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WASHINGTON PUBLIC POWER SUPPLY SYSTEM

P.O. Box 968 • 3000 George Washington Way • Richland, Washington 99352

January 23, 1991
G02-91-012

Docket No. 50-397

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station P1-137
Washington, D. C. 20555

Gentlemen:

Subject: NUCLEAR PLANT NO. 2, OPERATING LICENSE NO. NPF-21
NRC INSPECTION REPORT 90-29
RESPONSE TO NOTICE OF VIOLATION

The Washington Public Power Supply System hereby replies to the Notice of Violation contained in your letter dated December 24, 1990. Our reply, pursuant to the provisions of Section 2.201, Title 10, Code of Federal Regulations, consists of this letter and Appendices A and B (attached).

In Appendix A, the violation is addressed with an explanation of our position regarding validity, corrective action and date of full compliance. Appendix B addresses the issue pertaining to the capability of the Reactor Building Post-LOCA Grab Sampler to quantify post accident radioiodine concentrations in effluents. As requested in the cover letter to the subject inspection report, a schedule for correcting system performance is also contained in Appendix B.

Very truly yours,

G. D. Bouchey, Director
Licensing & Assurance

JDA/bk
Attachments

cc: JB Martin - NRC RV
NS Reynolds - Winston/Strawn
PL Eng - NRR
DL Williams - BPA/399
NRC Site Inspector - 901A

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RECIP.NAME RECIPIENT AFFILIATION
Document Control Branch (Document Control Desk)

SUBJECT: Responds to NRC 901224 ltr re violations noted in Insp Rept 50-397/90-29 on 901113-16.Corrective actions:separate procedure will be developed to provide method to document transfer of radioactive matl to another licensee.

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APPENDIX A

During an NRC inspection conducted on November 13-16, 1990, a violation of NRC requirements was identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Action," 10 CFR Part 2, Appendix C (1990), the violation is listed below:

10 CFR Part 71.5 requires, in part, that each licensee who transports licensed material outside of the confines of the plant shall comply with the regulations appropriate to the mode of transport of U.S. Department of Transportation (DOT) in 49 CFR Parts 170 through 189.

49 CFR Part 173.415(a) requires, in part, that each shipper of a DOT Specification 7A (Spec. 7A) package must maintain on file for at least one year after the latest shipment, a completed documentation of tests and an engineering evaluation showing that the construction methods, package design, and materials of construction comply with that specification (49 CFR 178.350).

Contrary to the above, March 21, 1990, the licensee shipped (WNP-2 shipment No. 90-20-02) radioactive materials in a DOT Spec. 7A package, and on November 13, 1990 a completed documentation of tests and engineering evaluations were not on file at the licensee's facilities.

This is a Severity Level IV violation (Supplement V).

Validity of Violation

The Supply System acknowledges the validity of this violation. The reason for the violation is less than adequate procedures in that there were no guidelines in Plant procedures for documenting a transfer of custody for radioactive material. In this particular situation, Battelle Pacific Northwest Laboratories requested a Reactor Water Cleanup (RWCU) resin sample from the Supply System to use in support of an EPRI Waste Management Study. The Supply System Health Physics person involved determined that Battelle would provide the shipping container, the transfer of material would take place within the confines of the WNP-2 Plant and that Battelle would transport the material offsite.

For these reasons, the Health Physics individual involved correctly viewed this transaction as a material transfer and not a shipment on the part of the Supply System. The individual then documented this transaction on an Offsite Radioactive Shipment Record. However, there was nothing on the form to indicate that this was not a Supply System shipment and, as such, documentation needed to be on file that a Department of Transportation, Specification 7A, container was used in the process. Efforts to obtain container test documentation from Battelle were unsuccessful.

This type of transaction with Battelle has occurred in the past to support research projects; however, the type of material was designated as Limited Quantity. Limited Quantity material does not require container specification test documentation and, as in this case, the transactions were transfers of radioactive material and not shipments. In those few instances where the Supply System has shipped Type A material to Battelle for research, a certified container has been supplied by WNP-2.

Corrective Steps Taken/Results Achieved

Plant Procedure (PPM) 11.2.23.6, "Shipping Other Than LSA Radioactive Materials," was modified on November 19, 1990 to include a signoff on the Radioactive Material Shipment Checklist that test results are on file for DOT, Specification A, containers.

Corrective Action to be Taken

A separate procedure will be developed to provide a method to document a transfer of radioactive material on Supply System property to another licensee authorized to transport such material on the Department of Energy Hanford Reservation.

Date of Full Compliance

Although full compliance was achieved on November 19, 1990 when PPM 11.2.23.6 was revised, the separate procedure for documenting the transfer of radioactive materials will be developed by March 31, 1991.

APPENDIX B

In the subject inspection report, the NRC followed up on its previous concern pertaining to the ability of the Reactor Building Post-Accident Grab Sampling System to representatively sample effluent releases under accident conditions involving steam, with attendant variable temperatures along the sample line due to halogen plateout and excessive condensation. As requested in the cover letter to the inspection report, this appendix provides our response to this concern and includes the current schedule for correcting system performance.

Reactor Building Post-Accident Grab Sampler REA-SR-48 implements NUREG-0737 Item II.F.1, Attachment 2, which requires the ability to representatively sample and quantify the effluent release rate for gaseous/particulate radioiodine under accident conditions. In the NUREG, the type and extent of the accident for which sampling must be provided is not defined.

Testing on the system was initiated and results indicated that a correction factor was needed to attain a representative sample due to sample flow rate and line size associated with REA-SR-48. The correction factor of 500 is large and considered to be undesirable for the long term. However, the factor is based upon independent offsite laboratory testing of the sample system (by means of a representative mockup) and is considered to be a valid and acceptable near-term compensation in the consideration of operability for the sample rack.

Furthermore, the identification of accident conditions involving steam, laden with radioiodine, venting through the elevated release path are beyond the WNP-2 design basis. Even in consideration of the requirement to evaluate the system to beyond-design-basis events, several failures must occur before such a condition would be present. Although NUREG-0737, Table II.F.1-2, does require entrained moisture to be considered in the context of adsorber degradation, it is also not clear that this represents a requirement to discount the moisture removal provisions already provided such as the Standby Gas Treatment System. When combined with the environmental requirements for the attainment of threshold conditions causing the potential for condensate accumulation, the beyond-design-basis accident scenarios are too complex to allow for design protection against all possible event paths.

Sampler REA-SR-48, with the correction factor applied and under conditions not causing condensate accumulation in the sample lines, is capable of reasonably quantifying post-accident radioiodine concentrations in effluents, even for many scenarios beyond the design basis. Although the system is considered operable with the correction factor applied, the decision to assume that the instrument is unreliable would be accident-scenario-based and would be evaluated under those conditions by the emergency response team. In the event of those beyond-design-basis conditions which would render the grab sampler inoperable, the high-range noble gas effluent radiation monitors would be utilized to quantify the radioiodine release rate using default values to reflect the presence in the stack of 25% of the core halogens that could be released based on the estimated percent fuel damage. This conservative approach is based on Regulatory Guide 1.3 source term guidance and existing WNP-2 fuel damage assessment procedures, and is consistent with current Protective Action Recommendation (PAR) determination guidelines.

However, it is also recognized that improvement in the performance of REA-SR-48 is advisable for the long term. An evaluation is currently in progress to determine the best option for improving system performance with regard to the requirements of NUREG-0737. Included in this evaluation is consideration of the use of on-line effluent monitors incorporating recent developments in pure germanium, helium-cooled sensor technology. On-line analysis of the spectrum of nuclides present in the stack, combined with existing flow monitoring from REA-SR-37, could exceed the guidance in NUREG-0737. The evaluation is currently scheduled for completion during July, 1991. At the completion of the evaluation, a supplemental response will be provided describing the action to be taken and its scheduled implementation date.