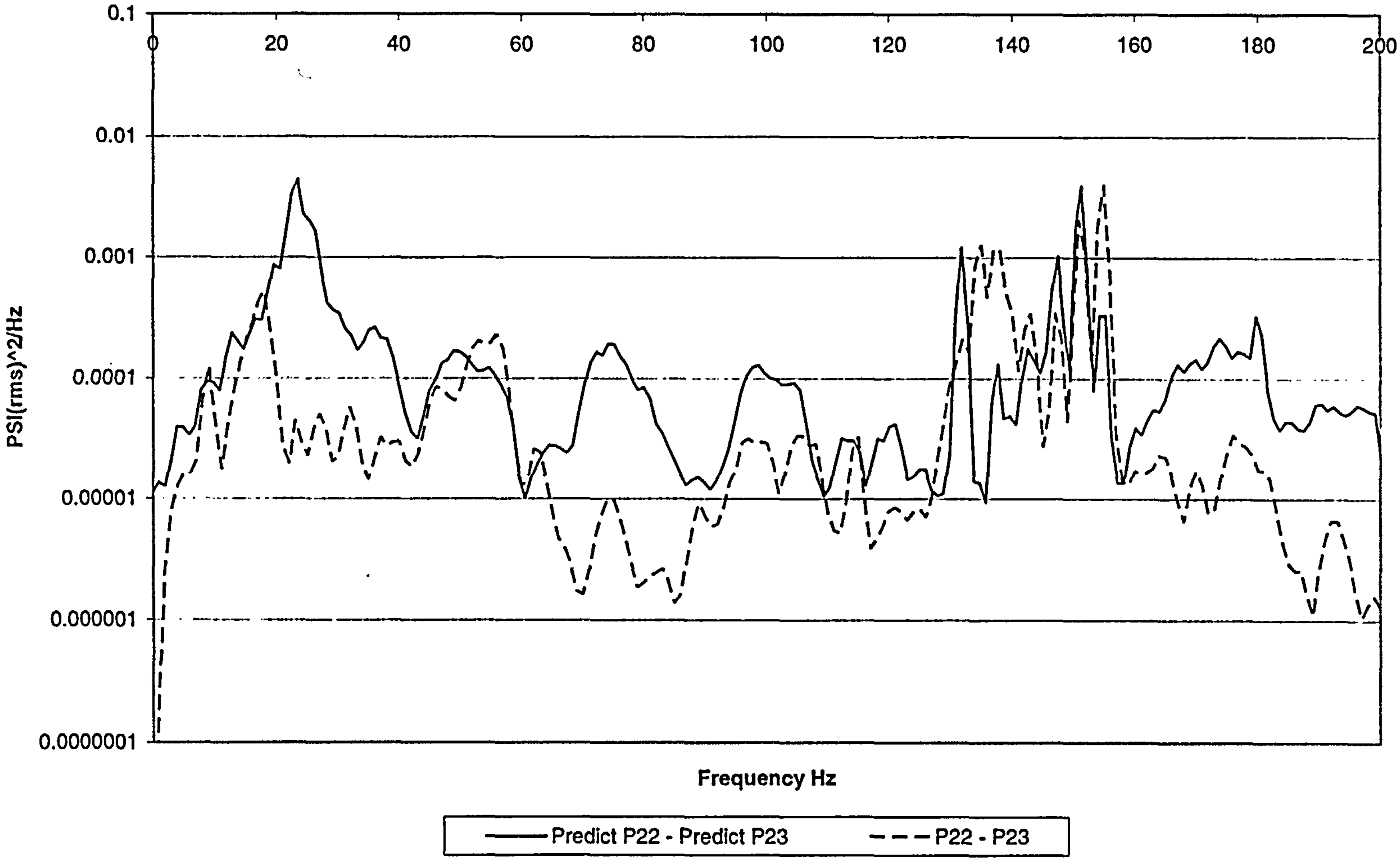

Figure EMEB-B-18-1-4-28

Exhibit EMEB-B-18-1, Rev. 1, VYNPS Steam Dryer Load Uncertainty
Attachment 4 (Rev 1)

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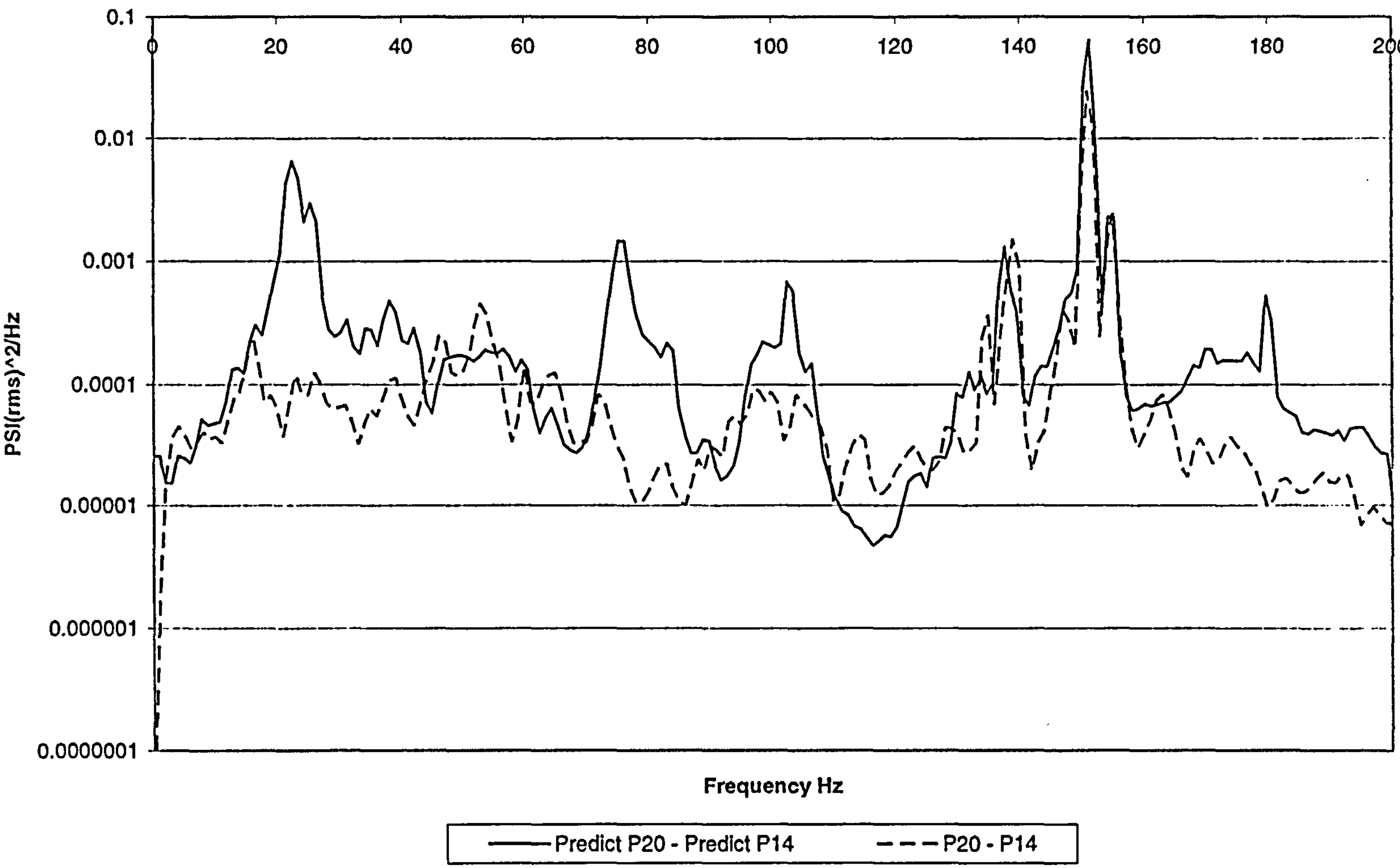

Figure EMEB-B-18-1-4-29

Exhibit EMEB-B-18-1, Rev. 1, VYNPS Steam Dryer Load Uncertainty
Attachment 4 (Rev 1)

PSD Comparison, QC2 Data vs. ACA Predictions plus Uncertainty

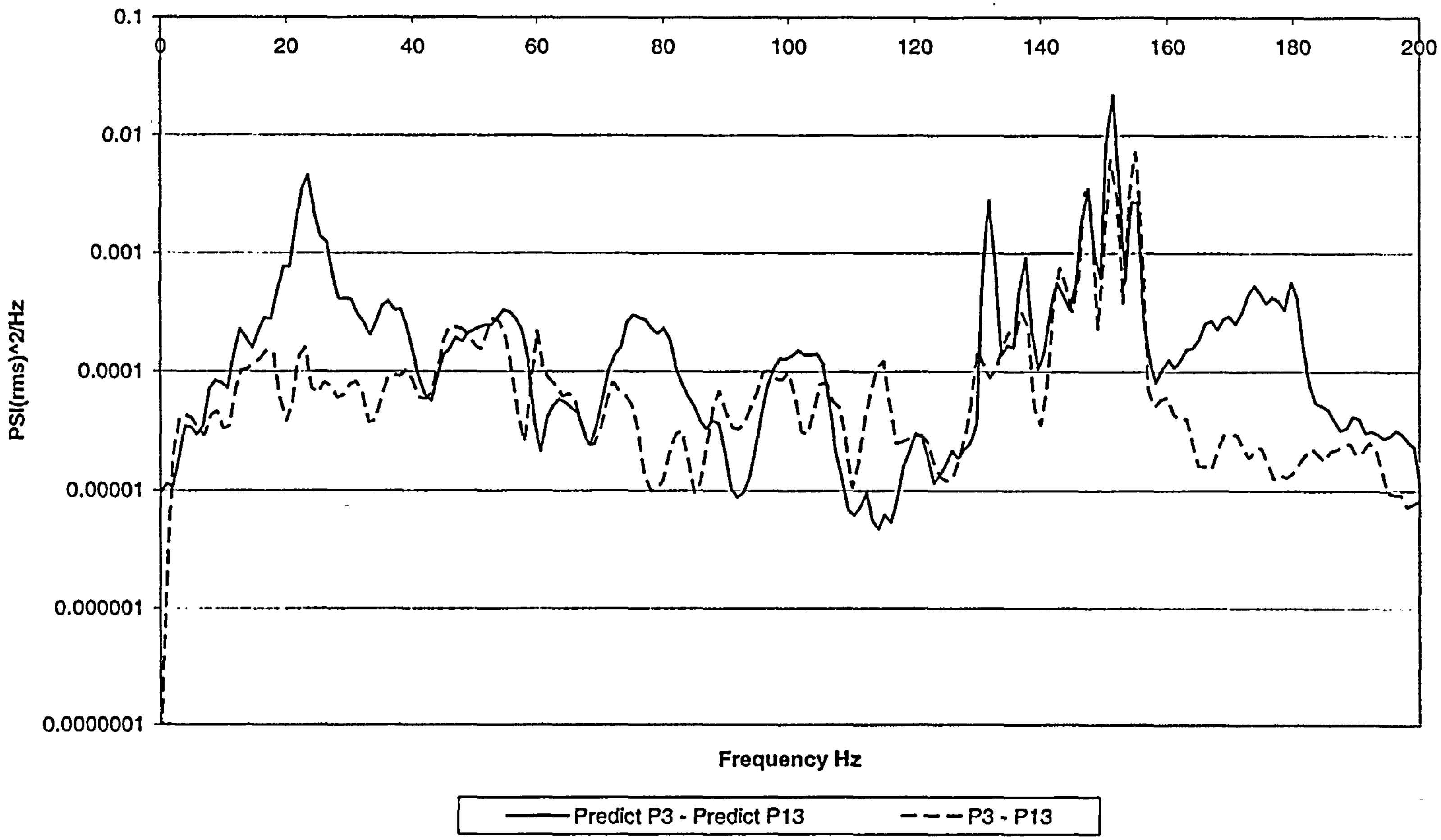

Figure EMEB-B-18-1-4-30

BVY 05-086
Docket No. 50-271

Attachment 2

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 263 – Supplement No. 34

Extended Power Uprate – Additional Information

RAI SRXB-A-66 Data

Total number of pages in Attachment 2 (excluding this cover sheet) is 1. Attachment 2 includes a compact disk (The CD contains Proprietary Information).

Vermont Yankee Nuclear Power Station
Data Supporting Response to RAI SRXB-A-66

Entergy's letter of September 10, 2005, responded to NRC RAI SRXB-A-66. Microsoft Excel spreadsheet files designated C4-TGBLA6_diff.Rev1.xls and Lat_7009_T6_C4_FD_Data.xls contain data supporting that response.

The file designated C4-TGBLA6_diff.Rev1.xls contains the basis of the RAI SRXB-A-66 plots of CASMO-4 and TGBLA06 data comparisons (K-inf, local peaking, plutonium isotopes, void coefficient, etc.) for a number of lattice designs, void history depletions, and instantaneous void cases. Void fraction definitions used in both methods are consistent and are based upon the heated channel area only with bypass and water rods at zero void. As noted in the response to RAI SRXB-A-66, these comparisons were within expectations for comparisons of different calculational methods. Larger differences are noted for 90% void history depletion cases, but consideration that only a small portion of any BWR core obtains these values for a small exposure window near beginning of life minimizes the impact. These cases are not currently used in any production or licensing basis applications. For the lower void fraction cases, the differences seen in K-inf and local peaking would be expected to be insignificant when incorporated into the homogenized nodal models in which the data are used. This has been demonstrated via comparisons of PANAC11 and SIMULATE-3 core-follow results.

The file designated Lat_7009_T6_C4_FD_Data.xls contains detailed pin-by-pin power data from CASMO-4 and TGBLA06 for a single lattice design for various void history depletions. These data were requested at the September 7, 2005, NRC audit to aid in the staff's evaluation of GNF's methods via an independent method. This information was taken from existing calculation files generated to support the response to RAI SRSB-A-66. Due to differences in the units of depletion (MWD/MT vs. MWD/ST), only a limited number of depletion points are close enough to the same exposure to provide meaningful comparisons at a pin level. Pin power differences (RMS for the entire lattice) are shown graphically and further detailed study is possible from the data available. The review performed of this information indicates that the difference in gadolinium treatment drives the difference in pin powers calculated by the two methods over the whole lattice. Peak pin agreement, as shown in the response to RAI SRXB-A-66, is generally good.

Attachment 3

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 263 – Supplement No. 34

Extended Power Uprate – Additional Information

Supplemental Response to SRXB-A-64

Total number of pages in Attachment 3 (excluding this cover sheet) is 2.

RAI SRXB-A-64

Provide the values for maximum bundle power and average power densities at VYNPS before and after the EPU.

Supplemental Response to RAI SRXB-A-64

Core thermal power information for VYNPS is provided in Table SRXB-A-64-1. The table provides the average power densities before and after EPU. The table also provides channel (bundle) power information requested by the RAI.

Table SRXB-A-64-1
Vermont Yankee Nuclear Power Station
Power Information

Parameter	Pre-EPU	Post-EPU	% Change
Total Core Thermal Power (MWt)	1593	1912	20
Power Density (kW/liter)	48.9	58.7	20
Channel Average Power (MWt)	4.33	5.20	20
Maximum Channel Power (MWt)	~7	~7	N/A

The channel average power is the total core thermal power divided by the number of fuel channels (368). The maximum channel powers shown in Table SRXB-A-64-1 are essentially unchanged by EPU operation. The values are presented as approximately 7 MWt in order to emphasize this point. The reason for this is that high power channels are limited by thermal limits. In other words, the peak LHGR and/or OLMCPR limits effectively put a ceiling on the maximum allowable bundle power. These limits are associated with the fuel and core designs, and are not a direct function of EPU. The actual pre- and post-EPU maximum bundle powers are 7.02 and 7.37 MWt, respectively. Again, the maximum values will likely change in the future depending on the particular reload core and bundle design. The maximum bundle power could also (potentially) be impacted by other design constraints, for example, the margin to the OLMCPR limit (i.e., how the peak bundles are projected to operate relative to the limit).

The NRC safety evaluation (SE) for constant pressure EPU documented in NEDC-33004P-A summarizes key elements related to the power uprate, including a discussion of power density. Section 1.3.3 of the SE contains the statements: "The CPPU approach achieves the power uprate by increasing the core average power density proportional to the core thermal power increase. This affects the reload core design and operating flexibility, the reactivity characteristics and the cycle energy requirements. No changes in fuel mechanical designs or fuel design limits are required to implement the CPPU process." From a core designer's point of view, the power uprate is effectively achieved by flattening the core radial power shape. More channels operate at or above the pre-uprate average bundle power level.

The next VYNPS operating cycle (i.e., cycle 25) core was designed to support operation under constant pressure power uprate (CPPU) conditions. The additional reactivity necessary to achieve the target power and cycle length is provided through the reload core design (i.e., the selection of bundle enrichments and the reload batch fraction).

Attachment 5

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 263 – Supplement No. 34

Extended Power Uprate – Additional Information

Responses to RAIs SRXB-A-65 and SRXB-A-67

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Total number of pages in Attachment 5 (excluding this cover sheet) is 28.

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RAI SRXB-A-65

Linear Heat Generation Rate (LHGR)

The NRC staff had previously asked whether any uncertainties were applied to the LHGR limit (curve) and the actual operating nodal steady state kilowatt/foot (kw/ft). The response to RAI SRXB-A-41 took credit for a reduced value in the gradient uncertainty. However, the power allocation and the pin power uncertainty values were increased to accommodate the lack of gamma scans of the current GE14 fuel designs as operated. The RAI response states that a local uncertainty of [[ ]] in LHGR is assumed in the development of the LHGR, implying that the [[ ]] kw/ft uncertainty addressed in the response to the staff RAI 5, associated with the NRC-approved safety limit minimum critical power ratio (SLMCPR) topical report NEDC-32694P-A, was intended for the generation of the LHGR limit. However, it is the staff's understanding that the uncertainty analyses provided in the RAI 5 response was addressing the uncertainty to be applied to the kw/ft calculated by the core monitoring system (e.g., 3D MONICORE) as opposed to a [[ ]] uncertainty assumed during the development of the LHGR curve.

The RAI 5 to NEDC-32694P-A stated that the process computer monitors peak kw/ft and maximum average planar linear heat generation rate (MAPLHGR). The peak kw/ft and the MAPLHGR depend on the bundle axial power distribution and, consequently, are significantly more sensitive to the 3-D MONICORE replacement of the traversing incore probe (TIP)/local power range montior (LPRM) axial power distribution. The RAI asked for uncertainty analysis for the 3-D MONICORE prediction of peak kw/ft and MAPLHGR. In the response, GE provided the following uncertainty analyses, which specified the uncertainty that would be applied to the peak kw/ft calculations:

<u>Nodal Power Uncertainty:</u> The nodal power uncertainty for 3D MONICORE is a combination of: 1) the uncertainty in the four bundle power at axial node k; 2) the uncertainty in the power allocation factor at node k; 3) the LPRM update uncertainty; and 4) the uncertainty in the TIP axial power distribution at node k. [[

]] The total nodal power uncertainty is, therefore, equal to:

[[

]]

<u>Pin Power Peaking Uncertainty:</u> The pin power peaking uncertainty can be determined from the factors outlined for the R-factor uncertainty summarized in Section 3 of NEDC-32601. Specifically, the pin power peaking uncertainty is a combination of 1) the model uncertainty, 2) the manufacturing uncertainty, and 3) the channel bow uncertainty. As in Section 3 of NEDC-32601P, the model uncertainty is a combination of the pin

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peaking uncertainty determined from Monte Carlo comparisons (1.44%) and an uncertainty due to flux gradients from neighboring bundles. [[

]] All of these pin power uncertainties have been combined in NEDC-32601P as:

[[
]]

The total LHGR uncertainty is the combination of nodal and pin power uncertainties:

[[ ]]

Staff Position

As shown in the NRC-approved SLMCPR methodology specified in NEDC 2694P-A, σ_{LHGR} changes with σ_{pal} and σ_{MC}. Accepting the reduction in the gradient uncertainty, a σ_{LHGR} of [[ ]] should be applied to the calculated kw/ft as discussed and specified in the NRC-approved licensing topical report. Because a [[ ]] uncertainty is assumed in the generation of the LHGR limit, this does not mean that the uncertainties due to the impact of modeling uncertainties on the operating kw/ft can be traded off with the [[ ]] uncertainty assumed in the development of the limit. The limit is developed based on the accuracy of the thermal-mechanical analytical models, methods and code systems. Therefore, any uncertainty currently applied in the development of the LHGR limit, can only be taken credit for or changed if it is demonstrated that for the current fuel designs and operating conditions additional nonconservatisims would not offset the "no cause" [[ ]] uncertainty.

The increase in the power allocation and pin power uncertainty applied to the SLMCPR does not directly lead to a proactive increase in the predicted steady state kw/ft. Therefore, potential underestimation in the nodal powers (bundle and peak pin) need to be accounted for. As evident in the RAI responses, the core-wide axial and nodal uncertainties determined through the TIP comparisons are not applied to the transient or accident analyses. The core-wide radial (e.g., bundle uncertainty σP4B) uncertainty is limited to the SLMCPR calculations. Therefore, there are no nodal or pin uncertainties that are applied to the predicted kw/ft. It is the staff's position that a [[ ]] kw/ft uncertainty be applied to the operating kw/ft calculated in the core simulator code, because of the following reasons:

1. Since there are no measurement data to validate the bundle and pin axial power, the uncertainties in the cross-sections and the pin powers are based on the TIP four bundle readings and the MCNP/TGBLA code-to-code comparisons. The four radial bundle uncertainty $\sigma_{P4B\ nodal}$ is derived from TIP comparisons and is applied to the SLMCPR. The power allocation between the four bundles $\sigma_{PAL\ nodal}$ derived from measurement data is also applied to the SLMCPR. The predicted operating kw/ft

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relies on the predicted axial bundle power and the pin powers. Although the 3D MONICORE adjusts the four bundle axial power peaking to the TIP reading, the adjusted axial power peaking is based on at least four bundle TIP response. Therefore, the power allocation in each bundle must be incorporated in the predicted kw/ft. Similarly, the uncertainty in the pin power needs to be included in the calculation of the peak kw/ft. Therefore, the calculated [[ ]] uncertainty needs to be applied to the predicted kw/ft, to account for the uncertainties in the cross-sections and the pin powers.

2. The [[ ]] power uncertainty bias, applied in the fuel rod internal pressure cited in the Alternative Approach (Supplement 30, Attachment 1), accounts for the differences between the design conditions the rod internal pressure calculations are based on and the rod internal pressures that would be obtained if actual operating history conditions were simulated. In other words, the [[ ]] uncertainty accounts for the difference between the as-designed and as-operated conditions.

3. The Alternative Approach cites an additional power uncertainty of [[ ]] power that is not specifically assigned to any cause. The Alternative Approach also states that separate experimental benchmarking information confirms that the model uncertainties remain valid. However, it is the NRC staff's understanding that, for the current fuel designs (GE14) as operated, no benchmarking of the fission gas inventory was performed. It is also the understanding that the [[ ]] "no cause" uncertainty is based on the original NRC-approval of the thermal-mechanical methodology and models. Therefore, it is not evident if a conservatism of [[ ]] would actually be available, if the operating and core design changes implemented since the initial development of the fuel thermal-mechanical models are evaluated. Neither the RAI response nor the Alternative Approach demonstrated this. The RAI response also did not discuss what uncertainties are assumed in the transient overpower kw/ft and if there is sufficient margin available.

4. The application of [[ ]] margin to the calculated kw/ft values would ensure that there are sufficient margins to the pellet exposure limits. The [[ ]] additional margin in the peak kw/ft would require a decrease in the nodal (bundle-wise) operating kw/ft, which would provide additional margin in bundle averaged accumulated exposure.

Response to RAI SRXB-A-65

The 3D Monicore surveillance system discussed in the RAI is intended to be [[

]] The following points are provided in response to items 1 – 4 under the *staff position* heading in the RAI.

1. As stated above, the GE objective is for the core monitoring methods to provide the most accurate [[ ]] quantification of the actual operating state. The

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uncertainty in that operating state calculation is addressed [[ ]], even when uprated conditions are considered, as discussed further below.

2. The [[ ]] bias applied to fuel rod internal pressure calculations is an allowance [[ ]]

Variations between the analyzed power history and actual power histories are addressed through the analysis assumption [[ ]] Figure SRXB-A-65-1 presents the [[ ]] (LHGR Operating Limit), as compared to an actual projected operating history for Bundle JLC505 Rod K4 Node 5 both under power uprate conditions and without power uprate. JLC505 experiences the highest bundle nodal exposure (Node 5) for any bundle in the VYNPS Cycle 25 core both with and without power uprate conditions. Rod K4 of JLC505 experiences the highest local exposure within that peak exposure bundle node. It is noted from Figure SRXB-A-65-1 that (1) the difference between the non-uprated and uprated nodal operating histories is relatively small, and (2) both operating histories are well bounded by [[ ]] the LHGR Operating Limit. It should be noted that at any point in time the local fuel rod power level could potentially momentarily approach or even be at the LHGR Operating Limit[[ ]]. The presented power history for JLC505 Rod K4 Node 5 provides a characterization of a typical operating history for a fuel rod node that operated at highest power, on the average over lifetime, of all fuel rods in the third cycle reload batch present in VYNPS Cycle 25. In this case, JLC505 Rod K4 Node 5 did not approach the LHGR Operating Limit prior to Cycle 25 and is not projected to approach the LHGR Operating Limit during VYNPS Cycle 25, although, again, it is recognized that any individual fuel rod, either JLC505 Rod K4 during actual Cycle 25 operation or any other fuel rod, could briefly operate at the LHGR Operating Limit. [[

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3. The basic fuel rod thermal-mechanical design analysis methodology currently used by GNF was implemented with GESTAR Amendment 7 with corresponding NRC approval as documented in Reference 65-1. Subsequent to the initial methodology approval, the NRC, in conjunction with NRC consultant and fuel rod thermal-mechanical analysis expert Dr. Carl Beyer (PNL), again reviewed the fuel rod thermal-mechanical design analysis methodology as documented in Reference 65-2. At the time of the original NRC review and approval of the fuel rod thermal-mechanical design and analysis methodology, the uncertainty in the fuel rod operating power level was addressed (1) directly through explicit consideration of the local power level variations that could develop [[

]].

In NEDC-32694-P-A, the response to RAI-5 (page A-10) identified an uncertainty in local LHGR of [[ ]]. Later revisions to the uncertainty treatment described in RAI-5 resulted in a slight increase to [[ ]] (page B-3 in the same topical report). Applying the adjusted uncertainty driven by lack of gamma scan data from RAI SRXB-A-41 for VYNPS would result in an uncertainty of [[

]]

For the fuel rod thermal-mechanical transient overpower analyses, again, the fuel rod is assumed [[

]]. This approach introduces considerable conservatism relative to the conditions that would be calculated for an actual operating history with a randomly placed transient event.

4. [[ ]] exposure limits are established for each product line. These limits are conservatively established with approved methods, including appropriate provisions for uncertainties. The limit established for GE14 fuel is applicable under the proposed CPPU conditions for VYNPS. The fuel rod thermal-mechanical performance consideration of greatest interest at exposures near the peak pellet exposure limit is the fuel rod internal pressure. As discussed above, a significant conservatism, most especially for the fuel rod internal pressure calculation, is [[

]]. Therefore, no additional conservatism in local exposure monitoring is required to maintain fuel integrity.

The discussion below supports items 1 – 4 above and contains additional information regarding the first paragraph of RAI SRXB-A-65.

As a point of clarification to the first paragraph of the RAI, the response to RAI II.5 in NEDC-32694P-A applies to uncertainties and core monitoring considerations. The LTR covers these topics, as well as their relevance to the SLMCPR methodology. The original RAI response provided a derivation of the uncertainty in the predicted peak LHGR. As discussed in the topical report, the same component uncertainties are incorporated into the SLMCPR. However, the LTR did not directly address how the uncertainties were incorporated [[ ]]. The responses documented in

NEDC-32694P-A accurately describe the uncertainties, but only in terms of their application in the SLMCPR.

The response to SRXB-A-41 indicated a slight increase in the predicted peak LHGR uncertainty. The response also indicated that power uncertainty is considered [[ ]]. The response included the statement "A local uncertainty of [[ ]]." This statement is accurate. This [[ ]] local power uncertainty is utilized with the application of the GESTR thermal-mechanical model [[ ]] for each fuel product line. Additional discussion concerning determination of the exposure-dependent LHGR Operating Limit is given below.

For each GNF fuel design, including GE14 as applied to VYNPS, LHGR operating limits are determined and specified in the form of allowable [[ ]] LHGR as a function of [[ ]] exposure. These fuel rod thermal-mechanical performance based operating limits are specified for each fuel rod type (UO_2 or $(U,Gd)O_2$ for various gadolinia concentrations) so that if each fuel rod type is operated within its respective exposure-dependent LHGR limit, all thermal-mechanical design and licensing criteria, including those which address response to anticipated operational occurrences, are explicitly satisfied.

The exposure-dependent LHGR operating limits are determined through the performance of a number of fuel rod thermal-mechanical analyses. As shown to the NRC staff during the GE Methods audit, an important assumption with these analyses is [[ ]]. This assumption represents a significant conservatism; [[ ]].

With this conservative [[ ]] assumption, the thermal-mechanical analyses are performed either on a worst tolerance basis or statistically. For those analyses performed statistically, such as the fuel rod internal pressure analysis, the uncertainty in each fuel rod fabrication parameter is determined and specifically addressed. The fuel rod thermal-mechanical model prediction uncertainty is also determined and addressed. [[

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For the GE14 fuel rod thermal-mechanical design and licensing analyses, the values of the preceding component uncertainties are: [[

]].

The LHGR Operating Limit is derived for an individual fuel design using the following basic procedure.

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Figure SRXB-A-65-2 is a chart presented to the USNRC in recent discussions to describe the results of the GE14 fuel rod thermal-mechanical design and licensing analyses, and is included here for documentation purposes. The primary result of the fuel rod thermal-mechanical design and licensing analyses is development of the LHGR Operating Limit. The analyses that contribute directly to the development of that limit are the analyses for [[

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In summary, with this methodology, the exposure-dependent LHGR Operating Limit is determined to ensure that the fuel rod thermal-mechanical design and licensing limits, such as the fuel rod internal pressure limit, will not be exceeded [[

]].

References

65-1. Letter from C. O. Thomas (NRC) to J. S. Charnley (GE), "Acceptance for Referencing of Licensing Topical Report NEDE-24011-P-A Amendment 7 to Revision 6, GE Standard Application for Reactor Fuel," March 1, 1985

65-2. Letter from Robert M. Gallo (NRC) to C. P. Kipp (GE), "NRC Inspection Report No. 99900003/96-01," September 10, 1996

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Figure SRXB-A-65-1
VYNPS Cycle 25 Projected Actual Operating History for JLC505 Rod K4 Node 5 - Comparison Between Uprated and Non-Uprated Conditions

(JLC505 Node 5 is the highest projected bundle nodal exposure in VYNPS Cycle 25; rod K4 is the highest exposure rod node in bundle JLC505 Node 5. See further description in Item 2 text above on page 4 of 28.)

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RAI SRXB-A-67

Shutdown Margin (SDM)

In the Alternative Approach and in the RAI responses, VYNPS SDM data was not provided as discussed in the July 12, 2005, telephone conference. As the NRC staff pointed out in the June 30, 2005 meeting, Figure 25-18, "Cold Critical Eigenvalues-All Cycles Studies," of the MFN-05-029 shows that the actual cold eigenvalue tracking of different plants show a scatter of the bias of each plant. However, the uncertainty applied to each plant is obtained by RMS averaging of bias from all plants. Thus, it seems that a bias of 0.38% Δk/k is applied to the calculated core-wide critical keff (in-sequence cold eigenvalue) although the bias from critical (keff = 1.0) may be larger for a given plant. Also, presenting the calculated cold critical eigenvalue alone does not indicate if the critical control rod positions were predicted.

a) Provide the VYNPS cold critical eigenvalues for at least two cycles. Include the recent mid-cycle startup cold critical eigenvalue. Include tables of the predicted keff with the CR withdrawals and indicate predicted critical eigenvalue and the calculated cold critical eigenvalue corresponding to when the core became critical. Evaluate the bias in the VYNPS cold critical eigenvalue data.

b) Provide the actual calculated SDM, with the correction for the period, temperature and peak reactivity.

c) The alternative approach states that for VYNPS "the standard design SDM is 1.1% Δk/k to provide additional flexibility in cycle length and operations." Clarify this statement. Is this an additional margin included to meet the cycle energy needs or is this additional conservatism that ensures SDM for any point in the cycle?

d) The Alternative Approach did not include impact of potential underprediction in reactivity and bundle and pin powers on the SLC system cold shutdown capability. Provide an evaluation of the SLC system shutdown capability and rod withdrawal error analysis.

e) Demonstrate that the [[

]] would not have an important impact when the [[ ]] void fraction and extrapolation to higher voids are used. Also, provide a discussion on what such an under-prediction would have on the accuracy of the local reactivity predictions and what impact, if any, it would have on the SDM, SLC system cold shutdown and rod withdrawal error calculations.

f) The RAI responses stated that the objective is for the eigenvalue trendline to remain constant and consistent from cycle to cycle for a given plant, unless significant change in core loading design results in some change in the trendline. However, the trendline is not a licensing parameter and can be adjusted according to a new trendline fitting a change in the data. The licensing parameter is the SDM.

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Therefore, from a licensing and safety perspective, the difference between the calculated keff for a critical reactor and the deviation from 1.0 is the most important parameter. Explain why it is not desirable for the keff bias and uncertainty to be derived on plant-specific bases. Thus, ensuring a better adjustment applied to the keff bias assumed in the SDM calculations would be based on individual plant's characteristic response and the accuracy of the neutronic methods.

Response to RAI SRXB-A-67

Response to Part (a)

The cold critical eigenvalues for the Vermont Yankee Nuclear Power Station (VYNPS) Cycles 23 and 24 are presented below. The results shown below are for the TGBLA06/PANAC11 set of methods. Because Cycle 24 was the first cycle at VYNPS to be designed and licensed with PANAC11, no predicted eigenvalues had been established for earlier cycles. The previous cycle cold criticals were analyzed with PANAC11 however in order to establish a data base from which the Cycle 24 predicted eigenvalues were developed. The mid-cycle Cycle 23 predicted eigenvalue was established by taking the actual beginning of cycle (BOC) eigenvalue and adjusting it by the standard reduction in cold eigenvalue with cycle exposure (used when sufficient mid-cycle information is not available for a plant). The process for determining predicted cold critical eigenvalues is discussed in the response to part (f) of this request.

The cold eigenvalues shown are very typical for other BWRs operating with GE fuel and analyzed with PANAC11 methods (Reference 67a-1). [[

]]

Cycle	Cycle Exposure (MWd/ST)	Predicted Eigenvalue	Critical Eigenvalue	Difference (Δk)
23	BOC	[[		
	7417			
24	BOC			
	961			]]

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Response to Part (b)

The VYNPS Technical Specification (TS) Shutdown Margin (SDM) is determined following a core reload, at the beginning of each cycle during plant startup. A copy of the demonstrated SDM calculation for the current operating cycle (cycle 24) is attached (see Attachment 8). The calculation involves correcting the SDM for the effects of temperature and period present at the critical measurement. As calculated, the SDM also includes a correction for any difference in peak reactivity at any point in the cycle, R. The period and temperature correction is obtained from the Cycle Management Report, as is the correction for the difference in peak reactivity, R. It should be noted that the temperature correction is a translation to the most reactive condition.

As shown in the accompanying worksheet, Cycle 24 SDM was demonstrated by test to be 1.291. The cycle was designed for Extended Power Uprate (115% CLTP) and a SDM of 1.1, which indicates that the SDM design criterion was easily met for this cycle.

During discussions related to this subject, some other issues were identified by the reviewers, and are addressed below:

The VYNPS TS require that the SDM, at any time there is fuel in the core, shall be greater than or equal to 0.38% $\Delta K/K$ with the analytically determined highest worth rod fully withdrawn. The 0.38% $\Delta K/K$ was determined based upon a statistical combination of allowed manufacturing tolerances and calculational uncertainties. The calculational uncertainties were determined from a statistical analysis of measured and calculated criticals performed at an operating BWR.

Procedurally, if the demonstrated SDM is less than 0.38% $\Delta K/K$, then the shift manager is immediately notified and SDM must be restored within 6 hours or the reactor must be in Hot Shutdown within the next 12 hours. If the corrected critical eigenvalue is different from the expected critical eigenvalue by more than 1% $\Delta K/K$, then the shift manager is immediately notified and the reactor must be shut down until the cause is determined. Additionally, if the corrected critical eigenvalue is different from the expected critical eigenvalue by greater than 0.75% $\Delta K/K$, then the reactor engineering superintendent is notified and a Condition Report is initiated.

Typically, the SDM demonstration is performed during the beginning-of-cycle (BOC) startup. Within the calculation of the demonstrated in-sequence SDM, there is a factor, R, that accounts for a decrease in SDM during the most reactive point in the cycle. This factor is zero when SDM is determined at the most reactive point in the cycle. For those situations when the SDM is not determined at the most reactive point in the cycle, the R factor is subtracted from the demonstrated SDM.

With regard to the effect of the assumed critical eigenvalue and its uncertainty on the demonstrated SDM, the following discussion is offered:

Per the Cycle Management Report (CMR), the equation for SDM is as follows:

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$$SDM = K_{Crit} - K_{SRO} + K_{Temp} - K_{Per} - R$$

where,

K_{Crit} = Eigenvalue when critical is achieved,
K_{SRO} = Eigenvalue with the strongest rod out (SRO),
K_{Temp} = ΔK temperature correction,
K_{Per} = ΔK period correction, and
R = Maximum decrease in SDM throughout the cycle.

But since,

K_{Crit} = K_{eff} with all rods in (ARI) + ΔK of the critical rod pattern (CRP)
= $K_{ARI} + \Delta K_{CRP}$, and

K_{SRO} = K_{eff} with all rods in + ΔK of the strongest rod out
= $K_{ARI} + \Delta K_{SRO}$, then

$$SDM = (K_{ARI} + \Delta K_{CRP}) - (K_{ARI} + \Delta K_{SRO}) + K_{Temp} - K_{Per} - R$$

which simplifies to:

$$SDM = \Delta K_{CRP} - \Delta K_{SRO} + K_{Temp} - K_{Per} - R$$

K_{ARI} is subject to the influence of the assumed critical eigenvalue and its uncertainty. It can be seen from the final equation that K_{ARI} cancels out and the demonstrated SDM is not influenced by the assumed critical eigenvalue or its uncertainty. However, it should be noted that ΔK_{SRO} includes a 0.003 $\Delta K/K$ adjustment to account for the methods bias which occurs when normalizing shutdown margin calculations to a cold eigenvalue derived from in-sequence critical benchmarking data.

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Response to Part (c)

The VYNPS TS 3.3.A.1 requires that any time fuel is the in the core, the core loading shall be limited to that which may be made subcritical in the most reactive condition during the operating cycle with the highest worth, operable control blade fully withdrawn and all other operable rods inserted.

The shutdown margin shall be:

(a) Greater than or equal to 0.38% $\Delta k/k$ with the highest worth rod analytically determined; or

(b) Greater than or equal to 0.28% $\Delta k/k$ with the highest worth rod determined by test.

Entergy confirms sufficient SDM for VYNPS at the BOC based upon greater than or equal to 0.38% $\Delta k/k$.

A failure to meet the Technical Specification SDM requirement is severe in that a redesign of the core loading and/or fuel design would be required to restart the plant. To ensure that $\geq$ 0.38% $\Delta k/k$ is always satisfied, a design margin of 1% SDM has been used by GE for many years. The additional margin between the Technical Specification SDM and 1% allows for the following factors to impact the prediction capability of the simulator:

1. Operation of the plant different than that projected
2. Fuel manufacturing tolerances
3. Control rod worth reduction due to depletion of control rod absorber material
4. Methodology approximations
5. Inexact tracking of actual plant parameters
6. Other unidentified factors

In all of these factors, the most significant factor is allowance for operation different from that projected. VYNPS must maintain sufficient operational flexibility to protect the core and fuel while maintaining acceptable economic objectives. Factors affecting the GE application methodology are quantified through the uncertainty in cold critical eigenvalue and deviation from expectations. These data are provided in the responses to RAIs SRXB-A-67 part (a) and SRXB-A-67 part (b).

The additional 0.1% $\Delta k/k$ that VYNPS requires results from consideration of inverted B_4C tubes in the core. Based upon a total of 82 inverted B_4C tubes in 44 control rods in 1975, a 0.07% $\Delta k/k$ SDM adder was required to compensate for the inverted B_4C tubes. [Reference 67c-1] While there are only 30 inverted B4C tubes in 13 peripherally located control blades, the 0.07% $\Delta k/k$ SDM adder is still being applied until all affected control rods are discharged.

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If the SDM demonstration at VYNPS results in a SDM less than Technical Specification requirement, the plant will take actions as specified in the Technical Specifications.

Response to Part (d)

The standby liquid control system (SLCS) calculation is performed on a cycle specific basis to assure that the plant will remain subcritical in the most reactive condition when the Technical Specification (Tech Spec) minimum requirement for soluble boron is introduced into the core. The calculation is performed as a function of exposure throughout the cycle to determine the minimum SLCS shutdown margin during the cycle. This is an analytical determination, and no actual demonstration of this shutdown capability is performed as is done in the one-rod-out shutdown margin. In order to provide a high degree of assurance that the analytically determined shutdown margin will indeed result in a subcritical condition, a SLCS shutdown margin criteria is established, requiring that the analytically determined shutdown margin be greater than or equal to this shutdown margin criteria. The criteria accounts for all of the biases and uncertainties inherent in the various components of the SLCS methodology.

It should be noted that unlike the one-rod-out shutdown margin, which must be demonstrated subsequent to any reconfiguration of the core, and which is highly sensitive to the local conditions in the four bundles surrounding the withdrawn blade, the SLCS shutdown margin is driven more by core-wide reactivity effects. This makes the calculation less sensitive to nodal uncertainties in exposure and isotopic content, and more dependent on the average exposure and reactivity behavior of the various fuel batches loaded in each cycle.

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NON-PROPRIETARY VERSION

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The severity of the RWE transient is largely dependent on the worth of the rod being withdrawn. The limiting bundle for the RWE for the VYNPS Cycle 25 analysis shows a controlled to uncontrolled $\Delta K\infty$ of approximately [[ ]] Of the four bundles face-adjacent to the error rod, two bundles are approximately [[ ]] including the limiting bundle. The other two bundles are approximately [[ ]]. The higher exposure bundles show a smaller $\Delta K\infty$, [[ ]], and a corresponding lower change in power and CPR during the RWE. The trend of reduction in $\Delta K\infty$, and corresponding lower change in power and CPR during the RWE, continues at exposures greater than [[

]]

<u>Response to Part (e)</u>

[[

]] as the impact on the 0, 40, and 70% void data is minimal. Consequently, this effect does not significantly impact the extrapolation using the 0, 40 and 70% void data to voids higher than 70%.

The above discussion indicates that there is potential for a change in the lattice reactivity of [[

NON-PROPRIETARY VERSION

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To demonstrate the reactivity impacts of this modification to the [[ ]] evaluation, a cycle of plant performance tracking using GE14 fuel in a high power density core was performed using both the current TGBLA production engineering computer program (ECP) and a version of TGBLA that was modified to correct this issue.

The hot core reactivity impact on the core tracking is shown in Figure SRXB-A-67-1 and the impact to Shutdown Margin (SDM) as a function of cycle exposure is shown in Figure SRXB-A-67-2. Table SRXB-A-67-1 and Table SRXB-A-67-2 provide the core reactivity and SDM detailed results comparisons, respectively.

As shown in the figures and tables, [[

]] These levels of impact are not significant compared to the historical uncertainty of these calculated parameters.

[[

]]

The response to NRC RAI SRXB-A-67 part (d) provides a discussion of the impact of this potential reactivity uncertainty on the SLCS SDM and Rod Withdrawal Error (RWE) analyses.

The [[ ]] (see response to NRC RAI SRXB-67d).

Response to Part (f)

The current process is consistent with the expressed concern ("Explain why it is not desirable for the k-eff bias and uncertainty be derived on plant-specific bases."). [[

NON-PROPRIETARY VERSION

]]

Cycle	Cycle Exposure (MWd/ST)	Critical Eigenvalue
21	BOC	[[
22	BOC	
23	BOC	
	7417	
24	BOC	
	961	]]

NON-PROPRIETARY VERSION

References:

67a-1 MFN 05-029, TAC No. MC5780

67c-1 Letter, Dennis L. Ziemann (NRC) to G. Carl Andognini (YAEC), Docket No. 50-271, June 6, 1975

NON-PROPRIETARY VERSION

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NON-PROPRIETARY VERSION

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NON-PROPRIETARY VERSION

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NON-PROPRIETARY VERSION

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Attachment 6

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 263 – Supplement No. 34

Extended Power Uprate – Additional Information

Response to RAI SRXB-A-71

Total number of pages in Attachment 6 (excluding this cover sheet) is 1.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING APPLICATION FOR EXTENDED POWER UPRATE LICENSE AMENDMENT VERMONT YANKEE NUCLEAR POWER STATION

PREFACE

This attachment provides a response to the NRC Reactor Systems Branch's (SRXB) request for additional information (RAI) SRXB-A-71 in NRC's letter dated September 7, 2005.[1] Upon receipt of the RAI, discussions were held with the NRC staff to further clarify the RAI. The intent of individual RAI is addressed based on clarifications reached during these discussions. The information provided herein is consistent with those clarifications.

The RAI is re-stated as provided in NRC's letter of September 7, 2005.

RAI SRXB-A-71

In the response to RAI SRXB-A-6, the licensee stated "the reactivity events are analyzed with the steady state tools and the results presented regarding steady-state methods in this response are directly applicable. There are some increases in power, which are significant but remain within the comparisons between the above plants for corresponding events." This RAI response does not provide sufficient detail. The response to RAI SRXB-A-57 requested clarification to the above quoted statement. The generic event sequence was described, rather than explaining the statement in the initial RAI response. Please explain the intent of the statement in the initial submittal.

Response to RAI SRXB-A-71

The intent of the statement in quotations was that the VYNPS events analyzed with the 3D core thermal-hydraulic PANACEA model, such as the Rod Withdrawal Error and Fuel Loading Error, started from conditions within the range in other analyses as shown in Figures 6-1 through 6-6 of the response to RAI SRXB-A-6[2]. No comparison was made against the events analyzed with the steady state methods for the other plants of Figures 6-1 through 6-6 because of differences in the plant size, core design and loading, rod block monitor setup, power distribution and control rod patterns, which result in inconsistent comparisons.

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[1] U.S. Nuclear Regulatory Commission (Richard B. Ennis) letter to Entergy Nuclear Operations, Inc. (Michael Kansler), "Request for Additional Information – Extended Power Uprate, Vermont Yankee Nuclear Power Station (TAC No. MC0761)," September 7, 2005

[2] Entergy letter to U.S. Nuclear Regulatory Commission, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263 – Supplement No. 24, Extended Power Uprate – Response to Request for Additional Information," BVY 05-024, March 10, 2005

Attachment 7

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 263 – Supplement No. 34

Extended Power Uprate – Additional Information

General Electric Affidavit

Total number of pages in Attachment 7
(excludina this cover sheet) is 3.

General Electric Company

AFFIDAVIT

I, George B. Stramback, state as follows:

(1) I am Manager, Regulatory Services, General Electric Company ("GE"), 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 Enclosure 2 of GE letter, GE-VYNPS-AEP-403, *Responses to NRC RAIs SRXB-64, 65, 67, and 71*, dated September 16, 2005. The proprietary information in Enclosure 2, *Responses to NRC RAIs SRXB-64, 65, 67, and 71*, is delineated by a double underline inside double square brackets. Figures and large equation objects are identified with double square brackets before and after the object. In each case, the superscript notation[3] refers to Paragraph (3) of this affidavit, which provides the basis for the proprietary determination.

(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), and 2.390(a)(4) for "trade secrets" (Exemption 4). The material for which exemption from disclosure is here sought 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 aspects of past, present, or future General Electric customer-funded development plans and programs, resulting in potential products to General Electric;

d. 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 paragraphs (4)a., and (4)b, above.

(5) To address 10 CFR 2.390 (b) (4), 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 detailed results and conclusions from analyses supporting the extended power uprate of the Vermont Yankee Power Station utilizing analytical models and methods including computer codes and methods of applying these for safety analyses, which GE has developed. The development of these models and computer codes and methods was achieved at a significant cost to GE, on the order of several million dollars.

The development of the analytical methods and 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.

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.

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information, and belief.

Executed on this 16 th of September 2005.

George B. Stramback
George B. Stramback
General Electric

Attachment 8

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 263 – Supplement No. 34

Extended Power Uprate – Additional Information

Demonstrated Shutdown Margin

Total number of pages in Attachment 8 (excluding this cover sheet) is 15.

IN-SEQUENCE CRITICAL/SDM WORKSHEET

1. Prerequisites Met	T.G. Stetson Printed Name	[signature] Signature	

2. Conditions at Criticality		
Date/Time of Criticality	10:58 / 5-3-04	
Moderator Temperature T_{mod}	152.6 °F	Computer point B023 preferred
Reactor Period P	186.4 seconds	
Critical Rod Location	26-15	
Notch Position	36	

3. Calculation of Control Rod Density		
Total Rod Positions Withdrawn, N (e.g. 1 rod 00→02 is 2 positions)	1044	
Control Rod Density (CRD) = (4272 - N)/4272	.756	Can be verified via Computer Point C116 ✓

4. Calculation of Uncorrected Critical Eigenvalue		
k_{eff} of critical rod at analyzed withdraw limit k_1	1.0042	Per core design vendor or qualified substitute
Fraction of analyzed step withdrawn at criticality Fr	1.0	
k_{eff} of rod prior to criticality at its withdraw limit k_2	1.0030	Per core design vendor or qualified substitute
Uncorrected Critical Eigenvalue $k_{un} = Fr * (k_1 - k_2) + k_2$	1.0042	

5. Calculation of Corrected Critical Eigenvalue		
Period Correction k_{per}	.00034 Δk	Per core design vendor or qualified substitute
Temperature Correction k_{temp}	-.00294 Δk	Per core design vendor or qualified substitute
Corrected Critical Eigenvalue $k_{corr} = k_{un} - k_{per} + k_{temp}$	1.00092 Δk	

6. Calculation of Demonstrated SDM		
Eigenvalue with strongest rod out k_{SRO}	0.9873 Δk	Per core design vendor or qualified substitute
Decrease in SDM with exposure over cycle R_1	0 Δk	Per core design vendor or qualified substitute. *The value of R_1 is equal to zero when the SDM test is performed at the most reactive point during the cycle (in other words, when BOC is the most reactive point in the cycle).*
Potential SDM loss from inverted CR tubes R_2	0.0007 Δk	Per Reference 3.a
Demonstrated SDM $SDM_{demon} = 100 * (k_{corr} - k_{SRO} - R_1 - R_2)/k_{corr}$	1.291 %Δk/k	

7. Calculation of Difference Between Corrected and Expected Critical Eigenvalue, Diff		
Expected Critical Eigenvalue k_{exp}	1.000	
Difference Between Corrected and Expected Critical Eigenvalue $Diff = 100 * \lvert k_{exp} - k_{corr} \rvert$	0.092 % Δk/k	

8. Comments, Remarks, or Discrepancies:

9. Calculations Performed By	T. G-S Letsen Printed Name	[signature] Signature	5/3/04 / 11:25 Date/Time
10. Calculations Verified By	Edward Lindsay Printed Name	[signature] Signature	5/3/04 / 11:50 Date

IN-SEQUENCE CRITICAL/SDM WORKSHEET (Continued)

VERIFICATION OF TECHNICAL SPECIFICATION ACCEPTANCE CRITERIA			
11. Verification of Tech. Spec. Acceptance Criteria, $SDM_{demon} \geq 0.38$			
$SDM_{demon} \geq 0.38$ (Check applicable box)	Yes ✓	No	
Verified By (Printed Name and Signature)	T.G. Stetkar Printed Name	T. S. Stetkar Signature	5/3/04 Date
Verified By (Printed Name and Signature)	Edward Lindsay Printed Name	E Lindsay Signature	5/3/04 Date

IF $SDM_{demon} < 0.38$, THEN notify Shift Supervisor immediately that SDM must be restored within 6 hours or the reactor must be in hot shutdown within the next 12 hours per Tech. Spec. 3.3.A.1

12. Verification of Tech. Spec. Acceptance Criteria, -1% ≤ Diff ≤1%			
-1% ≤ Diff ≤1% (Check applicable box)	Yes ✓	No	
Verified By (Printed Name and Signature)	T.G. Stetkar Printed Name	T. Stetkar Signature	5/3/04 Date
Verified By (Printed Name and Signature)	Ed Lindsay Printed Name	E Lindsay Signature	5/3/04 Date

IF Tech Spec Acceptance Criteria of -1% ≤ Diff ≤ +1% is not met, THEN notify Shift Supervisor immediately that reactor must be shut down until cause has been determined and corrective actions have been taken if such actions are appropriate, per Tech Spec 3.3.E.

VERIFICATION OF ADMINISTRATIVE LIMIT ACCEPTANCE CRITERIA			
13. Verification of Admin Acceptance Criteria, -0.75% ≤ Diff ≤0.75%			
-0.75% ≤ Diff ≤ 0.75% (Check applicable box)	Yes ✓	No	
Verified By (Printed Name and Signature)	T.G. Stetkar Printed Name	T. Stetkar Signature	5/3/04 Date
Verified By (Printed Name and Signature)	Edward Lindsay Printed Name	E Lindsay Signature	5/3/04 Date

IF Administrative Acceptance Criteria of -0.75 ≤ Diff ≤ +0.75% is not met, THEN notify Superintendent, Reactor Engineering immediately and initiate an ER.

SURVEILLANCE REVIEW			
14. Shift Supervisor Review By	Bob Faupel Printed Name	Robert Faupel Signature	5/3/04 Date
15. Superintendent, Reactor Engineering Review By	David J. Mannai Printed Name	DJ Signature	5/3/04 Date

6. Shutdown Margin Demonstration

6.1 Introduction

Technical specifications require that the core K-effective not exceed 0.9962 with the strongest operable control rod fully withdrawn and all other rods fully inserted at any time during the cycle. The BOC shutdown margin test demonstrates that this requirement is met after core alterations are made. The test is done in a cold xenon free state.

6.2 Evaluation of Shutdown Margin - In-Sequence Critical

Table 6-1 shows the order of in-sequence rod withdrawals for the A sequence (groups one through four) which were assumed for these calculations. Predicted eigenvalues corresponding to these in-sequence rod withdrawals are reported in Table 6-2. The amount of shutdown margin demonstrated can be calculated using the following equation:

$$SDM = K_{Crit} - K_{SRO} + K_{Temp} - K_{Per} - R$$

Where,

K_{Crit}	=	Eigenvalue when critical is achieved	Table 6-2
K_{SRO}	=	Eigenvalue with strongest rod out	0.9873[1]
K_{Temp}	=	ΔK temperature correction	See Figure or Table
K_{Per}	=	ΔK-period correction	See Figure or Table
R	=	Maximum decrease in Shutdown Margin (SDM) from BOC	0.0000
		Location of strongest rod at BOC	34-11

The predictive values shown below plus the measured point of criticality when used in the shutdown margin equation will determine the amount of shutdown margin demonstrated. The values reported assume a moderator temperature of 68 degrees F and an infinite period. Temperature and period corrections are provided both graphically and in tabular form as functions of exposure in subsequent figures and tables. These corrections are included in the shutdown margin equation shown above. The 'R' value provided is the calculated decrease in shutdown margin from BOC to the cycle exposure where shutdown is a minimum.

The last figure in this section shows the expected shutdown margin behavior through the cycle.

[1] The reported K_{SRO} includes a 0.003 ΔK adjustment to account for the methods bias which occurs when normalizing shutdown margin calculations to a cold eigenvalue derived from in-sequence critical benchmarking data.

Attachment to VYOPF 4430.03
(05/03/04)
Page 4 of 15

Table 6-1 Sequence A Rod Withdrawals
Assumed Order

First Group Pull Location		Second Group Pull Location		Third Group Pull Location		Fourth Group Pull Location	
1	26-23	1	34-39	1	22-27	1	6-35
2	18-7	2	34-7	2	38-19	2	38-35
3	2-23	3	10-7	3	22-3	3	38-11
4	18-39	4	10-39	4	6-19	4	6-11
5	34-31	5	34-23	5	14-35	5	22-35
6	34-15	6	10-23	6	30-35	6	22-11
7	10-15	7	26-31	7	30-11	7	14-27
8	10-31	8	18-15	8	14-11	8	30-19
9	26-39	9	18-31	9	6-27	9	14-19
10	42-23	10	26-15	10	22-43	10	30-27
11	26-7		-	11	38-27		-
12	18-23		-	12	22-19		-

Attachment to VYOPF 4430.03
(05/03/04)
Page 5 of 15

Table 6-2 Sequence A Reactivity During In-Sequence Critical

Rod Group[1]	Number of Rods Withdrawn	Notch Position[2] From - To	Predicted Eigenvalue		Pos w/D
ARI[3]	0		0.9485		
1	1	0 - 48	0.9805	26-23	
1	2	0 - 48	0.9830	18-07	
1	3	0 - 48	0.9830	02-23	
1	4	0 - 48	0.9832	18-39	
1	5	0 - 48	0.9853	34-31	
1	6	0 - 48	0.9868	34-15	
1	7	0 - 48	0.9875	10-15	
1	8	0 - 48	0.9881	10-31	
1	9	0 - 48	0.9936	26-39	
1	10	0 - 48	0.9937	42-23	
1	11	0 - 48	0.9938	26-07	
1	12	0 - 48	0.9950	18-23	576
2	1	0 - 4	0.9950		
2	1	4 - 8	0.9950		
2	1	8 - 12	0.9950		
2	1	12 - 16	0.9950	34-39	
2	1	16 - 20	0.9951		
2	1	20 - 24	0.9951		
2	1	24 - 48	0.9954		624
2	2	0 - 4	0.9954		
2	2	4 - 8	0.9954		
2	2	8 - 12	0.9954	34-07	
2	2	12 - 16	0.9954		
2	2	16 - 20	0.9954		
2	2	20 - 24	0.9955		
2	2	24 - 48	0.9956		672
2	3	0 - 4	0.9956		
2	3	4 - 8	0.9957		
2	3	8 - 12	0.9957		
2	3	12 - 16	0.9957	10-07	
2	3	16 - 20	0.9957		
2	3	20 - 24	0.9957		
2	3	24 - 48	0.9960		720
2	4	0 - 4	0.9960		
2	4	4 - 8	0.9960	10-39	
2	4	8 - 12	0.9960		

1 Rod Group as defined in Table 6-1.
2 Notch Position indicates the number of notches withdrawn (48 Notches / Control Rod).
3 ARI denotes all rods fully inserted.

Attachment to VYOPF 4430.03
(05/03/04)
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Table 6-2 Sequence A Reactivity During In-Sequence Critical

Rod Group[1]	Number of Rods Withdrawn	Notch Position[2] From - To	Predicted Eigenvalue	Rs w/o
2	4	12 - 16	0.9960	
2	4	16 - 20	0.9960	10-39
2	4	20 - 24	0.9960	
2	4	24 - 48	0.9962	768
2	5	0 - 4	0.9962	
2	5	4 - 8	0.9962	
2	5	8 - 12	0.9962	
2	5	12 - 16	0.9962	34-23
2	5	16 - 20	0.9962	
2	5	20 - 24	0.9963	
2	5	24 - 48	0.9967	816
2	6	0 - 4	0.9967	
2	6	4 - 8	0.9967	
2	6	8 - 12	0.9967	
2	6	12 - 16	0.9967	10-23
2	6	16 - 20	0.9967	
2	6	20 - 24	0.9968	
2	6	24 - 48	0.9971	864
2	7	0 - 4	0.9971	
2	7	4 - 8	0.9971	
2	7	8 - 12	0.9972	
2	7	12 - 16	0.9973	
2	7	16 - 20	0.9974	
2	7	20 - 24	0.9979	26-31
2	7	24 - 30	0.9990	
2	7	30 - 36	0.9996	
2	7	36 - 42	0.9998	
2	7	42 - 48	0.9998	912
2	8	0 - 4	0.9998	
2	8	4 - 8	0.9998	
2	8	8 - 12	0.9998	
2	8	12 - 14	0.9999	
2	8	14 - 16	0.9999	
2	8	16 - 20	1.0000	18-25
2	8	20 - 24	1.0001	
2	8	24 - 30	1.0006	
2	8	30 - 36	1.0009	
2	8	36 - 42	1.0010	
2	8	42 - 48	1.0010	960
2	9	0 - 4	1.0010	
2	9	4 - 8	1.0010	18-31
2	9	8 - 12	1.0010	

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Table 6-2 Sequence A Reactivity During In-Sequence Critical

Rod Group[1]	Number of Rods Withdrawn	Notch Position[2] From - To	Predicted Eigenvalue		Bs W/D
2	9	12 - 16	1.0011		
2	9	16 - 20	1.0013		
2	9	20 - 24	1.0017	18-31	
2	9	24 - 30	1.0025		
2	9	30 - 36	1.0029		
2	9	36 - 42	1.0030		
2	9	42 - 48	1.0030		1008
2	10	0 - 4	1.0030		
2	10	4 - 8	1.0031		
2	10	8 - 12	1.0031		
2	10	12 - 16	1.0031		
2	10	16 - 20	1.0032		
2	10	20 - 24	1.0034	26-15	
2	10	24 - 30	1.0039		
2	10	30 - 36	1.0042		
2	10	36 - 42	1.0042		
2	10	42 - 48	1.0042		1056
3	1	0 - 4	1.0043	22-27	
3	2	0 - 4	1.0043	35-19	
3	3	0 - 4	1.0043	22-03	
3	4	0 - 4	1.0043	06-19	
3	5	0 - 4	1.0043	14-35	
3	6	0 - 4	1.0043	30-35	
3	7	0 - 4	1.0043	30-11	
3	8	0 - 4	1.0043	14-11	
3	9	0 - 4	1.0043	06-27	
3	10	0 - 4	1.0043	22-43	
3	11	0 - 4	1.0043	38-27	
3	12	0 - 4	1.0043	22-19	1104
3	1	4 - 6	1.0044		
3	2	4 - 6	1.0044		
3	3	4 - 6	1.0044		
3	4	4 - 6	1.0044		
3	5	4 - 6	1.0044		
3	6	4 - 6	1.0044		
3	7	4 - 6	1.0044		
3	8	4 - 6	1.0044		
3	9	4 - 6	1.0044		
3	10	4 - 6	1.0044		
3	11	4 - 6	1.0044		
3	12	4 - 6	1.0046		1128
3	1	6 - 8	1.0050		

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(05/03/04)
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Table 6-2 Sequence A Reactivity During In-Sequence Critical

Rod Group[1]	Number of Rods Withdrawn	Notch Position[2] From - To	Predicted Eigenvalue
3	2	6 - 8	1.0050
3	3	6 - 8	1.0050
3	4	6 - 8	1.0050
3	5	6 - 8	1.0051
3	6	6 - 8	1.0051
3	7	6 - 8	1.0051
3	8	6 - 8	1.0051
3	9	6 - 8	1.0051
3	10	6 - 8	1.0052
3	11	6 - 8	1.0052
3	12	6 - 8	1.0060
3	1	8 - 10	1.0075
3	2	8 - 10	1.0075
3	3	8 - 10	1.0075
3	4	8 - 10	1.0075
3	5	8 - 10	1.0076
3	6	8 - 10	1.0077
3	7	8 - 10	1.0077
3	8	8 - 10	1.0078
3	9	8 - 10	1.0078
3	10	8 - 10	1.0078
3	11	8 - 10	1.0078
3	12	8 - 10	1.0095
3	1	10 - 12	1.0113
3	2	10 - 12	1.0113
3	3	10 - 12	1.0113
3	4	10 - 12	1.0113
3	5	10 - 12	1.0114
3	6	10 - 12	1.0114
3	7	10 - 12	1.0115
3	8	10 - 12	1.0115
3	9	10 - 12	1.0115
3	10	10 - 12	1.0115
3	11	10 - 12	1.0116
3	12	10 - 12	1.0132
3	1	12 - 48	1.0188
3	2	12 - 48	1.0188
3	3	12 - 48	1.0189
3	4	12 - 48	1.0190
3	5	12 - 48	1.0213
3	6	12 - 48	1.0230
3	7	12 - 48	1.0233

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Table 6-2 Sequence A Reactivity During In-Sequence Critical

Rod Group[1]	Number of Rods Withdrawn	Notch Position[2] From - To	Predicted Eigenvalue
3	8	12 - 48	1.0239
3	9	12 - 48	1.0247
3	10	12 - 48	1.0250
3	11	12 - 48	1.0256
3	12	12 - 48	1.0288
4	1	0 - 4	1.0288
4	2	0 - 4	1.0288
4	3	0 - 4	1.0288
4	4	0 - 4	1.0288
4	5	0 - 4	1.0288
4	6	0 - 4	1.0288
4	7	0 - 4	1.0288
4	8	0 - 4	1.0288
4	9	0 - 4	1.0288
4	10	0 - 4	1.0288
4	1	4 - 6	1.0288
4	2	4 - 6	1.0288
4	3	4 - 6	1.0288
4	4	4 - 6	1.0288
4	5	4 - 6	1.0288
4	6	4 - 6	1.0288
4	7	4 - 6	1.0288
4	8	4 - 6	1.0288
4	9	4 - 6	1.0289
4	10	4 - 6	1.0289
4	1	6 - 8	1.0289
4	2	6 - 8	1.0289
4	3	6 - 8	1.0289
4	4	6 - 8	1.0289
4	5	6 - 8	1.0289
4	6	6 - 8	1.0289
4	7	6 - 8	1.0289
4	8	6 - 8	1.0289
4	9	6 - 8	1.0290
4	10	6 - 8	1.0290
4	1	8 - 10	1.0290
4	2	8 - 10	1.0290
4	3	8 - 10	1.0290
4	4	8 - 10	1.0290
4	5	8 - 10	1.0290
4	6	8 - 10	1.0290
4	7	8 - 10	1.0291

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Table 6-2 Sequence A Reactivity During In-Sequence Critical

Rod Group[1]	Number of Rods Withdrawn	Notch Position[2] From - To	Predicted Eigenvalue
4	8	8 - 10	1.0291
4	9	8 - 10	1.0291
4	10	8 - 10	1.0292
4	1	10 - 12	1.0292
4	2	10 - 12	1.0292
4	3	10 - 12	1.0292
4	4	10 - 12	1.0292
4	5	10 - 12	1.0292
4	6	10 - 12	1.0292
4	7	10 - 12	1.0293
4	8	10 - 12	1.0293
4	9	10 - 12	1.0294
4	10	10 - 12	1.0295
4	1	12 - 48	1.0296
4	2	12 - 48	1.0298
4	3	12 - 48	1.0299
4	4	12 - 48	1.0301
4	5	12 - 48	1.0312
4	6	12 - 48	1.0318
4	7	12 - 48	1.0337
4	8	12 - 48	1.0349
4	9	12 - 48	1.0364
4	10	12 - 48	1.0375

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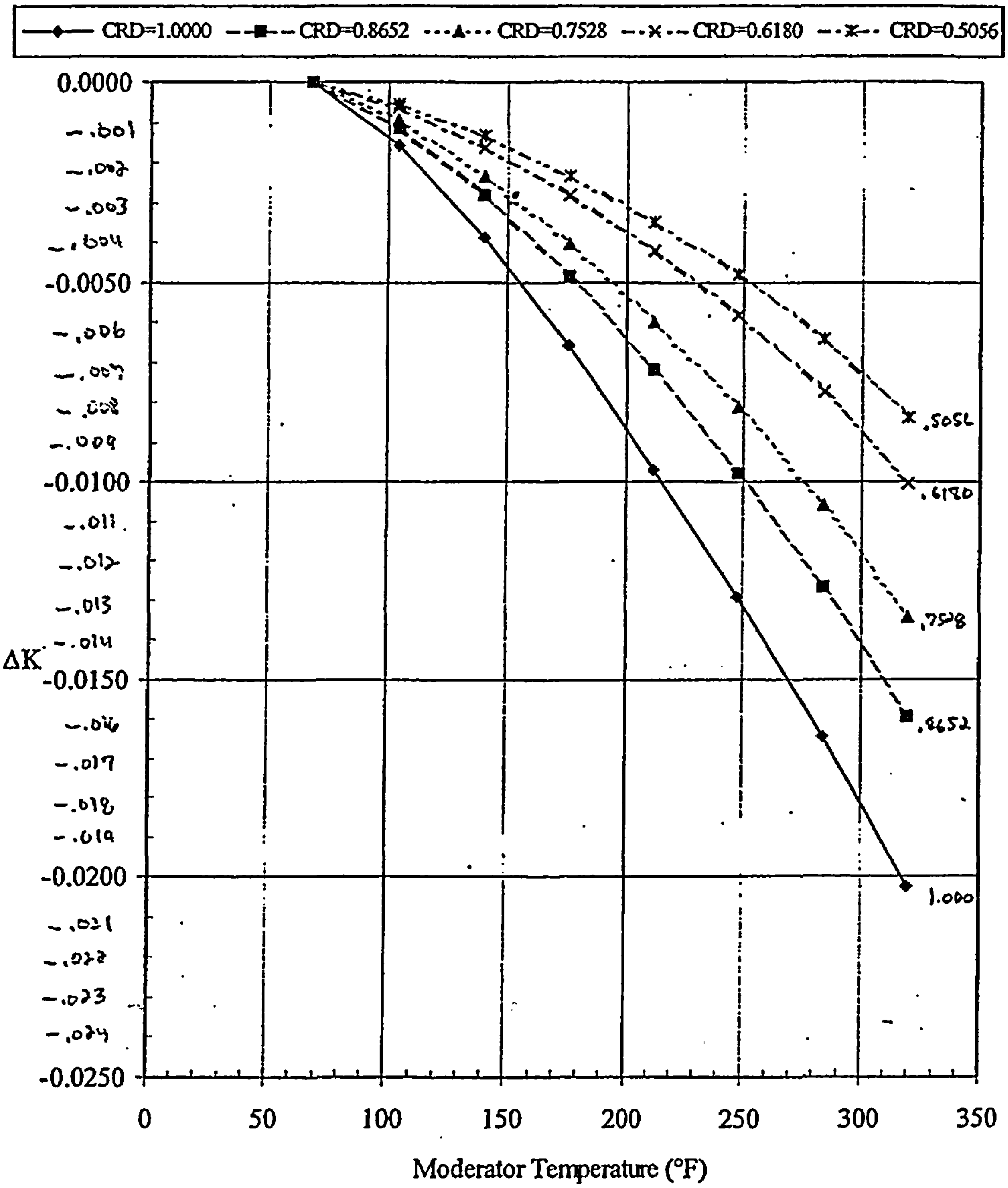
Figure 6-1 Moderator Temperature Correction ΔK[1]
(Cycle Average Exposure = 0.0 MWd/ST)
(Core Average Exposure = 13726.9 MWd/ST)



[1] See Table 6-3 for the numeric data.

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Table 6-3 Moderator Temperature Correction ΔK
(Cycle Average Exposure = 0.0 MWd/ST)
(Core Average Exposure = 13726.9 MWd/ST)

Moderator Temperature (°F)	CRD=1.0000	CRD=0.8652	CRD=0.7528	CRD=0.6180	CRD=0.5056
68.	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
104.	-0.1571E-02	-0.1135E-02	-0.9428E-03	-0.6560E-03	-0.5413E-03
140.	-0.3856E-02	-0.2805E-02	-0.2323E-02	-0.1621E-02	-0.1338E-02
176.	-0.6608E-02	-0.4855E-02	-0.4023E-02	-0.2816E-02	-0.2323E-02
212.	-0.9677E-02	-0.7198E-02	-0.5974E-02	-0.4217E-02	-0.3481E-02
248.	-0.1292E-01	-0.9759E-02	-0.8128E-02	-0.5812E-02	-0.4810E-02
284.	-0.1645E-01	-0.1267E-01	-0.1060E-01	-0.7722E-02	-0.6417E-02
320.	-0.2022E-01	-0.1595E-01	-0.1342E-01	-0.1002E-01	-0.8378E-02

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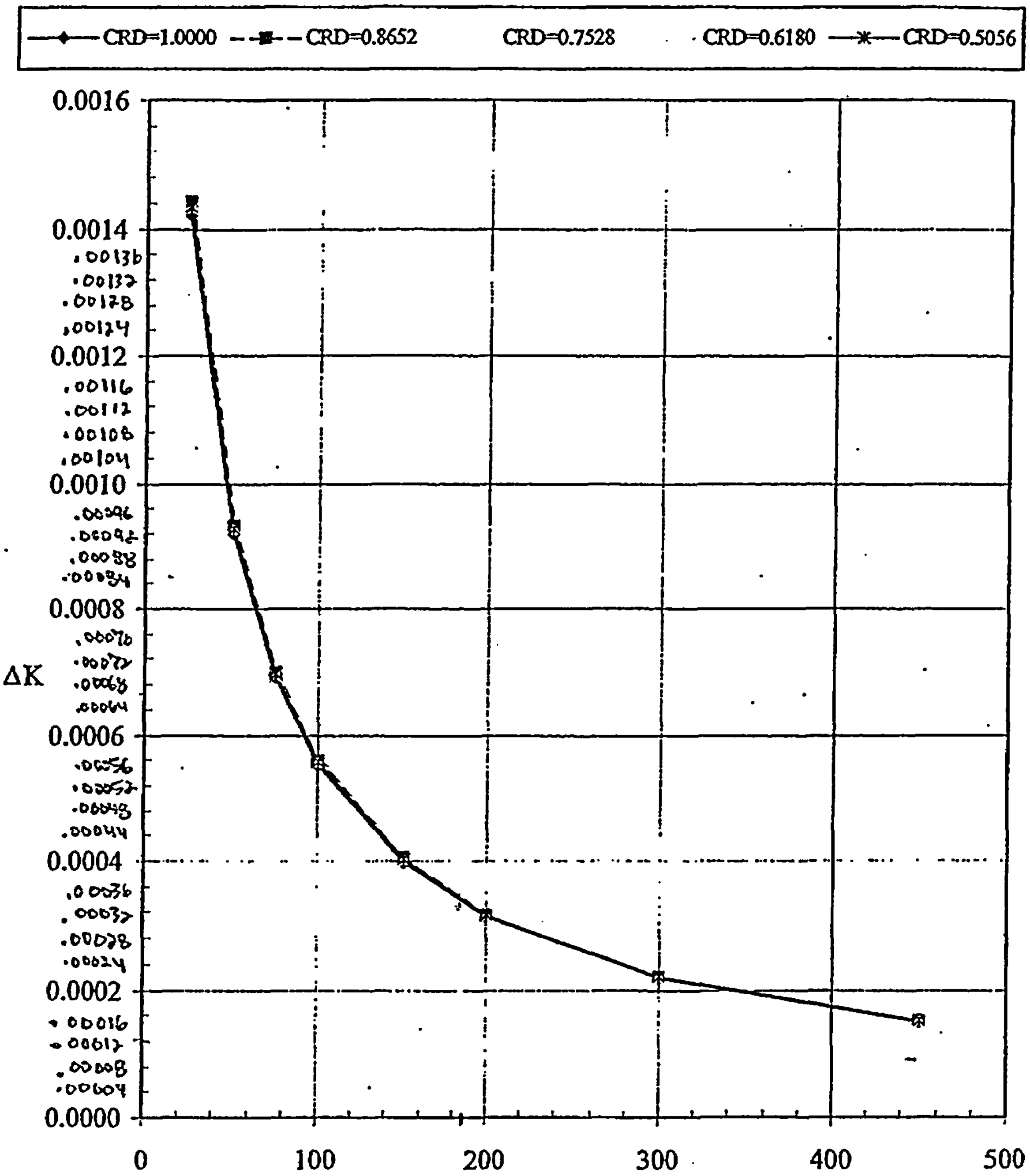
Figure 6-2 Reactor Period Correction ΔK[1]
(Cycle Average Exposure = 0.0 MWd/ST)
(Core Average Exposure = 13726.9 MWd/ST)



[1] See Table 6-4 for the numeric data.

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Table 6-4 Reactor Period Correction ΔK
(Cycle Average Exposure = 0.0 MWd/ST)
(Core Average Exposure = 13726.9 MWd/ST)

Period (sec.)	CRD=1.0000	CRD=0.8652	CRD=0.7528	CRD=0.6180	CRD=0.5056
25.	0.1421E-02	0.1442E-02	0.1434E-02	0.1432E-02	0.1434E-02
50.	0.9207E-03	0.9339E-03	0.9290E-03	0.9277E-03	0.9288E-03
75.	0.6898E-03	0.6997E-03	0.6960E-03	0.6950E-03	0.6958E-03
100.	0.5537E-03	0.5617E-03	0.5587E-03	0.5579E-03	0.5586E-03
150.	0.3986E-03	0.4043E-03	0.4021E-03	0.4016E-03	0.4021E-03
200.	0.3119E-03	0.3164E-03	0.3147E-03	0.3143E-03	0.3146E-03
300.	0.2177E-03	0.2209E-03	0.2197E-03	0.2194E-03	0.2196E-03
450.	0.1500E-03	0.1522E-03	0.1514E-03	0.1512E-03	0.1513E-03

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