

September 30, 2002

Dr. Jill Lipoti, Assistant Director
Radiation Protection Programs
NJ Department of Environmental
Protection and Energy
CN 415
Trenton, NJ 08625-0415

Dear Dr. Lipoti:

The purpose of my letter is to notify you that the U.S. Nuclear Regulatory Commission (NRC) staff intends to issue amendments to Facility Operating License Nos. DPR-70 and DPR-75 for the Salem Nuclear Generating Station, Unit Nos. 1 and 2 (Salem) that will allow PSEG Nuclear LLC (PSEG) to remove its Post Accident Sampling System (PASS) Program from the Salem Technical Specifications (TSs). I understand that this action is being taken despite the New Jersey Department of Environmental Protection's (NJ-DEP's) strong objections. First and foremost, I wish to convey that the NRC staff has taken your concerns seriously. However, in performing an entire review of our efforts, the staff believes there is a sound technical basis in granting PSEG's request. I am enclosing a summary of what we understand to be your concerns, and the staff's rationale for approving the TS change.

I understand that our stakeholders may not always agree with the decisions the NRC makes in conjunction with its duties. However, I am always willing to consider the concerns and insights expressed by interested parties. If you desire, we could have further discussions related to the regulation of nuclear power plants located in New Jersey. My telephone number is 301-415-1453.

Sincerely,

/RA/

John A. Zwolinski, Director
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Enclosure: As stated

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* See previous concurrence

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**RESPONSE TO COMMENTS MADE BY
NEW JERSEY DEPARTMENT OF ENVIRONMENTAL PROTECTION
CONCERNING THE ELIMINATION OF THE POST-ACCIDENT SAMPLING SYSTEM (PASS)
FROM TECHNICAL SPECIFICATIONS**

Comment:

- Offsite officials need to have a mechanism to know not only inferred source term, but also actual source term over time, and the stability and potential hazards that remain in the coolant.
- Radionuclide inventory and source term data would enhance the confidence of public health officials in their evaluation of subsequent protective actions in the ingestion phase. This is especially important in light of the feedback NJ-DEP received when it began providing potassium iodide (KI) tablets to members of the public.
- Offsite officials need to know the actual volume of radioactive material inside of containment (not just the inferred source term) to make additional protective action recommendations.

NRC Response:

The NRC staff understands that all organizations involved in responding to an accident at a nuclear power plant need to have the best information available in order to make the right decisions affecting public health and safety. The staff further respects NJ-DEP's responsibility to provide appropriate protective action recommendations (PARs) to the governor and other public officials. The staff recognizes that the ultimate decision to order the evacuation of citizens surrounding the plant or to recommend the prophylactic use of KI tablets must be based on timely and accurate information, and that the public needs to have confidence in the decisions made by public officials.

Core Damage Assessment Methodology

On November 22, 1996, the Westinghouse Owners' Group (WOG) submitted Topical Report WCAP-14696, "Westinghouse Owners Group Core Damage Assessment Guidance," to the NRC for review. In the topical report, a revised methodology was described that could be used by a licensee's emergency response organization staff for estimating the extent of core damage that may have occurred during an accident at a Westinghouse nuclear power plant. The new methodology was a revised calculational technique for estimating core damage and relies on real-time plant indications rather than samples of plant fluids. The revised post-accident core damage assessment (CDA) methodology in WCAP-14696 replaced the methodology approved by the staff in 1984.

The 1984 methodology was changed for two major reasons: (1) the 1984 methodology relied on radionuclide samples which did not effectively support emergency response decision-making due to the significant time delay in obtaining and analyzing these samples using the post-accident sampling system (PASS); and (2) the methodology did not reflect the latest

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understanding of fission product behavior, particularly the sequence-specific nature of fission product retention and hydrogen holdup in the reactor coolant system (RCS), and fission product deposition in the containment and sample lines.

Accuracy of PASS Sample Results

The NRC staff considers the difficulty of obtaining accurate and representative samples to be a major shortcoming of the PASS system. The deposition of iodine in PASS sample lines is particularly problematic since iodine is the best indicator of core damage and its potential significance of health consequences from a release of the containment atmosphere. The amount of deposition of iodine in the sample lines is also dependent on its chemical form. At the time that the PASS criteria were developed in NUREG-0737, "Clarification of TMI Action Plan Requirements," the majority of iodine in the containment atmosphere from a potential severe accident was believed to be in elemental form. Since that time, severe accident research has shown that the chemical form of iodine is expected to primarily be Cesium Iodide (CsI) in the form of an aerosol. The collection of accurate samples of CsI aerosols poses significant problems because there will be a tendency for the particles to deposit on the cooler walls of sampling lines due to thermophoresis and Stefan flow, if steam is present. These mechanisms will be present at all times during sampling operation, and it is not expected that an equilibrium state between deposition and removal of the CsI aerosols will ever be reached.

Timeliness of PASS Sample Results

Although radionuclide sampling information may be useful in estimating the degree of core damage, the time needed to obtain and analyze a PASS sample may unnecessarily delay decisions made by the emergency response organization. Decisions to order affected members of the public to be evacuated, or to recommend the prophylactic use of potassium iodide (KI) tablets, need to be made as early as possible in order to be most effective. NUREG-0737 specifies that the PASS has the capability to promptly (i.e., within 3 hours) quantify certain radionuclides that are indicators of the degree of core damage. However, the staff has concluded that the revised CDA guidance, described in WCAP-14696, provides the capability to assess the degree of core damage with a sufficient level of accuracy to support more timely emergency response decisions. The revised guidelines represented an improvement over the previous methodology that relies on PASS sampling. It is simpler, more timely, and accounts for improved understanding of fission product behavior inside containment. By making core damage information available earlier in an event, such that it can be used to refine dose assessments and confirm or extend initial protective action recommendations, implementation of the revised CDA guidance will increase the effectiveness of the emergency response organization's decisions. Because timeliness is critical in decisions of whether or not to evacuate the public, the NRC staff considers it more appropriate for emergency response purposes to estimate the degree of core damage based upon real-time indications. The staff believes that this will enhance public confidence in the decisions ultimately made by public officials.

Comment:

- NJ-DEP expects, that following an accident, the radionuclide mix will be characterized within a “reasonable length of time” (within 24 hours) by whatever means available.
- Can the NRC approve the Salem PASS amendment contingent upon the licensee providing the information within 24 hours after the accident?

NRC Response:

The NRC staff expects licensees to provide timely and accurate information on core damage estimates and dose assessments throughout the entire accident mitigation and recovery phases. To support timely and accurate dose assessments, the staff concluded that the methodology provided by WCAP-14696 allows licensees to characterize the radiological mix within a reasonable time and with sufficient accuracy. The staff further concluded that licensees should retain the capability to obtain samples from the reactor coolant, containment sump and containment atmosphere when deemed necessary.

As described in Supplement 3 to NUREG-0654, “Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants,” the staff maintains that initial protective action recommendations (PARs) should be based upon plant indications of actual or projected core damage. Following this initial PAR, the licensee should continue to assess the accident to determine whether the PAR should be modified. Relaxation of the PAR should not occur until the source of the threat is clearly under control. In NUREG-0654, the NRC indicated that licensees’ capability to perform this assessment should include the post-accident sampling capability.

Due to limitations associated with obtaining representative radionuclide samples, the NRC staff does not expect that information obtained from PASS samples would be a primary factor in licensee and offsite emergency response decisions regarding PARs during the early phases of an accident. However, the staff does believe that containment atmosphere sample information would provide the public additional confidence that the licensee understood the magnitude of any remaining threat that the accident may pose after plant conditions in the accident have stabilized. Therefore, the staff also concluded that licensees should be required to develop plans for radionuclide sampling; although the staff does not consider it necessary to have dedicated equipment to obtain this sample in a prompt manner. These contingency plans should detail the plant's existing sampling capabilities and what actions (e.g., assembling temporary shielding) may be necessary to obtain and analyze highly radioactive samples.

The NRC staff notes that contingency plans developed for Salem by PSEG do not preclude the ability to obtain radionuclide samples, as radiological conditions permit, at any time during the accident mitigation and recovery phases. If and when it is determined that samples are required, the staff believes that sufficient plans would be in place to obtain these samples at Salem. Therefore, the staff does not consider approval of the licensee’s amendment request contingent upon its ability to obtain samples within 24 hours following an accident to be necessary.

Comment:

- Radiochemical analysis of the coolant, containment sump and containment atmosphere is the most accurate method for performing CDA.
- A limitation of performing CDA based upon containment radiation monitor indication is that it is based upon the radiation monitor response to an assumed mixture of radionuclides. Since the nuclide mix varies greatly from one accident scenario to the next, the actual monitor response may vary by orders of magnitude.

NRC Response:

The staff recognizes that there are limitations with the individual indications used for CDA which is why current guidance relies on a number of instrument indications to diagnose and evaluate core damage. The staff agrees that radiochemical analysis can be more accurate than other available indications but it too has limitations. At the time of PASS design, the iodine chemical form was assumed to be predominantly in elemental gaseous form (91 percent). The staff's current understanding, documented in NUREG/CR-5732, indicates that iodine entering the containment is at least 95 percent particulate CsI. Once the iodine enters containment, however, additional reactions are likely to occur. In an aqueous environment, as expected for Light Water Reactors, iodine is expected to dissolve in water pools or plateout on wet surfaces. This can bias the radionuclide samples obtained from PASS and lead to underestimates of the extent of core damage.

The staff agrees that the nuclide mix varies greatly from one accident scenario to the next which affects radiation monitor response. Revised CDA guidance relies on core exit thermocouples (CETs), RCS pressure and containment spray system status to sufficiently narrow the accident scenario being assessed and the expected variation in the nuclide mix.

The approach for converting instrument readings into core damage estimates is consistent with the current understanding of clad and fuel damage characteristics, and accounts for fission product and hydrogen retention/holdup in an approximate fashion. Specifically, containment radiation monitor readings are compared to plant-specific radiation levels for 100 percent clad damage or fuel over-temperature damage, CET readings are compared to values typically associated with clad damage and fuel over-temperature damage, and containment hydrogen concentration is compared to the amount expected in containment for 100 percent over-temperature damage. CET readings that exceed the setpoints or the operating limits of the thermocouples are interpreted as core damage in that region of the core. The core damage estimates derived separately from different indicators (containment radiation, CET, and containment hydrogen concentration readings) are compared and reconciled, thereby improving the confidence in the core damage estimate.

The staff has concluded that the revised CDA guidance, that does not rely on PASS, provides the capability to assess the degree of core damage with a sufficient level of accuracy and timeliness to support emergency response decision-making. The revised guideline also represents an improvement over the existing methodology which relied on PASS sampling.

Comment:

- Field team measurements have inaccuracies associated with atmospheric transport, field team measurements may not be timely, and there is a large uncertainty associated with source term estimate based upon in-plant instrumentation.

NRC Response:

NRC guidance (NUREG-0654, Supplement 3) specifies that initial protective action recommendations should be based upon plant conditions which indicate that there is actual or projected severe core damage. This initial PAR is followed by dose assessments which may be used to expand the area covered by the initial PAR. Initial dose assessments will likely be based upon an assumed source term. This source term may be refined based upon plant indications or core damage assessments. This source term can be further refined based upon offsite field team measurements. A benefit of using field team measurements is that the source term being estimated is that released from containment rather than the source term in containment which could be altered prior to being released from containment. PASS results are another potential input to refinements to the source term. However, there are concerns with the accuracy of source term estimates based upon PASS because of the potential for the sample not to accurately represent the source term in containment and with the time needed to obtain and analyze these samples.

The NRC believes that PASS results will not have an important role in source term refinements for use in dose assessments because indications such as CET and containment radiation monitors (in conjunction with correlations of these indications to CDAs) will be more timely for refining source term estimates and indications such as field team measurements will be more accurate in refining the source term estimates.

The NRC considers that PASS may be useful in making subsequent protective action recommendations (or confirming the initial PAR) after the initial protective action recommendation has been made. However, the NRC considers that there is adequate information on the actual (or potential) consequences of a release of radioactive material from field team measurements and containment atmosphere radiation monitors to support assessment of protective action recommendations.

Comment:

- The revised CDA methodology relies primarily on plant instrumentation. Yet, we now understand that the NRC staff is contemplating changes that will allow licensees to remove hydrogen and radiation monitoring instrumentation from their plants. By allowing licensees to remove or downgrade the instrumentation the NRC says is necessary for CDA represents divergent viewpoints on that of the staff.
- The Radiation Monitoring System (RMS) at Salem has had a history of unreliable operation, as far back as 1991, and we believe that this should be factored in your review of PSEG's license amendment request.
- Why isn't the NRC taking more of an integrated approach in its decisions affecting issues such as PASS elimination, potential changes to hydrogen monitoring requirements, and the documented problems with the RMS at Salem?

NRC Response:

As described in the proposed rulemaking related to combustible gas control (see rule forum on the NRC web page), the NRC has proposed to maintain the existing requirement in 10 CFR 50.44(b)(1) for monitoring hydrogen in the containment atmosphere for all plant designs. The hydrogen monitors are required to assess the degree of core damage during a beyond design-basis accident and confirm whether random or deliberate ignition has taken place. Hydrogen monitors are also used, in conjunction with oxygen monitors in inerted containments, to guide response to emergency operating procedures. Hydrogen monitors are also used in emergency operating procedures of BWR Mark III facilities. If an explosive mixture that could threaten containment integrity exists, then other severe accident management strategies, such as purging and/or venting of containment, would need to be considered. The hydrogen monitors are needed to implement these severe accident management strategies.

The Commission proposes to reclassify the hydrogen monitors as not safety-related components. With the proposed elimination of the design-basis LOCA hydrogen release, the hydrogen monitors are no longer required to mitigate design-basis accidents and, therefore, the hydrogen monitors do not meet the definition of a safety-related component as defined in 10 CFR 50.2. The proposal to maintain but reclassify the hydrogen monitors was developed with full recognition of the activities related to the elimination of regulatory requirements for PASS. The staff's finding that the proposed rulemaking does not negate the previous conclusions on the elimination of PASS is explicitly stated in the supporting documentation for the proposed rulemaking.

Currently, Regulatory Guide (RG) 1.97 recommends classifying the hydrogen monitors in Category 1, defined as applying to instrumentation designed for monitoring key variables that most directly indicate the accomplishment of a safety function for design-basis accident events. The hydrogen monitors no longer meet the definition of Category 1 in RG 1.97 and, therefore, the Commission believes that licensee's current commitments are unnecessarily burdensome. The Commission believes that Category 3, as defined in RG 1.97, is an appropriate categorization for the hydrogen monitors because the monitors are required to diagnose the course of beyond design-basis accidents. Category 3 applies to high-quality, off-the-shelf backup and diagnostic instrumentation. As with the revision to oxygen monitoring, this

proposed relaxation may require a license amendment. Licensees would also need to update their final safety analysis reports to reflect the new classification and RG 1.97 categorization of the monitors in accordance with 10 CFR 50.71(e).

Regarding the condition of the RMS at Salem, the NRC believes that it has established the appropriate regulatory controls for the various monitors given their use in controlling effluents or providing information during an emergency. The various monitors come under the control of requirements defined in Technical Specifications, the maintenance rule, or other NRC-defined programs. These programs in turn establish, directly or indirectly, requirements on the availability and reliability of the specific monitors and actions to be taken when monitors are inoperable. The most recent report documenting an inspection of Salem's RMS is Inspection Report 50-272/2001-11 dated January 25, 2002. The inspection was performed in conjunction with an evaluation the effectiveness of PSEG's radioactive gaseous and liquid effluent control programs. The inspector reviewed a number of condition reports (notifications), recent channel calibration and functional test results, and performed system walkdowns to determine the availability and material condition of radioactive liquid/gaseous effluent RMS. The inspection resulted in no significant findings. The NRC staff recognizes that, because of its age, many components of the RMS may be a challenge to maintain; however, the staff believes that the condition of the system has not reached the point where it would not be able to support CDAs following an accident.

Therefore, the staff believes that it has taken an integrated approach in efforts associated with its regulatory responsibilities.

SUMMARY

The NRC staff believes there is a sound technical basis in granting PSEG's request:

- The revised post-accident core damage assessment methodology in WCAP-14696 allows PSEG to promptly, and with sufficient accuracy, make appropriate protective action recommendations. The staff believes this enhances public confidence.
- The revised methodology reflects the latest knowledge of core damage progression.
- The NRC believes that PASS sample results will not have an important role in refining source term and dose assessments.
- PSEG retains the ability to take samples as radiological conditions permit.
- Hydrogen monitoring will still be required; however, the quality level of hydrogen monitoring equipment will be commensurate with the reclassification of its safety function.
- The NRC has established programs and requirements to oversee the availability and reliability of the specific monitors used to support core damage assessments.