PSEG Nuclear LLC
P.O. Box 236, Hancocks Bridge, New Jersey 08038-0236

AUG 0 8 2003

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PSEG
Nuclear LLC

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

60-DAY RESPONSE TO GENERIC LETTER 2003-01
CONTROL ROOM HABITABILITY
SALEM UNIT NOS. 1 AND 2 AND HOPE CREEK
FACILITY OPERATING LICENSE NOS. DPR-70, DPR-75 AND NPF-57
DOCKET NOS. 50-272, 50-311, AND 50-354

On June 12, 2003, the NRC issued Generic Letter (GL) 2003-01, "Control Room Habitability." GL 2003-01 requires each plant to confirm that their facility's control room meets the applicable habitability regulatory requirements and that the Control Room Habitability Systems (CRHSs) are designed, constructed, configured, operated, and maintained in accordance with the facility's design and licensing basis. As stated in GL 2003-01, if a licensee cannot provide the information requested for the 180-day response, then the licensee should submit a written response within 60-days providing the alternative course of action, the basis for the acceptability of the proposed alternative course of action and the schedule for completion.

Item 1(a) of the GL 2003-01 specifically requests that each plant confirm that the most limiting unfiltered inleakage (and the filtered inleakage if applicable) into the plants Control Room Envelope (CRE) is no more than the value assumed in the plant's design basis radiological analyses for control room habitability. The Salem CRE is a common envelope for both Salem Units 1 and 2. With the exception of the Fuel Handling Accident (FHA), the dose analyses for Salem assume a CRE inleakage value of 60 cfm unfiltered inleakage. The FHA dose analysis was revised utilizing alternative source term (AST) in accordance with 10CFR50.67 and approved in Amendment 251 for Salem Unit 1 and 232 for Salem Unit 2. The revised FHA dose analysis assumes a CRE inleakage value of 4000 cfm. For Salem Units 1 and 2, tracer gas testing was performed from May 31 to June 4, 2003, to measure the inleakage to the Salem CRE. The Salem Control Room Emergency Air Conditioning System (CREACS) can be aligned to respond to a design basis accident with either a single train (either Unit 1 or Unit 2 train) operating by itself or with both trains operating. Three tests were performed in single train alignment of CREACS (two with the Unit 1 train operating by

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itself and one with the Unit 2 train operating by itself) and one test was performed with both CREACS trains operating. The preliminary results of the tracer gas testing identified that two of the four test results were below 60 cfm unflitered inleakage and two tests measured unfiltered inleakage in the range of 90 to 100 cfm. Based on the preliminary results of the tracer gas testing, a GL 91-18 operability determination was issued for the Salem dose analyses that assume a maximum inleakage of 60 cfm stating that the Salem CRE was operable but non-conforming. To terminate the operability determination, PSEG is planning to perform a full conversion of the Salem dose analysis to AST. Preliminary work performed on the conversion to AST indicates that the unfiltered inleakage assumed in the dose analysis can be increased to bound the measured inleakage value without exceeding the limits of 10CFR50 Appendix A General Design Criterion 19 (GDC-19). As noted above, the FHA analysis has already been converted to AST. A license change request to convert the dose analysis to AST is currently planned to be submitted by the end of December 2003.

Item 1(b) of the GL 2003-01 specifically requests that plants confirm that the most limiting unfiltered inleakage into the plant's CRE is incorporated into the hazardous chemical assessments. PSEG has performed a preliminary review of the hazardous chemical assessments for both Salem and Hope Creek and has determined that these evaluations have not incorporated the unfiltered inleakage measured from tracer gas testing performed for Salem in June 2003 and Hope Creek in July 2001. A preliminary assessment of the hazardous chemical evaluations has been performed using the measured unfiltered inleakage values. The preliminary indications of this assessment indicate that there will be no changes needed to the operation of Salem and Hope Creek as a result of including the measured unfiltered inleakage values into the hazardous chemical evaluations. The hazardous chemical evaluations for Salem and Hope Creek will be revised to incorporate the measured unfiltered inleakage values into the evaluations. The revision of these evaluations will be completed prior to submittal of the 180-day response to GL 2003-01.

PSEG will submit the remaining requested information in items 1, 2 and 3 of GL 2003-01 for Salem and Hope Creek in our 180-day response.

Should you have any questions regarding this submittal, please contact Mr. Brian Thomas at 856-339-2022.

Sincerely,

John Carlin
Vice President – Technical Support

C Mr. H. J. Miller, Administrator - Region I
U. S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

U. S. Nuclear Regulatory Commission
ATTN: Mr. R. Fretz, Licensing Project Manager - Salem
Mail Stop 08B2
Washington, DC 20555-0001

U. S. Nuclear Regulatory Commission
ATTN: Mr. R. Ennis, Licensing Project Manager – Hope Creek
Mail Stop 08B2
Washington, DC 20555-0001

USNRC Senior Resident Inspector - Salem (X24)

USNRC Senior Resident Inspector – Hope Creek (X24)

Mr. K. Tosch, Manager IV
Bureau of Nuclear Engineering
PO Box 415
Trenton, NJ 08625

OMB Control No.: 3150-0011

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, DC 20555-0001

June 12, 2003

NRC GENERIC LETTER 2003-01: CONTROL ROOM HABITABILITY

Addressees

All holders of operating licenses for pressurized-water reactors (PWRs) and boiling-water reactors (BWRs), except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel and more than 1 year has elapsed since fuel was irradiated in the reactor vessel.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this generic letter to:

(1) alert addressees to findings at U.S. power reactor facilities suggesting that the control room licensing and design bases, and applicable regulatory requirements (see section below) may not be met, and that existing technical specification surveillance requirements (SRs) may not be adequate,

(2) emphasize the importance of reliable, comprehensive surveillance testing to verify control room habitability,

(3) request addressees to submit information that demonstrates that the control room at each of their respective facilities complies with the current licensing and design bases, and applicable regulatory requirements, and that suitable design, maintenance and testing control measures are in place for maintaining this compliance, and

(4) collect the requested information to determine if additional regulatory action is required.

Background

The control room is the plant area, defined in the facility licensing basis, from which actions are taken to operate the plant safely under normal conditions and to maintain the reactor in a safe condition during accident situations. For most facilities, the habitability criteria of General Design Criterion 19 (GDC 19) in 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," apply to this area. The control room envelope (CRE) is the plant area, defined in the facility licensing basis, that encompasses the control room and may encompass other plant areas. The structures that make up the CRE are designed to limit the inleakage of radioactive and hazardous materials from areas external to the CRE. Control room habitability

systems (CRHSs) typically provide the functions of shielding, isolation, pressurization, heating, ventilation, air conditioning and filtration, monitoring, and the sustenance and sanitation necessary to ensure that the control room operators can remain in the control room and take actions to operate the plant under normal and accident conditions. The personnel protection features incorporated into the design of a particular plant's CRHSs depend on the nature and scope of the plant-specific challenges to maintaining control room habitability. In the majority of the CRHS designs, isolation of the normal supply and exhaust flow paths and pressurization of the CRE relative to adjacent areas are fundamental to ensuring a habitable control room.

During the design of a nuclear power plant, licensees perform analyses to demonstrate that the CRHSs, as designed, provide a habitable environment during postulated design basis events. These design analyses model the transport of potential contaminants into the CRE and their removal. The amount of inleakage of assumed contaminants is important to these analyses. Unaccounted-for contaminants entering the CRE may impact the ability of the operators to perform plant control functions. If contaminants impair the response of the operators to an accident, there could be increased consequences to the public health and safety.

There are two typical CRE designs. These designs are referred to as positive-pressure and neutral-pressure CREs. Both designs focus on limiting the amount of contaminants entering the CRE. For radiological challenges, the positive-pressure CRE intentionally pressurizes the CRE with air from outside the CRE. The pressurization air is treated by a high-efficiency particulate air filter and iodine adsorption media to remove contaminants. The neutral-pressure CRE does not intentionally pressurize the CRE, but limits inleakage of contaminants by isolating controlled flow paths into the CRE. Most plants with a positive-pressure CRE have a technical specification SR to verify that those ventilation systems serving the CRE can maintain the CRE at a positive differential pressure relative to adjacent areas. These surveillance tests (typically referred to as a ΔP surveillance) are generally implemented through a technical specification SR for the CRHSs. Plants with a neutral-pressure CRE design typically do not have a CRE integrity testing program. (The term "neutral-pressure" means only that the CRE is not intentionally pressured. The actual pressure of the CRE may be positive, neutral, or negative relative to adjacent areas.)

In addition to the ΔP surveillance described above, licensees have performed CRE integrity testing at approximately 30 percent of the power reactor facilities using the standard test method described in American Society for Testing and Materials (ASTM) consensus standard E741, "Standard Test Method for Determining Air Change in a Single Zone by Means of a Tracer Gas Dilution." Unlike the ΔP surveillance, the ASTM E741 test determines the total CRE inleakage from all sources. It is well suited for assessing the integrity of positive-pressure or neutral-pressure CREs. The test basically involves homogeneously dispersing a nontoxic tracer gas throughout the CRE and measuring the dilution of the tracer gas caused by inleakage.

The results of the ASTM E741 tests indicate that the ΔP surveillance is not a reliable method for demonstrating CRE integrity. For all but one facility tested using the ASTM E741 standard, the measured inleakage was greater than the inleakage assumed in the design basis analyses. In some cases, even though the licensees had routinely demonstrated a positive ΔP relative to adjacent areas at their facilities, the measured inleakage was several orders of magnitude greater than the value previously assumed. Affected facilities were subsequently able to achieve compliance with the control room radiation protection regulatory requirements by sealing, adding new ductwork, changing their CRE, or reanalyzing their control room habitability.

Use of the ΔP surveillance as an indicator of CRE integrity has two inherent deficiencies. First, it does not measure CRE inleakage. The ΔP surveillance infers that no contamination can enter the CRE if the CRE is at a higher pressure than adjacent areas. Second, the ΔP surveillance cannot determine whether there may be unrecognized sources of pressurization of the CRE that could introduce contaminants into the CRE under accident conditions. Two possible unrecognized contamination pathways are the CRHS fan suction ductwork that is located outside the CRE, and the pressurized ducts that traverse the lower pressure CRE en route to another plant area.

The ASTM E741 testing has helped to identify a spectrum of CRHS deficiencies that affect (1) system design, construction, and quality, (2) system boundary construction and integrity, and (3) technical specification SRs. Licensees have determined that the performance of the CRHSs can be affected by (1) the gradual degradation in associated equipment such as seals, floor drain traps, fans, ductwork, and other components, (2) the drift of throttled dampers, (3) maintenance on the CRHSs, and (4) inadvertent misalignments of the CRHSs. Since inleakage is influenced by pressure differentials between the CRE and adjacent areas, changes in ambient pressure in these adjacent areas can affect the CRE inleakage. These changes can be the result of a modification, the degradation of the ventilation systems serving these areas, or inadequate preventive and corrective maintenance programs.

Licensees and NRC staff have identified other deficiencies in CRHS design, operation, and performance from the review of license amendments, licensee event reports, and records and reports prepared pursuant to 10 CFR 50.59. These deficiencies showed that the licensees' CRHSs did not meet their design bases. Some of these deficiencies are discussed in Regulatory Issue Summary 2001-19, "Deficiencies in the Documentation of Design Basis Radiological Analyses Submitted in Conjunction with License Amendment Requests." For example, some licensees credited the operation of CRHSs based upon actuation of high-radiation signals from instrumentation. Further investigation revealed that for some licensees the system would not be actuated due to incorrect setpoints or placement of the instrumentation. Other CRHS designs appear not to have considered unfiltered or once-filtered inleakage through idle CRHS ventilation trains. Without adequate consideration of such design issues, design basis radiation exposure limits may be exceeded.

Previous to the ASTM E741 testing, a group of licensees had trouble meeting the control room criteria in Three Mile Island (TMI) Action Item III.D.3.4, "Control Room Habitability Requirements," that the NRC ordered most licensees to implement after the accident at TMI. At that time, radiological source term research suggested that the distribution of the chemical forms of iodine released during an accident could be different from the distribution in the traditional source term defined in U.S. Atomic Energy Commission Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactor Sites." Because of the possible differences, the staff allowed licensees to postpone changing their control rooms until the ongoing source term research was completed or until a generic letter on control room habitability was issued. The staff believed that postponing changes was reasonable since the source term research or improved methods of analyses might prove that the changes were unnecessary. Many of these licensees that postponed changes incorporated compensatory actions into their operating procedures to assure that the control room operators would be protected in case of an accident. Since then, some licensees have found that they could not meet the thyroid dose limits for habitability without using compensatory actions. The NRC also allowed these facilities to use compensatory actions until completion of the source term research. In August 2000, the NRC staff incorporated the results of the source term research into Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," which is now available for use by licensees.

Although many CRE integrity testing programs focus on radiological concerns, radiation is only one potential design basis challenge to the protection of the operators. The inleakage of other contaminants may have a greater impact on control room habitability. An inleakage rate that is tolerable for one contaminant may not be tolerable for another. The control room licensing basis describes the hazardous chemical releases considered in the CRE design, the design features, and the administrative controls implemented to mitigate the consequences of these releases to the control room operators. Smoke and other byproducts of fire within the CRE or in adjacent areas are among the contaminants that can have an adverse impact on control room habitability.

Discussion

Information obtained by the NRC indicates that some licensees have not maintained adequate configuration control over their CREs and have not corrected identified design and performance deficiencies. The primary design function of CRHSs is to provide a safe environment in which the operator can control the nuclear reactor and auxiliary systems during normal operations and can safely shut down these systems during abnormal situations to protect the health and safety of the public. It is important for the operators to be confident of their safety in the control room to minimize errors of omission and commission. Errors of omission and commission are more likely if CRHSs do not properly perform as intended in response to challenges from off-normal or accident situations. The control room must be safe so that operators can remain in the control room to monitor plant performance and take appropriate mitigative actions. This is an underlying assumption in both the design basis and severe accident risk analyses. It is, therefore, imperative to the health and safety of the public that operators are safe in the control room at all times.

The scope and magnitude of the problems that NRC staff and certain licensees have identified raise concerns about whether similar design, configuration, and operability problems exist at other reactor facilities. The NRC staff is particularly concerned about whether licensees' programs to maintain configuration control of CRHSs are sufficient to demonstrate that the physical and functional characteristics of CRHSs are consistent with and are being maintained according to their design bases. It is emphasized that the NRC's position has been, and continues to be, that it is the responsibility of individual licensees to know the licensing basis for the CRHSs. Licensees should also have appropriate documentation of the design basis and procedures in place, in accordance with NRC regulations, for performing necessary assessments of plant or procedure changes that may affect the performance of the CRHSs.

The technical specifications for about 75 percent of the control rooms (mostly positive-pressure CREs) have an SR to measure the ΔP from the CRE to adjacent areas. The bases of the Improved Standard Technical Specifications state that this SR demonstrates control room integrity with respect to unfiltered inleakage. The ASTM E741 integrated testing proves that it does not. Because 10 CFR 50.36 requires technical specifications to be derived from the safety analyses, the staff believes that the existing deficiency should be corrected. This correction is consistent with NRC Administrative Letter 98-10, "Dispositioning of Technical Specifications That Are Insufficient To Assure Plant Safety," which describes the staff's expectation that licensees correct technical specifications that are found to "contain non-conservative values or specify incorrect actions."

Because of the importance of ensuring habitable control rooms under all normal and off-normal plant conditions, the addressees are requested to provide certain information that will enable the NRC staff to verify whether addressees can demonstrate and maintain the current design bases for the CRHSs at their facilities. Addressees are encouraged, but not required, to work closely with industry groups on the coordination of their responses. Coordinating the responses promotes efficiency since it leads to a uniform approach to demonstrating compliance with the design bases of their CREs.

NEI 99-03, "Control Room Habitability Assessment Guidance," provides industry generic guidance on control room habitability. The NRC staff reviewed NEI 99-03, but rather than fully endorse NEI 99-03, the NRC staff developed its own guidance. Regulatory Guide 1.196 (formerly DG-1114), "Control Room Habitability at Light-Water Nuclear Power Reactors," endorses NEI 99-03 to the extent possible and provides additional guidance. Licensees are not required to comply with Regulatory Guide 1.196, but may find it useful in responding to this generic letter. Licensees that are unable to confirm item 1 under the Requested Information section may use Regulatory Guide 1.196 to develop and implement corrective actions.

<u>Requested Information</u>

Addressees are requested to provide the following information within 180 days of the date of this generic letter.

1. Provide confirmation that your facility's control room meets the applicable habitability regulatory requirements (e.g., GDC 1, 3, 4, 5, and 19) and that the CRHSs are designed, constructed, configured, operated, and maintained in accordance with the facility's design and licensing bases. Emphasis should be placed on confirming:

    (a) That the most limiting unfiltered inleakage into your CRE (and the filtered inleakage if applicable) is no more than the value assumed in your design basis radiological analyses for control room habitability. Describe how and when you performed the analyses, tests, and measurements for this confirmation.

    (b) That the most limiting unfiltered inleakage into your CRE is incorporated into your hazardous chemical assessments. This inleakage may differ from the value assumed in your design basis radiological analyses. Also, confirm that the reactor control capability is maintained from either the control room or the alternate shutdown panel in the event of smoke.

(c) That your technical specifications verify the integrity of the CRE, and the assumed inleakage rates of potentially contaminated air. If you currently have a ΔP surveillance requirement to demonstrate CRE integrity, provide the basis for your conclusion that it remains adequate to demonstrate CRE integrity in light of the ASTM E741 testing results. If you conclude that your ΔP surveillance requirement is no longer adequate, provide a schedule for: 1) revising the surveillance requirement in your technical specification to reference an acceptable surveillance methodology (e.g., ASTM E741), and 2) making any necessary modifications to your CRE so that compliance with your new surveillance requirement can be demonstrated.

If your facility does not currently have a technical specification surveillance requirement for your CRE integrity, explain how and at what frequency you confirm your CRE integrity and why this is adequate to demonstrate CRE integrity.

2. If you currently use compensatory measures to demonstrate control room habitability, describe the compensatory measures at your facility and the corrective actions needed to retire these compensatory measures.

3. If you believe that your facility is not required to meet either the GDC, the draft GDC, or the "Principal Design Criteria" regarding control room habitability, in addition to responding to 1 and 2 above, provide documentation (e.g., Preliminary Safety Analysis Report, Final Safety Analysis Report sections, or correspondence) of the basis for this conclusion and identify your actual requirements.

Requested Response

If an addressee cannot provide the information or cannot meet the requested completion date, the addressee should submit a written response indicating this within 60 days of the date of this generic letter. The response should address any alternative course of action the addressee proposes to take, including the basis for the acceptability of the proposed alternative course of action and the schedule for completing the alternative course of action.

The written response should be addressed to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001. A copy of the response should be sent to the appropriate regional administrator.

NRC staff will review the responses to this generic letter and, if concerns are identified, will notify affected addressees. The staff may conduct inspections to determine licensees' effectiveness in addressing this generic letter.

Applicable Regulatory Requirements

Several provisions of the NRC regulations and plant operating licenses (technical specifications) pertain to the issue of control room habitability. The general design criteria for nuclear power plants (10 CFR Part 50, Appendix A), or, as appropriate, the quality assurance requirements in the licensing basis for a reactor facility (stated in 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants"), and the technical specifications, are the bases for the NRC staff's assessment of control room habitability.

Appendix A to 10 CFR Part 50 and the plant safety analyses require or commit licensees to design and test safety-related structures, systems, and components (SSCs) to provide adequate assurance that they can perform their safety functions. The NRC staff applies these criteria to plants with construction permits issued on or after May 21, 1971, and to those plants whose licensees have committed to them. The applicable GDC are GDC 1, 3, 4, 5, and 19. GDC 1 requires quality standards commensurate with the importance of the safety functions performed. GDC 3 requires SSCs to be designed and located to minimize the effects of fires. GDC 4 requires SSCs to be designed to accommodate the effects of accidents. GDC 5 requires that an accident in one unit will not significantly impair orderly shutdown and cooldown of the remaining unit.

GDC 19 specifies that a control room be provided from which actions can be taken to operate the nuclear reactor safely under normal conditions and maintain the reactor in a safe condition under accident conditions, including a loss-of-coolant accident. There must be adequate radiation protection to permit personnel to access and occupy the control room under accident conditions without receiving radiation exposures in excess of specified values.

Before the issuance of the GDC, proposed GDC (sometimes called "principal design criteria") were published in the *Federal Register* for comment. As they evolved, several of the proposed GDC addressed control room habitability. A facility may have been licensed before the issuance of the GDC, but the licensee may have committed to the proposed GDC as they existed at the time of licensing.

Following the accident at TMI, TMI Action Plan Item III.D.3.4, "Control Room Habitability Requirements," as clarified in NUREG-0737, "Clarification of TMI Action Plan Requirements," required all licensees to assure that control room operators would be adequately protected against the effects of accidental releases of toxic and radioactive gases and that the nuclear power plant could be safely operated or shut down under design basis accident conditions. When licensees proposed modifications, the NRC issued orders confirming the licensees' commitments. As a result, most plants licensed before the GDC were formally adopted were then subsequently required to meet the TMI Action Plan III.D.3.4 requirements.

Appendix B to 10 CFR Part 50 establishes quality assurance requirements for the design, construction, and operation of those SSCs that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. Appendix B, Criterion III, "Design Control," requires that design control measures be provided for verifying or checking the adequacy of design. A suitable testing program is identified as one method of accomplishing this verification. Appendix B, Criterion XVI, "Corrective Action," requires measures to be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, defective material and equipment, and nonconformances are promptly identified and corrected.

The regulations in 10 CFR 50.36, "Technical Specifications," require plant technical specifications to be derived from the safety analyses.

If, in the course of preparing a response to the requested information, an addressee determines that its facility is not in compliance with the Commission's requirements, the addressee is expected to take appropriate action in accordance with requirements of Appendix B to 10 CFR Part 50 and the plant technical specifications to restore the facility to compliance.

Reasons for Information Request

This generic letter transmits an information request that is necessary to permit the assessment of plant-specific compliance with applicable regulatory requirements. Specifically, this information will enable the NRC staff to determine whether the control rooms at power reactor facilities comply with the current licensing bases and whether additional regulatory actions are required.

The habitability of the control room and the operability of the CRHSs in the event of adverse environmental conditions external to the CRE have a direct link to maintaining public health and safety. Plant design bases and severe accident risk analyses both assume that the control room operators can remain safely within the control room to monitor plant performance and take appropriate mitigative actions. It is essential that operators be confident of their safety within the control room at all times.

Backfit Discussion

This generic letter transmits an information request for the purpose of verifying compliance with existing applicable regulatory requirements (see the Applicable Regulatory Requirements section of this generic letter). This generic letter does not constitute a backfit as defined in 10 CFR 50.109(a)(1) since it does not impose modifications or additions to structures, systems, and components or to the design or operation of an addressee's facility. Nor does it impose an interpretation of the Commission's rules that is either new or different from a previous staff position. Therefore, no backfit is either intended or approved by this generic letter, and the staff has not performed a backfit analysis.

Small Business Regulatory Enforcement Fairness Act

The NRC has determined that this action is not subject to the Small Business Regulatory Enforcement Fairness Act of 1996.

Federal Register Notification

A notice of opportunity for public comment was published in the *Federal Register* on May 9, 2002 (67 FR 31385). Comments were received from three licensees, three industry organizations, and one individual. The staff considered all comments that were received. The staff evaluation of these comments is accessible electronically from the Agencywide Documents Access and Management System (ADAMS) at ML030780493.

Paperwork Reduction Act Statement

This generic letter contains information collections that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.) These information collections were approved by the Office of Management and Budget (OMB), approval number 3150-0011, which expires January 31, 2004.

The burden to the public for these information collections is estimated to average 200 hours per response, including the time for reviewing instructions, searching existing data sources, gathering and maintaining the data needed, and completing and reviewing the information collection. Send comments regarding this burden estimate or on any other aspect of these information collections, including suggestions for reducing the burden, to the Records Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by Internet electronic mail to INFOCOLLECTS@NRC.GOV; and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0011), Office of Management and Budget, Washington, DC 20503.

Public Protection Notification

The NRC may not conduct or sponsor, and a person is not required to respond to, an information collection unless the requesting document displays a currently valid OMB control number.

/RA/
David B. Matthews, Director
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Technical Contact: Mark Blumberg, NRR
301-415-1083
E-mail: wmb1@nrc.gov

Lead Project Manager: Michael Webb
301-415-1347
E-mal: mkw@nrc.gov

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David B. Matthews, Director
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Technical Contact: Mark Blumberg, NRR
301-415-1083
E-mail: wmb1@nrc.gov

Lead Project Manager: Michael Webb
301-415-1347
E-mal: mkw@nrc.gov

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