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SUBJECT: Application for amend to license NPF-21,reflecting change in name of Washington Public Power Supply Sys to Energy Northwest.Marked up copy of affected pages of OL for plant, encl.

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WASHINGTON PUBLIC POWER SUPPLY SYSTEM
P.O. Box 968 • Richland, Washington 99352-0968

June 3, 1999
GO2-99-102

Docket No. 50-397

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Gentlemen:

Subject: **WNP-2 OPERATING LICENSE NPF-21**
REQUEST FOR AMENDMENT
LICENSEE NAME CHANGE

Pursuant to 10 CFR 50.90, this letter transmits an operating license amendment request for the WNP-2 Operating License (OL). It is requested that an amendment be made to update the OL such that the name of the licensee "Washington Public Power Supply System" is changed to "Energy Northwest." The need for this request results from the change in the name of the Washington Public Power Supply System to Energy Northwest.

No impact on the status of the OL or the continued operation of WNP-2 is foreseen, since this request contains a proposed change that is solely administrative in nature.

The attachments to this letter are as follows:

Attachment 1 provides a description of the proposed change.

Attachment 2 documents, pursuant to 10 CFR 50.92, the determination that the proposed amendment contains No Significant Hazards Considerations.

Attachment 3 provides, pursuant to 10 CFR 51.22(c)(9) and (10), the basis for the categorical exclusion from performing an Environmental Assessment/Impact Statement.

Attachment 4 provides a marked up copy of the affected pages of the OL for WNP-2.

A001

This amendment request has been reviewed by the Corporate Nuclear Safety Review Board and approved by the WNP-2 Plant Operations Committee. In accordance with 10 CFR 50.91, the State of Washington has been provided a copy of this letter.

Should you have any questions or desire additional information regarding this matter, please contact Mr. PJ Inserra at (509) 377-4147.

Respectfully,

JV Parrish
Chief Executive Officer
Mail Drop 1023

Attachments: as stated

cc: EW Merschoff - NRC RIV
JS Cushing - NRR
NRC Sr. Resident Inspector - 927N
DL Williams - BPA/1399
PD Robinson - Winston & Strawn
DJ Ross - EFSEC

STATE OF WASHINGTON)
)
COUNTY OF BENTON)

Subject: Request For Amendment
Name Change

I, J. V. PARRISH, being duly sworn, subscribe to and say that I am the Chief Executive Officer for ENERGY NORTHWEST, the applicant herein; that I have the full authority to execute this oath; that I have reviewed the foregoing; and that to the best of my knowledge, information, and belief the statements made in it are true.

DATE June 3, 1999

J. V. Parrish
Chief Executive Officer

On this date personally appeared before me J. V. PARRISH, to me known to be the individual who executed the foregoing instrument, and acknowledged that he signed the same as his free act and deed for the uses and purposes herein mentioned.

GIVEN under my hand and seal this 3rd day of June 1999.

Notary Public in and for the
STATE OF WASHINGTON

JULIE MARBOE
COMMISSION EXPIRES
NOTARY
PUBLIC
9-5-02
STATE OF WASHINGTON

Residing at Kennewick, WA

My Commission Expires 9/05/02

ATTACHMENT 1

REQUEST FOR AMENDMENT
LICENSEE NAME CHANGE

Description of Proposed Change

[illegible]

[illegible]

[illegible]

[illegible]

DESCRIPTION OF PROPOSED CHANGE

Proposed Change

With this submittal, the NRC is being requested to replace references to the name Washington Public Power Supply System, with references to the name Energy Northwest in all applicable locations of the Operating License (OL) for WNP-2. Currently, the OL states that Washington Public Power Supply System is the NRC licensee. This request also applies to Appendix A (Technical Specifications) and Appendix B (Environmental Protection Plan) to the OL. The Operating License number for WNP-2 is NPF-21.

Similarly, in any pending applications or license amendments heretofore submitted by Washington Public Power Supply System, but not yet acted upon by the NRC, references to Washington Public Power Supply System, should also be replaced by Energy Northwest. This administrative name change will also be reflected in future correspondence with the NRC.

Discussion

The proposed change is solely administrative in nature and involves only a name change. This request is being submitted to the NRC pursuant to 10 CFR 50.90 only for the purpose of updating the affected OL documents. The proposed change does not alter any technical content of the OL or any technical content of the WNP-2 Technical Specifications requirements, nor do they have any programmatic effect on the Washington Public Power Supply System Operational Quality Assurance Program Description. The change will have no impact on the design, function, or operation of any plant structure, system, or component, either technically or administratively.

ATTACHMENT 2

REQUEST FOR AMENDMENT
LICENSEE NAME CHANGE

No Significant Hazards Consideration

NO SIGNIFICANT HAZARDS CONSIDERATION EVALUATION

Pursuant to 10 CFR 50.92, it has been determined that this request involves No Significant Hazards Considerations. The determination of no significant hazards was made by applying the NRC established standards contained in 10 CFR 50.92. These standards assure that any changes to the operation of WNP-2 in accordance with this request, consider the following:

1) Will the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   No. This request involves an administrative change only. The Operating License (OL) is being changed to reference the new name of the licensee. No actual plant equipment or accident analyses will be affected by the proposed change. Therefore, this request will have no impact on the probability or consequence of any type of accident previously evaluated.

2) Will the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   No. This request involves an administrative change only. The OL is being changed to reference the new name of the licensee. No actual plant equipment or accident analyses will be affected by the proposed change and no failure modes not bounded by previously evaluated accidents will be created. Therefore, this request will have no impact on the possibility of any type of accident: new, different, or previously evaluated.

3) Will the change involve a significant reduction in a margin of safety?

   No. Margin of safety is associated with confidence in the ability of the fission product barriers (i.e., fuel and fuel cladding, Reactor Coolant System pressure boundary, and containment structure) to limit the level of radiation dose to the public. This request involves an administrative change only. The OL is being changed to reference the new name of the licensee.

   No actual plant equipment or accident analyses will be affected by the proposed change. Additionally, the proposed change will not relax any criteria used to establish safety limits, will not relax any safety system settings, or will not relax the bases for any limiting conditions of operation. Therefore, this request will not impact margin of safety.

ATTACHMENT 3

REQUEST FOR AMENDMENT
LICENSEE NAME CHANGE

Environmental Assessment/Impact Statement

ENVIRONMENTAL ASSESSMENT/IMPACT STATEMENT

Pursuant to 10 CFR 51.22(b), an evaluation of this request has been performed to determine whether or not it meets the criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (10) of the regulations.

This request involves an administrative change only. The proposed change updates the Operating License (OL) such that references to the licensee name will be consistent with the new name, Energy Northwest.

Additionally, this request will have no adverse radiation impact upon the environment, since it only applies to the name of the licensee designated in the OL. It has been determined that the proposed change involves.

1) No significant hazards consideration,

2) No significant change in the types, or significant increase in the amounts, of any effluents that may be released offsite, and

3) No significant increase in individual or cumulative occupational radiation exposures.

Therefore, this request regarding the OL meets the criteria of 10 CFR 51.22(c)(9) and (10) for categorical exclusion from an environmental assessment/impact statement.

ATTACHMENT 4

REQUEST FOR AMENDMENT
LICENSEE NAME CHANGE

Marked-Up Operating License Pages

ENERGY NORTHWEST

~~WASHINGTON PUBLIC POWER SUPPLY SYSTEM~~

DOCKET NO. 50-397

WNP-2

~~WPPSS NUCLEAR PROJECT NO. 2~~

FACILITY OPERATING LICENSE

License No. NPF-21

1. The Nuclear Regulatory Commission (the Commission or the NRC) has found that:

   A. The application for license filed by ENERGY NORTHWEST ~~the Washington Public Power Supply System~~ (~~WPPSS,~~ also the licensee), complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I, and all required notifications to other agencies or bodies have been duly made;

   B. Construction of ENERGY NORTHWEST ~~Washington Public Power Supply System~~, Nuclear Project No. 2 (the facility) has been substantially completed in conformity with Construction Permit No. CPPR-93 and the application, as amended, the provisions of the Act, and the regulations of the Commission;

   C. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission (except as exempted from compliance in Section 2.D. below);

   D. There is reasonable assurance: (i) that the activities authorized by this operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I (except as exempted from compliance in Section 2.D. below);

   E. ENERGY NORTHWEST ~~The Washington Public Power Supply System~~ is technically qualified to engage in the activities authorized by this license in accordance with the Commission's regulations set forth in 10 CFR Chapter I;

   F. ENERGY NORTHWEST ~~The Washington Public Power Supply System~~ has satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements", of the Commission's regulations;

G. The issuance of this license will not be inimical to the common defense and security or to the health and safety of the public;

H. After weighing the environmental, economic, technical, and other benefits of the facility against environmental and other costs and considering available alternatives, the issuance of this Facility Operating License No. NPF-21, subject to the conditions for protection of the environment set forth in the Environmental Protection Plan attached as Appendix B, is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied; and

I. The receipt, possession, and use of source, byproduct and special nuclear material as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Parts 30, 40 and 70.

2. Based on the foregoing findings regarding this facility, Facility Operating License NPF-21 is hereby issued to ~~the Washington Public Power Supply System~~ ENERGY NORTHWEST (the licensee) to read as follows:

A. This license applies to ~~the WPPSS Nuclear Project No. 2~~ (WNP-2), a boiling water nuclear reactor and associated equipment, owned by ~~the Washington Public Power Supply System~~. The facility is located on Hanford Reservation in Benton County near Richland, Washington, and is described in the licensee's "Final Safety Analysis Report", as supplemented and amended, and in the licensee's Environmental Report, as supplemented and amended.

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses ~~Washington Public Power Supply System~~: ENERGY NORTHWEST

(1) Pursuant to Section 103 of the Act and 10 CFR Part 50, to possess, use, and operate the facility at the designated location on Hanford Reservation, Benton County, Washington, in accordance with the procedures and limitations set forth in this license;

(2) Pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(26) Progress of Offsite Emergency Preparedness (Appendix D, SER)

In the event that the NRC finds that the lack of progress in completion of the procedures in the Federal Emergency Management Agency's final rule, 44 C.F.R. Part 350, is an indication that a major substantive problem exists in achieving or maintaining an adequate state of preparedness, the provisions of 10 C.F.R. Section 50.54(s)(2) will apply.

(27) Effluent Radiation Monitors (Section 11.5, SSER #4)

Prior to July 1, 1984, the licensee shall provide the following information to the NRC staff for their review and approval:

1. Sensitivity of the effluent monitors.

2. Evaluation of response times of these instruments.

3. Evaluation of the instruments per criteria set forth in Section 5.4.7 of ANSI 13.10.

4. Compliance with Section 5.4.9 of ANSI 13.10

5. Evaluation of capability to provide a calibrated electrical signal to verify circuit alignment and, if used, a commitment that they be qualified.

(28) Environmental Qualifications (Section 3.11, SER, SSER #3, SSER #4)

Prior to November 30, 1985, the licensee shall environmentally qualify all electrical equipment according to the provisions of 10 CFR 50.49.

(29) Protection of the Environment (FES)

Before engaging in additional construction or operational activities which may result in a significant adverse environmental impact that was not evaluation or that is significantly greater than that evaluation in the Final Environmental Statement the licensee shall provide a written notification to the Director of the Office of Nuclear Reactor Regulation and receive written approval from that office before proceeding with such activities.

(30) Additional Concerns

The Additional Concerns contained in Appendix C, as revised through Amendment No. 153, are hereby incorporated into this license. ~~Washington Public Power Supply System~~ shall operate the facility in accordance with the Additional Concerns.

ENERGY NORTHWEST

ATTACHMENT 1 TO ~~WPPSS NUCLEAR PROJECT NO. 2~~
OPERATING LICENSE NPF-21

The licensee shall complete the following requirements within the schedule noted below:

1. Preoperational/Acceptance Tests

   a. The licensee shall, prior to loading of fuel in the core, complete the System 36 preoperational testing to assure that those monitors required for fuel load fully meet the Technical Specification requirements without reliance on action statements:

   b. The licensee shall successfully complete the following preoperational/acceptance tests before exceeding 5% power:

      PT 33.0-B Chemical Waste Processing
      PT 37.0-D Miscellaneous Radiation Monitoring Equipment
      PT 40.0-A Off-Gas System
      AT 65.0-A Sealing Steam System
      AT 66.0-A Condenser Air Removal
      PT 69.0-A Condensate System
      PT 70.0-A Condensate Storage Transfer
      PT 71.0-A Condensate Filter Demineralizer System
      PT 72.0-A Reactor Feedwater Turbine and Pumps
      PT 72.0-B Reactor Feedwater Controls
      AT 74.0-A Heater Vents and Drains
      AT 82.0-A Turbine Building Heating and Ventilating
      PT 92.0-A Off-Gas Vault HVAC
      AT 110.0-A Loose Parts Detection
      PT 201.0-A Primary Containment Integrated Leakage Rate Test
      AT 302.0-A Integrated Condenser In-Leakage Test

   c. The licensee shall complete PT 22.0-B, Nitrogen Interting System prior to six months after initial criticality.

2. Hangers Supports, and Restraints

   All QI-SI and QII-SI hangers, supports, and restraints needing installation and/or modification will be completed prior to exceeding 5% power.

3. Construction Completion (Master Completion List Schedule)

   The licensee shall restrain fuel loading, primary system steam pressurization, exceeding 5% power, and commerical operation* by prerequisite completion of the associated categories of items in accordance with the schedule shown on the Project Master Completion List dated December 19, 1983. The licensee shall not extend the completion categories for individual items on the list without prior notification and individual concurrence by a representative of the NRC Regional Office.

*Commerical operation is defined as the 100% power warranty run or July 1, 1984, whichever occurs first.

APPENDIX B

TO FACILITY OPERATING LICENSE NO. DPR

ENERGY NORTHWEST

~~WASHINGTON PUBLIC POWER SUPPLY SYSTEM~~

~~NUCLEAR PROJECT NO. 2~~ (WNP-2)

DOCKET NO. 50-397

ENVIRONMENTAL PROTECTION PLAN

(NONRADIOLOGICAL)

ENERGY NORTHWEST

~~WASHINGTON PUBLIC POWER SUPPLY SYSTEM~~

~~NUCLEAR PROJECT NO. 2~~ (WNP-2)

ENVIRONMENTAL PROTECTION PLAN

(NON-RADIOLOGICAL)

TABLE OF CONTENTS

APPENDIX C

ADDITIONAL CONDITIONS

ENERGY NORTHWEST FACILITY OPERATING LICENSE NO. NPF-21

~~Washington Public Power Supply System~~ shall comply with the following conditions on the schedules noted below:

Amendment Number	Additional Condition	Implementation Date
149	The licensee shall relocate certain technical specification requirements to licensee-controlled documents as described below. The location of these requirements shall be retained by the licensee. a. This license condition approves the relocation of certain technical specification requirements to licensee-controlled documents (e.g., UFSAR, LCS, etc.), as described in Attachment 1 to the licensee's letter dated January 14, 1997. The approval is documented in the staff's safety evaluation dated March 4, 1997.	Implementation shall be completed by June 30, 1997.
149	Regulatory Guide 1.160 commitments as described in Attachment 1 to the licensee's letter dated January 14, 1997.	Implementation shall be completed 90 days from the date of issuance of Amendment 149.
151	To ensure sufficiently conservative SPC 9X9-9 OLMCPRs, the calculation of ΔCPR will include a conservative adder based on the variability observed in the US96A7 comparison with the ANFB correlation. This adder will be at a minimum, the greater of two times the standard deviation in the mean error of the predictions relative to the calculated matrix values, or a factor of 0.975 applied to the ΔCPR calculation, and will be independent of the 0.975 factor included in the US96A7 correlation as a conservative bias to the US96A7 predictions of CPR for the SPC fuel.	Implementation shall be completed prior to exceeding 25% power for Cycle 13.

50-397
2/28/2000

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ENERGY NORTHWEST
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February 28, 2000
GO2-00-037

Docket No. 50-397

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Gentlemen:

Subject: **WNP-2 OPERATING LICENSE NPF-21
REQUEST FOR AMENDMENT, POST-ACCIDENT NEUTRON FLUX MONITORING, LICENSE CONDITION 2.C.(16), ATTACHMENT 2, ITEM 3(b)
(ADDITIONAL INFORMATION)**

Reference: Letter, dated February 15, 2000, Jack Cushing (NRC) to JV Parrish (Energy Northwest), "Request for Additional Information (RAI) for the Energy Northwest Nuclear Project No. 2 (TAC No. MA6165)

In the referenced letter, the staff requested that additional information be provided to support review of our pending request that License Condition 2.C.(16), Attachment 2, Item 3(b), Wide Range Neutron Monitor, be removed from the WNP-2 Operating License.

The additional information is included as an attachment. Should you have any questions or require additional information regarding this matter, please call me or PJ Inserra at (509) 377-4147.

Respectfully,

D.W. Coleman

DW Coleman
Manager, Regulatory Affairs
Mail Drop PE20

Attachment

cc: EW Merschoff - NRC RIV
JS Cushing - NRC NRR
NRC Resident Inspector - 927N
DL Williams - BPA/1399
TC Poindexter - Winston & Strawn

ML003689109

A001

Question 1 *In Section 2.2, Accuracy: NEDO Section 5.2.2, of your July 29, 1999, submittal, you state that due to inaccuracies in the detectors, amplifiers and recorders, the APRMs would slightly exceed the accuracy requirement of +/- 2% of rated thermal power. Please provide additional clarification of the APRM accuracy and justification if the criterion cannot be met.*

Response

A reanalysis of the Average Power Range Monitor (APRM) instrument loop accuracy has determined that WNP-2 meets the criteria with existing equipment. Specifically, a calculation was performed that determined the APRM instrument loop accuracy is 1% of 100% of the rated power range under pre-accident conditions.

WNP-2 calibration procedures for Local Power Range Monitor (LPRM) and APRM gain adjustment and trip setpoints provide channel calibration in accordance with the WNP-2 Technical Specifications. Additionally, weekly surveillances verify the APRMs are accurate to +/- 2% rated thermal power based on the power values calculated by a heat balance during Mode 1 (Power Operation) while operating ≥ 25% rated thermal power.

NEDO-31558, Section 5.2.2, specifies an accuracy requirement of 2% of rated power. This requirement is more restrictive than Regulatory Guide 1.97, which is silent on instrumentation accuracy. The WNP-2 APRM system may not meet the NEDO accuracy requirement under all post-accident conditions. This judgement is based on the effects of anticipated off-normal core conditions following an Anticipated Transient Without Scram (ATWS) event (power < 25%, asymmetric control rod patterns, xenon, etc.). Therefore, the total APRM power measurement uncertainties may be in excess of 2% during an ATWS event but the exact degree of inaccuracy cannot be determined.

WNP-2 has evaluated the impact of not conforming to NEDO-31558, Section 5.2.2 post-accident and concludes the deviation is acceptable. The justification for this conclusion is provided below and is consistent with the BWROG position on the subject.

WNP-2 uses the Emergency Procedure Guidelines (EPGs) to achieve shutdown during an ATWS event. When the ATWS condition potentially threatens containment, shutdown is accomplished by injecting boron via the Standby Liquid Control system. The decision to inject boron is not dependent on APRM indications and is predicated on degrading containment conditions (such as rising suppression pool temperature). As a result, an APRM system uncertainty beyond that specified in NEDO-31558 is acceptable and does not compromise plant safety.

This position was accepted by the NRC for LaSalle County Station, Units 1 and 2 by letter dated September 17, 1999, from D.M. Skay to O.D. Kingsley, 'Regulatory Guide 1.97 – Boiling Water Reactor Neutron Flux Monitoring – LaSalle County Station, Units 1 and 2' (TAC NO. M77660). The letter dated June 21, 1999, from J.A. Benjamin, to U.S. NRC, concerning 'LaSalle County Station, Units 1 and 2 Compliance with Regulatory Guide 1.97 – Boiling Water Reactor Neutron Flux

Monitoring,' provided the final LaSalle responses for paragraph 5.2.2 of NEDO 31558-A. This position was also accepted by the NRC for Quad Cities Nuclear Power Station Units 1 and 2 by letter dated December 31, 1998, from R.M. Pulsifer to O.D. Kingsley (TAC NOs. M51124 and M51125) as noted in the referenced LaSalle letter of June 21, 1999.

Question 2 *In section 2.8, Power Sources: NEDO section 5.2.8, you stated that the APRMs will lose power on a loss of offsite power until power is restored by the division 1 and 2 diesel generators and the motor generator breakers are manually reset. The NEDO criterion is for an uninterruptable and reliable power source. Please provide additional justification for not meeting this criterion.*

Response

The WNP-2 Neutron Monitoring System (NMS) is fed from highly reliable power sources. The LPRM/APRM subsystem is powered from redundant 480/120 Volt AC motor-generator (MG) sets configured in two Reactor Protection System (RPS) divisional buses (A and B). The MGs are fed from redundant and separate divisional (ESF Divisions 1 and 2) 480 Volt AC buses in separate motor control centers. Either RPS Division Bus A or B can be energized by a reserve feed from a non-divisional source via main control room operator action. Two Electrical Protection Assemblies (EPAs) are installed in series between each of the two RPS MG sets and RPS buses and between the reserve feed and the RPS buses. The EPA assemblies are packaged in enclosures that are mounted on Seismic Category I structures. EPAs provide redundant protection to the RPS buses by acting to disconnect the RPS from the power circuits.

Each MG set is equipped with a high inertia flywheel which is sufficient to maintain the voltage and frequency of generated voltage within -5% of the rated values for at least 1 second following a loss of power to the drive motor.

The MG set power sources are reliable and *uninterrupted* as required to properly perform all the functions discussed in the WNP-2 FSAR. Neutron Monitoring System power will not be lost due to load shedding logic or a single failure that would cause the loss of redundant RPS buses powering the NMS instrumentation. In the unlikely event of the loss of one RPS Division, the power level indication will be provided on the redundant Division of NMS.

The power sources for the NMS meet the NEDO requirement for uninterruptibility, because they are reliable and capable of providing continuous power so that NMS safety functions discussed in the FSAR are met.

However, for a Loss of Offsite Power (LOOP) event, WNP-2 deviates from the NEDO requirements because both RPS power sources will be lost temporarily. For this event, restoration of power to the APRM subsystem is dependent upon emergency diesel generator (DG) startup time

plus manual restart of the RPS MG sets and reset of the EPAs. In accordance with station procedures for loss of all offsite electrical power, immediate operator actions are to ensure that all automatic actions have occurred which include verifying reactor SCRAM (all rods inserted) and the diesel generators auto start and reenergize their respective buses. The subsequent operator action following verification of automatic actions is to restart the RPS MG sets and ensure neutron monitoring systems return to service. In accordance with design, the DGs are running and supplying power to safety buses in approximately 15 seconds. Reset of the EPAs and manual restart of the RPS MG set are in the same location (Rad Waste Bldg 467'), however, this location is remote from the main control room (Rad Waste Bldg 501') and operator dispatch is required.

During this period of time, the control room operator can determine if control rods inserted properly using the Control Rod Position Indication System (RPIS) which remains available to provide backup to the NMS. Source Range Monitor (SRM) and Intermediate Range Monitor (IRM) systems, utilize detectors that are withdrawn from the core during normal power operation. The drive motors for the SRM/IRM detectors are powered from Engineered Safety Feature (ESF) divisional sources which will be energized upon DG startup. The SRM and IRM systems have redundant channel capability. The system sensors and associated equipment are powered by a 24 Volt DC battery/charger system. The battery chargers for this system receive their power source from ESF divisional sources.

In summary, the present design of the WNP-2 NMS meets the intent of Section 5.2.8 of NEDO 31558-A in that the system is reliable and uninterruptible for NMS required safety functions. It should be noted that with a concurrent LOOP the RPS MG set would be interrupted, but can be manually restored as described above. The operator still has information available as described above to determine reactor status during RPS MG set restoration. This non-conformance with the NEDO is consistent with the BWROG Reg Guide 1.97 Neutron Monitoring System subcommittee for the RG 1.97 NMS – Power Supplies position that the existing MG set power supplies meet the intent of the BWROG post-accident monitoring functional criteria as described in paragraph 5.2.8 of NEDO 31558-A and does not compromise plant safety.

This position was accepted by the NRC for LaSalle County Station, Units 1 and 2 by letter dated September 17, 1999, from D.M. Skay to O.D. Kingsley, 'Regulatory Guide 1.97 – Boiling Water Reactor Neutron Flux Monitoring – LaSalle County Station, Units 1 and 2' (TAC NO. M77660). The letter dated June 21, 1999, from J.A. Benjamin, to U.S. NRC, concerning 'LaSalle County Station, Units 1 and 2 Compliance with Regulatory Guide 1.97 – Boiling Water Reactor Neutron Flux Monitoring,' provided the final LaSalle responses for paragraph 5.2.2 of NEDO 31558-A. This position was also accepted by the NRC for Quad Cities Nuclear Power Station Units 1 and 2 by letter dated December 31, 1998 from R.M. Pulsifer to O.D. Kingsley (TAC NOs. M51124 and M51125) as noted in the referenced LaSalle letter of June 21, 1999.

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February 7, 2000
GO2-00-022

Docket No. 50-397

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Gentlemen:

Subject: **WNP-2, OPERATING LICENSE NPF-21**
REQUEST FOR AMENDMENT
TECHNICAL SPECIFICATION 4.3.1.2.b
(ADDITIONAL INFORMATION)

Reference: Letter, dated January 3, 2000, Jack Cushing (NRC) to JV Parrish (Energy Northwest) "Request for Additional Information (RAI) for WNP-2, (TAC NO. MA7228)"

In the reference, the staff requested that additional information be provided to support review of our pending request for an amendment to revise Subsection 4.3.1.2.b of Technical Specification 4.3.1.

The additional information is included as attachments, which consists of a response to the RAI questions and a report from Asea Brown-Boveri (ABB) Combustion Engineering, Inc. Some of the material in Attachment B has been identified as proprietary and is marked accordingly (i.e., bracketed). Therefore, pursuant to the requirements of 10 CFR 2.790, an affidavit is enclosed to support the withholding of this information from public disclosure.

Should you have any questions or desire additional information regarding the matter, please call me or PJ Inserra at (509) 377-4147.

Respectfully,

D. W. Coleman

DW Coleman (Mail Drop PE20)
Manager, Regulatory Affairs

Attachments

cc: EW Merschoff- NRC RIV
JS Cushing- NRC NRR
NRC Sr. Resident Inspector-927N
DL Williams - BPA/1399
TC Poindexter - Winston & Strawn

0036850092 change: PDR ltr encl 1 w/o prop APO1

AFFIDAVIT

STATE OF WASHINGTON	)	Subject:	Report CE NSPSD-787-P, WNP-2
	)		SVEA-96 Fuel Assemblies Dry Fuel
	)		Storage Criticality Safety Evaluation,
COUNTY OF BENTON	)		Dated February, 1995

I, D.W. Coleman, being duly sworn, subscribe to and say that I am the Manager, Regulatory Affairs, for ENERGY NORTHWEST, the applicant herein; that I have the full authority to execute this oath; that I have reviewed the foregoing; and that to the best of my knowledge, information, and belief the statements made in it are true.

The attachment to this letter contains information [marked in brackets] that is considered by ABB Combustion Engineering, to be proprietary. Attached is an affidavit executed by I.C. Rickard, Director, Nuclear Licensing, of ABB Combustion Engineering Nuclear Power, Inc., dated January 25, 2000, which provides the basis on which it is claimed that the subject document should be withheld from public disclosure under the provisions of 10 CFR 2.790.

Energy Northwest treats the subject document as proprietary information on the basis of statements by the owner. In submitting this information to the NRC, Energy Northwest requests that the subject document be withheld from public disclosure in accordance with 10 CFR 2.790.

DATE February 7, 2000

D.W. Coleman

D.W. Coleman
Manager, Regulatory Affairs

On this date personally appeared before me D.W. Coleman, to me known to be the individual who executed the foregoing instrument, and acknowledged that he signed the same as his free act and deed for the uses and purposes herein mentioned.

GIVEN under my hand and seal this 7th day of February 2000.

[signature]
Notary Public in and for the
STATE OF WASHINGTON

Residing at W. Richland

My Commission Expires 3-29-01

Request for Additional Information Question # 1

Discuss briefly the types of analyses (including any seismic dynamic analysis) performed to determine the structural integrity of the various elements affected by the new geometrical limitations for storage of new fuel assemblies in the new fuel racks. This discussion should include the analyses related to the accidental drop of the fuel assemblies being supported by the pedestal.

Response to Request for Additional Information Question #1

A structural analysis has been performed to verify the structural integrity of the new fuel racks resulting from the new geometrical limitations of the new fuel rack support system. See attached diagrams of new fuel storage rack arrangement. The configuration control components used in the new fuel vault consist of a series of templates and working platform grating sections and a fuel support assembly (pedestal).

The templates and working platform grating sections allow only ¼ of the total design capacity of fuel assemblies to be installed (60 vs. 240 assemblies). The templates permit every other location in two fuel rack rows to be available for fuel assembly storage in a checkerboard pattern and then skips two rows in between where the working platform grating resides. The working platform grating prevents any fuel from being inserted. This pattern is repeated, as necessary, for the volume of fuel assemblies to be stored up to the limit of 60 fuel assemblies. Each template is fabricated from ¼ inch aluminum plate and is a non-structural element that adds no weight to the fuel rack beams or their supports. The template is securely mounted on and fastened to the working platform grating. The working platform is supported from the new fuel vault cover lip, independent from the fuel rack beams or their vault wall support box beams. Thus, neither the templates nor the working platforms provide any loading to the fuel rack or its support system.

The other component is a "fuel support assembly" or "pedestal" which is placed on the lower fuel rack and acts as a spacer to raise the fuel assemblies approximately 42 inches for ease of inspection, exchange of the shipping handle with the in-vessel bale handle, etc. The pedestal is constructed of 3½ inch stainless steel schedule 40 pipe. The strength characteristics of the pedestal are sufficient to support the fuel assembly in the receptor cell in the lower fuel rack beam. This pedestal securely fits into the lower fuel rack beam in a structurally similar manner as a fuel assembly and accepts the new fuel assembly into its tube section in a structurally similar manner as did the lower fuel rack. The pedestal employs a 3 inch stainless steel schedule 40 pipe section to achieve a slip fit design and a square plate to assure proper centering and fit into the fuel rack receptor cell. The pedestal is less than 5% of the weight of a fuel assembly.

The new fuel vault fuel rack support system is composed of three levels of fuel rack beams supported by box beams attached to the walls of the vault. The upper two fuel rack beams hold the fuel laterally and the lower fuel rack beam holds the fuel vertically and laterally when

Response to Request for Additional Information Question #1 (continued)

an assembly is fully inserted. To address the potential for uplift of a fuel assembly mounted in the pedestal, a review of the floor response spectra and a calculation of the lower fuel rack beam was performed. The vertical load on any lower fuel rack beam is reduced to less than 55% of the original load (i.e., one half of the weight of the original number of fuel assemblies, plus the weight of the pedestals). Based on a conservative natural frequency and response spectrum analysis, the forces from a safe shutdown earthquake for the new fuel assembly support system are below 1.0 g in the vertical direction. Thus, no vertical uplift of the fuel assembly will occur and the implementation of this tool (pedestal) will not have an adverse structural impact (vertically).

To address the potential for lateral load increases due to the elevated fuel assemblies (through the use of pedestals), an analysis was performed comparing the original and elevated fuel assembly configurations. The proposed fuel storage limitations and the template assure only half (every other one) of the designed number of new fuel assemblies are placed in a row. The original design used three levels of fuel rack beams to carry the lateral load of fuel assemblies resulting in ½ of the fuel assembly lateral load being carried by the center fuel rack beam and ¼ of the lateral load being carried by the other two fuel rack beams. When the 42 inch pedestal is used, only the center and upper fuel rack beams are assumed to carry the lateral load. However, since only half of the fuel assemblies are allowed (from that originally designed), the resulting lateral loads on each of the beams will be equal to (on the upper fuel rack) or less than (on the center fuel rack) the original load. This maintains adequate design margins for fuel rack loads.

Request for Additional Information Question # 2

Provide a summary of the results of the above analyses and confirm that the (strength) "capacities" of the various structural elements (e.g., the pedestal, vault floor, rack walls, cover plates, etc.) are adequate to satisfy the demand imposed on them by the new configuration of the fuel assemblies as per the applicable industry codes.

Response to Request for Additional Information Question #2

The proposed limitations result in a new configuration of fuel assemblies that consists of only ½ the number of original design fuel assemblies in an single fuel rack row and ¼ of the total number of original design fuel assemblies in the new fuel vault. This is a significant load reduction and a key to assuring that the strength capacities of various structural components are adequate. The templates and working platform gratings are supported independently from the fuel assembly racks and do not affect the strength of the new fuel rack structural elements. Therefore, the use of templates for loading configuration control and the use of pedestals do not result in load increases to the vault floor, rack walls, or other rack components. Furthermore, the strength capacities of FSAR Table 3.9-2s are maintained.

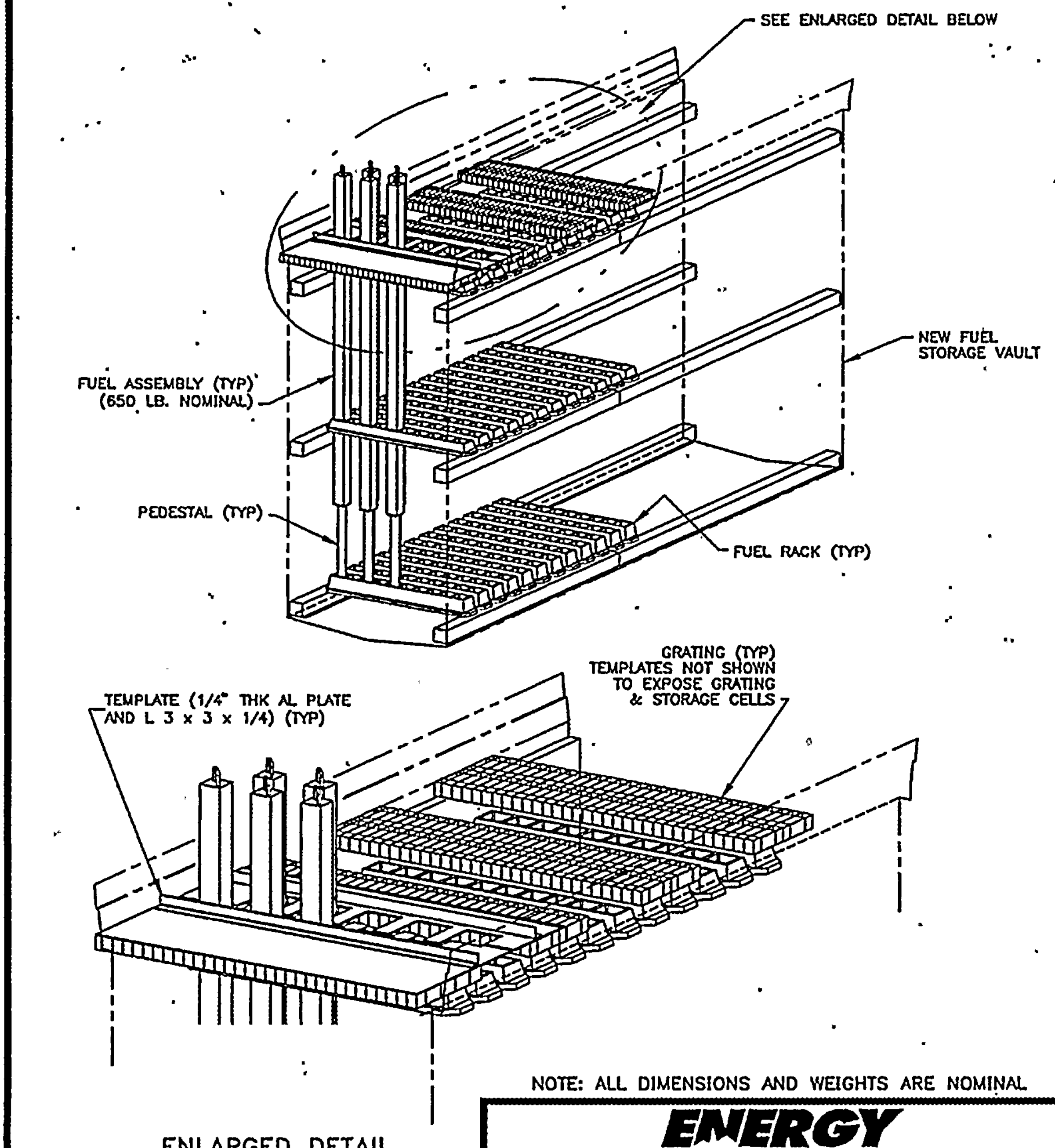
Calculations demonstrate that there is no mechanism resulting from the configuration control components that would adversely affect the configuration or integrity of the new fuel assembly and that they would not cause an accidental fuel drop. In addition, these configuration control components do not affect the previous testing results of accidental fuel drops on the fuel racks or vault floor described in FSAR Section 9.1.1.3.2.

Request for Additional Information Question # 3

Provide report CE NPSD-787-P, "WNP-2 SVEA-96, Fuel Assemblies Dry Fuel Storage Critical Safety Evaluation."

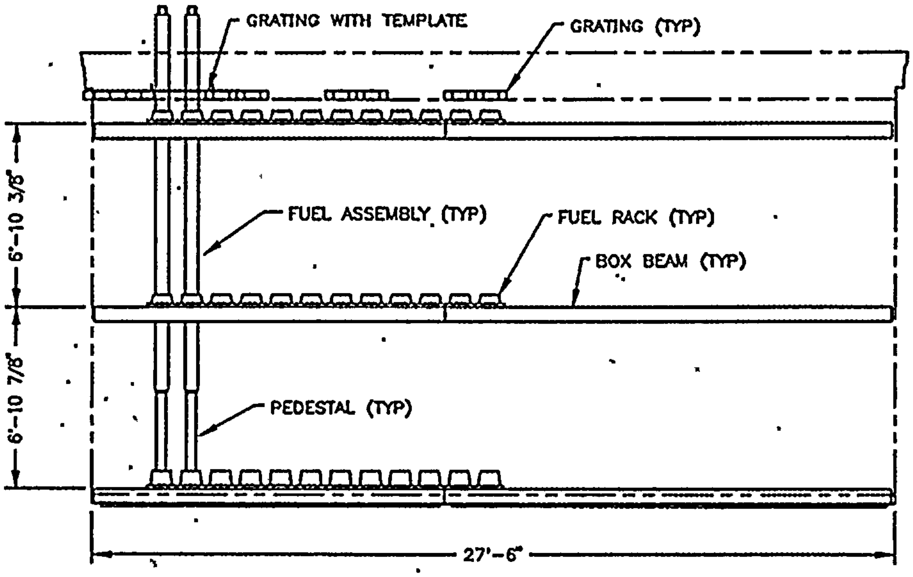
Response to Request for Additional Information Question #3

See attached proprietary report CE NPSD-787-P, "WNP-2 SVEA-96, Fuel Assemblies Dry Fuel Storage Critical Safety Evaluation."

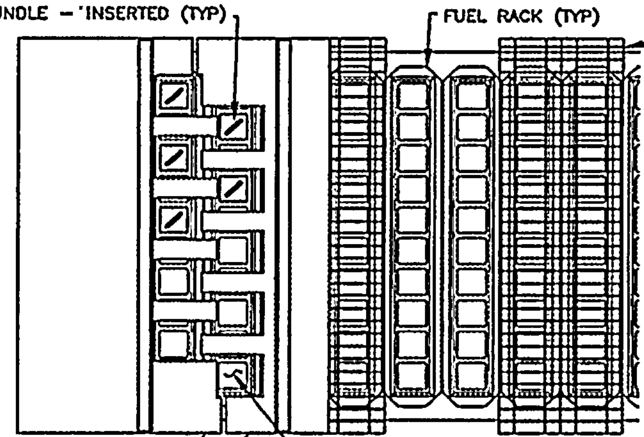


NOTE: ALL DIMENSIONS AND WEIGHTS ARE NOMINAL

ENERGY NORTHWEST
WNP-2, OPERATING LICENSE NPF-21 REQUEST FOR AMENDMENT TECHNICAL SPECIFICATION 4.3.1.2.b (ADDITIONAL INFORMATION)
NEW FUEL VAULT – DIAGRAM 1



SIDE ELEVATION VIEW



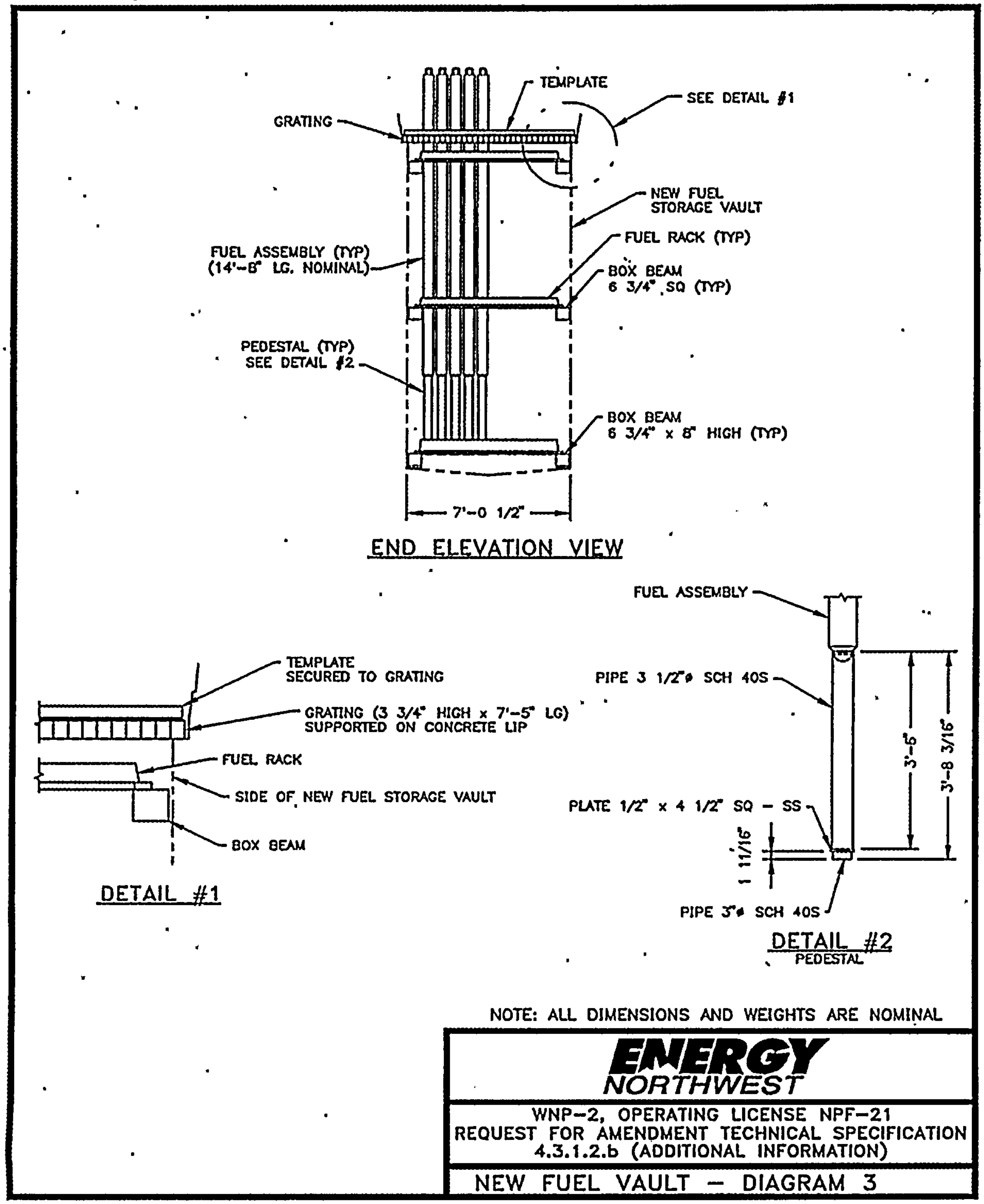
ENLARGED PARTIAL PLAN VIEW

NOTE: ALL DIMENSIONS AND WEIGHTS ARE NOMINAL

ENERGY NORTHWEST

WNP-2, OPERATING LICENSE NPF-21
REQUEST FOR AMENDMENT TECHNICAL SPECIFICATION
4.3.1.2.b (ADDITIONAL INFORMATION)

NEW FUEL VAULT - DIAGRAM 2


TEMPLATE
SEE DETAIL #1
GRATING
NEW FUEL STORAGE VAULT
FUEL RACK (TYP)
FUEL ASSEMBLY (TYP) (14'-8" LG. NOMINAL)
BOX BEAM 6 3/4" SQ (TYP)
PEDESTAL (TYP) SEE DETAIL #2
BOX BEAM 6 3/4" x 8" HIGH (TYP)
7'-0 1/2"
END ELEVATION VIEW
TEMPLATE SECURED TO GRATING
GRATING (3 3/4" HIGH x 7'-5" LG) SUPPORTED ON CONCRETE LIP
FUEL RACK
SIDE OF NEW FUEL STORAGE VAULT
BOX BEAM
DETAIL #1
FUEL ASSEMBLY
PIPE 3 1/2"ø SCH 40S
3'-6"
3'-8 3/16"
PLATE 1/2" x 4 1/2" SQ - SS
1 11/16"
PIPE 3"ø SCH 40S
DETAIL #2
PEDESTAL
NOTE: ALL DIMENSIONS AND WEIGHTS ARE NOMINAL
ENERGY NORTHWEST
WNP-2, OPERATING LICENSE NPF-21
REQUEST FOR AMENDMENT TECHNICAL SPECIFICATION 4.3.1.2.b (ADDITIONAL INFORMATION)
NEW FUEL VAULT - DIAGRAM 3

Proprietary Report CE NPSD-787-P,
"WNP-2 SVEA-96, Fuel Assemblies Dry Fuel Storage Critical Safety Evaluation"

I, Ian Rickard, depose and say that I am the Director, Nuclear Licensing, of ABB C-E Nuclear Power, Inc. (ABB), duly authorized to make this affidavit, and have reviewed or caused to have reviewed the information which is identified as proprietary and described below. I am submitting this affidavit in conformance with the provisions of 10 CFR 2.790 of the Commission's regulations for withholding this information.

I have personal knowledge of the criteria and procedures utilized by ABB in designating information as a trade secret, privileged or as confidential commercial or financial information. The information for which proprietary treatment is sought, and which document has been appropriately designated as proprietary, is contained in the following:

- CE NPSD-787-P, "WNP-2 SVEA-96 Fuel Assemblies Dry Fuel Storage Criticality Safety Evaluation," dated February, 1995.

Pursuant to the provisions of paragraph (b)(4) of Section 2.790 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure, included in the above referenced document, should be withheld.

1. The information sought to be withheld from public disclosure is owned and has been held in confidence by ABB. It consists of methodology and calculational results for nuclear criticality of SVEA-96 fuel contained in dry storage vaults.
2. The information consists of analytical data or other similar data concerning a process, method or component, the application of which results in substantial competitive advantage to ABB.
3. The information is of a type customarily held in confidence by ABB and not customarily disclosed to the public.
4. The information is being transmitted to the Commission in confidence under the provisions of 10 CFR 2.790 with the understanding that it is to be received in confidence by the Commission.
5. The information, to the best of my knowledge and belief, is not available in public sources, and any disclosure to third parties has been made pursuant to regulatory provisions or proprietary agreements that provide for maintenance of the information in confidence.
6. Public disclosure of the information is likely to cause substantial harm to the competitive position of ABB because:
   a. A similar product is manufactured and sold by major competitors of ABB.
   b. Development of this information by ABB required thousands of dollars and hundreds of manhours of effort. A competitor would have to undergo similar expense in generating equivalent information.
   c. The information consists of technical data and qualification information for ABB-supplied products, the possession of which provides a competitive economic advantage. The availability of such information to competitors would enable them to design their product to better compete with ABB, take marketing or other actions to improve their product's position or impair the position of ABB's product, and avoid developing similar technical analysis in support of their processes, methods or apparatus.
   d. In pricing ABB's products and services, significant research, development, engineering, analytical, manufacturing, licensing, quality assurance and other costs and expenses must be included. The ability of ABB's competitors to utilize such information without similar expenditure of resources may enable them to sell at prices reflecting significantly lower costs.

Ian C. Rickard
Director, Nuclear Licensing

Sworn to before me this
25th day of January, 2000

Catherine P. McCarthy
Notary Public
My commission expires: 1/31/03

1/31/2000

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Subject:
WNP-2, OPERATING LICENSE NPF-21 REQUEST FOR AMENDMENT TO TECHNICAL SPECIFICATION LCO 3.4.9, RESIDUAL HEAT REMOVAL SHUTDOWN COOLING SYSTEM - HOT SHUTDOWN (ADDITIONAL INFORMATION)

Body:

Docket: 05000397, Notes: N/A

AA2

ENERGY
NORTHWEST
P.O. Box 968 ▫ Richland, Washington 99352-0968

January 31, 2000
GO2-00-019

Docket No. 50-397

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Gentlemen:

Subject: **WNP-2, OPERATING LICENSE NPF-21**
REQUEST FOR AMENDMENT TO TECHNICAL SPECIFICATION
LCO 3.4.9, RESIDUAL HEAT REMOVAL SHUTDOWN
COOLING SYSTEM - HOT SHUTDOWN
(ADDITIONAL INFORMATION)

Reference: Letter, dated January 3, 2000, Jack Cushing (NRC) to JV Parrish (Energy Northwest), "Request for Additional Information (RAI) for WNP-2, (TAC NO. MA6166)"

In the reference, the staff requested that additional information be provided to support review of our pending request for an amendment to revise the Applicability of LCO 3.4.9 in the Technical Specifications.

The additional information is included as an attachment. Should you have any questions or desire additional information regarding the matter, please call me or PJ Inserra at (509) 377-4147.

Respectfully,

D.W. Coleman

DW Coleman
Manager, Regulatory Affairs
Mail Drop PE20

Attachment

cc: EW Merschoff - NRC RIV
JS Cushing - NRC NRR
NRC Sr. Resident Inspector - 927N
DL Williams - BPA/1399
TC Poindexter - Winston & Strawn

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[illegible]
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Question

In its July 29, 1999 submittal, the licensee stated that the basis for the requested technical specification (TS) change is that the original plant design operating temperature for the residual heat removal (RHR) shutdown cooling (SDC) piping and supports is less than the operational limit currently required by TS Limiting Condition for Operation (LCO) 3.4.9. During a conference call with the staff on November 17, 1999, the licensee stated that in 1988, an evaluation was performed to assess the condition of the RHR SDC piping system because of the potential of exposing the piping system to beyond original design operating temperature. The licensee is requested to provide details of the 1988 assessment (with respect to thermal stress limit and thermal fatigue cycle limit) and its findings.

Background

The 1988 system operating temperature discrepancy and resolution was documented in Non-Conformance Report (NCR) 288-028 (February of 1988). The NCR noted that the RHR piping downstream of the heat exchanger was designed for a normal operating temperature of 295°F, while by procedure it was possible to expose a portion of the piping to a maximum temperature of 320°F (saturation temperature for 75 psig) during shutdown. This was because the flow path for initiating RHR was through the heat exchanger bypass valve. A review of the past RHR shutdown cooling operation was completed to supplement the resolution of the 1988 NCR. Additionally, our current review noted that from February of 1984 through March of 1986, the system initiation was allowed at temperatures up to 355°F (saturation temperature for 125 psig). Thus, for our evaluation of the condition of the affected piping system a maximum temperature of 355°F at 125 psig was assumed for the initiation temperature for the RHR Shutdown Cooling (SDC).

Thermal Loads on Piping & Supports

The RHR SDC supply and return piping consists of a combination of ASME Code Class 1 and Code Class 2 piping.

ASME Class 1 piping primary (e.g. earthquake) plus secondary (e.g. thermal expansion) stress intensity range (Equation 10) has an allowable stress of 3Sm, which is based on the stress intensity defined as twice the maximum shear stress. If the Equation 10 allowable is exceeded then the alternative Equation 12 and 13 must be satisfied. Only Equation 12 includes stresses due to thermal expansion and thermal anchor movements. Additionally, ASME Class 1 piping and components are evaluated for cumulative damage caused by various stress cycles applied to systems. The cumulative usage factor shall not exceed 1.0.

[illegible]

Thermal Loads on Piping & Supports (continued)

The effects of thermal expansion on the ASME Class 2 piping system must meet the requirements of either Equation 10 (Sa) or Equation 11 (Sh+Sa). For ASME Class 2 piping, the allowable stress range for expansion stresses (Sa) is based on 7000 full range thermal cycles.

Based on the plant operating cycle history, the plant had been started up 34 times by the end of 1988. During the first year of operation, 1984, the plant experienced 13 startups. Although every shutdown did not include going into the shutdown cooling mode, for this evaluation it is assumed that 34 temperature cycles were experienced. The preferred loop, RHR-B, was normally used to initiate shutdown cooling, but it is possible that each loop would have had a portion of the maximum projected cycles. However, for this evaluation it was assumed that both loops had experienced 34 cycles of higher temperature.

The current ASME Class 1 and 2 stress analyses for the RHR return and supply piping meet the ASME Code allowable stress limits for the applicable operating conditions. These piping analyses were evaluated for the effect of the potential higher operating temperature. The new evaluation showed that the adjusted stresses remain within the ASME Code Class 1 and 2 allowable limits.

During the 1988 assessment, it was concluded that the limiting factor for thermal expansion beyond the analyzed system temperature was the pipe support system (e.g. hangers, anchors, etc.) of the return lines. Given the possibility of initiating the RHR SDC at higher than analyzed temperature, NCR 288-028 identified ten critical pipe supports that may have been loaded in excess of original thermal design load. Those critical supports were inspected and no damage was found. The highest loading would have occurred during 1984 to 1986, when temperatures possibly reached 355°F. From 1986 to 1988 the procedures limited system temperatures to a maximum of 320°F. Thus, the 1988 inspection was sufficient to demonstrate that no damage had occurred in the support system.

Thermal Fatigue Cycle

The ASME Class 1 piping fatigue limit is a cumulative usage factor less than or equal to 1.0. An evaluation was completed that accounted for the increased temperature for initiation of RHR SDC. The results demonstrated that the piping fatigue usage was still less than 1.0 for both RHR piping loops assuming that each loop had been used for all shutdowns. The occurrence of higher temperature RHR SDC injections was noted in the applicable system design calculations and will be accounted for in any future updates of the ASME Class 1 fatigue analyses or evaluations for plant life extension.

[illegible]

[illegible]

[illegible]

[illegible]

Thermal Fatigue Cycle (continued)

The ASME Class 2 piping thermal fatigue cycle limit of 7000 full range cycles is satisfied because the piping thermal expansion stresses, due to the increased temperature, meets the requirement of either Equation 10 or Equation 11 of ASME Code Sub-Section NC-3600.

Conclusion

Prior to 1988, plant procedures allowed for initiation of RHR SDC at temperatures in excess of the specified operating temperature in the RHR system design specification. An evaluation of the thermal fatigue cycles imposed on affected piping determined that ASME limits were not exceeded. Since the time of NCR 288-028, plant procedures were changed to limit RHR SDC operation to a reactor steam dome pressure of less than 48 psig (295° F). This limitation agrees with all current piping system analyses.

50-397
11/18/99

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Subject:
WNP-2, OPERATING LICENSE NPF-21 REQUEST FOR AMENDMENT TECHNICAL SPECIFICATION 5.5.7.c VENTILATION FILTER TESTING PROGRAM

Body:
PDR ADOCK 05000397 P

Docket: 05000397, Notes: N/A

AA2

ENERGY NORTHWEST

P.O. Box 968 ◻ Richland, Washington 99352-0968

November 18, 1999
GO2-99-203

Docket No. 50-397

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Gentlemen:

Subject: **WNP-2, OPERATING LICENSE NPF-21
REQUEST FOR AMENDMENT
TECHNICAL SPECIFICATION 5.5.7.c
VENTILATION FILTER TESTING PROGRAM**

Reference: NRC Generic Letter 99-02, dated June 3, 1999, "Laboratory Testing of Nuclear-Grade Activated Charcoal"

In accordance with the Code of Federal Regulations, Title 10, Parts 2.101, 50.59 and 50.90, and as requested by the referenced generic letter, Energy Northwest hereby submits a request for amendment to the WNP-2 Operating License. Specifically, we are requesting a revision to Technical Specification (TS) 5.5.7.c.

The changes would revise the requirements that: 1) a sample of the charcoal adsorber for the Standby Gas Treatment (SGT) System and the Control Room Emergency Filtration (CREF) System be tested in accordance with American Society for Testing and Materials (ASTM) D3803-1986, "Standard Test Method for Nuclear-Grade Activated Carbon"; 2) methyl iodide penetration be less than a value of .175% for the SGT System and 1.0% for the CREF System; and 3) charcoal adsorber testing be conducted at a relative humidity of greater than or equal to 70%. As requested by Generic Letter (GL) 99-02, Energy Northwest proposes that TS 5.5.7.c be revised so that: 1) testing of charcoal adsorber samples be in accordance with ASTM D3803-1989 at a specified temperature of 30° Centigrade (C) (86° Fahrenheit (F)); 2) methyl iodide penetration to be less than a value of 0.5% for the SGT System and 2.5% for the CREF System; and 3) testing be performed at 70% relative humidity.

Generic Letter 99-02 also requires that TS 5.5.7.c specify the face velocity of any system that has a face velocity greater than 44 feet per minute (fpm), so that charcoal testing will be conducted at that velocity. For this TS change, a face velocity of 75 fpm will be specified for the SGT System. The face velocity for the CREF System is below 44 fpm and need not be specified. In addition, the revision to TS 5.5.7.c will note that variations in testing parameters are permitted per the guidance in Table 1 and Section A5.2 of ASTM D3803-1989.

9913010150 A001

The engineered safety feature (ESF) filter ventilation systems are described in FSAR Section 6.5.1. The SGT System is designed to limit the release of airborne radioactive contaminants from secondary containment to the atmosphere per the guidelines of 10CFR100 in the event of a design basis accident (DBA). The safety-related SGT System is a standby system which consists of two fully redundant subsystems, each with its own set of ductwork, dampers, high efficiency particulate air (HEPA)/charcoal filters, and controls. Each charcoal filter train consists of a moisture separator, two electric heater banks, a prefilter, a HEPA filter bank, two four inch charcoal adsorber banks, a second HEPA filter bank, and two centrifugal fans. The CREF System provides a radiologically controlled environment from which the plant can be safely operated following a DBA. The safety-related CREF System is a standby system which is operated to maintain the control room environment during normal operation. Upon receipt of initiation signal(s) (indicative of conditions that could result in radiation exposure to control room personnel), the CREF System automatically switches to the pressurization mode of operation to prevent infiltration of contaminated air into the control room. A system of dampers isolates the control room (from the normal intake and exhaust), and control room outside air flow is redirected and processed through either of two filter subsystems. Each subsystem consists of an electric heater, a prefilter, a HEPA filter, an activated charcoal adsorber section, a filter unit fan, a control room recirculation fan, and the associated ductwork and dampers.

In GL 99-02 the NRC noted that testing nuclear-grade activated charcoal to standards other than ASTM D3803-1989, such as ASTM D3803-1986, does not provide assurance for complying with our current licensing basis as it relates to limiting dose to the public and control room staff during a DBA. The staff considers ASTM D3803-1989 to be the most accurate and realistic protocol for testing charcoal in ESF ventilation systems because it offers the greatest assurance of accurately and consistently determining the capability of the charcoal. Generic Letter 99-02 also noted that testing charcoal at an elevated temperature greater than 30° C results in an overestimation of the actual iodine-removal capability of the charcoal, while a 30° C test temperature is more representative of limiting accident conditions.

The proposed changes to TS 5.5.7.c are consistent with the sample technical specification provided in GL 99-02. Energy Northwest will replace the reference to ASTM D3803-1986, including associated testing methods A and B, with a requirement to test in accordance with ASTM D3803-1989. Testing will occur at a temperature of 30° C (86° F). Testing will also continue at a specified relative humidity of 70% because the SGT and CREF systems have humidity control. In addition, and as permitted by the generic letter, the limits for methyl iodide penetration will be changed to less than 0.5% for the SGT System and less than 2.5% for the CREF System. Because ASTM D3803-1989 is a more accurate and demanding test method than older test methods, Energy Northwest can use a safety factor of 2 rather than 5 for determining the acceptance criteria for charcoal filter efficiency. Also, because the SGT System has a face velocity greater than 44 fpm, its face velocity of 75 fpm will be included in the revision to TS 5.5.7.c.

As requested by GL 99-02, a recent laboratory charcoal test of the CREF System performed on August 5, 1999 used the guidance provided by ASTM D3803-1989. The results met the acceptance criterion derived from applying a safety factor of 2 to the charcoal filter efficiency assumed in our design basis analysis. The next laboratory charcoal test will be performed on the SGT System, and should be completed by December 1999. Energy Northwest will continue to test our ESF ventilation systems using the 1989 standard.

As previously discussed with the staff, this request for amendment to the WNP-2 Operating License suffices for the written response originally required by GL 99-02 within 180 days of the date of the generic letter.

Additional information has been attached to this letter to complete the amendment request. Attachment 1 describes an evaluation of the proposed changes in accordance with 10CFR50.92 and concludes they do not result in a significant hazards consideration. Attachment 2 provides the Environmental Assessment Applicability Review and notes that the proposed change meets the eligibility criteria for a categorical exclusion as set forth in 10CFR51.22(c)(9). Therefore, in accordance with 10CFR51.22(b), an environmental assessment of the change is not required. Attachment 3 provides marked up pages of the Technical Specifications. Attachment 4 consists of the typed Technical Specification pages as proposed by this amendment.

This request for amendment has been approved by the WNP-2 Plant Operations Committee and reviewed by the Energy Northwest Corporate Nuclear Safety Review Board. In accordance with 10CFR50.91, the State of Washington has been provided a copy of this letter.

Should you have any questions or desire additional information regarding this matter, please contact me or PJ Inserra at (509) 377-4147.

Respectfully,

DK Atkinson for RL Webring

RL Webring, Mail Drop PE08
Vice President, Operations Support/PIO

Attachments

cc: EW Merschoff - NRC RIV
JS Cushing - NRC NRR
NRC Senior Resident Inspector - 927N
DJ Ross - EFSEC
TC Poindexter - Winston & Strawn
DL Williams - BPA/1399

STATE OF WASHINGTON)
)
COUNTY OF BENTON)

Subject: Request for Amendment
Technical Specification 5.5.7.c
Ventilation Filter Testing Program

I, DK Atkinson, being duly sworn, subscribe to and say that I am the Acting Vice President, Operations Support/PIO, for ENERGY NORTHWEST, the applicant herein; that I have the full authority to execute this oath; that I have reviewed the foregoing; and that to the best of my knowledge, information, and belief that the statements made in it are true.

DATE November 18, 1999

D. K. Atkinson

DK Atkinson
Acting, Vice President, Operations Support/PIO

On this date personally appeared before me DK Atkinson, to me known to be the individual who executed the foregoing instrument, and acknowledged that he signed the same as his free act and deed for the uses and purposes herein mentioned.

GIVEN under my hand and seal this 18 day of November 1999

Lori A. Mix

Notary Public in and for the
STATE OF WASHINGTON

Residing at W. Richland

My Commission expires 3-29-01

Evaluation of Significant Hazards Considerations

Summary of Proposed Change

As requested by Generic Letter (GL) 99-02, Energy Northwest is requesting a revision to Technical Specification (TS) 5.5.7.c. This TS presently requires that a sample of the charcoal adsorber for the Standby Gas Treatment (SGT) System and the Control Room Emergency Filtration (CREF) System be tested in accordance with American Society for Testing and Materials (ASTM) D3803-1986, "Standard Test Method for Nuclear-Grade Activated Carbon." Technical Specification 5.5.7.c also specifies that methyl iodide penetration be less than a value of 0.175% for the SGT System and 1.0% for the CREF System, and that charcoal adsorber testing be conducted at a relative humidity of greater than or equal to 70%.

The staff has noted in GL 99-02 that testing nuclear-grade activated charcoal to standards other than ASTM D3803-1989, such as ASTM D3803-1986, does not provide assurance for complying with our current licensing basis as it relates to limiting dose to the public and the control room during a design basis accident (DBA). The staff considers ASTM D3803-1989 to be the most accurate and realistic protocol for testing charcoal in engineered safety feature (ESF) ventilation systems because it offers the greatest assurance of accurately and consistently determining the capability of the charcoal. Energy Northwest proposes a revision to TS 5.5.7.c that is consistent with the sample technical specification provided in GL 99-02. The change will replace the reference to ASTM D3803-1986, including associated testing methods A and B, with a requirement to test in accordance with ASTM D3803-1989. The change will also specify that: 1) testing will occur at a temperature of 30° Centigrade (86° Fahrenheit); 2) testing will occur at a relative humidity of 70% due to the SGT and CREF systems having humidity control; 3) the limits for methyl iodide penetration will be changed to less than 0.5% for the SGT System and less than 2.5% for the CREF System; 4) testing for the SGT System occurs at its design face velocity of 75 feet per minute; and 5) variations in the testing parameters (noted above) are permitted per the guidance in Table 1 and Section A5.2 of ASTM D3803-1989.

No Significant Hazards Consideration Determination

Energy Northwest has evaluated the proposed change to the Technical Specifications using the criteria established in 10CFR50.92(c) and has determined that it does not represent a significant hazards consideration as described below:

- **The operation of WNP-2 in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.**

  The SGT System is designed to limit the release of airborne radioactive contaminants from secondary containment to the atmosphere within the guidelines of 10CFR100 in the event of a DBA. The CREF System provides a radiologically controlled environment from

which the plant can be safely operated following a DBA. The proposed amendment will require that charcoal from these two ESF systems be tested to the more conservative standards of ASTM D3803-1989. Using the more conservative ASTM D3803-1989 testing standard will provide no increase in the probability of an accident previously evaluated.

The staff considers ASTM D3803-1989 to be the most accurate and most realistic protocol for testing charcoal in ESF ventilation systems because it offers the greatest assurance of accurately and consistently determining the capability of the charcoal. Using the more conservative ASTM D3803-1989 testing standard will provide greater assurance that the ESF ventilation systems will properly perform their safety function, thus assuring no increase in the radiological consequences of a DBA.

Therefore, operation of WNP-2 in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

- **The operation of WNP-2 in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.**

The proposed change will not create a new or different kind of accident since it only requires that charcoal from the SGT and CREF safety-related filtration systems be tested to the more conservative standards of ASTM D3803-1989. Using the more conservative ASTM D3803-1989 testing standard will provide even greater assurance that the ESF ventilation systems will properly perform their safety function, thus helping to minimize the radiological consequences of a DBA. The increased margin provided by the more conservative testing standard will assure no new or different kinds of accidents result from the proposed change.

Therefore, the operation of WNP-2 in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

- **The operation of WNP-2 in accordance with the proposed amendment will not involve a significant reduction in the margin of safety.**

The proposed amendment requires that more conservative ESF charcoal filter testing criteria be used to verify ESF ventilation systems are operable. More conservative testing criteria will provide greater assurance that the ESF ventilation systems will properly perform their safety function, thus helping to minimize the radiological consequences of a DBA. Using more conservative testing criteria will result in maintaining the current margin of safety.

In addition, the proposed methyl iodide penetration acceptance criteria include a safety factor of two as permitted by GL 99-02. This safety factor provides a degree of assurance that, at the end of the operating cycle, the charcoal will be capable of performing at a level at least as good as that assumed in the design basis accident dose analysis. The NRC found this factor of safety acceptable, based on the accuracy of test results obtained using the ASTM D3803-1989 standard, as noted in the NRC safety evaluation report enclosed in the letter dated May 13, 1998, NRC to OD Kingsley, "Issuance of Amendments (TAC NOS. M99726 AND M99727)."

Therefore, operation of WNP-2 in accordance with the proposed amendment will not involve a significant reduction in the margin of safety.

Environmental Assessment Applicability Review

Energy Northwest has evaluated the proposed amendment against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10CFR51.21.

The proposed change meets the criteria for categorical exclusion as provided for in 10CFR51.22(c)(9). The change request does not pose a significant hazards consideration nor does it involve an increase in the amounts, or a change in the types, of any effluent that may be released off-site.

Furthermore, this proposed request does not involve an increase in individual or cumulative occupational exposure.

REQUEST FOR AMENDMENT
TECHNICAL SPECIFICATION 5.5.7.c
VENTILATION FILTER TESTING PROGRAM
Attachment 3

Marked-Up Version of Technical Specification 5.5.7.c

5.5 Programs and Manuals

5.5.7 <u>Ventilation Filter Testing Program (VFTP)</u> (continued)

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-~~1986 (Method B for the SGT System and Method A for the CREF System)~~ 1989 at a temperature of 30°C (86°F) and the relative humidity ~~greater than or equal to the value~~ specified ~~below:~~ below. Testing of the SGT System will also be conducted at a face velocity of ~~65~~ 75 feet per minute.

ESF Ventilation System	Penetration (%)	RH (%)
SGT System	~~0.175~~ 0.5	70
CREF System	~~1.0~~ 2.5	70

d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters and the charcoal adsorbers is less than the value specified below when tested at the system flowrate specified below:

ESF Ventilation System	Delta P (inches wg)	Flowrate (cfm)
SGT System	< 8	4012 to 4902
CREF System	< 6	900 to 1100

e. Demonstrate that the heaters for each of the ESF systems dissipate the nominal value specified below when tested in accordance with ASME N510-1989:

ESF Ventilation System	Wattage (kW)
SGT System	18.6 to 22.8
CREF System	4.5 to 5.5

5.5.8 <u>Explosive Gas and Storage Tank Radioactivity Monitoring Program</u>

This program provides controls for potentially explosive gas mixtures contained in the Main Condenser Offgas Treatment System and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks.

The program shall include:

(continued)

Variations in the above testing parameters of temperature, relative humidity, and face velocity are permitted per Table 1 and Section A5.2 of ASTM D3803-1989.

REQUEST FOR AMENDMENT
TECHNICAL SPECIFICATION 5.5.7.c
VENTILATION FILTER TESTING PROGRAM
Attachment 4

Replacement Pages for Technical Specification 5.5.7.c

5.5 Programs and Manuals

5.5.7 Ventilation Filter Testing Program (VFTP) (continued)

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and the relative humidity specified below. Testing of the SGT System will also be conducted at a face velocity of 75 feet per minute.

ESF Ventilation System	Penetration (%)	RH (%)
SGT System	0.5	70
CREF System	2.5	70

Variations in the above testing parameters of temperature, relative humidity, and face velocity are permitted per Table 1 and Section A5.2 of ASTM 03803-1989.

d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters and the charcoal adsorbers is less than the value specified below when tested at the system flowrate specified below:

ESF Ventilation System	Delta P (inches wg)	Flowrate (cfm)
SGT System	< 8	4012 to 4902
CREF System	< 6	900 to 1100

e. Demonstrate that the heaters for each of the ESF systems dissipate the nominal value specified below when tested in accordance with ASME N510-1989:

ESF Ventilation System	Wattage (kW)
SGT System	18.6 to 22.8
CREF System	4.5 to 5.5

(continued)

5.5 Programs and Manuals (continued)

5.5.8 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Main Condenser Offgas Treatment System and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks.

The program shall include:

a. The limits for concentrations of hydrogen in the Main Condenser Offgas Treatment System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion); and

b. A surveillance program to ensure that the quantity of radioactivity contained in all outside temporary liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System is less than the amount that would result in concentrations greater than the limits of Appendix B, Table 2, Column 2 to 10 CFR 20.1001 - 20.2402, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program Surveillance Frequencies.

5.5.9 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program shall establish the required testing of both new fuel oil and stored fuel oil. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

   1. An API gravity, a specific gravity, or an absolute specific gravity within limits,

(continued)

5.5 Programs and Manuals

5.5.9 <u>Diesel Fuel Oil Testing Program</u> (continued)

2. A kinematic viscosity, if gravity was not determined by comparison with the supplier's certificate, and a flash point within limits for ASTM 2-D fuel oil,

3. A water and sediment content within limits or a clear and bright appearance with proper color;

b. Other properties for ASTM 2-D fuel oil are within limits within 31 days following sampling and addition to storage tanks; and

c. Total particulate concentration of the fuel oil in the storage tanks is ≤ 10 mg/l when tested every 31 days in accordance with ASTM D-2276, Method A-2 or A-3.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program test Frequencies.

5.5.10 <u>Technical Specifications (TS) Bases Control Program</u>

This program provides a means for processing changes to the Bases to these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:

1. A change in the TS incorporated in the license; or

2. A change to the FSAR or Bases that involves an unreviewed safety question as defined in 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

d. Proposed changes that meet the criteria of 5.5.10.b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

(continued)

50-397
11/18/99

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Subject:
WNP-2, OPERATING LICENSE NPF-21 REQUEST FOR AMENDMENT TECHNICAL SPECIFICATION 4.3.1.2.b FUEL STORAGE

Body:
pdr adock 05000397 p

Docket: 05000397, Notes: N/A

AA2

ENERGY NORTHWEST

P.O. Box 968 ▫ Richland, Washington 99352-0968

November 18, 1999
GO2-99-202

Docket No. 50-397

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Gentlemen:

Subject: **WNP-2, OPERATING LICENSE NPF-21**
REQUEST FOR AMENDMENT
TECHNICAL SPECIFICATION 4.3.1.2.b
FUEL STORAGE

Reference: NRC Administrative Letter 98-10, December 29, 1998, "Dispositioning of Technical Specifications that are Insufficient to Assure Plant Safety"

In accordance with the Code of Federal Regulations, Title 10, Parts 2.101, 50.59, and 50.90, Energy Northwest hereby submits a request for amendment to the WNP-2 Operating License. Specifically, Energy Northwest is requesting a revision to sub-section 4.3.1.2.b of Technical Specification 4.3.1 "Criticality," to revise the wording that defines the limitations for placement of fuel in the New Fuel Storage Facility.

The current wording of Technical Specification 4.3.1.2.b, adopted as part of the Improved Technical Specifications and documented by NUREG-1434, correctly describes the new fuel vault rack spacing associated with the original rack design. However, it does not accurately reflect the current design features and controls relied upon to adequately limit the spacing of new fuel assemblies in the new fuel vault as required to ensure compliance with Technical Specification 4.3.1.2.a under all postulated conditions; and, therefore constitutes a degraded or non-conforming condition pursuant to the guidance of the Reference. This correction should have been made as part of the review activities in preparation for submittal of the Improved Technical Specifications, but. was not. We are proposing an amendment to subsection 4.3.1.2.b of Technical Specification 4.3.1 to address this non-conforming condition.

A001

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Additional information has been attached to this letter to complete Energy Northwest's amendment request. Attachment 1 provides a detailed description and basis for acceptability of the proposed changes. Attachment 2 describes an evaluation of the proposed changes in accordance with 10CFR50.92(c), and concludes the changes do not result in a significant hazards consideration. Attachment 3 provides the Environmental Assessment Applicability Review and notes that the proposed change meets the eligibility criteria for a categorical exclusion as set forth in 10CFR51.22(c)(9). Therefore, in accordance with 10CFR51.22(b), an environmental assessment of the change is not required. Attachment 4 summarizes the proposed change and provides a marked up page of the Technical Specification. Attachment 5 submits the typed Technical Specification page as proposed by this request.

This request for an amendment has been approved by the WNP-2 Plant Operations Committee and reviewed by Energy Northwest's Corporate Nuclear Safety Review Board. In accordance with 10CFR50.91, the State of Washington has been provided a copy of this letter.

Should you have any questions or desire additional information regarding this matter, please contact me or PJ Inserra at (509) 377-4147.

Respectfully,

DK Atkinson for

RL Webring
Vice President, Operations Support/PIO
Mail Drop PE08

Attachments

cc: EW Merschoff -- NRC RIV
JS Cushing -- NRC NRR
NRC Resident Inspector -- 927N
DJ Ross -- EFSEC
TC Poindexter --Winston & Strawn
DL Williams -- BPA/1399

STATE OF WASHINGTON)
)
COUNTY OF BENTON)

Subject: Request for Amendment
Technical Specification 4.3.1.2.b
Fuel Storage

I, DK ATKINSON, being duly sworn, subscribe to and say that I am the Acting Vice President, Operations Support/PIO for ENERGY NORTHWEST, the applicant herein; that I have the full authority to execute this oath; that I have reviewed the foregoing; and that to the best of my knowledge, information, and belief the statements made in it are true.

DATE November 18, 1999

D.K. Atkinson

DK Atkinson
Acting Vice President, Operations Support/PIO

On this date personally appeared before me DK Atkinson, to me known to be the individual who executed the foregoing instrument, and acknowledged that he signed the same as his free act and deed for the uses and purposes herein mentioned.

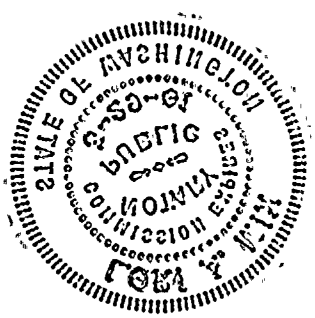
GIVEN under my hand and seal this 18 day of November 1999.

Lori A. Mix

Notary Public in and for the
STATE OF WASHINGTON

Residing at W. Richland

My Commission Expires 3-29-01



Evaluation of Significant Hazards Consideration

Summary of Proposed Change

Energy Northwest is proposing an amendment to sub-section 4.3.1.2.b of Technical Specification 4.3.1, "Criticality." We propose to change the current wording, which describes the new fuel racks, with wording that would limit the number of fuel assemblies that may be stored in the facility, and establish geometrical limitations for storage of new fuel assemblies in the racks. The proposed wording is as follows (changes are <u>underlined</u>):

4.3.1.2 The new fuel storage racks are designed and, <u>with fuel assemblies inserted</u>, shall be maintained with:

a. (no change)

b. <u>A maximum of 60 new fuel assemblies stored in the new fuel storage racks, arranged in 6 spatially separated zones. Within a storage zone, the nominal center-to-center distance between cells for storing fuel assemblies is 14 inches. The nominal center-to-center distance between cells for storing fuel assemblies in adjacent zones is 37 inches. Design features relied upon to spatially limit the placement of fuel bundles within the new fuel vault are required to be installed prior to placement of new fuel bundles in the vault.</u>

No Significant Hazards Consideration Determination

Energy Northwest has evaluated the proposed change to Technical Specifications using the criteria established in 10CFR50.92(c), and has determined that it does not represent a significant hazards consideration as described below:

- **The operation of WNP-2 in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.**

  The proposed change does not increase the consequences of any previously analyzed accident or transient, since the arrangement of new nuclear fuel in storage racks maintains the effective neutron multiplication factor much less than 0.95. The change in configuration requirements will not increase the probability of any previously analyzed accident, because physical constraints are installed in the storage racks when new fuel assemblies are inserted, assuring that only certain cells can be used for storage of new fuel.

  Therefore, operation of WNP-2 in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

- **The operation of WNP-2 in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.**

  The proposed change is consistent with a new fuel criticality analysis performed in support of a previously implemented change to Section 9.1 of the FSAR. A variety of accidents were considered in that analysis, and it was determined that the effective neutron multiplication factor was well below specified limits for any normal or accident case.

  Therefore, operation of WNP-2 in accordance with the proposed amendment will not create the possibility of a new or different kind of accident previously evaluated.

- **The operation of WNP-2 in accordance with the proposed amendment will not involve a significant reduction in the margin of safety.**

  The current wording of Technical Specification 4.3.1.2.b was determined to not provide sufficient margin of safety to assure that the requirements of Technical Specification 4.3.1.2.a would be maintained. The proposed amendment modifies the requirements for new fuel storage configuration for Technical Specification 4.3.1.2.b, to assure the margin of safety is maintained for optimum moderation conditions.

  Therefore, operation of WNP-2 in accordance with the proposed amendment will not involve a significant reduction in the margin of safety.

Description of Proposed Changes

Summary Of Proposed Technical Specification Change

Energy Northwest is proposing an amendment to sub-section 4.3.1.2.b of Technical Specification 4.3.1, "Criticality." We propose to change the current wording, which describes the new fuel racks, with wording that would limit the number of fuel assemblies that may be stored in the facility, and establish geometrical limitations for storage of new fuel assemblies in the racks. The proposed wording is as follows (changes are underlined):

4.3.1.2 The new fuel storage racks are designed and, with fuel assemblies inserted, shall be maintained with:

a. (no change)

b. A maximum of 60 new fuel assemblies stored in the new fuel storage racks, arranged in 6 spatially separated zones. Within a storage zone, the nominal center-to-center distance between cells for storing fuel assemblies is 14 inches. The nominal center-to-center distance between cells for storing fuel assemblies in adjacent zones is 37 inches. Design features relied upon to spatially limit the placement of fuel bundles within the new fuel vault are required to be installed prior to placement of new fuel bundles in the vault.

Basis for the Proposed Technical Specification Change

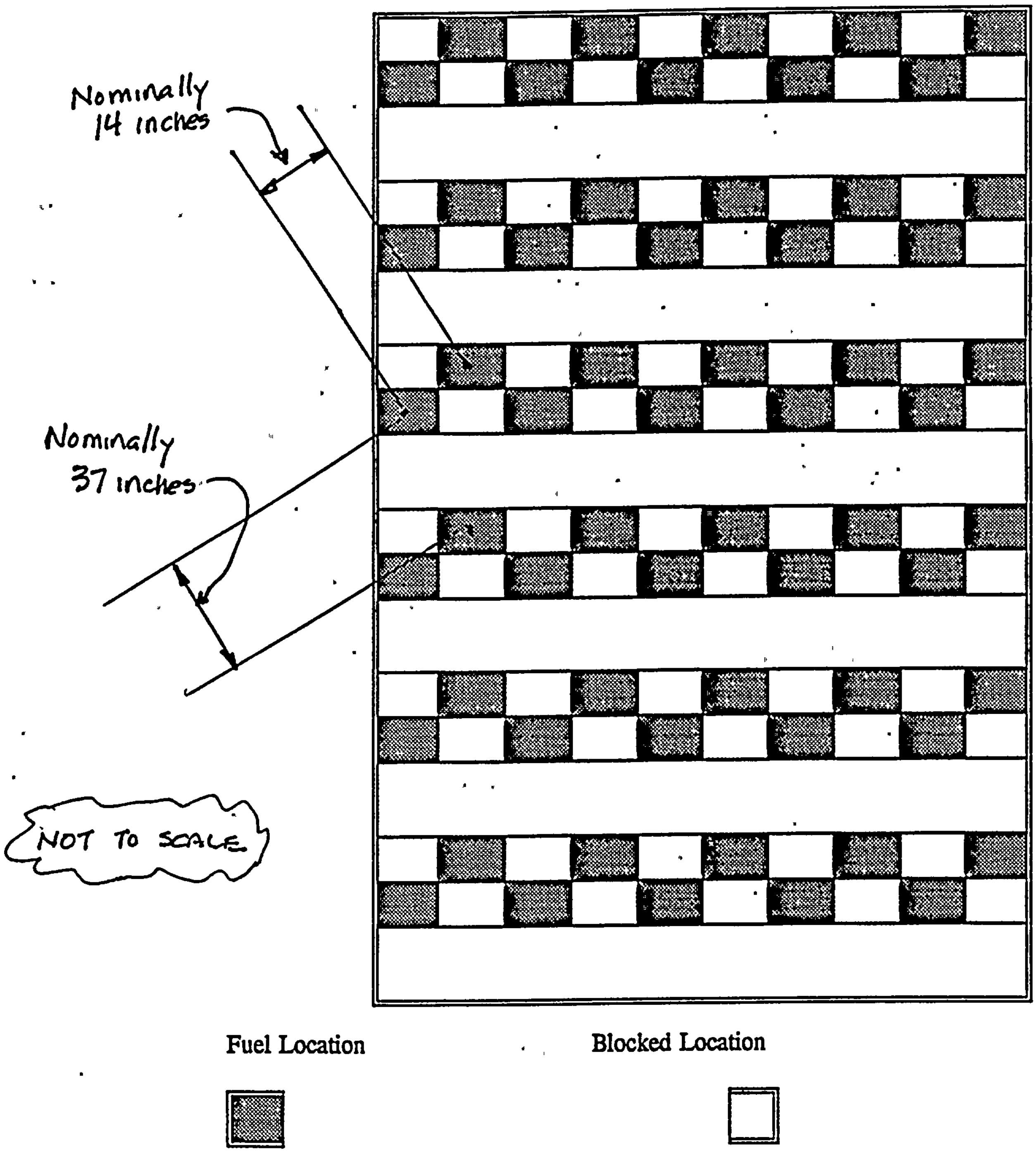
The New Fuel Storage Facility is a dry storage facility with air as the medium surrounding stored fuel. The facility is a concrete vault; both the vertical and horizontal cross-sections are rectangular. The floor of the vault includes a drain to remove water that may accidentally or unknowingly be introduced into the vault.

The cell utilization pattern for the fuel consists of 2 contiguous rows in which fuel assemblies may be stored, alternating with 2 contiguous rows in which fuel storage is prohibited. Within a 2-row set in which fuel is stored, alternate cells are physically blocked, in a checkerboard pattern, to prevent inadvertent cell usage. This results in a nominal center-to-center distance between cells for storing fuel assemblies of 14 inches. The nominal center-to-center distance between cells used to store fuel, across the 2-row set in which fuel storage is prohibited, is 37 inches. A sketch of this utilization pattern is included on Page 3 of this attachment.

The above configuration was analyzed to determine the effective neutron multiplication factor, k_{eff}, for (1) geometrical variations resulting from tolerances for the installation, (2) air as the vault atmosphere, and (3) water as the vault atmosphere in a range of densities varying from 1 to 0.02 gm./cc.

Additionally, postulated accidents were included in the analysis: assemblies dropped on the vault floor, and insertion patterns that varied from the baseline configuration described above. No credit was taken for the neutron absorptive effect of metals comprising the storage rack, the gadolinium and the zirconium cladding in the fuel assemblies, and any metal in the concrete structure of the vault. The analysis was performed using the computer code KENO, with neutron cross-sections calculated using the PHOENIX code. The NRC has approved both codes. The conclusion of the analysis of this configuration is that k_{eff} ranges between 0.64 and 0.86 for normal geometry and is 0.898 for a worst-case accident involving an insertion pattern that varied from the specified baseline configuration. The dropped fuel bundle accident resulted in a range of k_{eff} of 0.87 to 0.88. Technical Specification 4.3.1.2.a specifies a limiting value of 0.95 for k_{eff} when fully flooded with unborated water. In short, the KENO analysis shows a considerable margin of safety for the configuration described above, graphically presented on Page 3 of this attachment, and for configurations resulting from accidents involving dropped fuel assemblies and insertion errors.

NEW FUEL STORAGE VAULT FUEL LOADING PATTERN



Environmental Assessment Applicability Review

Energy Northwest has evaluated the proposed amendment against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10CFR51.21.

The proposed change meets the criteria for categorical exclusion as provided under 10CFR51.22(c)(9) because the change does not pose a significant hazards consideration nor does it involve an increase in the amounts, or a change in the types, of any effluent that may be released offsite.

Furthermore, this request does not involve an increase in individual or cumulative occupational exposure.

REQUEST FOR AMENDMENT
TECHNICAL SPECIFICATION 4.3.1.2.b
FUEL STORAGE
Attachment 4

Marked-Up Version of Technical Specification 4.3.1.2.b

4.0 DESIGN FEATURES (continued)

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1.2 of the FSAR; and

b. A nominal 6.5 inch center to center distance between fuel assemblies placed in the storage racks.

, with fuel assemblies inserted,

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

a. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1.1 of the FSAR; and

~~b. A nominal fuel assembly center to center spacing of 7.0 inches within rows and 12.25 inches between rows in the new fuel storage racks.~~

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 583 ft 1.25 inches.

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 2658 fuel assemblies.

b. A maximum of 60 new fuel assemblies stored in the new fuel storage racks, arranged in 6 spatially separated zones. Within a storage zone, the nominal center-to-center distance between cells for storing fuel assemblies is 14 inches. The nominal center-to-center distance between cells for storing fuel assemblies in adjacent zones is 37 inches. Design features relied upon to spatially limit the placement of fuel bundles within the new fuel vault are required to be installed prior to placement of new fuel bundles in the vault.

REQUEST FOR AMENDMENT
TECHNICAL SPECIFICATION 4.3.1.2.b
FUEL STORAGE
Attachment 5

Replacement Page for Technical Specification 4.3.1.2.b

4.0 DESIGN FEATURES (continued)

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1.2 of the FSAR; and

b. A nominal 6.5 inch center to center distance between fuel assemblies placed in the storage racks.

4.3.1.2 The new fuel storage racks are designed and, with fuel assemblies inserted, shall be maintained with:

a. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1.1 of the FSAR; and

b. A maximum of 60 new fuel assemblies stored in the new fuel storage racks, arranged in 6 spatially separated zones. Within a storage zone, the nominal center-to-center distance between cells for storing fuel assemblies is 14 inches. The nominal center-to-center distance between cells for storing fuel assemblies in adjacent zones is 37 inches. Design features relied upon to spatially limit the placement of fuel bundles within the new fuel vault are required to be installed prior to placement of new fuel bundles in the vault.

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 583 ft 1.25 inches.

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 2658 fuel assemblies.

10/20/99

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Subject:
REQUEST FOR AMENDMENT TO TECHNICAL SPECIFICATIONS SR 3.8.4.6 and SR 3.8.5.1 (REPLACEMENT PAGES)

Body:

Docket: 05000397, Notes: N/A

AA2

ENERGY NORTHWEST

P.O. Box 968 ▫ Richland, Washington 99352-0968

October 20, 1999
GO2-99-184

Docket No. 50-397

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Gentlemen:

Subject: **WNP-2, OPERATING LICENSE NPF-21**
REQUEST FOR AMENDMENT
TECHNICAL SPECIFICATIONS SR 3.8.4.6 and SR 3.8.5.1
(REPLACEMENT PAGES)

Reference: Letter GO2-99-146, dated July 29, 1999, RL Webring (Energy Northwest) to NRC, "Request for Amendment, Technical Specifications SR 3.8.4.6 and 3.8.5.1"

The purpose of this letter is to resubmit the typed Technical Specification pages as they would be revised by the referenced amendment request. The original pages, which were included as Attachment 5 in the reference, contained an incorrect page numbering sequence.

The replacement pages associated with the proposed changes are included as an attachment and reflect the corrected page numbering. No other changes were made.

Should you have any questions or desire additional information regarding this matter, please call me or PJ Inserra at (509) 377-4147.

Respectfully,

Paul Inserra
for
DW Coleman
Manager, Regulatory Affairs
Mail Drop PE20

1/1

A001

Attachment

cc: EW Merschoff - NRC RIV
JS Cushing – NRC NRR
NRC Sr. Resident Inspector - 927N
DL Williams - BPA/1399
TC Poindexter - Winston & Strawn

290040
9910260091

REQUEST FOR AMENDMENT
TECHNICAL SPECIFICATIONS SR 3.8.4.6 and SR 3.8.5.1
(REPLACEMENT PAGES)

Attachment

Replacement Pages

Technical Specifications SR 3.8.4.6 and SR 3.8.5.1 Amendment Request

9930600092

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.4.5	Verify battery connection resistance is $\leq$ 24.4 E-6 ohms for inter-cell connectors of the Division 1 and 2 batteries, $\leq$ 169 E-6 ohms for inter-cell connectors of the Division 3 battery, and $\leq$ 20% above the resistance as measured during installation for inter-tier and inter-rack connectors.	12 months
SR 3.8.4.6	--------------------NOTE------------------- This Surveillance shall not be performed in MODE 1, 2, or 3. However, credit may be taken for unplanned events that satisfy this SR. --- Verify each required battery charger supplies the required load for $\geq$ 1.5 hours at: a. $\geq$ 126 V for the 125 V battery chargers; and b. $\geq$ 252 V for the 250 V battery charger.	24 months

(continued)

[illegible] [illegible]

[illegible] [illegible]

[illegible] [illegible]

[illegible] [illegible]

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.4.7	------------------NOTES------------------ 1. The modified performance discharge test in SR 3.8.4.8 may be performed in lieu of the service test in SR 3.8.4.7 once per 60 months. 2. This Surveillance shall not be performed in MODE 1, 2, or 3. However, credit may be taken for unplanned events that satisfy this SR. --- Verify battery capacity is adequate to supply, and maintain in OPERABLE status, the required emergency loads for the design duty cycle when subjected to a battery service test.	24 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.4.8	-------------------NOTE-------------------- This Surveillance shall not be performed in MODE 1, 2, or 3. However, credit may be taken for unplanned events that satisfy this SR. ---	
	Verify battery capacity is ≥ 80% of the manufacturer's rating for the 125 V batteries and ≥ 83.4% of the manufacturer's rating for the 250 V battery, when subjected to a performance discharge test or a modified performance discharge test.	60 months <u>AND</u> 12 months when battery shows degradation or has reached 85% of expected life with capacity < 100% of manufacturer's rating <u>AND</u> 24 months when battery has reached 85% of the expected life with capacity ≥ 100% of manufacturer's rating

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.3 Initiate action to suspend operations with a potential for draining the reactor vessel. AND A.2.4 Initiate action to restore required DC electrical power subsystems to OPERABLE status.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.5.1	--------------------NOTE------------------- The following SRs are not required to be performed: SR 3.8.4.6, SR 3.8.4.7, and SR 3.8.4.8. --- For DC electrical power subsystems required to be OPERABLE the following SRs are applicable: SR 3.8.4.1, SR 3.8.4.2, SR 3.8.4.3, SR 3.8.4.4, SR 3.8.4.5, SR 3.8.4.6, SR 3.8.4.7, and SR 3.8.4.8.	In accordance with applicable SRs