03-4313

UNITED STATES COURT OF APPEALS for the SECOND CIRCUIT

---X

Riverkeeper, Inc.,

Petitioner,

- against -

SAMUEL J. COLLINS, Director, Office of Nuclear Reactor Regulation; DR. WILLIAM TRAVERS, Executive Director for Operations of the Nuclear REGULATORY COMMISSION; the UNITED STATES OF AMERICA; ENTERGY NUCLEAR INDIAN POINT 2 LLC; ENTERGY NUCLEAR INDIAN POINT 3, LLC; and ENTERGY NUCLEAR OPERATIONS, INC.

Respondents.

---X

On Petition for Review of a Decision of the Nuclear Regulatory Commission

Joint Appendix: Volume 2 of 3

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April 10, 2001

EA No. 01-055

Mr. John Groth
Senior Vice President - Nuclear Operations
Consolidated Edison Company of
New York, Inc.
Indian Point 2 Station
Broadway and Bleakley Avenue
Buchanan, NY 10511

SUBJECT: INDIAN POINT UNIT 2 - NRC SUPPLEMENTAL INSPECTION 05000247/2001-002

Dear Mr. Groth:

The Nuclear Regulatory Commission conducted a supplemental inspection from January 16th through February 9th, 2001, at your Indian Point 2 (IP2) facility. This inspection was conducted in accordance with the guidance contained in NRC Manual Chapter 0305 and inspection procedure 95003 and was performed in response to your facility's designation as having multiple degraded cornerstones, as defined by the NRC's reactor oversight process.

The results of our inspection indicate that your facility is being operated safely. However, the team identified problems similar to those that have been previously identified at the IP2 facility, particularly in the areas of design control, human and equipment performance, problem identification and resolution, and emergency preparedness. Senior management has raised performance expectations, increased accountability and emphasis on training, and taken steps to establish improvement programs that are aligned with the station's business planning process. While some performance improvements were noted, as a result, progress has been slow overall and limited in some areas, indicating the need for you to maintain, and in some areas consider accelerating, the ongoing performance improvement program which has been in place. One such area is that of design control where recurrent problems were found in the translation of important design assumptions into plant operating procedures, drawings, calculations, and testing programs.

The inspection team assessed its findings together with the results of similar, previous inspections in order to provide insight into the overall root and contributing causes of performance issues at the site. The NRC's effort at summarizing potential causes is not intended to be a substitute for a more focused root cause study or self-assessment on your part. We found that most performance issues could be attributed to one or more of the following:

000486

- Weaknesses in the ability to retrieve, verify, and assure the quality of engineering products, particularly design basis information.

- Inconsistent reinforcement of existing management standards with respect to staff performance, particularly in the areas of procedural quality and adherence and in implementation of the corrective action program;

- A tendency, in some instances, for the plant staff to accept degraded conditions;

- Some limitations in the application of resources leading to, for example, staffing issues and training weaknesses.

We observed that your current performance improvement plan, developed within the framework of your business plan, appears to envelope the areas needing improvement. The team determined that an alignment existed between the business plan and actions necessary to address performance issues. However, the plan is general in nature and relies heavily on department level implementation strategies that vary in quality and depth. We note previous improvement plans similarly covered the issues broadly, but were not fully effective. In that regard, you are requested to respond to this inspection report by May 7, 2001, highlighting both changes made to your business plan, based on the issues raised during this inspection, and measures you will use to monitor the effectiveness of your performance improvement efforts.

We will continue heightened oversight of Indian Point 2 until we gain confidence that your performance improvement program has substantially addressed the performance weaknesses identified in this and previous NRC inspections. This will include inspection of targeted areas of weakness, periodic site visits and public management meetings, and quarterly assessments by senior regional management. A more detailed oversight plan will be published in late May 2001, following receipt and assessment of your response.

We are planning two public meetings to discuss your performance improvement efforts. The first meeting, tentatively scheduled for April 30, 2001, will cover your response to this inspection focusing principally upon design control activities to provide confidence that appropriate actions are being taken and planned in this important area. Secondly, we are finalizing plans for an annual review meeting (as prescribed in the Agency Action Matrix), which will occur in the local area of the plant in June 2001; this will provide opportunity for broader discussion on your improvement program.

The details of our inspection findings are provided in the enclosed inspection report and were discussed with you and members of your staff throughout the inspection and at a public meeting held on March 2nd, 2001. The issues identified in the enclosed inspection report have, individually, been evaluated under the risk significance determination process as being minor in nature or having very low safety significance (Green). However, the issues provide evidence of some program and process weaknesses similar to those which contributed to previous plant events. We have determined that violations of regulatory requirements are associated with several of these issues. These violations are being treated as Non-Cited Violations, consistent with Section VI.A.1 of the NRC's Enforcement Policy. If you deny the non-cited violations, you should provide a response with the basis of your denial, within 30 days of the date of this inspection report to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region I, 475 Allendale Road, King of Prussia, PA 19406-1415; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Indian Point 2 facility.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room). Should you have any questions regarding this report, please contact Mr. Brian Holian at 610-337-5128.

Sincerely,

/RA/

Hubert J. Miller
Regional Administrator

Docket No. 05000247
License No. DPR-26

Enclosure: Inspection Report 05000247/2001-002

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.:	05000247
License No.:	DPR-26
Report No.:	2001-002
Licensee:	Consolidation Edison Company of New York, Inc.
Facility:	Indian Point 2 Nuclear Power Plant
Location:	Buchanan, New York 10511
Dates:	January 16, 2001 to February 9, 2001
Inspectors:	J. Shackelford, Team Leader, Region IV L. Scholl, Senior Reactor Inspector, Region I W. Schmidt, Senior Reactor Inspector, Region I J. Yerokun, Senior Reactor Inspector, Region I R. Gibbs, Senior Resident Inspector, Region II R. Pelton, Human Performance Specialist, NRR G. Morris, Reactor Inspector, Region I S. Pindale, Reactor Inspector, Region I G. Cranston, Reactor Inspector, Region I N. McNamara, Reactor Inspector, Region I G. Wertz, Resident Inspector, Region I G. McCoy, Resident Inspector, Region II D. Prevatte, Contractor O. Mazzoni, Contractor J. Kottan, Radiological Safety Program Manager, Region I (3rd on-site week) L. Peluso, Health Physicist, Region I (3rd on-site week)
Approved By:	Brian Holian, Deputy Director Division of Reactor Safety

Executive Summary

The NRC designated Indian Point 2 (IP2), owned and operated by Consolidated Edison Company of New York, Inc. (the licensee), a "multiple degraded cornerstone" facility in October 2000. As a result, a supplemental inspection was performed in accordance with the guidance in NRC manual chapter 0305 and inspection procedure 95003. A multi-disciplinary team of 14 NRC inspectors conducted the inspection over the course of approximately two months, with a total of three weeks of onsite effort. This report contains the results of that inspection. The objectives of the inspection included the following:

1) To provide the NRC additional information to be used in deciding whether the continued operation of the facility is acceptable and whether additional regulatory actions are necessary to arrest declining plant performance;
2) To provide an independent assessment of the extent of risk significant issues to aid in the determination of whether an unacceptable margin of safety exists;
3) To independently assess the adequacy of the programs and processes used by the licensee to identify, evaluate, and correct performance issues;
4) To independently evaluate the adequacy of programs and processes in the affected strategic performance areas; and,
5) To provide insight into the overall root and contributing causes of identified performance deficiencies.

The results of this inspection indicated that the licensee was operating IP2 safely, with an acceptable margin of safety, and that continued operation was acceptable. However, the team identified problems similar to those that have been previously identified at the IP2 facility, particularly in the areas of design control, human and equipment performance, problem identification and resolution, and emergency preparedness. In general, some progress has been observed in improving previously identified performance problems at the facility; however, progress has been slow overall, and limited in some areas. The team identified a number of performance weaknesses in programs and processes at the facility which indicate the need to maintain, and in some areas consider accelerating, the ongoing performance improvement program which has been in place.

The team determined that the overall program for problem identification and resolution was adequate. It was noted that some improvements had been made, in particular, an improved emphasis on problem identification and a metrics and tracking system for corrective action program issues. However, the team identified several continuing challenges to the program. It was observed that the effectiveness of some of the corrective actions for previously identified deficiencies was mixed. Additionally, the overall timeliness of corrective actions continued to be a significant challenge, and longstanding issues persisted with respect to prioritizing issues for resolution and trending causal factors. Additionally, the corrective action backlog presents an ongoing challenge to the station. Finally, as noted in previous assessments, weaknesses continue to exist in the operating experience review program, although some improvements have been made in this area. While performance difficulties continue to exist with respect to the review and disposition of technical issues, the site has made progress in areas related to industry outreach and bench-marking efforts.

Executive Summary

In the assessment of the reactor safety strategic performance area, the team selected the service water system and the 480 Vac system (including the emergency diesel generators) for in-depth reviews. These systems were selected primarily based on their overall importance to plant risk (the service water system is an important cooling water system and the 480 Vac/emergency diesel generators provide the emergency power source for the facility). Additionally, neither of these systems had received recent in-depth reviews from either the NRC or the licensee. With respect to these systems, the inspection focused heavily on the important design aspects, the quality of procedures, configuration control, and equipment performance. Additionally, the team reviewed the licensee's programs and processes associated with human performance and emergency preparedness.

It was determined that the licensee's overall performance was acceptable in the reactor safety strategic performance area. However, the team identified a number of issues in the areas of design control, equipment and human performance, and emergency preparedness which indicated weaknesses in these areas as well as the need for continued improvement.

Specifically, in the design control area, a number of performance issues were identified with respect to weaknesses in translating important design assumptions into plant operating procedures, drawings, calculations, and testing programs, including acceptance criteria. In some cases these deficiencies called into question the operability of the affected equipment. However, subsequent analyses demonstrated that the equipment would have been able to perform its safety function. The team also determined that difficulties existed in retrieving the design basis information necessary to support design control, system testing, and plant modification efforts. This particular issue had been previously identified, during NRC inspections as well as by the licensee in self-assessment efforts, and slow progress has been made to improve in this area. Additionally, this deficiency appears to have had additional impact in that some inconsistencies in the review of certain technical issues by the plant staff were observed.

In the area of equipment performance, the team determined that the reliability, material condition, and overall performance was acceptable for the reviewed systems. However, a number of other equipment issues presented challenges to both the plant and the operators. For example, emergent equipment failures in secondary plant systems continue to challenge the plant operators and have required numerous plant power changes. Examples included the feedwater pump oscillations during the recent plant startup, the heater drain pump flow element leak, and the feedwater system leak. In addition, the team noted that there had been some history of failures associated with the service water system strainers and boundary valves. The team also noted that a decrease in reliability and a concurrent increase in unavailability of the gas turbine generators occurred in the final quarter of 2000. This appears to be partly attributable to a decrease in the emphasis on maintenance for this equipment. Finally, the team concluded that the station work backlog continued to pose a significant challenge to the plant. It was also determined that due to oversights, a number of important work items had not been accurately captured in the accounting for the backlog, indicating that it may have been even somewhat larger than stated. Examples of this included the procedure changes required by the "communications to staff" program and the issues associated with verifying the comprehensiveness of the testing of various instrumentation and control components.

In the area of human performance, the team noted an increased emphasis on overall improvement and a recognition of the need for an improved training program. However, a number of program and process issues were identified. In particular, a challenge existed with respect to the number of licensed operators. This issue presented difficulties with respect to overall scheduling as well as overtime considerations. During the course of the inspection, the team witnessed a number of both planned and unplanned deviations from the overtime policy. However, the team also noted that licensee management recognized this problem and took steps to increase the number of licensed operators at the site.

The team also observed that operator performance issues have contributed to previous events and that some performance problems continue to occur. Performance errors were observed in the August 1999 reactor trip, the February 2000 steam generator tube failure, and again recently in the January 2001 turbine trip. Additionally, inconsistencies continued to exist with respect to procedural quality and adherence. Examples were also observed whereby the control room staff was unnecessarily challenged with maintenance planning efforts (in the control room) rather than having these same planning activities conducted by the work control organization outside the control room. However, the team did observe that overall crew performance was acceptable, and in particular, crew communications were good, indicating that some improvements had been made in this area.

In the area of emergency preparedness, the team determined that the overall program was adequate and provided reasonable assurance that the emergency response organization could respond effectively to an emergency. Additionally, while issues were identified that indicated the need for continued improvement, improvements were noted in a number of areas where performance issues had been previously identified. Notwithstanding, the team observed that the remediation for some of the previously identified performance issues in the technical support center, emergency operations facility, and joint news center had not been fully effective. Examples included weaknesses in technical support center assessment activities and communication, and information dissemination and coordination activities in the emergency operations facility and the joint news center. The team acknowledged that while some corrective actions had been taken in these areas, the training program had not been fully effective in preventing the recurrence of these issues. The team also found minor examples of performance issues associated with implementation of the emergency plan and the associated implementing procedures.

The team integrated these supplemental inspection findings and the results of previous similar efforts to develop the overall root and contributing causes to performance issues at the site. However, this effort was not intended to be a substitute for a more focused root cause study or self-assessment on your part.

The team determined that weaknesses existed with:

- The ability to retrieve, verify, and assure the quality of engineering products, particularly design basis information. These weaknesses contributed to problems in developing and validating calculations, testing methodologies, and acceptance criteria.

000496

Executive Summary

- An inconsistent reinforcement of existing management standards with respect to staff performance, particularly in the areas of procedural quality and adherence and in implementation of the corrective action programs. The team concluded that although adequate standards existed, inconsistent application of these standards appeared to cause performance issues to continue in those areas.

- A tendency, in some instances, for the plant staff to accept degraded conditions. This was true for both equipment issues and the quality of technical information. However, the team concluded that improvement has been made in this area.

- Some limitations in the application of resources which led to, for example, staffing issues and training weaknesses.

The team noted that station management identified similar root causes. Further, the team determined that, while a number of program and process issues existed at Indian Point 2 (some of a longstanding nature), some improvements have been made. While progress has been somewhat slow overall and limited in some areas, the business plan appeared to envelope the major performance issues which have been identified, and if executed properly, should result in continued station performance improvements. Previous site improvement plans had shown similar promise, but were not fully effective in improving overall plant performance. The NRC will continue heightened oversight of Indian Point 2 until we gain confidence that the performance improvement program has substantially addressed the performance weaknesses identified in this and previous NRC inspections.

000497

SUMMARY OF FINDINGS

IR 05000247-01-02, on 01/16 - 02/09/2001; Consolidated Edison; Indian Point 2 Nuclear Power Plant. Supplemental Inspection, Multiple Degraded Cornerstones - 95003, Problem Identification and Resolution, Human Performance, Safety Systems, Chemistry, Emergency Preparedness.

The inspection was conducted by Region I, Region II, Region IV regional and resident inspectors and NRC Headquarters and contract personnel. The significance of issues is indicated by their color (green, white, yellow and red) and was determined by the Significance Determination Process (SDP). This inspection identified all green issues.

Cornerstone: Mitigating Systems

The team identified the following issues concerning design control. The four individual findings are being treated as a non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control." (NCV 2001-002-002)

> **Green.** The design temperature ratings of electrical components in the emergency diesel generator (EDG) building, including ventilation fan thermal overloads, cabling, and control power transfer switches had not been verified. These issues were of very low significance because the as-found thermal overload settings would not have resulted in the loss of ventilation at the maximum building temperatures, the effects of elevated temperature on the cabling voltage drop calculation would have been negligible, and information obtained from the vendor indicated that the control power transfer switch circuitry would have remained functional at the elevated temperature. (Section 2.A.1.b.1)
>
> **Green.** The results of the EDG loading calculation had not been transmitted to the operations department for inclusion into appropriate operating and test procedures. These issues were of very low safety significance since the ability of the EDGs to provide emergency power was not affected and the procedure issues would not have impacted safe operation of the affected systems. (Section 2.A.1.b.1)
>
> **Green.** The ability of the service water system to supply adequate flow to all safety-related components based on existing service water low header pressure alarm setpoint and the control room log limits was not supported by engineering calculations. The licensee performed a preliminary analysis and determined that the alarm setpoint of 53 psig was adequate to ensure adequate flows. However, if pressure decreased to the control room log limit of 48 psig the system would not have had sufficient capacity to supply adequate flow to all components. The licensee increased the control room log limit to 58 psig, giving a 5 psig margin to the 53 psig low pressure alarm design limit. This issue was of very low safety significance because there was no indication that the service water system had been operated below a header pressure of 53 psig. (Section 2.A.2.b.3)

000498

Summary Of Findings

Green. Controls were not in place to prevent damage to components in the service water strainer room given an external flood caused by high river water level and a concurrent internal flood due to a potential single failure of a service water pump vacuum breaker valve. The licensee implemented a temporary procedure change to address this issue. This issue was of very low safety significance because it involved the relatively low probability of an internal flooding event coupled with the low probability of an external flooding event. (Section 2.A.2.b.3)

The team identified the following issues concerning the quality and use of procedures. The four individual findings are being treated as a non-cited violation of procedures required by Technical Specification 6.8.1 (NCV 2001-002-003).

Green. Abnormal Operating Instruction (AOI) 27.3.1, "Emergency Fuel Oil Transfer Using the Trailer," Rev. 0, did not provide adequate instructions for filling the trailer. This issue was of very low safety significance because the use of this procedure has never been required and would require minor changes to resolve the discrepancies. (Section 2.A.2.b.1)

Green. Addendum VI to SAO 100, "Indian Point Station Procedure Policy," Rev. 3, which describes the process for implementing temporary procedure changes (TPCs), was not followed when alarm response procedure ARP AS-1 (Accident Assessment Panel 1; windows 5-4 and 6-4) was changed with TPC 00-0853. This TPC was implemented because a temporary modification disabled the associated alarm inputs; however, the alarm inputs had already been disabled and the change was not required for immediate operation of the plant. This issue was of very low safety significance because the use of a TPC did not have any actual detrimental affect on plant operations. (Section 2.A.2.b.1)

Green. The reactor coolant loop Delta-Temperature alarm was received during power ascension as a result of having an incorrect setpoint value in calibration procedure. This issue was determined to be of very low safety significance since the instrument does not have any automatic protective function, only an alarm function. (Section 2.A.4.b.1)

Green. Leaving two oil absorbent pads inside the EDG 21 instrumentation cabinet following repairs to a leak did not comply with SAO-701, "Control of Combustibles and Transient Fire Load," Rev. 8. This issue was of very low safety significance because it did not represent a fire impairment nor a degradation of a fire protection feature or defense in depth issue. (Section 2.A.4.b.1)

The team identified the following other findings concerning design, testing, and maintenance rule issues.

000499

Summary Of Findings

Green. Design bases information was not translated into electrical systems testing and operating procedures acceptance criteria or operating limits. This issue was of very low safety significance because none of the test results or operating data reviews identified instances where equipment was operating outside of its design limits. This failure to include appropriate acceptance in the procedures and drawings to ensure activities have been satisfactorily accomplished is being treated as a non-cited violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings." (NCV 2001-002-004) (Section 2.A.2.b.2)

Green. The plant testing program did not include a verification that the safety-related service water strainer room drain line check valve, MD-500, could open to prevent internal strainer pit flooding. The licensee demonstrated operability by manually cycling the valve from the full open to full closed position and observing that the valve opened with minimal effort and that there was no restriction in movement. . This failure to test a valve by periodically exercising it to its safety function position is being treated as a non-cited violation of 10 CFR 50.55a, "Codes and Standards," paragraph (f), "Inservice Testing Requirements." (Section 2.A.2.b.3) (NCV 2001-002-005)

Green. Corrective actions were not taken to resolve reliability and availability performance issues with the alternate AC power sources, gas turbines (GTs) -1, -2 and -3. The GTs had not been meeting the licensee developed maintenance rule reliability and availability performance goals since 1995. The team did an independent calculation of the change in core damage probability associated with the unavailability of GT-2 for an estimated repair length of 60 days and determined that the risk increase to be within the very low safety significance band (<1E-6). This issue was of very low safety significance because the Technical Specifications relative to GT availability were met. This failure to effectively implement corrective actions to ensure that the established maintenance rule goals would be met is being treated as a non-cited violation of 10 CFR 50.65 (a)(1). (Section 2.A.3.b.1) (NCV 2001-002-006)

Cornerstone: Emergency Preparedness

Green. The team found that the Emergency Response Data System (ERDS) was found inoperable during an exercise in November 2000 and again during a test conducted in the 1st quarter 2001. The NRC conducted an ERDS test during this inspection and found both the system and it's backup to be operable. This issue was determined to be of very low safety significance because the licensee retained capability to communicate via the telephone system. The failure to correct a deficiency identified during a drill/exercise is being treated as a non-cited violation of 10 CFR 50.47(b)(14). (Section 2.D.1.b) (NCV 2001-002-007)

000500

Summary Of Findings

Green. The licensee could not locate Emergency Operations Facility inventory records for the third quarter 2000 nor verify those inventories were actually conducted and a review of available quarterly inventory records identified cases where the records were not properly filled out. This issue was determined to be of very low safety significance because notwithstanding the discrepancies which were identified, the licensee had sufficient resources in the facilities to properly respond to an event. The failure to properly maintain emergency facilities and equipment is being treated as a non-cited violation of 10 CFR 50.47(b)(8) and the licensee's E-Plan, Section 8.3 which states quarterly inventories will be conducted. (Section 2.D.4.b) (NCV 2001-002-008)

Green. The licensee was not able to produce the 3rd quarter records for the operational check of the emergency communications links between facilities and could not verify that the tests had been conducted. This issue was determined to be of very low safety significance because the licensee had installed spare operable telephone lines. The failure to conduct and/or document the performance of quarterly communications tests is being treated as a non-cited violation of 10 CFR 50.54(q) and Section 8.1.3 of the licensee's E-Plan. (Section 2.D.4.b) (NCV 2001-002-009)

Green. The team found that ten individuals assigned to the offsite and onsite monitoring teams had let their respirator qualifications lapse. This issue was determined to be of very low safety significance because there were sufficient responders with respiratory qualifications to fill the positions. The failure to maintain qualifications necessary to maintain proficiency as an emergency responder is being treated as a non-cited violation of 10 CFR 50.54(q) and Section 8.1.2 of the licensee's E-Plan. (Section 2.D.5.b) (NCV 2001-002-010)

Green. The licensee continued to identify exercise deficiencies that are repetitive performance issues and are reflective of past performances, particularly in the area of plant assessment and the dissemination of the information to the general public. The team determined that the training program was not fully effective in preventing recurrence of repetitive exercise issues to ensure consistent emergency response organization performance. This issue was determined to be of very low safety significance because these performance issues did not deal with the risk significant planning standards (classifications, notifications, PARs). The failure to establish an effective training program to train employees and exercising, by periodic drills to ensure that employees maintain the proficiency of their specific emergency response duties, is being treated as a non-cited violation of 10 CFR Part 50.54(q) and Appendix E.IV.F.2.g. (Section 2.D.5.b) (NCV 2001-002-011)

C00501

Summary Of Findings

Cross-Cutting Issues: Problem Identification and Resolution

The team identified the following findings which are being treated as a non-cited violation of 10 CFR 50, Appendix B, Criteria XVI, "Corrective Action." (NCV 2001-002-001)

> **Green.** The licensee failed to identify and correct the cause of repetitive failures of the service water strainers and motor operated service water isolation valve SWN-7. These items were determined to be of very low safety significance because the strainer failures did not have more than a minimal impact on system operability and the valve failures were identified when the valve was out of service for maintenance. (Section 1.A.b)
>
> **Green.** The licensee failed to initiate condition reports for three failures to meet the acceptance criteria for service water strainer blowdown flow rates during the performance of procedure PT-93 on July 13, 2000. This issue was determined to be of very low safety significance because the operability of the system was not affected. (Section 1.A.b)

000509

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Report Details

1. Review of Licensee Control Systems for Identifying, Assessing, and Correcting Performance Deficiencies

The team evaluated the ability of Consolidated Edison of New York (the licensee) to identify, assess, and correct performance problems within the corrective actions program. The evaluation focused on the programmatic performance of the condition reporting system and on the identification and resolution of plant performance issues.

A. Significant Deficiencies Review

a. Inspection Scope

The team conducted a review of the licensee's condition reporting system and related programs focusing on evaluating the ability to identify, assess, and effectively correct performance deficiencies. The review focused primarily on evaluations and assessments associated with program performance issues and organizational deficiencies. Additionally, the team reviewed licensee actions taken to address identified program performance issues (e.g., the effectiveness reviews conducted for the August 1999 loss of offsite power and reactor trip event). The team reviewed performance aspects associated with the January 2, 2001, turbine trip and other important issues associated with the plant systems and processes described in section 2 of this report.

b. Findings

Program Issues

In most cases, the team found that the licensee's condition reporting system was effective in identifying program performance issues and organizational deficiencies and that the individual site department business plans included the long term corrective actions for the identified performance issues within their respective organizations.

The overall ability to easily access and use the condition reporting system had been previously identified as a performance issue, and the team observed that this problem continued to challenge the plant staff. The quality assurance (QA) organization had attributed the usage problems to inadequate training and an overall lack of familiarity.[1] To address this issue, approximately one-half of the plant employees received training on the system during 2000. However, the team concluded that implementation of this corrective action was slow, because a previous condition report (CR) [2] had been initiated to document this same knowledge weakness in November 1999.

[1]These conclusions were documented in condition report (CR) 200000994.

[2] CR 199908802

Additionally, the team observed that the condition reporting system exhibited several computer based weaknesses. As examples, on several occasions during the inspection, the system was unavailable due to plant computer problems and the program made it difficult to track the status of corrective actions. The licensee had recognized these deficiencies and included an initiative in the 2001 business plan to purchase new condition reporting system software.

Line management ownership of the corrective action program had also been previously identified as an important performance issue and the team found that challenges continued to exist in this area. The team noted that the licensee had implemented measures to improve accountability (i.e., an improved metrics report, condition report quality reviews, and quarterly departmental reviews) but more improvement was needed to assist in managing and reducing the backlog and provide more effective condition report responses. Line management ownership of the program was expected to become even more important because the proposed revision 4 (Rev. 4) to the corrective action program procedure[3] would result in a significant increase in the backlog since individual items would not be closed until their associated work orders were completed. The team noted that the 2001 business plan addressed insufficient line management ownership as one of the most significant contributing causes for corrective action program problems.

The licensee's ability to trend condition reporting causal factors continued to be a challenge. This item had been identified by the NRC as an issue in 1998, and more recently in the 2000 problem identification and resolution inspection. To address this longstanding issue, the corrective action group had recently begun assigning causal factors to condition reports because prior efforts by the line organizations to perform this function had not been successful. The licensee indicated that the complicated nature of the condition reporting system software and unfamiliarity of the program by the plant staff were the primary reasons for this continuing deficiency. The licensee had initiated measures to address this issue by evaluating a less complicated software and assigning a specific individual for assigning causal factors. Additionally, plans to improve this deficiency were included in the licensee's 2001 business plan. The team determined that the inability to trend causal factors was a weakness of a longstanding nature and one for which there had been little measurable progress.

The licensee continued to face challenges in the area of condition report response effectiveness. The licensee had initiated a number of condition reports (as a result of audits and self-assessments in this area) which pointed out various problems related to this issue. For example, CR 200003865 identified that the extent-of-condition assessments were better developed for significance level (SL) 3 CRs when compared to the more significant SL1 and SL2 CRs.[4] Additionally, with respect to the quality and effectiveness of corrective actions, several deficiencies were identified. For example, CR 200004854 identified that several SL2 CRs did not meet management expectations for quality, primarily due to insufficient line management ownership for corrective action

[3] Station Administrative Order (SAO)-112 Corrective Action Program, Rev. 4

[4] The licensee's system assigned a significance level to each CR, with SL3 having the lowest significance and SL1 having the highest.

evaluations. As a result, the licensee required that all SL2 CRs receive a quality review by the Corrective Action Review Board (CARB). However, the team observed that the CARB's review of at least one SL1 CR was of mixed quality. Specifically, one CARB member was not fully aware of all the key elements to be considered during the review. Another member expressed concern that if the board assigned lower quality scores, then that would require the SL1 report to be revised. The team was concerned that this attitude indicated a potential hesitancy to score CR quality responses low to avoid required revisions. Additionally, the quality review process observed by the team was informal and lacked critical assessment on some issues.

The team noted that the licensee's effectiveness reviews continued to indicate difficulties with the corrective actions taken to address problems identified following the August 1999 loss of offsite power and reactor trip event. The licensee used outside contractors to conduct several independent assessments such as a review of common cause trends in the condition reporting system, a review of the closure of condition reports, and a review of corrective action effectiveness for actions taken following the event. These reviews were self-critical and provided valuable information with respect to improving plant performance. However, these reviews also identified areas where previous corrective actions have not been fully effective.

Implementation Issues

In the review of the implementation of the corrective action program, the team identified a number of issues related to weaknesses in implementing effective corrective actions and in identifying repetitive failures of certain plant components. Additionally, several examples were identified where condition reports were not promptly initiated for plant and equipment deficiencies.

For example, the team discovered instances of repeated equipment failures that were not identified in the condition reporting system. While the issues were individually raised in separate condition reports that were subsequently closed to work orders, the repetitive nature of the failures were not questioned relative to the adequacy of previous corrective actions. Examples included:

- Repetitive service water strainer failures were identified through the review of maintenance activities performed since early 1998. The strainers had failures caused by issues such as: tripping overloads, binding, and a damaged arm shaft.[5] As part of an effort to address the unavailability caused by the failures in December 1998, the licensee added a preventive maintenance work scope that involved a periodic overhaul or replacement of a strainer with a rebuilt internals package every six months. However, additional failures subsequently occurred, caused by issues such as binding, tripping, and high differential pressure. There was no indication that the problems were being pursued as repetitive failures to

[5] CRs 199905026, 199902815 and 199902586, respectively

ascertain their root causes or to perform broader corrective actions to preclude repetition. This issue was determined to be of very low significance (Green) because each failure had a minimal impact on system operability.

- Repeated failures of service water valve 7 (SWN-7) were identified during a review of condition reports. SWN-7 is the isolation valve for the service water supply to turbine building loads and provides a barrier between the essential and non-essential loads. CR 200002700, written in April 2000, identified that the sector gear on the operator for SWN-7 required replacement and was closed out to a work order to complete the repair. On May 1, 2000, CR 200003085 was written to clarify that this was the second failure of this valve due to a damaged sector gear. This CR also noted that the worm gear on the valve operator was damaged, and had not been repaired even though the licensee attempted to return the valve to service. Although this worm gear had been determined to be damaged, the condition report identified that a new worm gear was on order and as of May 2000 had not been received. The team questioned why post maintenance testing had been attempted on the valve while it still contained damaged components and why this issue had not been raised by any of the condition reports in the system. After reviewing the condition reports and work orders involved with this issue, the licensee agreed that the condition reports had been inappropriately closed without an engineering evaluation to address the repetitive failure. This issue was determined to be of very low significance (Green) because the deficiency had been discovered when the valve was out of service for preventive maintenance and had not been returned to service.

Contrary to 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," the licensee failed to take adequate measures to properly identify and correct several instances of repeated failures and degradation of the service water strainers and valve SWN-7. As a result the licensee failed to determine the root causes and to take appropriate corrective action to preclude repetition of these issues. This violation is being treated as a non-cited violation consistent with Section VI.A of the NRC Enforcement Policy, issued May 1, 2000 (65FR25368). The two specific issues were entered into the corrective action program as CRs 200101388 and 200101125 **(NCV 05000247/2001-002-001).**

The team also identified several examples where the licensee failed to promptly issue a CR upon the discovery of an adverse condition or deficiency. For example, during the performance of PT-R93, "Essential Service Water Header Flow Balance," in July 2000, the team identified three cases where the as-found service water strainer blowdown flows exceeded the 215-235 gpm acceptance criterion, and no condition report had been generated as required by the corrective action program.[6] The affected strainers were: pump 21 strainer (277 gpm as-found flow), pump 23 strainer (305 gpm as-found flow), and pump 26 strainer (254 gpm as-found flow). It was also noted that the procedure required blowdown flows to be adjusted to within the acceptable range prior to obtaining

[6] SAO-112, Rev. 3

the as-found flows for the remaining components. Even though the remaining components' as-found flows might be acceptable, this premature adjustment of blowdown flows had the potential to mask unacceptable flows to the other loads.

It was estimated that, before the adjustments, flow to the other loads were approximately 1.15% lower than recorded. This would have resulted in only one of the other components, a containment fan cooler unit, to have flows less than its acceptance criterion. The fan cooler's flow would have been 10 gpm below the 1,740 gpm acceptance criterion. However, since the actual required flow for operability was 1,600 gpm, it would have still been able to perform its safety function.

The licensee failed to generate CRs for three failures to meet the acceptance criteria for service water strainer blowdown rates in procedure PT-R93 on July 13, 2000. This issue was determined to be of very low significance (Green) because the operability of the system was not affected. This issue is considered an additional example of the non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Actions." This issue was entered into the corrective action program as CR 200100568 **(NCV 05000247/2001-002-001)**.

The team identified other examples, of a more minor nature, of the failure to initiate required CRs. Although, each of these issues warranted correction, none presented an operability concern and were therefore considered to be minor violations of regulatory requirements, not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy. Representative examples included:

- The licensee failed to initiate condition reports for instrumentation and control preventive maintenance procedures 1775-1 and 1778-1 when the alarms could not be verified (as required by the procedure) due to tagout 98-10993 which removed dc control power. Also, the licensee was slow to initiate condition reports after the team identified this on January 18, 2001. CRs 200101467 and 200101468 were written, but not until February 8, 2001.

- During the walkdown of the service water pump intake bay, the team identified several issues that did not meet foreign material exclusion requirements. The conditions involved: (1) the presence of spalling concrete, (2) peeling epoxy coating on SW piping, (3) a 3/4 inch carbon steel nut in the service water strainer pit drain valve MD-501, and (4) degraded valve assembly nuts on the drain valve. The spalling condition had been previously identified in CR 199808290, but there was little evidence of any meaningful corrective action beyond installing a tarp in the area. Following the team's identification of these issues, the licensee generated CRs 200101433, 200101464, and 200101431 to address these conditions.

- During a walkdown of the service water system, the team noted several conditions that demonstrated a lack of attention to detail by maintenance personnel. Specifically, instances of the use of fasteners fabricated from dissimilar materials, inconsistent use of washers in bolted arrangements, improper nut thread engagement, and physical differences between the fasteners used on similar equipment. The licensee issued CRs 200100565, 200100560, and 200100510 to address these issues.

- During a review of the Temporary Facility Change (TFC) process, the team noted that the licensee failed to conduct the quarterly review of TFCs for the fourth quarter of 2000 as required by station procedures.[7] The purpose of this review is for the Generation Support Manager to determine the compliance of each individual change with respect to the procedure requirements and to determine whether individual open TFCs should remain in effect. The team considered this to be a minor violation of administrative controls. The licensee initiated CR 200101456 to address this issue.

- During the review of the 480 Vac Design Basis Document (DBD), the team found that only 2 of 101 open items had been entered into the corrective action program for resolution. The remaining 99 open items contained conditions such as missing or unapproved calculations and specifications. In response, the licensee grouped the 99 items into 13 general categories and generated a separate condition report for each category.

Additionally, the team identified a weakness in documentation and in initial efforts to establish root and contributing causes of the January 2, 2001, turbine trip. In CR 200100048 the licensee indicated that a contributing cause for the event was an off-normal system line-up leading to the operator having to start a second condensate pump to address a lower that normal feed pump suction pressure. Additionally, the report described untimely actions by a reactor operator which caused overfeeding of the steam generators and an associated steam generator high level turbine trip. However, in the resolution of the CR, there were no specific corrective actions to address the root and the contributing causes. The licensee noted in the interim action section of the report, that the operations manager was completing crew briefings on the event and that procedures were to be changed. However, the CR did not address any potential operator knowledge deficiencies in the operation of the condensate and feed system. After significant interaction with NRC staff, ConEd ultimately developed a reasonably comprehensive assessment of the event and took additional corrective actions.

B. Quality Assurance, External Audits, and Self-Assessments Review

a. Inspection Scope

The team reviewed selected audits and assessments performed by the line organizations, the quality assurance group, and external sources to determine whether the licensee had demonstrated the capability to identify performance issues before they resulted in undesired consequences. The team evaluated management support of these assessments and also evaluated the effectiveness of management systems to process and act upon identified performance issues.

[7] SAO-206, "Temporary Facility Changes," Rev. 20, section 6.1

b. Findings

In general, the audits and self-assessments reviewed by the team were well conducted and provided sufficient detail and recommendations for improvement. Also, the corrective actions taken were generally effective. The condition reporting system was used to identify and track the closure of issues from the audits and self-assessments. Some examples of effective self-assessment activities included the following:

- Audit 00-09-C, "Corrective Action - 1st Half 2000," dated September 28, 2000, was thorough and self-critical in identifying areas of needed corrective action program improvements. These improvements included a revision to the program procedure, enhanced metrics for timeliness and quality of condition report responses, and improved training for new employees. The team reviewed the condition reports for the significant audit findings and determined that the licensee's response to the performance deficiencies was acceptable. The team noted that continued efforts for further improvements in these areas was also included in the corrective action program 2001 business plan.

- The team reviewed several audits and condition reports associated with plant procedures. In particular, Quality Assurance Audits 98-08-L (January 5, 1999) and 00-08-A, (February 2001) assessed station instructions, procedures and drawing control. The team determined that the audits and associated condition reports were of good quality and provided the proper emphasis on station improvement.

- The system engineering self-assessment on engineering work control interface completed in February 1999 identified weaknesses.[8] The team reviewed the completed corrective actions for these condition reports and interviewed several system engineers and work week managers with respect to the findings. The team determined that the corrective actions were adequate.

Notwithstanding these positive observations, the team identified a number of performance weaknesses in the self-assessment program. The following examples are representative:

- The quality assurance (QA) department self-assessment of the audit program dated March 6, 1999, contained no substantive assessment of QA's ability to evaluate plant problems and effectively communicate those problems to plant management. The purpose of the assessment was to evaluate the audit program against industry practices and identify areas for improvement. However, the assessment primarily focused on elements such as training, staffing, audit report detail, procedures, and office space.

[8] CR 199902791 and CR 199902792

- QA's 2000 self-assessment dated September 14, 2000, concluded that the organization's program elements were not adequate for effectively promoting performance-based continuous improvement. The assessment identified that plant risk assessment data needed to be more effectively used in the audit process. The assessment also identified that individual auditor training plans needed to be developed to provide better technical skills. These were good findings. However, the training matrix developed to address the assessment's findings did not include plant risk training. The team considered that not including risk training in the matrix was a weakness with respect to the ability to integrate priority assessment results into effective corrective actions. The team noted that continued efforts for further improvements were included in the 2001 business plan.

- The primary purpose of the engineering third party self-assessment issued on August 14, 2000, was to review the quality of engineering output documents. However, the assessment did not document any reviews of actual engineering calculations or other output documents. The team also reviewed another assessment,[9] and found it had covered numerous engineering work product areas. The discussions provided appeared to be self-critical and constructive and represented meaningful assessments.

- It was recognized in the February 2001 audit of "Plant Operations and Operations Performance, Training, and Qualification," that the corrective actions associated with a similar audit in January 1999 had not been fully effective. Specifically, several issues associated with procedure quality and adherence were identified, but the subsequent effectiveness review concluded that the station still had problems with procedural compliance and accountability. As a result of this issue, the licensee issued CR 200005446.

C. Work Authorization and Allocation of Resources Process

a. Inspection Scope

The team reviewed the corrective action and maintenance backlogs for the systems selected for detailed review to assess the extent of the backlog and determine if there was open work which would prevent the systems from performing their safety functions and reviewed the prioritization and timeliness of corrective action program items. For the systems selected for review by the team, there were a total of approximately 40 open requests for engineering services and modifications.

[9] IP2 Engineering Document Quality Review, January 5, 2001

b. Findings

Backlog Review

The team found that the overall backlog of open CRs and work orders had increased, however there had been some improvement in the timeliness of completing condition report evaluations.

The total number of open corrective maintenance work orders for all plant systems was reported by the licensee to be approximately 875 at the end of the inspection and had gradually increased over the past several months. The number of temporary facility changes and control room deficiencies had trended upward and continued to exceed the plant goals. A recent reduction in the number of operator work-arounds had been achieved but the number had also continued to exceed the plant goal.

With respect to the maintenance backlog in the service water, 480 Vac, and emergency diesel generator (EDG) systems the team did not identify any open issues that appeared to challenge the functionality of the system. There were no overdue preventive maintenance work orders for the service water system. However, six were overdue for the 480 Vac and emergency diesel generator systems; none appeared to have a potential effect on equipment operability.

The team also noted that some open work items had not been accurately captured in the accounting for the backlog. Examples of this included approximately 99 open items from the recently completed design basis review of the 480 Vac system, a significant number of issues related to the instrumentation and control preventive maintenance program, and a large number of procedure changes associated with the "communications to staff" program. These observations indicated that the actual plant work backlog was somewhat larger than previously believed.

The team observed that the licensee continued to face challenges with respect to the use of plant risk information for condition report and corrective action prioritization. This had been identified in the recent NRC problem identification and resolution inspection, as well as in other previous NRC inspections and licensee self-assessments. The team concluded that this was another example of a longstanding weakness in the corrective action program and one for which limited progress had been achieved.

Finally, the team observed that the licensee's average time to close corrective actions was significantly outside station goals. The average as of the January 2001 data was approximately 256 days (i.e., time from identification to problem correction). The station goal for this metric, which was based on industry bench marking data, was in the 90 - 180 day range. It was noted that the configuration management and controls backlog appeared to the leading contributor to driving the average in the upward direction with a 560 day closure time as of the January 2001 data.

D. Review of Station Performance Goals and Strategic Plans

a. Inspection Scope

The team evaluated the licensee's performance goals to assess whether these goals and associated strategic plans were aligned with the actions needed to correct the known performance issues at Indian Point 2. The team conducted numerous management and plant staff interviews and specifically reviewed the departmental business plans for the following organizations: corrective action program, configuration management, work management, emergency planning, operator training, engineering, operations, and maintenance.

b. Findings

Performance Goals

The team reviewed the 2001 business plan for the site organizations as noted above. It was determined that the business planning process had adequately provided for the integration of efforts and provided an appropriate allowance for resources and associated funding. However, it was also noted that the details provided in the individual department plans varied significantly. Several of the plans lacked proposed completion dates for certain items and others were somewhat general in the description of areas of needed improvement. Some representative examples of individual department business plan observations are listed below.

- Several weakness in the documentation associated with the configuration management and control business plan were noted. For example, several business plan items associated with Technical Specification setpoint calculation issues contained question marks as place holders. Additionally, items related to staff training in the updated final safety analysis report, licensing basis and design basis documents contained provisions for funding, yet did not contain justification or support for station organizational goals. A similar example existed with a business plan goal associated with "Operating Equipment Staff Augmentation." The business plan listed the next seven design basis documents to be updated in the continuing design basis document upgrade project. The team noted that two of the systems, main steam and the emergency diesel generators were scheduled to have been started on October 1, 2000, but no current status appeared in the plan. Interviews with plant staff indicated that the projects had not yet been started.

- The maintenance department business plan was detailed and comprehensive. Major improvement areas were identified and included the maintenance backlog reduction plan, the work control process improvement plan, and instrumentation and control preventive maintenance program upgrade plan. The team noted that, with a few minor exceptions, the plan identified managers responsible for required actions, along with expected completion dates.

- The corrective actions program business plan was not fully developed and none of the plan's initiatives had schedule dates for completion. Additionally, the plan did not specifically address the resources required to complete the planned initiatives. The team also noted instances whereby items were closed prematurely. However, the team noted that even though the approved plan was not fully developed, the plan's major elements appeared to address needed improvement areas such as human error reduction, operating experience, trending, and line management ownership for corrective actions.

- The operations training business plan contained proposed budgets, staffing, and schedules for completing major department initiatives. Additionally, it was noted that effectiveness reviews of major actions taken were scheduled for later in 2001. The plan contained initiatives associated with major areas of operator knowledge weaknesses and referenced performance improvement programs established to improve the skills, knowledge, and abilities of licensed operators.

- The design and site engineering business plans included appropriate areas for improving engineering processes, design bases documentation, and equipment reliability. Backlog reduction efforts were also included in the plans. However, specific project details and schedules were not included within the business plans.

<u>Management Interviews</u>

The team conducted extensive interviews of licensee managers throughout the organization including the chief nuclear officer, site vice presidents, and many department managers. The management consensus was that the current plant performance problems started as experienced staff began to leave site in the early 1990s. This, combined with a lack of infrastructure improvements, and a successful extended plant run in 1996 led to an organization that lost a significant portion of its knowledge base, did not seek out external perspectives, and did not recognize the need for continued improvement due to demonstrated high capacity generation.

The team concluded that the station management was in general agreement with respect to the performance problems which existed at the site and in the areas requiring improvement. Additionally, the station management was in almost unanimous agreement in the belief that the 2000 business plan was a success and had allowed for focus on areas for improvement and in planning for and obtaining needed resources to complete the required tasks. The managers also believed that the 2001 business plan provided an adequate scope and method of documenting needed areas of future improvement along with the resources to accomplish the activities. Several managers indicated that the use of an approved, resource-loaded business plan was the first time that the organization had such a detailed plan for which they had been held accountable.

E. Employee Concerns Program Review

a. Inspection Scope

The team performed a review of the licensee's employee concerns program (ECP), also known as the Ombudsman Program. This review focused on the adequacy and responsiveness to employee concerns and included an assessment as to whether a safety conscious work environment existed at the facility. The team interviewed numerous personnel at various levels of the organization and reviewed the program files and documentation associated with the program. The team also reviewed a self-assessment of the Ombudsman Program to evaluate whether appropriate action was taken for deficiencies which had been identified.

b. Findings

The team noted that the ECP appeared to provide an acceptable means for employees to raise safety concerns to management without fear of retaliation. In addition, the licensee's condition reporting system allowed employees to raise safety issues anonymously and was viewed as an alternate process to the ECP. The team did note that the number of anonymous CRs initiated could be an indication that some employees were reluctant to identify themselves with concerns raised. In most cases, the team found the licensee's response to employee concerns was acceptable and interviews with site employees indicated that a safety conscious work environment existed at the facility.

Notwithstanding the overall adequacy of the program, the team identified several minor deficiencies. It was determined that the ECP procedure, SAO-123, "Employee Concerns Program," Rev. 10, lacked specificity with respect to several important program elements. These elements included (1) how employees access the ECP, (2) methods for employees to report safety concerns, (3) program assurance of maintaining employee confidentiality, and (4) measures to protect employees against retaliation. The team reviewed other aspects of the program such as general employee training information, bulletin board information about the program, and posted information at drop boxes where employees submit concerns. As a result of this finding, the responsible manager, otherwise known as the Ombudsman, initiated CR 200100619 to correct the deficiency. The team determined that, even though the governing procedure lacked the desired specificity, sufficient information regarding these program elements were included in the program.

The team reviewed the 2001 business plan for the ECP and found that it provided the expected degree of specificity for program improvements. In particular, the team noted that more detailed training for managers and other plant personnel was scheduled for 2001. Also, the plan included initiatives for updating the program procedure, preparations for the annual self-assessment, documentation improvements, and program improvements for the classification and tracking of concerns.

F. Operating Experience Review Program

a. Inspection Scope

The team conducted a review of the operating experience review program to determine if appropriate actions were taken to address potential plant problems identified as a result of industry operating experience. The team reviewed the licensee's governing operating experience review procedure, program assessments, and backlog of open items. Interviews were conducted with program personnel as well as the line organizations. The team also reviewed selected 10 CFR Part 21 reports and NRC Information Notices from 1998 thru 2000 to determine if the program had adequately assessed the issues for applicability at the site.

b. Findings

Previous NRC inspection efforts as well as licensee assessments had identified weaknesses in the licensee's operating experience review program. The team determined that while some limited progress had been made, primarily in the area of industry bench marking and outreach efforts, that weaknesses continued to exist in the program. The team observed that some progress had been made by the advent of enhanced electronic access and by increased resource allocation. However, the overall implementation of the program, particularly by the line organizations, continued to be a problem. Additionally, the team determined that while there had been progress in the reduction of the backlog associated with operating experience items, continued emphasis was needed. The following observations are representative of the team's findings with respect to the program:

- Surveillance Report 99-SR-040, "Operating Experience Review," dated November 11-18, 1999, was performed by the site quality assurance organization. The team determined that the audit was self-critical and identified several needed program improvements. The audit concluded that plant personnel did not effectively use operating experience. The team reviewed the results and found that no action had been taken on the audit findings until June 2000. The team concluded that based on the significant programmatic nature of the findings that the licensee's response was untimely. However, the team verified that the corrective actions were eventually included in the corrective action program 2000 business plan and were completed by the end of the year.

- The team reviewed the licensee's self-assessment, "Operating Experience Peer Evaluation," dated September 5-7, 2000. The assessment concluded that the program needed improvement in that the "observed performance did not indicate an active program or that individuals were sufficiently engaged with respect to the usage of operating experience." The team reviewed a number of condition reports that were initiated as a result of the assessment. For example, CR 200006619 was initiated to address operating experience training because as the assessment stated "station personnel are passive with respect to obtaining operating experience information in support of their day-to-day activities." However, the corrective actions did not address the need to train personnel on the value of operating experience as it relates to their daily work, but established a focus group with departmental points of contact. The team determined that no site wide training had been provided on operating experience and none had been provided for specific target audiences such as engineering, operations and maintenance personnel.

- The team reviewed nine selected operating experience review evaluations. Of the nine which were reviewed, the team found thoroughness issues with four of the evaluations. For example, CR 200009927 evaluated a 10 CFR Part 21 notification of a defective Foxboro relay module. The licensee verified that the defective relay was not installed in the plant but failed to place an in-stock spare on administrative hold until verification that the spare relay was not defective. The reviewer had intended to place the spare relay on hold and communicated this intent by e-mail versus using the condition reporting system for tracking the action. Subsequently, the individual did not follow through with his intentions and the verification was not performed until the inspection team discovered the problem. The spare relay was later checked and found to be satisfactory. The licensee initiated CR 200100904 to address this problem. An additional example of an inadequate response to an operating experience review item involved the failure to evaluate a residual heat removal system operating procedure. Specifically, CR 200004907 evaluated an industry notification which addressed the need to evaluate the system fill and vent procedure for certain specific problems described in the notification. The individual who performed the review misunderstood the process and failed to initiate a corrective action item or communication to staff item, consequently no procedure review was performed. The licensee initiated CR 200100894 for this problem.

- The team noted problems in the timeliness associated with completing operating experience reviews and corrective actions. For example, the evaluation for CR 199802561 took two years to complete. This item concerned NRC Information Notice 95-52 Supplement 1 which was related to fire protective systems. Interviews indicated that the delays in addressing this issue were related to resource limitations. Another example involved CR 199810884 which took 17 months in order to complete the needed corrective actions. This item was related to pipe weld failures in the chemical volume and control system that had occurred in the industry. The corrective action involved a radiograph of the suspect flow orifice in the piping to determine if cavitation damage had occurred.

- The team attended a CARB meeting on February 8, 2001. The meeting focus was to approve a SL1 condition report regarding the failure to maintain containment integrity calculations provided by a vendor. The presenter failed to address operating experience in the report, however, this shortcoming was identified by the board co-chair.

- The team reviewed the backlog of operating experience review items. In January 2001 the total backlog of open items was 133 with 38 items being overdue. The team noted that the backlog had gradually decreased from 366 in October 1999. A significant reduction in the backlog had occurred in June 2000 when the backlog decreased from 205 to 118. The licensee attributed this reduction to an increase in resources in the this area. The team concluded that progress had been made in the operating experience review backlog but continued emphasis was warranted in this area.

G. Supplemental Inspection - Emergency AC Power Unavailability, >2EDG Performance Indicator

a. Inspection Scope

The Indian Point 2 performance indicator (PI) for "Emergency AC Power Unavailability, >2EDG" exceeded 2.5% (white band) starting in the 2nd quarter of 1999. The AC power system availability declined due to the failure of the 23 emergency diesel generator (EDG) to operate on demand during the reactor trip event with complications on August 31, 1999. EDG 23 failed because the overcurrent trip device (amptector) on its supply breaker to emergency bus 6A had been improperly calibrated. The improperly calibrated amptector added 1444 hours of unavailability and increased the fault exposure hours in the calculated PI for EDG 23.

The NRC review of the performance of the emergency AC power supplies during the August 31, 1999, event was previously described in NRC Augmented Inspection Report 05000247/1999-08, Followup to the Augmented Inspection Team Report 05000247/1999-013, and the Enforcement Followup Inspection to the Augmented Inspection Team Report 05000247/1999-014. The corrective actions related to testing of the safety related breakers and other issues were described in a licensee letter to the NRC dated June 5, 2000.

b. Findings

During these reviews, the NRC verified that the licensee's evaluations provided assurance that the root and contributing causes for the EDG failure were understood, that the extent of condition on other safety-related breakers was identified, and that corrective actions to correct weaknesses in the calibration of overcurrent devices were sufficient to address the causes for the event and to preclude recurrence. As such, the NRC removed this issue from consideration in future Agency actions, per the Action Matrix, in accordance with the guidance in Inspection Manual Chapter 0305, "Operating Reactor Assessment Program."

H. Conclusions Associated with Licensee Control Systems for Identifying, Assessing, and Correcting Performance Deficiencies

The team determined that the overall program for problem identification and resolution was adequate. It was noted that some improvements had been made, in particular, an increased emphasis on problem identification and an improved metrics and tracking system for corrective actions program issues. However, the team identified several continuing challenges to the program. In particular, it was observed that the effectiveness of some of the corrective actions for previously identified deficiencies was of somewhat mixed quality. Additionally, significant challenges existed with respect to the timeliness of corrective actions and longstanding issues remained with respect to prioritizing issues for resolution and in trending causal factors. Further, the backlog associated with open corrective actions presented an ongoing challenge to the station. Finally, as noted in previous assessments, weaknesses continued to exist in the operating experience review program, although some improvements had been made in this area. While performance difficulties continued to exist with respect to the review and disposition of technical issues, the site has made considerable progress in areas related to industry outreach and bench-marking efforts.

2. Assessment of Performance in the Reactor Safety Strategic Performance Area

A. Emergency Diesel Generator, 480 Vac and Service Water Systems

1. System Design

a. Inspection Scope

The team selected the emergency diesel generator (EDG), 480 Vac and service water systems for detailed reviews. The selection was based on these systems' importance to overall plant risk and also due to the fact that these systems had not received recent, in-depth reviews by either the NRC or the licensee. The team reviewed licensing and design basis documents for these systems, including the Updated Final Safety Analysis Report (UFSAR), calculations, engineering analyses, and system descriptions (when available) to determine the functional requirements of the systems for normal, abnormal and accident operating conditions. The team reviewed a sample of risk significant plant modifications for the selected systems, including those that involved vendor supplied products and services to verify that the design changes did not negatively impact the ability of the systems to perform their design bases functions and that the changes would not cause initiating events. During this review, the team evaluated the effectiveness of the licensee in controlling design and licensing information, in providing necessary calculations to support plant changes and in developing and implementing thorough post-modification testing. The team assessed the adequacy of the licensee in evaluating applicable system and support system design attributes and regulatory requirements. The team also reviewed system modifications to ensure that original design and accident analyses assumptions were not invalidated by the changes. Additionally, the team reviewed the modifications to confirm that the licensee had properly evaluated any required changes or additions to plant procedures.

The team conducted general walkdowns of the systems. Also, recent changes to plant maintenance and operating procedures were also reviewed to ensure that they did not result in inadvertent design changes to the systems. For procedures that involved design changes, the team verified that the change was subjected to the appropriate design change processes, including review in accordance with 10 CFR 50.59, "Changes, Tests, and Experiments."

The team assessed the adequacy of communications between the site departments during the performance of design related activities such as the updating of training programs, updates of design related materials and the performance of operability evaluations. The team verified that the appropriate departments were involved in the evaluation and concurrence process for the approval of activities that included non-routine maintenance, temporary modifications, and field change requests. The team also assessed the adequacy of the licensee's control of vendor supplied services and products, including the process for communicating identified deficiencies to the vendor.

Finally, the team reviewed a sample of condition reports to assess the effectiveness of corrective actions for deficiencies involving design activities.

b. Findings

b.1 480 Vac and Emergency Diesel Generator System

The 480 Vac system provides power to safety and non-safety related equipment. The safety-related equipment is powered by a four bus, three train arrangement normally supplied from off-site power through the 6.9 kV buses. Upon loss of the normal off-site supply, the safety-related buses are powered from three emergency diesel generators. An alternate source of power to the buses is also available from three gas turbine generators that connect to the electrical system at the 13.8 kV level. The 480 Vac system is supported by the 125 Vdc system for switchgear and EDG control power and the 118 Vac system provides power for the safety injection initiation instrumentation.

The team reviewed the important design control aspects of the 480Vac and emergency diesel generator system. A number of performance issues and weaknesses were identified. The following observations are representative of the issues identified by the team.

EDG Building Ventilation System

The team reviewed the ventilation system for the three site EDGs. The EDGs occupy a common building. Calculation GMH-00006-00 determined the maximum building temperature under worst case conditions, assuming three of the six EDG building exhaust fans were unavailable, to be 126°F. In response to the team's questions on the capability of the electrical equipment in the building to operate at the maximum calculated building temperature, the licensee found that the control power auto-transfer switches for the diesels had not been qualified for the maximum building temperature.

The team also reviewed the settings of the thermal overload devices for the ventilation exhaust fan motors and found that the thermal overload ambient compensation had not been designed for the maximum building temperature. As a result, the trip point required derating for the higher temperature. The team also noted that the thermal overload calculation was based on a different device than what was actually installed in the circuits and did not account for the manufacturing tolerance which the team later found to be ± 20%. The team also observed that the thermal overloads were not periodically checked as part of the preventive maintenance program. In addition, the team found that the voltage drop calculation for the exhaust fan power circuits did not consider the maximum possible building temperature.

The above errors were a result of the licensee failing to confirm the adequacy of these components in a maximum ambient temperature of 126°F, which was 22°F above their nominal rating of 104°F. The licensee performed calculation FCX-00421-00 and determined that there was no immediate operability concern since, with two fans operating the building temperature would not exceed 104°F with an outside temperature up to 73°F.

The licensee subsequently revised the thermal overload calculation using derating factors obtained from the manufacturer for the higher room temperatures. The calculation indicated that the specified dial setting of 9 would have been satisfactory because the original setting included a 15% margin for the motor service factor. However, the calculation also concluded that a dial setting of 10 would be implemented to provide additional margin to the trip point. The team later found that the licensee had not verified the as-built settings of the overloads prior to revising the calculation and a field verification determined that five of the six fans were set at a dial setting of approximately 8.66 and the sixth fan, added by modification CPC-91-06847-H, was set at a dial setting of 9.0. The licensee reviewed the operability of the fans for the setting of 8.66 and concluded that there was sufficient margin to prevent tripping at an ambient temperature of 126°F.

The team determined these issues to be of very low significance (Green) because the as-found thermal overload settings would not have resulted in the loss of ventilation at the maximum building temperatures, the effects of elevated temperature on the voltage drop calculation would have been negligible and information obtained from the vendor indicated that the control power transfer switch power circuitry would have remained functional at the elevated temperature.

The team considered the failure to verify the adequacy of the design temperature ratings of components in the EDG building to be a violation of 10 CFR Part 50, Appendix B, Criterion III," Design Control." This violation is being treated as a non-cited violation consistent with Section VI.A of the NRC Enforcement Policy, issued May 1, 2000 (65FR25368). These issues were entered into the corrective action program as CRs 200100780, 200101447, 200101852 and 200102336 **(NCV 05000247/2001-002-002)**.

<u>EDG Manual Load Control</u>

The team reviewed the EDG loading calculation, FEX 000148-00, and observed that the sizing of the diesels was acceptable, but that little design margin was available when the required design basis assumptions were applied. The team also found that some of the assumptions and conclusions of the calculation regarding operator actions had not been formally transmitted to operations procedures.

The team reviewed the assumptions for frequency tolerance and individual motor load data.[10] The EDG vendor instruction manual, (VIM)-2351, included a section on setpoints which indicated a frequency tolerance of +/- 0.5 % which was included in the loading calculation. However, the team found that the surveillance tests for the EDGs either failed to include an acceptance criterion for frequency (Procedure PT-R14) or contained an acceptance criterion different than that assumed in the EDG loading calculation (Procedures PT-M21 and PT-R84).

The calculation also contained an assumption that the auxiliary feedwater pump flow would be throttled by operators during the accident (versus in a runout condition) prior to the transition to the recirculation phase following a loss-of-coolant accident (LOCA). However, that assumption had not been formally transmitted to operations for inclusion in plant procedures. The team also found that the emergency operating procedures (EOPs) had been recently updated to include revised motor loads but the update failed to include the correct loading values from the EDG load calculation. In many cases, the errors observed were non-conservative.

The team determined these issues were of very low safety significance (Green) because the ability of the EDGs to provide emergency power was not affected and the procedure issues would not have impacted safe operation of the affected systems.

The failure of the licensee to translate the design requirements for EDG loading into appropriate procedures and instructions is considered an additional example of the non-cited violation of 10 CFR 50, Appendix B, Criterion III, "Design Control." These issues have been entered into the corrective action program as CRs 200100777, 200100599, and 200100943 **(NCV 05000247/2001-002-002)**.

<u>Alternate AC Power Source Voltage</u>

The team reviewed the capability of the gas turbine (GT) generators to power the safety-related shutdown loads. The licensee was unable to locate a voltage drop calculation to demonstrate that adequate voltage could be supplied to the required loads. Subsequently, the licensee performed an evaluation to address this issue. The team reviewed this evaluation and found that the licensee failed to confirm the actual tap setting of the 13.8 kV to 6.9 kV transformer which connects the alternate AC source to the plant. This resulted in a non-conservative input to the evaluation. The team also noted the evaluation was performed for GT-1 which is located on site and did not initially evaluate the voltage available from GT-2 or GT-3 which are located offsite and may have been more limiting due to voltage drop considerations.

The team determined this issue did not have a credible impact on safety because the load assumed in the evaluation was significantly higher than actual expected safety bus loads. Even with this resultant voltage drop, sufficient voltage would be available to power the safety-related loads. Although this issue should be corrected, it constitutes a

[10] Frequency affects motor speed for the driven loads; a higher frequency results in additional load to the EDGs

violation of minor significance that is not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy. This issue was entered into the corrective action program as CR 200101298.

480 Vac Load Ampacity Calculations

The team reviewed the ampacity rating for selected 480 Vac feeders, including the feeds to the 480 Vac switchgear and the service water pump motors. The licensee's calculation EPG-00027-00 indicated that the loss-of-coolant-accident (LOCA) load, with offsite power available, could be 2,420 kVA or 2,911 Amps. The team found that the calculation for the feeder to Bus 6A contained an incorrect input for the rating of the bus connection and used incorrect units. Based on the information supplied, it appeared that the bus would have been overloaded by 400 Amps. The licensee was subsequently able to demonstrate that the connection from the EDG to the bus had been analyzed for the re-rating of the EDG to carry 3,300 Amps.

The licensee could not produce a calculation for the service water pump motors that evaluated the adequacy of the feed from the Unit 2 buses (original design) or from the Unit 1 alternate supply. The licensee subsequently identified relevant correspondence from the original architect engineer from the 1969 time frame and also evaluated the cable size using the guidance in Okonite Engineering Bulletin EHB-98. Although a formal calculation had not been completed by the completion of the inspection, it appeared there was an acceptable basis for the original design. The team determined this issue did not have a credible impact on safety because the design was subsequently determined to be acceptable to support plant operations. Although this issue should be corrected, it constitutes a violation of minor significance that is not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy. These issues have been entered into the corrective action program as CRs 200101463, 200100584 and 200100796.

Design Inputs for Load Flow and Voltage Drop

The licensee's design basis calculations included voltage drop or load flow studies for the 480 Vac, 118 Vac, and 125 Vdc systems to demonstrate sufficient voltage at the safety-related loads. The team found that the 480 Vac load flow calculation, FEX-000144-00, included a number of unverified assumptions and inputs. These included the lack of a controlled basis for the impedance diagram and conflicting motor data. Also, the offsite system operating conditions were inconsistent with those used in the degraded voltage studies.

These issues did not have a credible impact on safety because the team reviewed a sample of assumptions and inputs and found that the variations in input data would not have affected the conclusion of the calculation. Although this issue should be corrected, it constitutes a violation of minor significance that is not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy. These issues have been entered into the corrective action program as CR 200100583 and CR 200100591.

Instrument Power Supply Voltage Automatic Transfer Point

The team reviewed the operation of the 118 Vac system safety-related inverters which power the safety-related instrument buses. The inverters have a solid state transfer switch on their outputs that transfers the output from the inverter to a transformer supply in the event of a degraded input or output voltage. The team found that there was no engineering evaluation to support the transfer set point for the inverters.

The team determined this issue did not have a credible impact on safety because the inverter output is periodically monitored and verified to operating at an acceptable value specified in the daily log. Although this issue should be corrected, it constitutes a violation of minor significance that is not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy. This issue has been entered into the corrective action program as CR 200100908.

Auxiliary Feedwater Pump Motor Loading

The team reviewed the sizing of the auxiliary feedwater pump motor and found that the as-built rating of 400 horsepower at a 1.15 service factor would be exceeded with an assumed runout load of 490 horsepower as indicated in the loading calculation. The licensee could not locate correspondence from the motor manufacturer that was referenced in the loading calculation. However, the licensee had a manufacturer's performance test of the motor at 500 horsepower and a thermal stress calculation that indicated there would be an acceptable operating life at 500 horsepower. The failure of the licensee to clearly document the design bases for this pump was considered a design control weakness. The licensee initiated CR 200100972 to further evaluate this issue.

Alternate AC Supply Transformer Replacement Modification

The team reviewed safety evaluation 99-339-MD associated with the modification that replaced the GT-1 transformer. The team found that the safety evaluation failed to document that, while the transformer was non-safety related, it did in fact perform a function important-to-safety as the alternate ac power source. The modification package also lacked any references to important bases documents, including the calculations for the no-load tap setting. The team determined that these issues represented weaknesses in the licensee's design control process.

b.2 Service Water System

The service water system provides cooling to safety-related and non-safety-related components through two separate main supply headers. Flow to each header is provided by three pumps, each rated at 5,000 gallons per minute (gpm) at 220 feet of water discharge head. The pumps take suction from a common intake bay supplied from the Hudson River through two parallel traveling screens. In addition to the traveling screens, there are rotating strainers installed between the pump and the main headers to remove any particles or debris that could obstruct the flow paths through the components.

The main headers are aligned and designated as "essential" and "non-essential" headers. The essential header supplies cooling to all of the safety components except the component cooling water system heat exchangers. The non-essential header supplies the component cooling water system heat exchangers and the non-safety related components. The system design ensures that both headers will be able to perform their safety functions following any single active failure in the system.

In the event of a LOCA, operators are required to isolate the non-safety components from the non-essential header prior to entering the recirculation phase. The system can also be aligned for three header operation during which both the essential and non-essential headers supply only their respective safety-related components and the non-safety-related components are supplied by a separate river water system. The team reviewed the important design control aspects of the service water system. A number of performance issues and weaknesses were identified. The following observations are representative of the issues identified by the team.

Non-Essential Header Flow

The team identified that the licensee did not have a documented analysis or test that verified the ability of the service water system to supply the post-accident design flow to the component cooling water (CCW) heat exchangers. The licensee had a hydraulic model, Calculation PGI-00371, Rev. 0, which addressed the normal system lineup with the non-essential header supplying both the non-safety related components and the CCW heat exchangers. However, the analysis did not confirm the ability of the system to provide the required 2,500 gpm to each heat exchanger following an accident.

In response to this finding, the licensee used the flow model to evaluate the adequacy of flow to the heat exchangers under design basis accident conditions while assuming the service water pump was at the maximum degraded condition of 7%. This analysis showed that one of the CCW heat exchangers would receive 2,725 gpm and the other 3,054 gpm. Although this analysis was preliminary, it was determined that the service water system and CCW heat exchangers were operable.

The team found the licensee's immediate actions to address this issue, including the operability determination, to be acceptable. The system would have been able to perform its intended functions, as such, the team determined this issue did not have a credible impact on safety. Although this issue should be corrected, it constitutes a violation of minor significance that is not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy. This issue was entered into the corrective action program as CR 200100566.

Containment Fan Cooler Radiation Detector Analysis

The containment fan coolers were equipped with two radiation detectors in the service water system outlet flow paths to provide for monitoring effluent discharge paths for radioactivity that could be released from postulated accidents. This feature was incorporated into the design since the service water system pressure at locations inside the containment with the system in the incident mode alignment could be below the containment post-accident design pressure of 47 psig. These detectors were designed to actuate an alarm in the control room whenever their set points were exceeded. The team reviewed the detector set point calculation, RS-92, Rev. 2, to verify that it was appropriate to prevent exceeding the allowable accident radiation exposure limits specified by the regulations. The team found that the analysis had been performed for normal operating conditions assuming a total service water flow of approximately 16,000 gpm and a 600,000 gpm dilution flow from the circulating water system. The team noted therefore under design basis accident conditions the circulating water system may not be operating and that this assumption was non-conservative.

The licensee acknowledged this finding and performed another calculation that credited other conservative assumptions in the original calculation. The results of the revised calculation showed that the setpoint would have ensured that the accident exposures would have remained within regulatory limits, as such, the team determined this issue did not have a credible impact on safety. Although this issue should be corrected, it constitutes a violation of minor significance that is not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy. This issue was entered into the corrective action program as CR 200100879.

Essential Header Flow Verification

The team reviewed test procedure PT-R93, "Essential Service Water Header Flow Balance," Rev. 3, which performed an operational test of the essential service water header to verify that design flow was provided to all system components. The test is normally performed at the end of each refueling outage on the header that is aligned as the essential header and using the two lowest performing pumps to simulate worst case design basis accident conditions.

The team noted that during plant operation the system was realigned every six months to equalize the time each header functioned as the essential or non-essential header to more evenly distribute pump wear. However, the team also noted that there were no requirements in the test procedure, or other plant procedures, to ensure that the refueling interval testing would alternate between the two headers. The licensee was able to verify from operating records that both headers would function properly as the essential header. The team considered the lack of directions to alternate headers during testing to be a weakness with the flow testing procedure. The licensee initiated CR 200100511 to address this issue.

Strainer Blowdown Flow Safety Evaluation

The team reviewed test procedure PT-R93, "Essential Service Water Header Flow Balance" that was performed on August 24, 1998, following the replacement of all six service water pumps during 1997 and 1998 (Modification Number FMX-96-10376-M). During the test, the pumps were unable to deliver the design basis flows to all of the safety-related components and CR 199807295 was generated. In reviewing this issue the licensee discovered that the service water strainer blowdown flow was at approximately 600 gpm. The flow was adjusted to the required value of 225 ± 25 gpm and the test was re-performed successfully. The licensee then implemented a temporary facility change (98-222) to maintain the blowdown valves at the new throttled setting.

The team reviewed the documents associated with the temporary modification and determined that safety evaluation 98-322-EV, Rev. 2, did not clearly address the required strainer blowdown flows. The safety evaluation indicated that UFSAR Table 9.6-1 specified the minimum essential service water pump strainer blowdown flow as 100 gpm. The safety evaluation further identified that service water operability could be maintained with as little as 0 gpm and as much as 250 gpm without reconciling these differences with the UFSAR specified minimum flow. In addition, the strainer supplier recommended a blowdown flow rate of 2 to 3% of the through-strainer flow.[11] Calculation FFX-00713, Rev. 0, documented that the maximum through-flow was approximately 6,923 gpm. Using 2% of this value would yield a minimum allowable blowdown flow of 138 gpm. The calculation showed that with the throttle valves set at the new normal operating minimum flow of 200 gpm, the actual blowdown for worst case accident conditions would be 164 gpm, thereby meeting the vendor's recommended minimum flow. Therefore, the team determined that, although the 225 gpm ± 25 gpm setting for normal operating blowdown flow was adequate to maintain strainer operability, the safety evaluation was weak since this value had not been evaluated against the correct basis provided by the vendor (138 gpm). Additionally, the safety evaluation did not identify that the 100 gpm UFSAR minimum value was inadequate and would have incorrectly allowed 0 gpm blowdown flow. The licensee initiated CR 200101133 to address this concern.

b.3 General Design Control Observations

The team observed that there appeared to be a general difficulty in retrieving design basis information to support design control, testing and plant modification efforts. This issue had been previously identified and slow progress has been made to improve in this area. Additionally, this deficiency appeared to have had additional plant staff impact in that some inconsistencies in the review of certain technical issues were observed. The team noted that the licensee's business plan incorporated long-term initiatives to address this issue.

2. Procedure Quality

a. Inspection Scope

The team reviewed licensee event reports, NRC inspection reports, self-assessments, and condition reports to evaluate the extent that procedure quality has contributed to previous performance issues. The team reviewed a sample of procedures involved in performance problems to assess the technical adequacy of those procedures. The reviews included a verification that the procedure steps would achieve the required

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[11] The lowest blowdown flow would occur at maximum through-strainer flow conditions that would correspond to the lowest pump discharge pressure

system performance for normal, abnormal, remote shutdown and emergency operating conditions. Procedures were also reviewed to ensure the activity was accomplished within the plant design bases and regulatory requirements, and that procedure inadequacies did not exist that would cause an initiating event. The team reviewed maintenance procedures to ensure they were sufficient to perform the task, that they included independent quality verification of important attributes, and that they resulted in the task being performed consistent with the equipment vendor instructions and specifications. A sample of important vendor manuals were also reviewed to ensure they were complete and up-to date. The team reviewed the effectiveness of the licensee in ensuring current copies of documents were in place in the working files and that procedures affected by modifications or industry experience were updated in a timely manner.

The team reviewed the procedure change process to ensure it was in accordance with regulatory requirements and that appropriate personnel were involved in the development, review and approval of procedure changes. The team also reviewed the adequacy of controls for developing special or complex procedures to ensure that they were adequately validated and discussed with the plant personnel prior to implementation.

The team evaluated a sample of temporary procedure changes to ensure the changes were reviewed and approved in accordance with technical specification requirements and that the changes were consistent with the plant design and licensing bases. The team reviewed night orders, work orders and other documents to ensure that they did not result in uncontrolled procedure changes. The team also reviewed a sample of condition reports involving procedure quality to assess the effectiveness of corrective actions.

b. Findings

b.1 General Procedure Issues

Emergency Fuel Oil Transfer Procedure

The team reviewed AOI 27.3.1, "Emergency Fuel Oil Transfer Using the Trailer," Rev. 0, and found that the instructions for filling the trailer from the gas turbine fuel oil storage tank were deficient. This procedure is used to transfer fuel oil from the gas turbine fuel oil storage tank to replenish the fuel oil supply to the onsite emergency diesel generators. The procedure improperly directed the operator to connect the trailer fill hose to a drain line on the tank connection manifold rather than the fill line. Further, the precautions and limitations of the procedure stated that a flush of the trailer fuel lines may be required to remove ethylene glycol used for freeze protection. However, there were no instructions for performing this task and an operator interviewed by the team was unaware of how that particular flush evolution would be accomplished.

The team considered this issue to be of very low safety significance (Green) because the use of this procedure has never been required and would require minor changes to resolve the discrepancies. The failure to establish adequate procedure directions is considered an additional example of the non-cited violation of TS 6.8.1. This issue was entered into the corrective action program as CR 200100944 **(NCV 05000247/2001-002-003)**.

Temporary Procedure Change Process

Addendum VI to SAO 100, "Indian Point Station Procedure Policy," Rev. 3, described the process for implementing temporary procedure changes (TPCs). A TPC provides guidance for plant operations when existing plant procedures cannot be performed as written. The procedure stated that if not required for immediate operation of the plant, then the procedure shall be revised in accordance with SAO 100. The team reviewed TPC 00-0853 which was implemented to change alarm response procedure (ARP) AS-1 (Accident Assessment Panel 1; windows 5-4 and 6-4) because a temporary modification had disabled the associated alarm inputs. Since the alarm inputs had already been disabled and the change was not required for immediate operation of the plant, the team determined that a TPC was not the appropriate mechanism to change the procedure.

The team considered this issue to be of very low safety significance (Green) because the use of this TPC had minimal affect on plant operations. However, the failure to implement the requirements of SAO 100 for the use of TPCs is considered an additional example of the non-cited violation of TS 6.8.1. This issue was entered into the corrective action program as CR 200100866 **(NCV 05000247/2001-002-003)**.

Biennial Procedure Reviews

The team found that the licensee did not implement biennial procedure reviews in a manner consistent with existing administrative guidance. SAQ 100, "Indian Point Station Procedure Policy," Rev. 31, stated that biennial procedure reviews apply to documents which implement the regulations of 10 CFR 50, Appendix B. The procedure also stated that procedures which are used routinely (at least every two years), may be excluded from biennial reviews. Examples included calibration procedures, check-off lists (COL), maintenance procedures, plant operating procedures (POP), surveillance test procedures, system operating procedures (SOP), alarm response procedures (ARP), and abnormal operating instructions (AOI). The team found that the generation support department personnel interpreted this guidance to mean that all COLs, POPs, SOPs, ARPs, and AOIs are exempted from biennial procedure reviews. However, the team noted that there was no mechanism to identify procedures that are not used within a two year interval, and would therefore require a biennial review. The licensee researched the basis for this interpretation and found that the quality assurance program description stated that routine plant procedures that have not been used for two years shall be reviewed before use to determine if changes are necessary or desirable.

The failure to implement the SAO-100 procedure was not subjected to a cornerstone significance determination process. Although this issue should be corrected, it constitutes a violation of minor significance that is not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy. This issue was entered into the corrective action program as CR 200101449.

Incomplete Plant Operating Procedures

Operations Administrative Directive (OAD) 33, "Procedure Adherence and Use," Rev. 15, requires that operators verify the completion of steps in POPs. While reviewing a controlled procedure binder in the control room, the team identified that two POPs used for the recent plant startup (December 2000) contained several procedure steps that were not properly signed off. Specifically, POP 1.1, "Plant Restoration From Cold Shutdown to Hot Shutdown Conditions," Rev. 55, and POP 1.2, "Reactor Startup," Rev. 30, had numerous procedure steps that were apparently completed, but not initialed by licensed operators. This was considered to be an example of a minor violation of a failure to follow procedures since it appeared that the affected procedure steps had actually been performed and only the associated signatures were missing.

The failure to implement the OAD 33 procedure was not subjected to a cornerstone significance determination process. Although this issue should be corrected, it constitutes a violation of minor significance that is not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy.

Environmental Qualification Engineer Review of Work Orders

Station procedure SAO-430, "Environmental Qualification (EQ) Program," Section 2.2.12 required that the EQ engineer review all work packages on EQ equipment to assure that EQ considerations have been addressed. The team identified that this review was not performed for work order NP-99-06573. The team interviewed an EQ engineer, who stated that he was not aware of this procedure requirement and did not review all the completed work packages. The EQ engineer stated that he had reviewed and approved the general procedures that were used during the performance of the associated work. He also noted that he did not review all completed packages as a routine matter.

The team determined this issue did not have a credible impact on safety because there were no actual equipment deficiencies identified that were due to a lack of the EQ engineer review. Although this issue should be corrected, it constitutes a violation of minor significance that is not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy. This issue was entered into the corrective action program as CR 200100872.

Procedure Change Backlog

The team reviewed the backlog of operations procedure changes and noted there were about 650 Communications to Staff (CTS) items in the backlog. Many of the CTS items represented change requests for multiple procedures. Accordingly, the backlog of affected procedures requiring changes was substantially higher than 650. The team discussed the backlog with licensee personnel in the generation support department (operations procedure writers) and reviewed the klog and found that there was no formal mechanism to prioritize individual items. only prioritization occurred when CTS items were received, and judgement calls made as to whether immediate changes were necessary. The team identified a mber of items which should have received elevated priority. The following examp are representative of the team's findings in this area:

- CTS 98-1248, dated October 21, 1998, ferred to an Abnormal Operating Procedure (AOI 29.6) that implemented n operating principle that was inconsistent with current practice.

- CTS 99-0265, dated April 14, 1999, documented that a procedure check-off list (PCO 3.2) did not properly reposition two valves (residual heat removal heat exchanger motor-operated valves) following a safety injection.

- CTS 99-0535, dated July 28, 1999, identified that operations log sheet DSR-8M, associated with the gas turbine north and south fuel oil storage tanks, did not accurately reflect the proper minimum and normal tank levels.

The items listed above had been in the system for some time (nearly 2 ½ years for CTS 98-1248), and were more than minor editorial changes. The team considered the extent and age of the procedure change backlog to be a weakness in the maintenance of plant procedures. The team also noted that nearly all of the operations procedures had not received biennial reviews due to the misinterpretation of SAO 100 as discussed earlier, contributing to the time it takes for incorporating proposed changes by way of periodic procedure reviews and revisions.

Document Control

The team identified several minor document control issues associated with station procedures. For example, uncontrolled, and out-of-date copies of the post-run attachments of the diesel generator operating procedures (SOP 27.3.1.1, 27.3.1.2, and 27.3.1.3) were found in the EDG building. However, it did not appear that any out-of-date attachments had been used for obtaining and recording actual EDG data. The licensee promptly removed the uncontrolled attachments from the EDG building and initiated CR 200101382 to further review this issue.

The team also found that there was no mechanism or instruction to remove expired temporary operating instructions (TOI) from the controlled, active TOI binder located in the control room. Previously, the generation support supervisor removed outdated TOIs during routine tours. During the course of this inspection the team identified two expired TOIs that were still in the control room binder. The licensee promptly removed the expired TOIs from the control room binder and initiated CR 200101383 to further review these issues.

Procedure Use and Quality

The team determined that OAD 33, "Procedure Adherence and Use," Rev. 15, allowed broad flexibility for place keeping while using implementing procedures. The procedure recommended, but did not require, place keeping for continuous use procedures and operating instructions by placing a mark on the sign off line upon completion of the step (marks can be made in pencil and then erased). The team observed that during the power ascension on January 19, 2001, the status of ongoing evolutions was not apparent because place keeping within an active procedure was not consistently conducted. Although a panel walk down by the team did not identify any mis-positioned components or missed procedural steps, the team concluded that place keeping guidance and implementation was a weakness and made it difficult for operators to ascertain accurate system configurations.

The team also identified quality weaknesses associated with the procedure associated with scheduling, approving and assessing overtime. The team determined that procedure OAD 9, "Operations Section Organization," Rev. 27, did not institute maximum limits for excessive overtime. Rather, the procedure allowed workers to surpass the overtime limits for planned overtime with the advance approval of the assistant operations manager or higher. Further, excessive unplanned overtime required only the approval of the shift manager. The team also found that excessive overtime approvals did not require any assessment with respect to worker fitness for duty. The team reviewed overtime request and approval records, and did not identify instances where procedure requirements were violated. However, the team concluded that the procedure weaknesses represented the potential for inappropriate overtime hours being worked without including an assessment for fitness for duty concerns.

b.2 480 Vac and Emergency Diesel Generators Procedure Issues

Procedure Acceptance Criteria

The team reviewed various procedures associated with the 480 Vac and EDG systems and identified a number of performance issues. The following examples are representative of the team's findings in this area:

- The team noted that the EDG loading calculation assumed a frequency variation of +/- 0.5% based on the vendor setpoint tolerance. The team found that the safety injection with loss of off-site power surveillance test did not contain an acceptance criteria for EDG frequency. Based on the available design data the acceptance criterion should have been 60 Hz, +/- 0.3 Hz. Although the procedure did not specify an acceptance criterion, the team found that the results of the most recent testing performed during the 2000 outage confirmed that the frequency was within the values assumed in the calculation. The team also noted that the monthly EDG surveillance procedure and the 24 hour load test procedures specified an acceptance criteria tolerance of +/- 1.5 Hz which was not consistent with the loading calculation. In addition the team noted that the procedure for verifying the capacity of the EDGs did not include considerations of instrument uncertainty for the maximum loading (2300 kW) condition testing.

- The team reviewed control room operator log, DSR-1, and found that the minimum and maximum ranges specified for the instrument bus voltage were not bounded by the 118 Vac instrument power system voltage calculations.

- The team found that the vendor requirement to restrain the end cells of battery 23 had not been adequately translated into installation drawings.

- The team reviewed instrumentation and control preventive maintenance package for the undervoltage relays (ICPM 1741) for the 125 Vdc control power automatic transfer switches that supply EDG and 480 Vac switchgear control circuits. The team observed that the specified acceptance criteria of 100 +/- 2.0 volts was not consistent with the 125 Vdc voltage drop calculations FEX-00044-02 through FEX-00046-02 and FEX-00048-02 and would not ensure acceptable voltage at the dc loads prior to transfer.

The team determined these issues were of very low safety significance (Green) because none of the test results or operating data identified instances where equipment was operating outside of its design limits.

The team considered the failure of the licensee to include appropriate acceptance in the procedures and drawings to ensure activities have been satisfactorily accomplished to be a violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings." This violation is being treated as a non-cited violation consistent with Section VI.A of the NRC Enforcement Policy, issued May 1, 2000 (65FR25368). These issues were entered into the corrective action program as CRs 200100777, 200100531, 200100908, 200101576 and 200100750 **(NCV 05000247/2001-002-004)**.

b.3 Service Water System Procedure Issues

Service Water Header Pressure Analyses

The team reviewed Alarm Response Procedure (ARP) Window 4-6, "Service Water Hdr 21, 22, 23, 24, 25, 26 High/Low Press," Rev. 25, and DSR 1, "Unit 2 Central Control Room Log," Rev. 77, and found that the service water header low pressure alarm set point was 53 psig and the minimum acceptable header pressure in the control room log was 48 psig . The team found that the bases for the low pressure alarm set point was to ensure there would be adequate pressure to supply flow to the main turbine lube oil coolers. The control room log minimum appeared to have been based on the same requirement but without an elevation head correction that should have been considered. The licensee did not have an engineering analysis to demonstrate that all safety-related components would receive adequate flow if header pressure was controlled based on these limits.

The licensee performed a preliminary analysis assuming a header pressure of 53 psig and it was determined that acceptable flows would be delivered to the system. However, the control room log limit of 48 psig was found to be inadequate, and it was raised to 58 psig by Revision 78 during the inspection to provide a 5 psig margin above the set point.

This issue was of very low safety significance (Green) because the team did not identify any instances of operation at less than 53 psig.

The failure to properly translate the header pressure design bases into plant procedures is considered an additional example of the non-cited violation of 10 CFR 50, Appendix B, Criterion III, "Design Control." This issue was entered into the licensee's corrective action program as CRs 200100707 and 200101410 **(NCV 05000247/2001-002-002).**

Service Water Strainer Pit Flooding

The team reviewed the service water system for potential failure modes. It was noted that an event that requires the automatic starting of the service water pumps results in the potential for one of the service water pump vacuum breaker valves to fail open. These valves were located in the strainer room and would discharge directly into the space whose floor elevation (5' - 9") is several feet above normal Hudson River elevation. As a means of relieving an internal flood in the strainer pit, there was an eight inch drain line that discharges to the service water pump bay. This line included butterfly valve MD-501 that was maintained normally open by procedure COL 24.1.1, "Service Water and Closed Cooling Water Systems," Rev. 30.

Procedure AOI 28.0.4, "Plant Flooding-Conventional Side," Rev. 2 required closing MD-501 if river water level reached 5' - 8" to prevent flooding the room from the river (external flood). However, in this configuration, an internal flood from a failure, such as a vacuum breaker valve, could cause failure of all of the service water strainer motor operators. In response to this finding, the licensee initiated TPC 01-0039, dated January 24, 2001, which revised Procedure AOI 28.0.4. to continuously monitor the service water strainer pit for evidence of water in-leakage when the river water level reaches 5' - 8" and valve MD-501 is closed.

The team determined this issue was of very low risk significance (Green) because it involved the relatively low probability of a valve failure coupled with the low probability of an external flooding event.

The failure to properly translate the design bases into plant procedures is considered an additional example of the non-cited violation of 10 CFR 50, Appendix B, Criterion III, "Design Control." This issue was entered into the corrective action program as CR 200100878 **(NCV 05000247/2001-002-002).**

Service Water Strainer Pit Drain Check Valve

The team noted that in addition to the manually operated valve discussed above, the strainer room drain line also contained check valve, MD-500, located on the outboard side of the room in the service water pump bay. This valve had safety-related functions to close to prevent river water from entering the room in the event of high river level and to open to prevent internal strainer pit flooding. The valve has a counter-balanced disk designed to assure opening at the very low differential pressure that would be associated such flooding. The team discovered that valve MD-500 was not included in the plant testing program to verify its ability to fulfill its function. In response to this finding, the licensee took immediate action to demonstrate operability by manually cycling the valve from the full open to full closed position and observing that the valve opened with minimal effort and that there was no restriction in movement. The team considered this issue to be of very low safety significance (Green) because the valve was confirmed to be operable.

The failure to test the valve by periodically exercising it to its safety function position is considered a violation of 10 CFR 50.55a, "Codes and Standards," paragraph (f), "Inservice Testing Requirements." This violation is being treated as a non-cited violation consistent with Section VI.A of the NRC Enforcement Policy, issued May 1, 2000 (65FR25368). This issue was entered into the corrective action program as CR 200101466 **(NCV 05000247/2001-002-005)**.

Inservice Testing Procedure

The team reviewed the results of performance test PT-Q26A, Rev. 7, "21 Service Water Pump," performed on September 13, 2000, and found that the test acceptance criteria reflected the original Aurora pump criterion for operability of ≥ 253 feet differential pressure at 1,500 gpm. The team noted that the licensee had not revised the acceptance criteria following the replacement of the Aurora pumps with Johnston pumps in 1997 and 1998 to properly reflect the characteristics of the new pumps.

The licensee indicated that the basis for the acceptance criteria corresponded to the 10% degraded head point for the Aurora pumps as documented in Calculation PGI-00371, Rev. 00. The calculation demonstrated that, with 10% degradation, the Aurora pumps could still provide the required design basis flow to all of the safety-related components. Although the replacement Johnston pumps' vendor curves showed better performance than the Aurora pumps at the 1,500 gpm test point, they showed somewhat lower performance at the 5,000 gpm design point. The team noted that there were several missed opportunities for the licensee to discover and correct this discrepancy. Preliminary analyses by the licensee during the inspection showed that if the pumps had been allowed to degrade to the acceptance criteria values in this test procedure and the other service water pumps' corresponding IST procedures, their performances would not have been adequate to meet the design basis requirements.

The licensee evaluated this issue and determined that if individual pump performance remained above the 95% "alert" value in the test procedures, the pumps would be capable of providing the design basis flows. The licensee also confirmed that all of the pump actual test results remained above the alert values and as a result all were considered operable.

The system would have been able to perform its intended design functions, as such, the team determined this issue did not have a credible impact on safety. Although this issue should be corrected, it constitutes a violation of minor significance that is not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy. This issue was entered into the corrective action program as CR 200100170.

Service Water System Test Correction Factors

The team reviewed procedure PT-R93, "Essential Service Water Flow Balance," Rev. 3, and identified that the acceptance criteria for minimum flows to the various safety-related components had not been adjusted to compensate for several factors that could result in accident flows being less than design basis requirements. These factors included test instrument uncertainty, actual river levels versus the design basis minimum level, and the effect of pump strainers at design basis maximum differential pressure. The team also noted that the procedure directed the installation of temporary flow instrumentation without provisions to ensure consistent installation from one test to the next.

The licensee evaluated this issue and determined that, although the factors discussed above were not accounted for in the procedure, there were sufficient margins in the established flows to ensure that all components were operable. The team determined this issue did not have a credible impact on safety because the system was capable of performing its design function. Although this issue should be corrected, it constitutes a violation of minor significance that is not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy. This issue was entered into the corrective action program as CR 200100970.

Service Water Strainer Differential Pressure

The team reviewed design documents and operating and test procedures associated with the service water system strainers. Several procedures reflected a 9 psid design differential pressure limit across the strainer, and the strainer vendor manual documented 15 psid as the structural differential pressure limit. The team observed that during normal operation the flows in both the essential and non-essential headers were significantly lower than design basis accident flows due to flow throttling for temperature control. In an accident, however, the flow control valves would be either full open or bypassed in order to maximize heat removal. The differences between normal and accident flows were at the maximum in winter when throttling was maximized. An example of the difference was observed on February 5, 2001, when, with ice in the river, in three-header operation, the non-essential header flow was observed to be 3,250 gpm. The licensee had determined that the minimum accident flow would have been 5,780 gpm. Since the differential pressure is proportional to the square of the flow rate, for this particular day the strainer differential pressure would have increased by a factor of 3.2 for accident flow conditions. Since the actual differential pressure was 1.3 psid on this date the non-essential header would not have exceeded the design limit of 9.0 psid as a result of expected post-accident flow rates. However, higher normal strainer differential pressure, well below the procedure limit would result in strainer differential pressures in excess of the design limit or the vendor's structural limit after accident flow conditions were established. Therefore, these normal operation procedural limits were inadequate.

The team also identified a weakness in the alarm response procedure, "Service Water Strainers Trouble," Rev. 25, which had an alarm set point at 8.5 psid. The alarm response procedure stated, "IF differential pressure remains above 15 psid, PLACE standby service water pump in service and shutdown service water pump associated with affected strainer." This direction would allow strainer operation above 15 psid for a limited period, which was contrary to the vendor's direction and could cause permanent damage. The licensee had no basis or analysis to demonstrate that its operating limit was adequate to prevent exceeding the strainer structural limit of 15 psid for accident conditions.

The team determined that these issues did not have a credible impact on safety because the differential pressure across the strainers was low enough that the design limit would not have been challenged even at the higher accident flow rates. Although this issue should be corrected, it constitutes a violation of minor significance that is not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy. This issue was entered into the corrective action program as CR 200101404.

Service Water System Operating Procedure

The team found that procedure SOP 24.1, "Service Water System Operation," Rev. 40, contained a precaution which stated "Do not operate 23 and 24 SWPs simultaneously, if it can be avoided by existing operational considerations, due to the potential for creating vortexing in the service water bay." The procedure contained a similar note following step 4.1.1.

The ability of these pumps to operate together safely was further called into question by a July 1994 evaluation report on a 1-to-6.4 scale model hydraulic study of the service water pump intake. The study had been commissioned by the licensee in response to three pump failures that occurred over a period of a few weeks. The report indicated that there were severe sub-surface vortices for almost all pump combinations tested, and because of the large length-to-diameter ratio, the pump columns were sensitive to flow imbalances and fluctuations. The report also indicated that the hydraulic performance of the existing service water intake did not meet the acceptance criteria selected for the study because of adverse sub-surface vortices. The most severe vortexing was noted with pumps 2,3,4, and 6 operating.

The licensee initiated CR 200100912 to document and further review this issue and determined that the procedure statements associated with vortexing were added by a procedure change in response to the report. This change had been reviewed by the Station Nuclear Safety Committee on August 25, 1994. The reason stated in the meeting minutes for the changes was "only because of the possible long-term effects of potential vortexing." The licensee also informed the team that the pump configurations were in accordance with the Hydraulic Institute Standards and that the new Johnston pumps, installed in 1997 and 1998, were more heavily constructed than the original Aurora pumps. In addition, the three pump failures that precipitated the original study had ultimately been attributed to improper coupling assembly and foreign object ingestion. Based on this information, and the fact that during normal operation no excessive wear or vibration had been observed in any of the pumps, the licensee concluded that the precaution and note were unnecessary and planned to revise the procedure to remove the procedure statements. The team considered the failure of the licensee to correct the procedure to be a weakness, in that it unnecessarily restricted operators from certain operating configurations.

3. Equipment Performance

a. Inspection Scope

The team reviewed various maintenance related issues for the selected systems to determine the licensee's effectiveness in identifying the causes and extent of equipment problems as well as in developing and implementing corrective actions. Additionally, an assessment of the implementation of maintenance rule (MR) requirements was conducted. The team reviewed maintenance related documents, observed maintenance activities and conducted plant tours to assess the effectiveness of the licensee in entering maintenance issues into the corrective action program. The team also reviewed open condition reports and corrective maintenance work orders for the selected systems to assess their potential impact on operability.

The review also included surveillance and post-maintenance tests to assess the effectiveness of the licensee in specifying appropriate acceptance criteria and to verify the effectiveness of controls to restore equipment to operation following testing. The team also reviewed the scope of the calibration program for the selected systems and sampled system instrumentation loops to ensure instrumentation important to safety was included. Additionally, the team reviewed the preventive maintenance programs for the selected systems to assess the program adequacy and to verify that design document, vendor manual and generic communication information were incorporated into the maintenance program. Observations of in-progress maintenance and testing on the selected systems were conducted.

b. Findings

b.1 480 Vac and Emergency Diesel Generators

Gas Turbine Performance

The team reviewed the performance of the GTs that provide a backup electrical supply in the event of a station blackout condition and for alternate safe shutdown in the event of a fire. Based on these functions, the GTs were included within the scope of the licensee's 10 CFR 50.65 maintenance rule program. The licensee established an availability goal of 80% (less than 3,504 hours unavailability in a 24 month period) and a reliability goal of less than 2 maintenance preventable functional failures (MPFF) and zero repetitive MPFF's in a 24 month period. The team noted that the GTs had not been meeting these goals since 1995. In addition, a review of the performance history documented in the existing site maintenance rule basis document for the gas turbines indicated that none of the goals (availability and reliability) were being met at that time and that the GTs remained classified as (a)(1) under the MR.

The team reviewed the system health report for the gas turbines for the 4th quarter of 2000 and noted that GT-2 was still not meeting the goals for availability and none of the GTs were meeting the goal for reliability due to numerous failures. Discussions with licensee personnel indicated that several outstanding issues impacted the station's ability to adequately maintain the GTs. For example, the preventive maintenance program lacked specificity and rigor and there was poor design information, such as electrical schematics and mechanical drawings available to the staff. The team also noted that there was a significant decline in performance of the GTs during the 4th quarter of 2000 that included several repetitive maintenance preventable failures. The licensee attributed these problems, in part, to a lack of preventive maintenance during the 2000 steam generator replacement outage.

The team determined these issue were of very low safety significance (Green) because the technical specification requires only one GT to be operable. In addition, the team did an independent calculation of the change in core damage probability associated with the current unavailability of GT-2 for an estimated repair length of 60 days and determined that the risk increase to be within the very low safety significance band (<1E-6).

The failure of the licensee to effectively implement corrective actions to ensure that the established maintenance rule goals would be met is considered a violation of 10 CFR 50.65 (a)(1). This violation of 10 CFR 50.65(a)(1) is being treated as a non-cited violation (**EA-01-055**), consistent with Section VI.A of the NRC Enforcement Policy, issued on May 1, 2000 (65FR25368). This issue was entered in the corrective action program as CR200100233 **(NCV 05000247/2001-002-006).**

480 Vac and Emergency Diesel Generator Performance

The team reviewed the maintenance history, equipment performance and maintenance rule program aspects associated with the emergency diesel generators and 480 Vac systems. The review focused on system performance in the post-1999 time period since extensive follow-up was performed following the August 1999 loss of offsite power and reactor trip. The team determined that while minor equipment problems had been observed, the overall performance of the systems had been adequate.

b.2 Service Water

Instrumentation and Controls Preventive Maintenance

The team reviewed several EDG instrument calibrations which were performed using instrumentation and controls preventive maintenance (ICPM) packages and found that in several cases the entire instrumentation circuit was not tested. For example, several packages[12] were completed without control power available to test the resultant circuit actuations. The specified sensors were tested through verification of relay contacts, but in some cases, the resultant actuations such as alarm and annunciation were not tested. The incomplete PMs referenced a condition report, however, the inability to test the specific condition was not included in the report.

The team also reviewed ICPM package 1350, Rev. 3, that tested instrumentation associated with service water flow control valves FCV 1176 and 1176A. These valves control the flow of cooling water from the EDGs. The control circuitry includes contacts to open the valves if a high jacket water or high lube oil temperature is sensed on an operating EDG. Although the ICPM checked and calibrated the setpoint of the temperature switches, there was no testing to verify that the associated relay and circuitry would open the valves on a high temperature condition. The team reviewed CR 199900576 which documented that the licensee had identified this same issue during the development of the component function matrix. The CR recommended testing to improve plant reliability but also stated the devices are not important to nuclear safety since the valves also open on a safety injection signal, which was routinely tested. However, the team noted that a single failure of flow control instrumentation for the valves could result in a close signal to both valves. Consequently, during operation of the EDGs without the presence of a safety injection signal, the high temperature circuitry was important to nuclear safety since it was necessary to prevent the loss of the emergency power safety function due to a single failure that could isolate all cooling water to the diesels. The licensee reviewed the issue further and concluded that the high temperature circuitry was not tested but also identified a previous modification

[12] PM Packages No. 1779-1, Diesel Generator 22 Lube Oil System, Rev. 2, PM package No. 1778-1, Diesel Generator Jacket Water System, Rev. 2, and PM package no. 1776, Diesel Generator 21 Fuel Oil System, Rev. 4

which added a mechanical stop to prevent full closure of the 1176A valve. While the purpose of this modification was to provide sufficient flow velocity to prevent fouling of the system, the licensee was also able to show that with the valve closed to the mechanical stop, adequate flow would be provided to the EDGs.

The team discussed these findings with licensee personnel and found that the station had recognized the need to improve the ICPM program and developed a program to convert the packages to procedures that used the surveillance test procedure format. Further, the team also noted that the ICPM program did not include all of the various safety and non-safety related instruments. There were approximately 650 existing ICPM packages requiring action and approximately 600 instruments not included in the ICPM program scope. As a result of the team raising this issue, the licensee subsequently reviewed a random sample of approximately 100 ICPMs to assess the adequacy of testing and identified 7 additional discrepancies. Based on these results, the licensee completed a review of all safety-related instrumentation ICPM packages and verified that there were no concerns with equipment operability due to inadequate testing.

The team determined this issue did not have a credible impact on safety because none of the deficiencies affected any component operability. Although this issue should be corrected, it constitutes a violation of minor significance that is not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy. This issue was entered into the corrective action program as CRs 200100974, 200101411, 200101467 and 200101468.

Service Water Pump and Motor Replacement

Between July 1997 and January 1998, all six service water Aurora pumps were replaced with Johnston pumps. Also, in 2000, the motors for pumps 21 and 24 were replaced. Each pump and motor replacement was followed by a post-maintenance test (PMT) in accordance with procedure TP-SQ-11.016, "Post Maintenance Test Program." The test involved a performance of the applicable quarterly test procedure, PT-Q26A - F, which involved a single point (low-flow, high-head) pump test. The team reviewed this guidance and found it to be in accordance with the licensee's commitments to ASME OM Part 6. The team also noted that following a pump or pump motor disassembly or replacement, the procedure requires a single point capacity test for flow verification as well as checks for vibration levels, operating temperature and fluid leakage. The team further observed, that subsequent to the pump replacements, the pump vendor identified a nonconformance associated with pump performance curves in that the curves could be in error up to 3.8% due to a failure to take into account instrument uncertainties during the development of the curves. Capacity testing at more than one point would have increased the potential for identifying this discrepancy since at the test point (1,500 gpm at 307 ft) the original curve had negligible deviation from the curves that were subsequently adjusted for the potential error. Although the testing was in accordance with the station procedure, the team considered flow testing at a single point to be a weakness in the test program since it may not be adequate to verify pump performance over the full range of flows that would be experienced during normal and post-accident operation.

Emergency Diesel Generator Heat Exchanger Flow Measurements

The team reviewed PT-R93, "Essential Service Water Header Flow Balance," performed in July 2000, and noted that the procedure did not have an acceptance criteria for the flow through the individual emergency diesel generators. Instead, it contained an acceptance criteria for the combined flow of 1,200 gpm for all 3 EDGs. The team found that the licensee had previously initiated CR 200005646 to address deficiencies associated with the test and included the issue described above. The licensee had determined that, based on factors such as regular inspection and cleaning of the heat exchangers and the similarity of the parallel flow paths to the EDGs, that there was adequate assurance that each EDG had adequate flow. The team considered this item to be another example of testing program weaknesses. The licensee planned to improve the test procedure.

Motor Operated Valve "T" Drains

During a plant walkdown the team noted that the "T" drains for motor operated valve (MOV) SWN-44-4A were not installed at the low point of the motor as required. The licensee reviewed this condition and determined that the environmental qualification of this particular valve was not affected based on the expected post-accident pressure and temperature conditions. However, the licensee also found that the maintenance procedures for the MOVs were weak in that they did not include directions to ensure the drains were installed at the low point and the procedure did not specify the number of drains to be installed. CR 200101007 was initiated to further evaluate this issue.

Service Water System Performance

The team reviewed the maintenance history, equipment performance and maintenance rule program aspects associated with the service water system. The team determined that while minor equipment problems had been observed, the overall performance of the system had been adequate and that adequate flows would be delivered to important system components.

4. Configuration Control

a. Inspection Scope

The team reviewed operability evaluations performed for the selected systems to assess their thoroughness, technical adequacy and to ensure that they did not result in plant operation outside of the design and licensing bases. The team reviewed temporary modifications for the systems to evaluate whether they had been reviewed and approved by the appropriate personnel and that controls were in place to limit the duration of the installation. Additionally, the team reviewed whether procedures and drawings were updated where necessary. The assessment included a review of selected configuration control issues from the corrective action program data base to assess the adequacy of the licensee's problem identification and resolution program.

The team performed detailed walkdowns of the systems to determine whether the as-built configurations and lineups were consistent with plant procedures, drawings, UFSAR and design basis documents. The team also assessed the material condition of the system and support system components to determine if any conditions existed that could adversely impact operability. Additionally, the team performed a verification that system components were properly labeled, cooled and lubricated to support the performance of their design function requirements and that power was available and correctly aligned to support automatic activations where appropriate. The team also reviewed selected system instrumentation to verify it was properly installed and calibrated. The team reviewed overall cleanliness, control of ignition sources and flammable material in the vicinity of the systems and control of temporary storage of materials and equipment to determine whether they impacted equipment operation or access by plant operators.

The team reviewed the backlog of corrective and preventive maintenance for the systems to assess whether any items or combinations thereof could impact equipment operability. The team assessed the process for controlling maintenance, including the assessment of risk and the inclusion of emergent work into the schedule. A sample of tag-outs were reviewed to assess the adequacy of the configuration for the planned work and the methods for controlling equipment status changes, including the control of entry and exit from Technical Specification (TS) action statements. A walkdown was performed to independently verify a sample of tag placements and component alignments. Long term tag-outs, control room deficiencies, operator work-arounds and equipment deficiencies were reviewed to assess the significance of these conditions. The review included an assessment of work control procedures for the control of hot work (welding, open flame, etc.) and the control of scaffolding in the vicinity of safety related and important operating equipment. The team also reviewed the process for performing maintenance using the Fix-It Now (FIN) team.

The team reviewed primary and secondary system chemistry controls to assess their effectiveness in preventing degradation of the reactor coolant system (RCS) pressure boundary. The inspection included a review of chemical analyses records, trends of water quality data and corrective actions taken when chemical variables exceeded established limits. The adequacy of the licensee's measures to prevent the introduction of chemical contaminants into the primary and secondary coolant water and measures to detect any inadvertent contamination were also reviewed.

The team further assessed the adequacy of the fission product barriers by verifying a selected portion of the containment isolation lineup, including attributes such as component positions and power availability to ensure that components were properly controlled in accordance with Technical Specifications. The team also reviewed a reactor coolant system leak rate determination and reviewed procedures for ensuring the containment atmosphere met design basis assumptions.

The team reviewed the operating performance history for the selected systems and components and compared the out-of-service time to the assumed time in the individual plant examination. The team also reviewed the licensee's efforts to integrate preventive and corrective maintenance to minimize unavailability.

The team performed a walkdown of the containment spray system to independently verify the system configuration. Temporary modifications for the system were also reviewed to ensure proper installation in accordance with design information.

b. <u>Findings</u>

b.1 480 Vac and Emergency Diesel Generators

Control of Setpoints for Delta - Temperature Annunciation

During power ascension, the control room alarm for abnormal Delta-Temperature (Delta-T) between reactor coolant loops was received. The operators took appropriate actions as specified in the alarm response procedure for the deviation. However, it was determined that the actual physical reactor coolant temperature differential was below the setpoint for the alarm. The operators stopped the power increase and contacted maintenance to investigate the alarm. Upon further investigation, it was determined that the setpoint for the delta-T deviation loop 2 channel was incorrect which resulted in the alarm actuating prematurely. Additionally, the preventive maintenance procedure used to calibrate the instrument contained incorrect setpoint values.

Although the setpoints were incorrect for the delta-T deviation alarm, there was minimal safety significance associated with the event. The delta-T deviation alarm prompts the operators to investigate a possible core flux distribution or instrument problem and is not part of any protective circuitry. Accordingly, this issue was determined to have very low safety significance (Green). The licensee took corrective actions which included adjustment of the setpoint to the proper setting.

The team considered the failure to properly adjust the setpoints of the Delta-Temperature circuitry as required by procedure an additional example of the non-cited violation of TS 6.8.1. This issue was entered into the corrective action program as CR 200100669 **(NCV 05000247/2001-002-003).**

Oil Pads in EDG Instrumentation Cabinet

The team identified two oil absorbent pads inside the emergency diesel generator (EDG) 21 instrumentation cabinet. The system engineer indicated that the pads were used on October 26, 2000, to contain the oil from a leaking oil pressure switch (PC-5440-S). The leak had been repaired but the pads were not removed. The oil soaked pads represented an ignition hazard due to the presence of 120 volt direct current. Several components in the cabinet could fail in the presence of heat and flame and result in diesel unavailability. Technical Specification 6.8.1 specifies that written procedures shall be implemented which cover the Fire Protection Program. Portions of the Fire Protection Program are implemented at Indian Point 2 by procedure SAO-701, "Control of Combustibles and Transient Fire Load," Rev. 8. The finding was determined to have very low safety significance (Green) because the issue did not represent a fire impairment, degradation of a fire protection feature, or a reduction in defense in depth.

The team considered the failure to remove the oil pads from EDG 21 gauge panel as required by procedure SAO-701 an additional example of the non-cited violation of TS 6.8.1. This issue was entered into the corrective action program as CR 200101448 **(NCV 05000247/2001-002-003).**

Drawing Errors

The team identified a number of minor configuration control errors related to component labeling and drawing discrepancies. Representative examples included:

- Drawing 9321-F-4046, "Diesel Generator Building Floor Drains & Ventilation Control Air Piping Plans and Sections," did not show the 6th building exhaust fan which had been added to the system. Additionally, another drawing had mislabeled the exhaust fan.

- Drawing 243683, Revision 2, showed SOV-7215 as a two way solenoid valve whereas the installed valve was a three way solenoid valve. The installed valve also did not match the bill of materials listed on the drawing.

- Drawing 9321-F-3278, for heat trace panel 21, was not updated following a modification.

- Loop diagram 252686 had an error involving the depicted valve type.

- One line diagram 208088 contained an error associated with the service water cable size.

The team considered these to reflect weaknesses in the area of drawing controls.

Temporary Power Cord

The team discovered that an uncontrolled, temporary power cord was plugged into an energized power source outside the EDG building and fed under the building door to power a maintenance air compressor. The compressor had not been used recently nor had the power cord been disconnected as specified by Station Administrative Order (SAO) 218, "Housekeeping Policy," Rev. 14. The temporary power cord was disconnected and CR 2900100786 was initiated to document this issue. The team concluded that this represented a weakness in the configuration control process.

Control of Licensing Basis Information

The team identified examples of incomplete or inaccurate licensing basis information. It was noted that Technical Specification 4.6.D.1 indicated the gas turbine generator would provide a minimum of 750 kilowatts (KW) for alternate safe shutdown loads. The team questioned the basis for the 750 KW load rating and determined from a review of the station's fire protection analysis that in fact, approximately 1,700 KW was required. The system engineer concurred that TS 4.6.D.1 appeared incorrect and initiated several CRs[13] to prompt further engineering investigation. This apparent Technical Specification discrepancy did not appear to be a safety concern since the GT load ratings were well above (> 10,000 KW) the necessary loads required for the plant to achieve a cold shutdown condition. In addition, they are tested monthly in accordance with station test procedures PT-M38A, B & C.

The team also identified incomplete licensing basis information associated with UFSAR Section 8.2.3.2. This section of the analysis dealt with the emergency fuel supply for the diesels and stated that "19,000 gal of storage ensures that at least two diesels can operate to power the minimum engineered safeguards load for 73 hr." However, unless

[13] CRs 200101386, 20011386, and 200101486

one diesel fails following a demand signal, all three EDG's would start and load their respective emergency buses. The calculation which determined the minimum EDG operation of 73 hours did not account for the fuel consumption from the third diesel. The team estimated that if all three diesels were operating, the fuel storage capacity would provide for only approximately 50 hours of diesel operation. The licensee initiated CR 200100782 to revise this incomplete UFSAR description and include the fuel supply given all three EDGs are operating. This issue did not present a safety concern as adequate fuel monitoring capability was available to the operators when the EDGs are operating and an adequate supply of fuel oil was available on-site with the necessary transfer capability.

b.2 Service Water System

Systems not Operated as Designed

The team identified equipment related to the service water system in which the automatic controls were degraded or long-term temporary fixes were installed. For example, following the replacement of the service water pumps, the blowdown flow for the strainers had to be reduced to ensure sufficient flow was provided to the service water loads. This was accomplished using TFC 98-222 to throttle the blowdown stop valves. The team noted that although these were ball valves which are not designed to be used as throttle valves, a permanent modification has not yet been implemented and the temporary change has remained installed since 1998.

The team also found that the EDG temperature control valves, FCV-1176 and FCV-1176A, are usually operated in automatic but are periodically placed in manual when one or more of the valves begin to hunt. This problem was documented in CR 200006702 but had not been resolved at the time of the inspection. This issue was determined to be of minor safety significance because at the time of the inspection one valve was in manual and the other was in automatic and in the event of a high temperature condition on any diesel generator or a safety injection signal the valves receive open signals which override the automatic controls.

The team also reviewed a similar control problem associated with two automatic control valves which control service water flow to the hydrogen cooler. Pressure control valve PCV-1180 is on the inlet side of the hydrogen cooler and limits flow such that service water pressure inside the cooler is always below the hydrogen pressure. Temperature control valve, TCV-1101, is on the outlet of the hydrogen cooler and automatically controls the outlet temperature of the cooler. The team found that the temperature control valve for the generator hydrogen cooler could not always be operated in the automatic mode because of interactions between the two valves.

The team noted an additional example of problems with automatic control of the service water traveling screen 27. When the screen was actuated by the automatic control system the control room incorrectly received a loss of spray water pressure alarm. This condition was created when valve FCV-6983 and its actuator were replaced with a different model valve and actuator. The newly installed valve operated slower than the previous valve, resulting in the alarm circuitry actuating just prior to system pressure being reached. Although operation of the screen system was not affected, the change has resulted in unnecessary nuisance alarms.

These are examples of operating with known degraded conditions for extended periods of time. While these issues are individually of very low safety significance, they present a burden to operators.

EDG Temporary Facility Change

The team identified several administrative deficiencies associated with TFC 99-083 installed on the EDGs including: a caution tag on valve SWN 77-6 with an incorrect tag number, an unsigned TFC tag on valve SWN 77-6, absence of a date and signature on the deficiency tag on the 22 EDG raw water pressure gauge, and absence of a date on the tag hanging on valve SWN 77-5. In addition, TPC 2000-0055 was incorporated into SOP 27.3.1.3, "23 Emergency Diesel Generator Manual Operation," but was not documented on the TFC. These issues were of minor significance and did not affect the safe operation of the plant.

Drawing and Document Discrepancies

The team identified UFSAR descriptions of radiation monitoring on the service water outlets from the containment fan coolers that did not accurately describe the arrangement of these devices. UFSAR Section 6.4.2.1.4 stated that the cooling water discharge from the cooling coils flows to the discharge canal and is monitored for radioactivity by routing a small bypass flow from each through a common radiation monitor. The team noted that the bypass flow did not come from the discharge of each cooling coil, but rather from common headers into which coolers discharged, and the bypass flow was monitored by two monitors and not one common monitor. Also, UFSAR Section 9.6.1.2 stated that the ventilation cooler and motor cooler discharge lines will be monitored by routing a small bypass flow from each through redundant radiation monitors. The team noted that the bypass flow did not come from the discharge of each cooling coil, but rather from common headers into which the coolers discharged. The licensee initiated CR 200100849 to address these inaccuracies.

The team also identified that service water system drawing, 9321-2722 Rev. 99, showed valve SWN-68-1 which could not be located in the plant. The licensee investigated this discrepancy and determined that this valve was associated with a service water flow instrument that was retired in place in 1991 when an improved flow instrument was installed. In 1993, a generic piping modification removed this valve and capped the elbow tap. However, this modification was never updated in the system drawings. The licensee initiated CR 200100910 to address this deficiency.

The team also identified six strainer drain valves which were not reflected on the system drawings. The licensee investigated this issue and determined that these drain lines had been installed by a modification in August 2000. The control room did not receive an as-built marked-up version of the drawing until January 23, 2001, after the team questioned the condition of these valves. The licensee initiated CRs 200101483 and 200101488 to address this issue.

The team noted discrepancies in the Service Water System Lineup, COL 24.1.1. The check off list required that the seal[14] numbers on the strainer blowdown stop valves be checked by comparing the number on the seal with the number recorded in the most recent documentation of acceptable flow. During the system walkdown, the team noted that the seals installed on these valves did not contain specific identification numbers. The licensee indicated numbered seals are no longer used at the plant, however, the plant procedures had not been updated to reflect this fact. The team also noted that the last service water system lineup performed on December 21, 2000, did not identify the problem with a lack of numbers on the seals. The licensee initiated CR 200100923 to address this issue. The team also noted that COL 24.1.1 had two entries for a valve identified as "Service Water Cooling Water to R-46, R-49 and R-53 (Header 4) Stop" labeled with two different numbers, once as SWN-5 and the other time as SWN-56. The team verified with that both situations referred to the same valve, and that the number should have read SWN-56 in both cases. The team determined this issue did not have a credible impact on safety. Although this issue should be corrected, it constitutes a violation of minor significance that is not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policies. However, they demonstrate a lack of attention to detail on the part of the licensee staff and weaknesses in the control of design drawings and documents. The licensee initiated CR 200100774 to address this issue.

b.3 Findings - Fission Product Barrier Control

During the walkdown of the containment spray system the team noted that a portion of the suction pipe between the refueling water storage tank and the containment spray pumps was outside of the building and above grade. The team reviewed the adequacy of the freeze protection on the exposed piping and noted that there could be an undetected loss of freeze protection in the event the neutral wire connection was lost. Further, it was determined that minimal measures were in place to ensure the continued

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[14] The seal are installed to ensure that the valves remain in the required throttled positions

reliability and availability of the freeze protection circuitry for this portion of the system. The licensee acknowledged this potential and initiated CR200100858 to document and further review this issue.

B. Chemistry Controls

a. Inspection Scope

The team reviewed primary and secondary system chemistry controls to assess their effectiveness in preventing degradation of the reactor coolant system (RCS) pressure boundary. The inspection included a review of chemical analyses records, trends of water quality data and corrective actions taken when chemical variables exceeded established limits.

A confirmatory measurements radio-chemistry inspection was performed to review the licensee's ability to measure radioactivity in plant systems and effluent samples and the ability to demonstrate the acceptability of analytical results through implementation of a laboratory quality assurance program. Water, charcoal cartridge, (particulate) filter, and gas samples were analyzed by both the licensee and by the NRC Region I Mobile Radiological Measurements Laboratory.

Inspection of this area included a review of the licensee's internal laboratory quality program as described in Procedure No. CH-SQ-13.003, "Quality Assurance/Quality Control of Analyses," Rev. 5. This procedure, as well as other licensee procedures, provided for the control of analytical results through a number of mechanisms including: definition of personnel responsibilities, the use of traceable standards, implementation of instrument control checks, and participation in an interlaboratory quality control program.

b. Findings

During a review of the secondary chemistry data sheets in the control room, the team found an out-of-specification reading for feedwater hydrazine concentration that was not circled in red and not noted by the control room supervisor who had reviewed the logs. It was later determined that the actual value was not out-of-specification due to the fact that the limits had been recently changed by a temporary procedure change. In reviewing this issue the team found that TPC 01-0015 changed the acceptable hydrazine requirement in the chemistry administrative procedure to greater than 100 ppb. This change was carried into the control room chemistry log book but not into the chemistry administrative procedure or the watch chemist logs. As a result, the apparent out-of-specification (70 ppb) readings were not red circled or noted in the control room log book since the watch chemist's log sheet still indicated that the 70 ppb reading was acceptable. Further, the team's review of watch chemist logs showed numerous red circled readings. These included: in-line instruments out-of-service, in-line sample temperatures high, low hydrazine levels and low primary lithium concentrations. The team noted that there were no condition reports written to document these out-of- specification conditions. The team determined that these issues were of minimal safety significance; the out-of-specification conditions were of short duration and properly corrected. These issues represented minor violations of regulatory requirements.

The team conducted a comparison of the split sample results of various radio-chemistry samples. It was concluded that the licensee was able to accurately quantify concentrations of radioactive material in effluent and in-plant samples. The comparisons for the sample results indicated that all of the measurements were in agreement under the criteria for comparing results. The comparison data associated with the sampling activities are presented in Table I.

The licensee's primary and secondary chemistry procedures and analysis were found to be satisfactory and in accordance with the Electric Power Research Institute guidance. The team concluded that the licensee had an adequate internal laboratory quality assurance and quality control program and had appropriately participated in an acceptable interlaboratory program.

C. Human Performance

1. Organizational Practices

a. Inspection Scope

The team conducted in excess of 50 hours of control room observations, including a 24 hour continuous coverage period. Operators were observed performing evolutions, tests, and responding to annunciators. The team also accompanied operators during the performance of operator rounds. Written logs and shift status reports or updates were reviewed for completeness and accuracy to ensure they provided sufficient detail.

Additionally, the team observed the performance of six operating crews in the simulator (on-shift, initial license, and staff crews). The team evaluated shift communications and turnover, operator knowledge of plant conditions and activities in progress, and operator response to alarms.

The team observed scheduled and non-scheduled maintenance activities, the control room command function, and implementation of compensatory measures as required by risk and safety evaluations. The team observed pre-job and pre-evolution briefings, evaluated communication between operations and other departments, and interviewed operators to determine their awareness and understanding of ongoing activities.

Activities of field support supervisors and nuclear plant operators were observed to determine whether operations personnel were knowledgeable about the status of systems, structures, and components, equipment performance, and the impact of ongoing work activities.

b. Findings

The team determined that a resource limitation existed with respect to the number of licensed operators. There were 6 shift managers one of whom is the assistant operations manager, 5 control room supervisors, and 5 watch engineers at the site. The team noted that this level of staffing had the potential to increase the amount of planned and unplanned overtime deviations. In fact, several instances of planned as well as unplanned deviations from the administrative overtime limits were observed since January 1st, 2001. The team noted that the licensee had initiated efforts to requalify several individuals holding inactive senior operator licenses. Additionally, nine individuals were currently enrolled in a senior operator licensing class and were expected to be evaluated for operating licenses by the NRC in July 2001. Additional licensing classes were scheduled to start in April 2001 and early in 2002.

The team reviewed a number of self-assessments and third party assessments of operations training. It was observed that these assessments were self-critical and had identified a number of training weaknesses. The team concluded that although a number of significant challenges existed with respect to the operator training program, that the licensee had recognized these challenges and had initiated measures to improve the overall training program. However, progress in this area has been slow and the effectiveness of these measures had yet to be realized.

The team observed a weakness with respect to management reinforcement of standards associated with the use of plant operating procedures. It was observed that during the preparations to reduce power to repair a leak on the heater drain pump, that plant management believed that the abnormal operating instruction (AOI 21.1.1) for the loss of the drain pump provided an adequate basis for the ultimate power level to be achieved. However, the AOI guidance conflicted with the more conservative guidance contained in plant operating procedure (POP) 3.1 which governed a plant load decrease.[15] The team observed control room discussions concerning which procedure should be used. Ultimately, after discussions with the Chief Nuclear Officer, the licensee determined that the power should be reduced in accordance with POP 3.1. However, a night order written that evening to the plant operators suggested that it would have been acceptable to have terminated the load reduction at 900 MW. The team determined that the guidance in the abnormal operating instructions, while suggesting that an acceptable basis for the power level may exist at 900 MW, did not necessarily establish the most desirable plant conditions to conduct corrective maintenance. Rather, the abnormal operating instructions were written to place the plant in a safe and stable configuration from which additional actions and assessment can be made. The team determined that the management standards regarding the use and adherence to procedures were weak in this case. The team noted an additional weakness in that the planning and discussions associated with this evolution were concentrated in the control room versus being planned by engineering and maintenance with operations support.

In general, the command and control function in the control room was adequate. However, the team observed several problems in this area. For example, the team noted in one instance that shift management had difficulty prioritizing actions in response to multiple, simultaneous alarms. In another instance, the operating crew was not aware

[15] AOI 21.1.1 would lead to a power level of 900 MW whereas POP 3.1 would have led to a level of 650 MW

of post-maintenance testing being conducted. Additionally, during the start of a main boiler feedwater pump (MBFP), the control room supervisor exhibited weak operational oversight of activities when he became directly involved in the restart of the pump rather than directing overall activities.

On one occasion, the control room operators and maintenance personnel did not display conservative actions following erratic behavior of the main feedwater pump control system. On January 21, 2001, the 'B' MBFP flow oscillated and the 'A' MBFP control system and pump responded accordingly. Operators promptly placed the 'B' MBFP control system in manual, which stabilized the flow oscillations. On January 22, the team observed that the 'B' MBFP had been returned to automatic. When questioned, the operators stated that no troubleshooting work had been performed and the suspected control system inputs had not been instrumented. The operators felt that if the flow oscillations occurred again, they would be able to quickly respond. A second flow oscillation occurred the evening of January 23. System traces were not available to evaluate the pump's response or to positively identify the cause of the flow oscillation. Subsequent troubleshooting isolated the suspected channel but the failure to instrument the channel represented a missed opportunity and demonstrated the willingness of operators to accept a potential operational challenge.

During the 24 hour continuous control room coverage, a period when the plant was engaged in power ascension activities, minimal senior station management presence was observed in the control room. Lack of management involvement in control room activities had been identified in previous licensee self-assessments and NRC inspection efforts.

The team also observed during the control room observations that maintenance personnel suggested a potentially disadvantageous approach to repairing a service water leak on the generator hydrogen cooler. The recommended approach involved introducing a vulnerability of losing the only inservice hydrogen cooler, increasing the probability of a plant shutdown. After discussions between the operating crew and maintenance personnel, the crew conservatively determined that the alternate cooler should be placed in service prior to maintenance. The control room staff effectively managed the risk of the evolution. However, poor maintenance planning in this instance resulted in additional burden on the control room operating crew.

Problems in control room logkeeping were noted for the 1999 reactor trip with complications, the 2000 tube failure, the fall 2000 operator requalification inspection, and the recent turbine trip. It was again noted during the continuous control room coverage that the operating logs in the control room do not consistently contain an appropriate level of detail to allow a reconstruction of many operational activities.

In most cases, licensed operators were observed to use self-checking and peer checking in both the simulator and the control room. However, one instance was noted in which the balance of plant operator did not self-check during a valve manipulation. Instead of waiting for the valve to fully stroke, the operator walked away while the valve was in mid-stroke.

On one occasion, weak teamwork was exhibited by a shift crew when repeated alarms for a failed main steam line radiation monitor occurred simultaneously with repeated alarms associated with an in-progress post-maintenance test. These simultaneous alarms challenged the crew's effectiveness in prioritizing their actions to respond. In addition, the performance of the post-maintenance testing was not communicated to the crew, further contributing to the confusion. Also during this period, the crew was visibly frustrated with respect to a separate issue related to the power ascension ramp rate. The reactor engineer's instructions were to increase power at a maximum rate of 3% per hour. Some crew members wanted to be more conservative and proceed at a rate of about 2%. The shift manager, however, informed the crew that they were being overly conservative and the reactor engineer's instructions were meant to be an average ramp rate versus a maximum rate. This disagreement was eventually settled and discussed during the pre-evolution brief for the power ascension.

Several instances of a weak accounting of the status of ongoing evolutions were observed. For example, it was noted that place keeping within active procedures was not consistently conducted. During the power ascension it was not apparent which actions in SOP 21.1, "Main Feedwater System," had been completed. For example, several pages had missing signoffs and other pages were incomplete with respect to the steps which had been completed.

2. Training and Qualification

a. Inspection Scope

The team verified the training and qualifications of station personnel with respect to the level of work assigned. The team conducted observations of training using the guidance and checklists found in NUREG-1220 Rev. 1, "Training Review Criteria and Procedures." The team conducted interviews of trainees, supervisors, and instructors. The team assessed whether personnel were able to evaluate hypothetical conditions or data, identify respective emergency action levels, evaluate or perform dose calculations, classify emergencies, and recommend appropriate protective actions. Personnel were interviewed to determine their awareness and understanding of procedure changes, and whether they had received adequate training for their use.

b. Findings

Interviews were conducted with plant operators with respect to the quality of the site training program. Many operators stated that they believed that licensed operator continuing training was improving. Many of the operators noted that, while the overall industry operating experience level of the licensed instructors was good, the site specific experience level of the instructors warranted improvement.

The licensee had issued SL1 CR 200004471 as result of an adverse trend in the quality of nuclear training lesson plans. This trend was identified when initial licensed operator training was rescheduled due to inadequate lesson plans. The team reviewed the condition report and associated root cause assessment. It was determined that the overall assessment was adequate and that the corrective actions identified, if properly implemented, should address this significant issue. The actions planned to improve the lesson plans were scheduled for March and August 2001. Additionally, the team reviewed the licensee's assessment of the 2000 operator requalification examination. The licensee's evaluation included a root cause assessment of examination performance difficulties. The team concluded that the root cause assessment appeared to be adequate and that the corrective actions, if properly implemented, should address issues related to improving the fundamental knowledge level of the licensed operators. The licensee indicated that a review of the effectiveness of the actions taken will be conducted during the next licensed operator requalification examination.

A third party assessment of the simulator was conducted in March 1999 using the criteria in ANSI/ANS-3.5-1985, "Nuclear Power Plant Simulators for Use in Operator Training." The conclusion of the assessment was that the simulator appeared to meet the requirements of the standard. Five weaknesses related to the simulator were identified and entered into the condition reporting system. Four of the five condition reports had been satisfactorily completed. The actions for the fifth weakness associated with the computer were in progress.

The fuel handler's training provided to licensee personnel during the Fall 2000 outage was evaluated by the team. The training program included the refuel equipment course conducted by Westinghouse training and operational services at the Waltz Mills facility. The refuel equipment course was the same course for licensee and Westinghouse personnel and was conducted at the same facility, using the same course materials and instructors. In addition to the refuel equipment course, the fuel handler's training program included site-specific crane training and qualification, based on the existing site crane operator training program. As part of the site-specific training, the fuel handler candidates completed a spent fuel tool, bridge crane, and upender refueling operator qualification guide containing three tasks and two refueling job performance measures. The three tasks were "operate the fuel storage building bridge crane," "operate the spent fuel handling tool," and "operate the upender." The two job performance measures involved moving dummy assemblies and operation of the upender. The fuel handler training program was designed using systems approach to training techniques and should ensure that employees are satisfactorily qualified to safely move and handle nuclear fuel.

3. Communications

a. Inspection Scope

The team assessed the quality of communications and whether communications were consistent with the licensee's procedures during the conduct of operations, maintenance, and testing activities. The team also evaluated the communications between various site departments and licensee management.

b. Findings

The team observed that overall crew communications were adequate. In most cases, operators announced expected and unexpected alarms, used three-way and, when appropriate, two-way communications. During the power ascension, communications between the control room supervisor and the operator at the controls were adequate.

The quality of pre-job and pre-evolution briefings was mixed but the briefings generally described expected indications and potential problems that could be encountered during the evolution.

4. Control of Overtime and Fatigue

a. Inspection Scope

The team reviewed the process for controlling overtime. Interviews were conducted with personnel who had worked overtime to determine how management ensures that personnel are not assigned to safety related duties while in a fatigued condition. A review of records was conducted to identify indications of recurrent or routine use of overtime.

b. Findings

The hours worked for operations personnel were reviewed. The team noted that while there did not appear to be an excessive use of overtime, that several instances of both planned and unplanned deviations from the overtime policy had occurred in recent months. During the continuous control room coverage, two operator trainees were observed to have worked a significant amount of overtime in order to acquire needed qualification requirements. A review of the audits conducted in calendar year 2000 through September 16, 2000, did not identify any working hour deviations that were not approved.

5. Human System Interface

a. Inspection Scope

The team conducted an evaluation of human-system interfaces, including work area design and environmental conditions. During both the control room coverage and simulator observations, the team walked down control panels and evaluated displays, controls, and alarms. The team assessed whether panels and equipment were correctly labeled and evaluated work areas.

b. Findings

The team did not identify any human-system interface problems with control room displays, controls, and alarms.

D. Emergency Preparedness

1. Problem Identification and Resolution

a. Inspection Scope

The team evaluated the effectiveness of corrective actions for emergency preparedness (EP) performance issues to determine whether identified problems were appropriately reviewed, prioritized, and resolved in a technically adequate and timely manner. The review included an assessment of 120 action items in the licensee's condition report system, QA audit report No. 00-05-A, and various self-assessments and exercise reports. In addition, interviews were conducted with the EP Manager and individuals responsible for overseeing the corrective action program within the EP group.

b. Findings

The team found that the licensee was self-critical of the EP program and had generated a number of condition reports to address identified performance issues. In particular, a number of thorough self-assessments were generated following the February 15, 2000, steam generator tube failure event. With respect to the overall program for identifying and correcting deficiencies in the EP area, the team determined that most condition reports were concise and well-written and that corrective actions had been appropriately specified. However, the team found several examples where the condition report responses were not sufficiently descriptive, or did not describe the actual corrective action taken.

The team reviewed surveillance test records for the Emergency Response Data System (ERDS) and found the system was operable in the 2nd and 3rd quarter of 2000. However, the system was found inoperable during an exercise in November 2000, and also during a test conducted in the 1st quarter of 2001. The system engineer stated that the cause of this failure was that the modem assigned to the ERDS had been borrowed and reconfigured prior to both tests. The NRC conducted an ERDS test during the inspection and found both the system and the backup to be operable. However, the team noted there were no procedures for activating the backup system. The licensee generated CR 200100964 to address this issue. Overall, the team concluded that the corrective actions taken as a result of a drill deficiency were inadequate to prevent a recurrence with respect to the failure of the ERDS. The finding was determined to have very low safety significance (Green) because the licensee retained capability to communicate via the telephone system. 10 CFR 50.54(q) states that licensees will follow and maintain in effect an E-Plan which meets the planning standards of 10 CFR 50.47(b) and the requirements of 10 CFR Part 50, Appendix E. This is considered a Severity Level IV violation of 10 CFR 50.47(b)(14), which states that deficiencies identified during a drill/exercise will be corrected. This violation is being treated as a non-cited violation consistent with Section VI.A of the NRC Enforcement Policy, issued May 1, 2000 (65FR25368) **(NCV 05000247/2001-002-007).**

The team noted that the licensee was responsive in resolving most identified issues. However, in some cases the licensee was not effective in diagnosing the underlying causes for the problems to prevent recurrence. Some examples of this included the ERDS issues discussed above and issues involving qualification lapses in the emergency response organization. Additionally, the licensee had identified several deficiencies in exercises that appeared to be repetitive (section D.5). The corrective actions focused on conducting an additional exercise, post-exercise critiques and lessons learned with emergency response organization emergency facility leads. However, the actions did not include an assessment of, for example, the effectiveness of training for resolving these issues, qualifications of the responders, or lessons-learned from discussions with the affected individuals.

During drills conducted in the past two years, the licensee consistently identified problems with the site public address system. After several attempts by EP to have engineering address this issue, a contingency measure was established to use a bullhorn in areas determined to be inaudible. The licensee indicated that the system needed to be upgraded and that repairing the system had not been considered a priority and entered into the corrective action system. While the EP work around was an adequate temporary corrective action, the team considered the continual delays by engineering to fix this issue a weakness.

The team identified a weakness with respect to the process for conducting the 2000 nuclear quality assurance audit in the emergency planning area. The team determined the audit report met the 10 CFR 50.54(t) requirements; however, the licensee did not maintain checklists for the team to verify the conduct of the audit and for supporting the conclusions in the audit report. In addition, the audit report did not include an assessment of the adequacy of corrective actions for previously identified deficiencies listed in the corrective action system. The team concluded that due to the number of emergency planning weaknesses in the past year, an independent assessment of ongoing corrective actions would have been appropriate.

Interviews with the EP manager indicated that he was knowledgeable of the corrective actions taken for identified performance issues. However, an EP staff member was delegated the responsibility for maintaining the condition reporting system. The site corrective action program manager stated that the use of a "surrogate" is considered to be an acceptable practice at the site. However, the EP manager did not routinely review the narrative of how condition reports were closed. This issue is considered a weakness and was entered into the licensee's corrective action system (CR 200101416) for resolution.

2. Emergency Response Staffing

a. <u>Inspection Scope</u>

The team reviewed the licensee's emergency response organization to ensure the minimum on-shift staffing met the applicable regulatory requirements and that staffing was sufficient to fill positions needed in the emergency facilities. The team also reviewed drill records and call-in procedures to determine if augmentation and off-hour drills were held as required by the E-Plan, whether augmentation goals were met, and that off-shift personnel were available if needed. In addition, interviews were conducted with emergency response organization responders to verify their understanding of the call-out process and their responsibilities for reporting to their facilities during an event.

b. Findings

The team verified that the emergency response organization assignment roster met the minimum on-shift staffing requirements as stated in the E-Plan. Key positions were divided into three teams with most positions having alternates as additional backups. Although the licensee designated a team per week to be on-call, they required all teams to report during an event to ensure complete coverage. Weekly pager tests were performed for the on-call team. A review of records indicated acceptable pager performance. The licensee conducted an unannounced off-hours augmentation drill in April 2000 and met the 60 minute requirement in all emergency facilities. The licensee had been conducting off-hours testing of a new automated dialer system (section D.4), and test records indicated that they would have been able to fill all key positions should there have been a real event. The EP manager stated that an unannounced off-hours drill would be conducted in 2001 to further verify that changes made to the notification system were adequate. During the planned drill, the ability to staff the Joint News Center will also be verified. Interviews with individuals who were recently added to the emergency response organization indicated they were knowledgeable of the call-out process and understood their responsibilities during an event.

3. Emergency Plan and Procedure Quality

a. Inspection Scope

The team performed a review of E-Plan changes since June 2000 to determine if any changes had decreased the effectiveness of the plan. In addition, a review of the plan's implementing procedures relative to the significant planning standards was performed. The team evaluated the 10 CFR 50.54(q) review documentation and applicable procedures to assess the adequacy of the method for reviewing the E-Plan and implementing procedure changes.

b. Findings

The team noted an instance where the licensee's review of changes made to the E-Plan and implementing procedures was not thorough. The issue involved a change to implementing procedure IP-1035, "Technical Support Center," Attachment 2. The change stated that prior to activation, a minimum staffing level of three individuals was required. This change appeared to contradict the E-Plan which stated that a minimum staffing level of seven people was needed for activation. The licensee continued to commit to the 60-minute activation staffing level (seven people), as set forth in the E-Plan. However, the licensee stated that the intent of IP-1035, was that a minimum of three people could begin to assist the control room. The licensee acknowledged that the word "activation" may have been misused in the implementing procedure relative to its use in the E-Plan. This issue was entered into the corrective action system (CR 200100813) and the discrepancy was corrected.

4. Emergency Facility Equipment

a. Inspection Scope

The team reviewed surveillance test records and maintenance procedures for offsite sirens, emergency pagers and communication equipment to determine if the tests were performed in accordance with regulations and E-Plan commitments. In addition, the team conducted an inventory of the emergency equipment located in the emergency facilities using the appropriate inventory checklists.

b. Findings

The team found a number of discrepancies with respect to the equipment inventories. These included: (1) five radiological instruments were out of calibration at the Emergency Operations Facilities (EOF); (2) the monthly inspection of full face respirators was not conducted in April and June 2000; (3) a radiological instrument located in one of the field kits had low batteries, and no batteries were found in the kit; (4) an expired calibration sticker on a meter was not replaced when calibrated the previous month; and, (5) inventory lists were not updated to reflect the addition of several radiological check sources.

According to Section 8.3 of the E-Plan, facility inventories are to be conducted on a quarterly basis. The licensee could not provide inventory records for the third quarter nor verify that those inventories were actually conducted. The EP manager stated that due to limited resources, the responsibility for conducting the inventories was given to another department within the past year. The team concluded that the emergency planning organization was not proactive in making sure the inventories were being conducted and properly documented. These issues were entered into the corrective action system (CR 200100815) and out-of-calibration instruments were immediately replaced. The team considered this issue to be of very low safety significance (Green) because notwithstanding the discrepancies which were identified, the licensee had sufficient resources in the facilities to properly respond to an event. 10 CFR 50.54(q) states that licensees will follow and maintain in effect an E-Plan which meets the planning standards of 10 CFR 50.47(b) and the requirements of 10 CFR Part 50, Appendix E. This is considered a Severity Level IV violation of 10 CFR 50.54(q) and the licensee's E-Plan, Section 8.3 which states that quarterly inventories will be conducted. This violation is being treated as a non-cited violation consistent with Section VI.A of the NRC Enforcement Policy, issued May 1, 2000 (65FR25368) **(NCV 05000247/2001-002-008)**.

In July 2000 the licensee's system performance group began an extensive initiative to address emergency response organization pager problems. These actions included: (1) evaluation of the current vendor for compatibility; (2) consolidation of pagers under one vendor; (3) installation of a repeater system to ensure pager operability in "dead" zones; and, (4) establishment of specific testing criteria. The work was completed by October 2000, and since that time, weekly pager test records indicated significant improvements in reliability. The licensee had installed and was testing an automated telephone system which would backup the pager system by simultaneously telephoning responders. The responders would call back the system which would log and track the number of responders needed to fill ERO positions. The licensee stated that this system would be operational by April 1, 2001.

Finally, the inspectors noted that Section 8.1.3 of the E-Plan stated that emergency communication links between facilities will be operationally checked on a quarterly basis. The communication tests would include the dedicated NRC communication links used in each facility. The team reviewed communication records for the year 2000 and found that the licensee was not able to produce the 3rd quarter records and could not verify that the required tests had been conducted. This issue was entered into the licensee's corrective action system (CR 200101776). The team determined this issue to be of very low safety significance (Green) because the licensee had installed spare operable telephone lines. 10 CFR 50.54(q) states that licensees will follow and maintain in effect an E-Plan which meets the planning standards of 10 CFR 50.47(b) and the requirements of 10 CFR Part 50, Appendix E. This is considered a Severity Level IV violation of 10 CFR 50.54(q) and Section 8.1.3 of the E-Plan. This violation is being treated as a non-cited violation consistent with Section VI.A of the NRC Enforcement Policy, issued May 1, 2000 (65FR25368) **(NCV 05000247/2001-002-009).**

5. Emergency Response Organization Performance

a. Inspection Scope

A review was conducted of the licensee's training program to ensure it was in compliance with the applicable regulations and the E-Plan. The team reviewed the following: (1) EP-AD-03, "ERO Training Program"; (2) various lesson plans; (3) conduct of training; (4) experience and qualifications of instructors; and (5) ERO qualification training records. The team also conducted interviews and observed training to identify any observed weaknesses. In addition, the team reviewed reports for several recent training exercises to determine the adequacy of training and the ability to identify and correct exercise deficiencies in a timely manner.

The team evaluated four mini-evaluation drills of simulated events that tested the performance of key members of the emergency response organization in understanding their assignments, responsibilities and authority. These drills provided an independent assessment of the licensee's capabilities to make and assess emergency classifications, dose assessment calculations and protective action recommendations (PAR). In addition, the team reviewed the documentation generated as a result of the exercises and evaluated the licensee's critique process.

b. Findings

The team observed that the licensee had recently revised their training program. The revision included procedure and exam development, classroom training, and a tracking process for qualifications. However, the team found that the program procedure did not describe if a drill or exercise was needed for initial qualifications or for requalification. Additionally, the procedure lacked specificity regarding the tracking of deficiencies.

The team reviewed the critique comments from classroom training conducted in December 2000 and found that while the comments were primarily administrative in nature, several had some technical significance. For example, comments involved confusion with terminology, questions on activation, request for additional practice for making classifications, and confusion regarding what procedures are current (versus changes expected to be made). The team further noted that there was no formal mechanism for reviewing critique comments and documenting their resolution. The team concluded that this represented a weakness with respect to documenting and tracking training issues.

The team interviewed a number of staff in key emergency response organization positions. There was a consensus that training had improved and that the EP staff were receptive to critical feedback and program enhancement suggestions. The team also observed an operations support center facility walkthrough class and noted the instructor was knowledgeable of the facility. The team further observed that the training appropriately emphasized the use of procedures and that the participants were actively involved in the training session.

The team reviewed qualification records and the training matrix listed in the licensee's administrative procedures. Overall, the team found that emergency responder qualifications were current. However, ten individuals assigned to the offsite and onsite monitoring teams had let their respirator qualifications lapse. It was determined that there was confusion between the EP and the health physics organizations regarding the necessity for maintaining respirator qualifications for emergency responders. Upon further review, the EP manager determined that all individuals that would be expected to wear respirators must be respirator qualified. This issue was entered into the licensee's corrective action system (CR 200100290) and at the end of the inspection the issue had been resolved. The team determined this issue to be of very low safety significance (Green) because there were sufficient responders with respiratory qualifications to fill the positions. 10 CFR 50.54(q) states that licensees will follow and maintain in effect an E-Plan which meets the planning standards of 10 CFR 50.47(b) and the requirements of 10 CFR Part 50, Appendix E. This is considered a Severity Level IV violation of 10 CFR 50.54(q) and E-Plan Section 8.1.2 of the licensee's E-Plan which describes the qualifications necessary to maintain proficiency as an emergency responder. This violation is being treated as a non-cited violation consistent with Section VI.A of the NRC Enforcement Policy, issued May 1, 2000 (65FR25368) **(NCV 05000247/2001-002-010)**.

Since the June 2000 NRC evaluated exercise, the licensee conducted four exercises[16] with the "blue" and "red" emergency response teams. The exercise reports were found to be self-critical and had identified areas for improvement. The NRC team trended the deficiencies identified in the four exercise reports and found repetitive issues in the exercises that were reflective of past performance, particularly in the area of plant assessment and the dissemination of the information to the general public.

[16] August, November (2), and December 2000

The team reviewed the condition report generated following the August 2000 exercise and found it to be descriptive; however, the corrective actions were general, simply indicating that more exercises were needed and lessons learned should be discussed with the facility leads. In this case, the affected team had one additional exercise and the lessons learned discussion was not performed until November. The condition reports associated with the second exercise did not capture the deficiencies in the joint news center and the corrective actions were only generally described and not pertinent to all the significant issues. The licensee provided two lesson plans for classes conducted in November 2000 and the instructor notes indicated some of the repetitive issues were addressed, but the classes were limited to only the facility leads and not the organization as a whole. Further, the team noted that the licensee did not retain any original player or controller comments, or trend and assess exercise performance. The emergency planning organization expressed their belief that significant improvement in the TSC has been observed, but that other facility personnel were not fully aware of the improvements and tend to be overly critical. However, the team noted that irrespective of the adequacy of the TSC, that a lack of confidence on the part of other key organizations could limit the effectiveness of the TSC.

While it appears the licensee implemented some corrective actions, the team determined that the licensee's training program was not fully effective in preventing recurrence of issues to ensure consistent emergency response organization performance. The team determined this issue to be of very low safety significance (Green) because these performance issues did not deal with the risk significant planning standards (classifications, notifications, PARs). The licensee entered this issue into the corrective actions system (CR 200101775). 10 CFR 50.54(q) states that licensees will follow and maintain in effect an E-Plan which meets the planning standards of 10 CFR 50.47(b) and the requirements of 10 CFR Part 50, Appendix E. Section 8.1.2 of the licensees E-Plan states a training program is established to train employees and exercising, by periodic drills to ensure that employees maintain the proficiency of their specific emergency response duties. This is considered a Severity Level IV violation of 10 CFR Part 50.54(q) and Appendix E.IV.F.2.g for inadequate training. This violation is being treated as a non-cited violation consistent with Section VI.A of the NRC Enforcement Policy, issued May 1, 2000 (65FR25368) **(NCV 05000247/2001-002-011).**

The team observed and evaluated the licensee's performance in response to two drills and four separate scenarios. The licensee used a limited emergency operations facility staff and simulated support from the technical support center to conduct the drill. The scenarios both required an upgrade to the protective actions recommendations due to a wind shift in one case, and increased radiological release in another. Dose assessment calculations were performed with the two shift managers and their control room supervisors and was independent of the training class. The team determined that the licensee effectively evaluated plant conditions and the emergency classifications. The required notifications and protective actions recommendations were accurate and timely. The licensee conducted an adequate critique of each performance and identified areas for improvement.

6. Emergency Preparedness Off-site Interface

a. Inspection Scope

The team evaluated the licensee's interface with off-site state and county agencies. This included a review of documentation of off-site state and county meetings, letters of agreement with offsite organizations and training drills. Also, the team conducted telephone interviews with the lead contacts from the New York State Emergency Management Agency, Orange County Office of Emergency Management, Rockland County Office of Fire and Emergency Services, Westchester County Office of Emergency Management and the Putnam County Office of Emergency Management.

The team reviewed documentation of radiological orientation training provided to the media as required by the regulations and the E-Plan. An interview was conducted with the site communications manager regarding the status of corrective actions from deficiencies identified during the Alert Event on February 15, 2000, and the June 1, 2000, exercise at the joint news center.

b. Findings

Following the steam generator tube failure event of February 15, 2000, the licensee has met with state and county officials on numerous occasions to gain a better understanding of their needs and requirements. While expressing concerns about the extent of past overall communications, most of the state and county officials indicated that the licensee has made an effort to improve communications and address their needs with respect to emergency preparedness. The team verified that all required offsite training and drills had been conducted and that letters of agreement for offsite assistance were current. The team also observed that the licensee conducted the required annual training session for the local media as required in Section 8.4 of the E-Plan.

E. Conclusions Regarding Performance in the Reactor Safety Strategic Performance Area

The team determined that overall performance was acceptable in the reactor safety strategic performance area. However, a number of issues were identified in the areas of design control, procedures, equipment and human performance, and emergency preparedness which indicated weaknesses in these areas as well as the need for continued improvement. The issues identified by the team have, individually, been evaluated under the risk significance determination process as being minor in nature or having very low safety significance (Green). However, the issues provide evidence of some program and process weaknesses similar to those which contributed to previous plant events.

In the design control area, the team identified several examples of performance issues related to weaknesses in translating important design assumptions into plant operating procedures, drawings, calculations, and testing programs. These examples point to weaknesses in the design control process which indicate the need for continued improvement in this area. Additionally, the team observed that there appeared to be difficulties in retrieving design basis information necessary to support design control, testing and plant modification efforts. This issue had been previously identified and slow progress has been made to improve in this area. Notwithstanding the performance issues identified, the team determined that while weaknesses, some of a longstanding nature, existed in the design control area, that the 480 Vac/emergency diesel generator and service water systems were capable of performing their safety functions.

In the area of procedures, the team found that while overall procedure quality was adequate, performance weaknesses in both procedure quality and usage existed at the facility. The team found deficiencies related to procedure clarity, consistency, and accuracy in administrative and implementing procedures. The team also noted that flexible guidance in some administrative procedures allowed for wide variation in procedure use and interpretation and there were several instances where the team identified that design, vendor, or modification information was not properly translated into procedures.

In the area of equipment performance, the team determined that the reliability, material condition and overall performance was acceptable for the systems which were reviewed. However, a number of equipment issues were observed which presented challenges to both the plant as well as the operators. It was observed that emergent equipment failures in secondary plant systems continue to challenge the plant operators and require plant power changes. The team also noted a decrease in reliability and a concurrent increase in unavailability of the gas turbine generators which appeared to be partly attributable to a decrease in the emphasis on maintenance for this equipment. Finally, the team noted that the station work backlog continued to pose a significant challenge to the station. It was also determined that a number of important work items had not been accurately captured in the accounting for the backlog, indicating that the backlog may be somewhat larger than stated.

In the area of human performance, the team noted an increased emphasis on overall improvement and a recognition of the need for an improved training program. However, a number of program and process issues were identified. In particular, a challenge existed with respect to the number of licensed operators which posed complications with respect to overall scheduling and overtime considerations. The team observed that there was a management recognition of this problem and that steps have been undertaken to increase the number of licensed operators. The team also observed that operator performance issues have contributed to recent events and that some performance problems continue to occur. Specifically, performance errors were observed in the August 1999 reactor trip, February 2000 steam generator tube failure and as recently as the January 2001 turbine trip. Additionally, inconsistencies continue to exist with respect to procedural quality and adherence, owing, in large measure, to inconsistent reinforcement of management expectations in this area. However, the team did observe that during the inspection, overall crew performance was acceptable, and in particular, crew communications were good, indicating some improvements.

In the area of emergency preparedness, the team determined that the overall program was adequate and provided reasonable assurance that the emergency response organization could respond effectively to an emergency. Additionally, while issues were identified that indicated the need for continued improvement, improvements were noted in a number of previously identified problem areas. Notwithstanding the improvement which was observed, the team concluded that the remediation for some of the previously identified performance issues in the technical support center, emergency operations facility and joint news center had not been fully effective. The team acknowledged that although some corrective actions had been implemented, the licensee's training program has not been fully effective in preventing the recurrence of issues to ensure consistent emergency response organization performance. However, risk significant planning standards continue to be met.

3. Root and Contributing Cause Assessment

The team, in accordance with Inspection Procedure 95003, integrated the inspection findings, with the results of similar, previous efforts in order to provide insight into the upper level causes of performance issues at the site. It should be noted, however, that this effort was not intended to be a substitute for a more focused root cause study or self-assessment by the licensee.

The team identified four specific causes:

- Inconsistent management application and reinforcement of existing standards with respect to staff performance, particularly in the areas of procedural quality and adherence and in implementation of the corrective actions programs.

- Weaknesses existed with respect to the ability to retrieve, verify, and assure the quality of engineering products, particularly design basis information. These weaknesses contributed to problems in developing and validating calculations, testing methodologies and acceptance criteria.

- The plant staff tended to accept degraded conditions. This was true of both equipment and documentation issues. However, it was noted that improvement has been made in this area, in particular, the increased emphasis on problem identification.

- A number of performance problems may have been influenced by resource issues. In particular staffing issues (in operations and instrumentation and control) and training resources.

4. Management Meetings

Exit Meeting Summary

The team conducted a detailed debriefing with the licensee on February 15, 2001.

An exit meeting, open for public observation, was conducted on March 2, 2001, at the Cortlandt Town Hall, Cortlandt, New York. The inspection results were presented to Mr. J. Groth and other members of the licensee staff who acknowledged the findings. This exit meeting was followed by a public question and answer session with elected officials and members of the public.

ITEMS OPENED, CLOSED AND DISCUSSED

Opened And Closed During This Inspection

05000247/2001-002-001	NCV	10 CFR 50 Appendix B, Criteria XVI, Corrective Action
05000247/2001-002-002	NCV	10 CFR 50 Appendix B, Criteria III, Design Control
05000247/2001-002-003	NCV	Technical Specification 6.8.1, Procedures
05000247/2001-002-004	NCV	10 CFR 50 Appendix B, Criteria V, Instructions, Procedures, Drawings
05000247/2001-002-005	NCV	10 CFR 50.55.a, Inservice Testing
05000247/2001-002-006	NCV	10 CFR 50.65(a)(1), Maintenance Rule
05000247/2001-002-007	NCV	10 CFR 50.47(b)(14), EP Drill Deficiencies
05000247/2001-002-008	NCV	10 CFR 50.47(b)(8), Emergency Equipment
05000247/2001-002-009	NCV	10 CFR 50.54(q), E-Plan 8.1.3, Communication Tests
05000247/2001-002-010	NCV	10 CFR 50.54(q), E-Plan 8.1.2, Emergency Responder Proficiency
05000247/2001-002-011	NCV	10 CFR 50.54(q), Appendix E.IV.F.2.g, Inadequate Training

TABLE I
INDIAN POINT 2 RADIOCHEMISTRY TEST RESULTS

SAMPLE	RADIONUCLIDE	NRC VALUE	Con Ed VALUE	COMPARISON
Liquid Radwaste 0945 hrs 2-8-01 (Detector NUC3) (Results in microCuries per milliliter)	Co-60 Cs-137 Co-58 Sb-125	(2.81±0.09) E-6 (6.00±0.10)E-6 (1.76±0.08)E-6 (2.62±0.04)E-5	(2.71±0.10) E-6 (5.81±0.11)E-6 (1.81±0.07)E-6 (2.60±0.04)E-5	Agreement Agreement Agreement Agreement
Reactor Coolant Particulate Filter (Crud Filter) 1200 hrs 1-31-01 (Detector NUC3) (Results in microCuries per milliliter)	Co-60 Co-58 Mn-54 Cr-51 Zr-95 Sb-124	(3.62±0.02)E-4 (5.16±0.02)E-4 (3.74±0.09)E-5 (1.522±0.008)E-3 (1.158±0.016)E-4 (6.6±0.6)E-6	(3.50±0.03)E-4 (5.04±0.03)E-4 (3.85±0.16)E-5 (1.553±0.014)E-3 (1.15±0.03)E-4 (6.1±0.7)E-6	Agreement Agreement Agreement Agreement Agreement Agreement
Reactor Coolant (First Count) 0828 hrs 2-8-01 (Detector NUC2) (Results in microCuries per milliliter)	I-132 I-133 I-134 I-135	(1.46±0.06)E-3 (7.8±0.3)E-4 (2.41±0.11)E-3 (1.50±0.14)E-3	(1.54±0.07)E-3 (8.3±0.7)E-4 (3.14±0.10)E-3 (1.80±0.16)E-3	Agreement Agreement Agreement Agreement
Reactor Coolant (Second Count) 0828 hrs 2-8-01 (Detector NUC2) (Results in microCuries per milliliter)	I-131 I-132 I-133 I-135	(1.1±0.2)E-4 (1.8±0.2)E-3 (7.9±0.2)E-4 (1.82±0.14)E-3	(9±2)E-5 (1.52±0.16)E-3 (8.7±0.3)E-4 (1.7±0.2)E-3	Agreement Agreement Agreement Agreement

SAMPLE	RADIONUCLIDE	NRC VALUE	Con Ed VALUE	COMPARISON
Waste Gas Decay Tank 1409 hrs 2-8-01 (Detector NUC2) (Results in microCuries per milliliter)	Xe-133 Xe-135	(2.63±0.03)E-5 (1.68±0.06)E-6	(2.48±0.04)E-5 (1.62±0.06)E-6	Agreement Agreement
Plant Vent Charcoal Cartridge 1235 hrs 2-7-01 (Detector NUC2) (Results in microCuries per milliliter)	I-131 I-133	<6E-13 <1E-12	<9E-13 <1E-12	No comparison, no radionuclides were detected in this sample.
Plant Vent Particulate Filter 0948 hrs 2-6-01 (Detector NUC2) (Results in microCuries per milliliter)	Co-60 I-131 I-133	<1E-13 <9E-14 <7E-13	<2E-13 <2E-13 <8E-13	No comparison, no radionuclides were detected in this sample.
Air Ejector 1308 hrs 2-7-01 (Detector NUC3) (Results in microCuries per milliliter)	Kr-85 Xe-133 Xe-135	<6E-6 <6E-8 <3E-8	<1E-6 <9E-9 <4E-9	No comparison, no radionuclides were detected in this sample.

SAMPLE	RADIONUCLIDE	NRC VALUE	Con Ed VALUE	COMPARISON
Steam Generator Blowdown (Water) 0900 hrs 2-7-01 (Detector NUC2) (Results in microCuries per milliliter)	Mn-54 Co-58 Co-60 I-131 I-133 Cs-137	<8E-8 <8E-8 <1E-7 <9E-8 <9E-8 <1E-7	<9E-8 <9E-8 <1E-7 <6E-8 <7E-8 <9E-8	No comparison, no radionuclides were detected in this sample.
Service Water 0900 hrs 2-9-01 (Detector NUC3) (Results in microCuries per milliliter)	Mn-54 Co-58 Co-60 I-131 I-133 Cs-137	<9E-8 <8E-8 <1E-7 <9E-8 <8E-8 <1E-7	<2E-7 <5E-8 <1E-7 <1E-7 <1E-7 <2E-7	No comparison, no radionuclides were detected in this sample.

NOTE: Reported uncertainties are ± 1 Standard Deviation counting uncertainties for both NRC and licensee results.

ATTACHMENT TO TABLE I

CRITERIA FOR COMPARING ANALYTICAL MEASUREMENTS

This attachment provides criteria for comparing results of capability tests and verification measurements. The criteria are based on an empirical relationship which combines prior experience and the accuracy needs of the program.

In these criteria, the judgement limits are variable in relation to the comparison of the NRC Reference Laboratory's value to its associated uncertainty. As that ratio, referred to in this program as "Resolution," increases, the acceptability of a licensee's measurement should be more selective. Conversely, poorer agreement must be considered acceptable as the resolution decreases.

Resolution[1]	Ratio for Comparison[2]
<4	No Comparison
4 - 7	0.5 - 2.0
8 - 15	0.6 - 1.66
16 - 50	0.75 - 1.33
51 - 200	0.80 - 1.25
>200	0.85 - 1.18

1. Resolution = (NRC Reference Value/Reference Value Uncertainty)

2. Ratio = (Consolidated Edison Value/NRC Reference Value)

ATTACHMENT 1

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety

- Initiating Events
- Mitigating Systems
- Barrier Integrity
- Emergency Preparedness

Radiation Safety

- Occupational
- Public

Safeguards

- Physical Protection

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.

ATTACHMENT 2

LIST OF ACRONYMS USED

AAC	Alternate AC
AFW	Auxiliary Feedwater
AOI	Abnormal Operating Instruction
ARP	Alarm Response Procedure
ASSD	Alternate Safe Shutdown
CARB	Corrective Action Review Board
CCHX	Component Cooling Heat Exchanger
CCR	Central Control Room
CCW	Component Cooling Water
CFR	Code of Federal Regulations
COL	Check-Off List
CR	Condition Report
DBD	Design Basis Document
ECP	Employee Concern Program
EDG	Emergency Diesel Generator
EOP	Emergency Operating Procedure
FMEA	Foreign Material Exclusion Area
GT	Gas Turbine Generator
GPM	Gallons Per Minute
ICPM	Instrument & Controls Preventive Maintenance
IMC	Inspection Manual Chapter
IPE	Individual Plant Examination
KVA	Kilo Volt Ampere
KW	Kilo Watt
LOCA	Loss Of Cooling Accident
MCC	Motor Control Center
MOV	Motor Operated Valve
MPFF	Maintenance Preventable Functional Failure
MR	Maintenance Rule
NCV	Non-Cited Violation
OAD	Operations Administration Directive
P&ID	Piping and Instrumentation Diagram
PM	Preventive Maintenance
PMT	Post Maintenance Test
POP	Plant Operating Procedures
QA	Quality Assurance
RCS	Reactor Coolant System
SAO	Station Administration Order
SDP	Significance Determination Process
SGRO	Steam Generator Replacement Outage
SL	Significance Level
SOP	System Operating Procedures
SOV	Solenoid Operated Valve
SSC	Structures, Systems and Components
SW	Service Water

SWSOPI	Service Water System Operational Performance Inspection
TFC	Temporary Field Change
TOL	Thermal Overload
TP	Test Procedure
TPC	Temporary Procedure Change
UFSAR	Updater Final Safety Evaluation Report
VAC	Volts AC
VDC	Volts DC
VMI	Vendor Manual Index

ATTACHMENT 3

LIST OF DOCUMENTS REVIEWED

The following is a list of licensee documents reviewed during the inspection, including documents prepared by others for the licensee. Inclusion on this list does not imply that NRC inspectors necessarily reviewed the documents in their entirety but rather that selected sections or portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any part of it, unless this is stated in the body of the inspection report.

Calculations/Studies/Engineering Analyses

NSL-EDG-900430A, Emergency Diesel Generator Fuel Oil Minimum Storage Requirements, Rev. 0
Con Edison study, "Update of the Indian Point Unit 2 Emergency Diesel Generator Loading Study," dated December 18, 2000
FEX-00152-00, Revision 0, 1/22/01, EDG Generator Ratings Analysis
Westinghouse Motor Company Engineering Report WMC-EER-90-005, dated October 23, 1990
FEX-00143-00, IP2 LOAD FLOW ANALYSIS OF THE ELECTRICAL DISTRIBUTION SYSTEM, 12/14/00
FEX-00120-01, Analysis of EDG Load Sequencing for Blackout & Unit Trip with and without an SI
FEX-00029-02, MINIMUM VOLTAGE ANALYSIS FOR INSTRUMENT BUSES 21 THRU 24 &21A THRU 24A, dated 2/3/98
FEX-00019-01, FEX-00020-01, FEX-00021-01, FEX-00022-01 INSTRUMENT BUS LOADING FOR INSTRUMENT BUSES
FEX-00025-02, Minimum Voltage Analysis for the Loads on Instrument Buses 21 & 21A, dated 2/3/98
EGE-00001-02, Indian Point - Class 1E Motor Minimum Starting Voltage and Acceleration Time Calculations, Rev. 2, 6/24/98
FEX-00101-00, revision 01, 4/21/00, 13.8 kV and 6.9 kV cable ampacity for primary and secondary leads of the new GT-1 transformer
125Vdc Protective Device Coordination Study No. SGX-00007-03 - Ebasco - Original, date 9/25/91, revision 3, approved 4/16/98
EPG-00006-00, Verify Adequacy of 480 Volt DB-50 Switchgear to interrupt Worst Case Short Circuit, Rev. 0, 9/5/91
SGX-00013-04, Setpoint Change for Undervoltage Relays on 480 Volt Buses 2A, 3A, 5A and 6A, Modification EGP-91-06786-E, Revision 4, dated 9/10/99
SGX-00004-00, Indian Point 2 - Calculate Fault Current at 480V Switchgear including 6.9 kV Motor Contributions, Rev. 0, 5-28-92
DA-EE-93-107-07, 480 Volt Coordination and Circuit Protection Study, Rev. 2
FFX-00822-01, Stress Analysis of Jacket Water Header for EDG JW Expansion Tank due to Replacement of Valve JW-5 (CR 200007667).
FMX-00107-00, EDG-JW/LOC Bundle Replacement - Seismic Evaluation.

Calculations/Studies/Engineering Analyses (Cont.)

MEX-00041-00, Seismic Evaluation of EDG Jacket Water and Lube Oil Coolers.
GMS-00014-01, Pipe Stress Analysis of Diesel Fuel Oil System to Determine if Piping is Over stressed due to Replacement of FO Valves to Day Tanks.
FFS-00131-00, Evaluation of Diesel Gen 21, 22 & 23 Air Compressors
FFX-00408-01, Evaluation of Diesel Generator Starting Air Line and "Supports Due to Installation of Hose at Motor.
FPX-00009-01, Installation of Check Valves in Discharge Lines from EDG 21, 22, and 23, Seismic Support Evaluation.
GCC-00155-00, Compressor Mounting in EDG Building - Seismic.
MMM-00014-00, IP Sluice Gate Flow, 1/29/92
PE-SW-910830A, SWP Submergence & NPSH, 8/30/91
PGI-00111-01, EDG JW and LO Heat Exchanger Tube Velocity, 3/10/95
(No document number),Update of the Indian Point Unit 2 Emergency Diesel Generator Loading Study, Final Report, Rev 0
Technical Report No. 97222-TR-28, Indian Point Unit 2 GL 98-13 Heat Exchanger Performance Assessment Program, Rev 1, June 2000
(No document number), Hydraulic Model Study of Service Water Intake by Alden research Laboratory, Inc., July 1994
PGI-00354, Generic Letter 89-13 Heat Exchanger Performance Assessment Program, Rev 1
PGI-00371-00, Service Water System Hydraulic Model, 7/29/98
MAA-00001, Service Water DBD Item 035, CFCU Outlet Flashing, Rev 00
FFX-00713, Evaluation of Service Water Strainer Minimum Blowdown Flow Through Throttled Valves, Rev 0
FFX-00300, Evaluation of Line 405, New & Existing Supports Due to t he Replacement of Valves SWN-35 & 35-1, Rev 2
FMX-00102, EDG Jacket Water Cooler & Lube Oil Cooler Performance, Rev 00
PGI-00162, 22 EDG Jacket Water Heat Exchanger Performance, Rev 0
PGI-00163, 22 EDG Lube Oil Heat Exchanger Performance, Rev 0
SMX-00005, FCU Service Water Flow Transmitter Replacements, Rev 1
FMX-00128, EDG-JWC/LOC Bundle Replacement: Vendor Thermal and Mechanical Design Calc., 4/29/99
GE Report NBR DER-1703, Emergency Diesel Flow Test, 9/19/91
RS-92, Service Water System Radiation Detector Alarm Set point, Rev 2
FEX-00003-00, Heat Trace of Lines 155, 161 and 181 for RWST, Rev. 0
EGE-00001-02, Class IE Motor Minimum Starting Voltage and Acceleration Time
EGE-00006-00, EDG Upgrade DB-75 and Switchgear Testing
EGE-00022-01, DB-75 Overload Capability During Degraded Voltage Conditions
EGP-00018-00, Service Water Improvement / Electrical Power Supply Ampacities
EGP-00027-00, Power Cable Ampacities for 480 VAC and 125 Vdc Systems
EGP-00110-00, Summary of Degraded Voltage Study
EGP-S36-001-00, EDG Bldg. Ventilation System Upgrade Control Panel Feeder Sizing
EGP-S36-002-00, EDG Bldg. Ventilation System Upgrade Ampacity & Voltage Drop
EPG-00006-00, Verify Adequacy of DB-50 Switchgear to Interrupt Worst Case Short Circuit

Calculations/Studies/Engineering Analyses (Cont.)

FCX-00421-00, Maximum Outside Ambient Air Temperature to Maintain 104°F Inside EDG Bldg.
FEX-00019-xx, 118VAC Instrument Bus Loading
FEX-00025-02, Minimum Voltage Analysis for Loads for Instrument Buses 21 &21A
FEX-00048-02, Minimum Voltage Analysis for 125 Vdc Power Panels
FEX-00066-00, Auxiliary Feedwater Pump Operability at 500 HP
FEX-00087-00, EDG 21, 22 & 23 KW Meter Accuracy
FEX-00139-00, EDG Loading
FEX-00143-00, Load Flow Analysis of the Electrical Distribution System
FEX-00148-00, Plant Startup with Pending EDG Load Study Revision
FEX-00152-00, EDG Generator Ratings Analysis
GMH-00006-00, Ventilation System for the EDG Building
SGX-00004-00, Fault Current at 480 Volt Switchgear Including 6.9 KV Motor
SGX-00005-00, EDG Bldg. Ventilation System Upgrade Protective Device Selection
SGX-00005-01, EDG Bldg. Ventilation System Upgrade Protective Device Selection
SGX-00013-04, Setpoint Change for Undervoltage Relays on 480 V Buses
SGX-00048-00, 480 V Protective Devices Coordination Review

Condition Reports

CR 199802561 Response to Information Notice 95-52
CR 199802596, 21EDG Took 17.5 Seconds to Come Up to Voltage
CR 199802858, 21EDG Failed to Start on Right Hand Air Start Motor
CR 199802979, 21EDG Air Start Motors Lack of Lubrication
CR 199803069, 21EDG Failed to Start Within Required Time
CR 199805606, Analysis of Service Water Header Cross-Tie Requires Procedure Revision
CR 199807295, ESW flow balance fails its acceptance criteria, 8/24/98
CR 199807530, 22EDG Declared Inoperable Due to Failed Start Time.
CR 199807706, EDG Start Time Measurement Methods Not Very Accurate
CR 199807866, 22EDG Failed to Start Within Required Time
CR 199809212, No Procedure for Program/Procedure Changes Following TS Amendments
CR 199810682, EDG system walkdown deficiencies
CR 199810840 Degradation of Fire Protection Foam Under Freezing Conditions
CR 199810884 CVCS Weld Failures Due to Cavitation Erosion
CR 199810933, 24 SW strainer blowdown valve indicator 90 degrees out of alignment, 12/22/98
CR 199810988 Part 21 Review for Valcor Valve Model V70900-11
CR 199811021, 22EDG Jacket Water Exp Tank Level Control Valve Leaks.
CR 199900210, SW strainer pit access hatch leaks, 1/10/99
CR 199900216, RWST instrumentation heat trace alarm
CR 199900327, 25 service water pump in alert range, 1/14/99
CR 199900401, Shaft stop on valve SWN-617 not consistent with other similar valves, 1/19/99
CR 199900470, EDG 21 overspeed trip reset lever pin broken

Condition Reports (Cont.)

CR 199900499, EDG 21 overspeed trip reset lever pin hole oversized
CR 199900536, Multiple problems with 24 SW strainer, 1/25/99
CR 199900576, No procedure for checking function of DG SW outlet valves FCV-1176 & 1176A
for DG jacket water high temperature, 1/26/99
CR 199900600 Loss of RHR During Maintenance
CR 199900653, New DG heat exchanger titanium tube bundles do not fit, 1/28/99
CR 199900698, 21EDG SW to lube oil cooler pressure indicates 0 reading, 130/99
CR 199900719, SWN-618 indication is backwards, 1/31/99
CR 199900830, SPIN database missing setpoints, dated 02/04/1999
CR 199900851, Valve SWN-41-2B Dual Indication
CR 199900869, Request for TS interpretation on failure of containment isolation valve leak test failure, 2/5/99
CR 199901326, EDG ICPM discovered loose wire on lube oil heater temperature switch
CR 199901424, Conduct of training
CR 199901438, Use of controlled procedures
CR 199901816, Lack of feedback to simulator students
CR 199901818, Lack of controlled procedures in simulator
CR 199901819, Simulator CPU weaknesses.
CR 199901821, Communications between training and computer applications
CR 199901822, Simulator operator performed surveillance testing.
CR 199901856, Chipped epoxy coating in 21 CWHX, 3/9/99
CR 199901944, UFSAR Table 6.2-12 discrepancy, 3/11/99
CR 199902505, EDG Jacket Water Exp Tank Float Valve Leaks
CR 199902527, EDG 50.54f identified discrepancies
CR 199902586, 23 SW strainer knocking and slipping in rotation, 3/27/99
CR 199902626, Point Beach cold weather freeze event
CR 199902675, Retire or Resolve Issues with TSC Diesel Generator Alarm Panel
CR 199902815, Knocking sound in 23 SW strainer getting worse, 4/6/99
CR 199903103, 21EDG Jacket Water Exp Tank Level Control Valve Leaks.
CR 199903369, Requirement for Second CCW Pump not Modeled in EDG Study
CR 199903467, 21, 22 & 23 EDG Over Speed Trip Reset Lever Resting On Pin Which Could Cause Premature Failure of Trip Reset Pin.
CR 199904088, 480V cable spreading room smoke detector testing adequacy review
CR 199904447 Fire Induced Failure of VCT Outlet Valve LCV-112C
CR 199905093, New 25 SW pump had only four holddown bolt holes drilled, 6/29/99
CR 199905487, EDG 21 inappropriate mechanical governor venting
CR 199905843, Lack of procedure Guidance to Initiate Data Archive During GT-3 Operation
CR 199906210, 21 SW pump discharge pipe expansion joint is cracked, 8/11/99
CR 199906411, EDG load sequencing relays single failure analysis
CR 199906681, EDG 23 unexpected load reduction from 900kW to 100kW
CR 199906815, 480v bus undervoltage relays without reset values
CR 199906901, Self Identified and Corrected Procedure Violation
CR 199907198, 480v breaker current transformer configuration
CR 199907277, Ability to hear public address systems during emergency

Condition Reports (Cont.)

CR 199907506, TSC DG Room Has an Alarm Panel But No Alarm Response Procedure
CR 199907665, 480v 3A to 6A crosstie breaker bent cell switch
CR 199907767, Concern about questioning attitude, 10/13/99
CR 199908666, EDG engine analysis PM deferral
CR 199908715 Operating Experience Program Enhancements
CR 199908743, Management review of contractor developed lesson plans
CR 199908802 CRS Training Deficiencies
CR 199908817, Timing of Project Completion and Filing of Report Installation
CR 199908826, Drawing and Procedure Discrepancies Associated with Fuel Oil Shipments
CR 199908884, EDG 21 overspeed trip reset lever pin missing
CR 199908999, Technical accuracy of contractor developed lesson plans
CR 199909125, Roll up of deficiencies found during various audits and self-assessments
CR 199909153, ICPM program
CR 199909417, Common Cause Analysis of Events at IP-2
CR 200000128, Qualification record keeping
CR 200000285, NRC Severity Level IV violations for inadequate exercise critiques
CR 200000288, Emergency exercise weakness due to overall poor performance in the TSC
CR 200000289, Emergency exercise weakness due to poor performance in the OSC
CR 200000290, Lapse of ERO Qualifications
CR 200000634, Operations Training extent of condition
CR 200000968, Questions retarding the backup methods for notifying offsite authorities
CR 200000994 CRS Training Needs
CR 2000010694, Service Water Traveling Screen 27 Stops on Zero Speed Alarm
CR 200001093, Logkeeping standards were not met during the Alert of 2/15/2000
CR 200001126, A 50.54(q) review may not have been done on changes made to PI-1023 & IP-1035
CR 200001183, Questions Deleted from Re-qual Test without EP Manager Approval
CR 200001221, Some phones in the OSC/TSC were inoperable during the Alert of 2/15/2000
CR 200001229, Changes to EOF IP were a hindrance to ERO operations regarding step-off pads
CR 200001240, Initial lesson plans not reviewed and updated to reflect plan changes
CR 200001241, Self study modules have not been revised to reflect plan changes
CR 200001301, Failure to conduct event critique with county and State following Alert
CR 200001356, ERO Training Program did not ensure Personnel were Trained in all Positions
CR 200001361, Accountability deficiencies identified during the Alert of 2/15/2000
CR 200001366, 6 year requirement to test off-hours emergency drill not conducted
CR 200001521, 480V undervoltage panel dc power indicating lights not lit
CR 200001621, 21EDG Over Speed Trip Reset Lever Slips to Tripped Position but EDG Remains Reset.
CR 200001874 CAG Procedures for Routine Activities
CR 200002109, Issues concerning off-site monitoring and post accident sampling
CR 200002247, Onsite contractors raising concerns with being in the trailers and not hearing alarms or announcements and what they do in an evacuation

Condition Reports (Cont.)

CR 200002274, 25 SW strainer not rotating smoothly, 3/30/00
CR 200002329, EP Pre-restart plan includes action that could potentially impact the restart of IP2
CR 200002522, Station failed to meet 30 minute requirement for completing accountability
CR 200002591, Employee concern regarding message left at his home for a pager test
CR 200002618 Continued Problems with the OE Program
CR 200002713, Deficiencies identified with the ERO notification system and process
CR 200002788, Deficiencies identified during a drill on 4/17/2000
CR 200002924 Response to Information Notice 2000-06
CR 200002952, Concerns of the PA system and evacuation during an event
CR 200002968, Internal SW piping inspection found shells, 4/24/00
CR 200003182, Compliance with SAO-112 for CR Closure
CR 200003560, Drill weaknesses identified from 5/10/2000 drill
CR 200003568, CR system training attendance
CR 200003578, 22 FUC inspection found tubercles in waterbox, 6/4/00
CR 200003838, Questions regarding Accountability process
CR 200003865 Extent of Condition Information for CRs
CR 200003868 Root Cause Determination Deficiencies
CR 200003890, Deficiencies identified during a 5/14/2000 drill
CR 200003891, Drill weaknesses identified from 5/25/2000 drill
CR 200003945, EDG 21 overspeed trip reset lever pin found on floor
CR 200003978, EDG21 unexpected load change from 750kW to 2300kW
CR 200003987, No page system in NSB location
CR 200004008, EDG prints didn't match as-found wiring
CR 200004012, Valve SWN-44-5B failed leak test, 5/30/00
CR 200004059, Unable to hear alarm or announcement
CR 200004142, Simulator problem noted during the 6/1/00 evaluated exercise
CR 200004149, During 6/1/00 exercise, personnel were walking around and in between the new
simulator building and the energy education center because they had not heard any announcements in the building concerning the drill
CR 200004153, JNC did not demonstrate the ability to coordinate clear, accurate and timely information to the news media during the 6/1/00 exercise
CR 200004181, 23 FCU inlet SW relief valve failed Appendix J leak test, 6/3/00
CR 200004265, Training and Drill weaknesses observed during 6/1/00 exercise
CR 200004311 Self-Assessments for the CAP
CR 200004312, failure of supply cable to MCC 21 due to damage to underground duct bank, dated 6/7/00
CR 200004345, Adequacy of offsite monitoring kits was questioned
CR 200004374, Siren 317 failed growl test
CR 200004393, Weaknesses identified in the JNC during the 6/1/00 exercise
CR 200004471, Contractor developed lesson plans
CR 200004545, 6/14/00 E-Plan training did not meet red team EOF participant's standards

Condition Reports (Cont.)

CR 200004578, EDG 21 mechanical governor mis-adjustment
CR 200004759 Roles and Responsibilities of the CAG
CR 200004766 No Action Plan for CAP
CR 200004839, Frisker failed source check in OSC locker
CR 200004907 Review of INPO SEN 214
CR 200005014 Contract Security Personnel Involvement with Condition Reporting System
CR 200005032, 21EDG Over Speed Trip Reset Lever Will Not Remain Locked in Reset Position.
CR 200005040, Maintaining respirator qualifications
CR 200005153, Training section computer upgrades
CR 200005260, Program deficiencies identified as a result of an root causes analysis
CR 200005332, Procedures, processes and training for the JNC do not allow for adequate information dissemination
CR 200005371, NPSH calculation not adequate, 7/19/00
CR 200005446, Re-evaluation of 1999 Common Cause Analysis corrective actions
CR 200005491, NRC identifies three white findings from Alert event of 2/15/2000
CR 200005516, Valve SWN-71-2B failed stroke test, 7/25/00
CR 200005585 Statement Regarding Technical Specifications
CR 200005640, On 7/28/2000, lost two phone circuits which service Reuter-Stokes at EOF
CR 200005646, PT-R93 doesn't assure design requirements of UFSAR Table 9.6-1 are met, 7/31/00
CR 200005704, TCV-1113 plugged with shells and sediment, 8/2/00
CR 200005815, Questions not trending beeper problems previous to 8/99 may have prevented current problems. Questions continual approval by CARB for extensions of due dates
CR 200005975, Several beepers did not activate during test
CR 200006021, 22 SW strainer not rotating, 8/15/00
CR 200006057, Heat trace functional tests
CR 200006156, Equipment deficiencies found during 8/16/2000 drill, including at JNC
CR 200006157, Deficiencies identified from August 16, 2000 emergency exercise
CR 200006170, Containment Recirculation Pump Effects on EDG Study
CR 200006180, 21 SW strainer dp switch reads 2.5 # when secured and drained, 8/21/00
CR 200006345, 24 SW strainer not rotating, 8/28/00
CR 200006357, LOR-08-00, Operations Training Section Training Program Self-Assessment
CR 200006369, 24 SW strainer tripped on thermals, 8/30/00
CR 200006377, Could not hear message in stairwell
CR 200006381 Noted Decrease in CRs Initiated
CR 200006501, Personnel in VC should not hear alarm
CR 200006508, Personnel unable to clearly understand announcement
CR 200006556, Page speaker in screen well house does not work
CR 200006565, High Head Safety Injection Pump HP Increase
CR 200006619 Training Personnel on Use of Operating Experience
CR 200006658 QA Auditor Training Needs
CR 200006663 Use of Risk Significance in QA

Condition Reports (Cont.)

CR 200006674, Heat trace panel discrepancies
CR 200006702, Flow instability with both FCV-1176 and 1176A in automatic, 9/10/00
CR 200006764, Inadequacies found with facilities and equipment implementing procedures
CR 200006794, Incorrect operability call on CR 200004534, 9/13/00
CR 200006944, FC-5032-A Alarm will not clear
CR 200006965, EDG Fuel Oil Transfer Pump Level Switch Tolerances Were Incorrect
CR 200007026, HPES training
CR 200007070 SAO-112 Procedure Deficiencies
CR 200007072 Effectiveness and Timeliness of Corrective Actions
CR 200007073 Training Needs to Prevent Recurrence
CR 200007078 Engineering Manager Understanding of CR Threshold
CR 200007108, EDG 21, 23 GE CR120A relay failure analysis report
CR 200007265, Johnson SW pumps do not meet hydraulic requirements, 9/27/00
CR 200007418, Relief valve SWN-86 IST failure
CR 200007509, 26 SWPS auto blowdown valve failed to stroke, 10/4/00
CR 200007600, Increase in Service Water Pump Load on EDGs
CR 200007667, Yoke Bushing Broke While Closing Valve JW-5.
CR 200007718, Stranded issues from "inappropriately" closed CR
CR 200007740, 480 V DBD Missing Reference
CR 200007742, 480 V DBD Missing Reference
CR 200007815, SWPS 24 not rotating, 10/13/00
CR 200007923, During the monthly notification drill, CAN was found inoperable
CR 200008089, Water Hammer Potential on Non-Essential SW Header, 10/23/00
CR 200008090, SW System flow model calculation deficiencies, 10/23/00
CR 200008156, EDG Loading Study Requires Revision
CR 200008249, Instrument Air Compressor smoke detector indicating light failure
CR 200008293, Licensed Operator Requalification Program
CR 200008448, Pager vendor inadvertently activated all ERO pagers while testing two. Used wrong test code and caused confusion
CR 200008472, Operator requalification examination results
CR 200008478, 21 SWP oil sample trending toward dilution of oil., 11/2/00
CR 200008487 Use of Circular Logic in CR Closure
CR 200008774, Radiological equipment deficiencies found during drill on 11/9/2000
CR 200008786, Valves SWN-6 and SWN-7 appear to not be properly supported, 11/9/00
CR 200008813, Deficiencies identified from November 9, 2000 emergency exercise
CR 200008829, SW Zurn strainer dp greater than 4 psid acceptance criterion, 11/10/00
CR 200008854, Oil in 24 SW pump appears to be emulsified, 11/11/00
CR 200008981, ERDS Inoperable During Training Session
CR 200009752, Inadequate Safety Evaluation 98-402-PR Regarding Changing EDG Start Time From 10 seconds to 10.5 seconds.

Condition Reports (Cont.)

CR 200009753, Inadequate Operability Determinations 98-012 and 98-013 Regarding Exceeding the EDG 10 second Start Time.
CR 200009927 Part 21 Review of Foxboro Relay for RWST Level Alarm
CR 200009963, Offsite Monitor procedure inadequacies regarding TLDs
CR 200009972, Instrument Air Compressor smoke detector timer failure
CR 200010025, Offsite Monitor procedure inadequacies
CR 200010120, Qualification to operate a crane
CR 200010268, SOP 1.7 discrepancy'
CR 200010277, QA audit finding regarding procedure for making an "emergency repair" under a
declared emergency condition
CR 200010278, QA audit finding regarding EP-AD-02 containing inadequacies and ambiguities
CR 200010279, QA audit finding regarding the adequacy of JNC procedure for preparing initial news releases during an event
CR 200010284, QA audit finding regarding alternating ERO requalification exams
CR 200010322, Alternate location for decontamination and applicable procedures
CR 200010476, Emergency Alarms & pagers are inaudible in Plant Cafeteria
CR 200010490, Does E-Plan training use the systematic approach to training which is used in operator training and technical training programs?
CR 200100170, No basis calculation for SW pumps IST quarterly tests' acceptance criteria, 1/5/01
CR 200100201, Maintenance planning area page
CR 200100290, Respirator qualification lapses for Onsite and Offsite monitors
CR 200100487 Automatic Self Locking Door for Employee Concerns Program Office
CR 200100499, pipe wrench left above instrument air compressors
CR 200100502, Heat trace circuit light intermittent
CR 200100510, Concern with 21 CCW heat exchanger holddown bolts, 1/17/01
CR 200100511, Balance of SW flows through DG heat exchangers, 1/17/01
CR 200100512, Corrosion on stainless steel line in CCW Heat exchanger
CR 200100513, Nuts on 21CCHX do not have full thread engagement, 1/17/01
CR 200100520, Leak rate program
CR 200100533, Page party speaker in NPO office
CR 200100545 Employee Concern Regarding Discontinuance of Posting CRs on Intranet
CR 200100549, NRC Found Instrument Out of Calibration.
CR 200100566, No test of non-essential SW header, 1/18/01
CR 200100577, unfastened deck plates in EDG building
CR 200100586, No condition report generated for failed acceptance criteria in PT-R93, 1/18/01
CR 200100599, Conclusions for Calculation FEX-00148-00
CR 200100606, Dwg 9321-F-4046, EDG Building Control Air Did Not Show 6th Building Exhaust Fan.
CR 200100611, Dwg 9321-F-1460-11, EDG Building Incorrectly Labeled 6th Building Exhaust Fan as #322 (number for the 5th fan) Versus #323.
CR 200100619 Employee Concerns Program Deficiencies

Condition Reports (Cont.)

CR 200100657, Loop 2 Delta-T Deviation Alarm
CR 200100663, scaffolding around instrument air compressor unsupported at base
CR 200100667, housekeeping items in EDG building
CR 200100669, ICPM 1508, Delta -T Deviation Alarm Setpoints
CR 200100700, oil pad fire protection assessment
CR 200100702, untimely generation of CR for instrument air scaffolding operability question
CR 200100714, past operability of instrument air scaffolding
CR 200100749, EDG 22 control room undervoltage annunciator alarming
CR 200100759, Field operator confusion over 125v DC control power indication
CR 200100773, 480V work orders incorrectly categorized (CM vs. other)
CR 200100782, EDG Fuel Oil Storage Issues
CR 200100783, Reduced SW flow to instrument air coolers, 1/23/01
CR 200100786, Temporary power cord connected to Air Compressor in EDG building
CR 200100788, EDG building sump backflow valves dirty
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NP-00-18111, Repair EDG 21 Day Tank Transfer Switch
NP-00-18162, Repair EDG 23 Cylinder Thermocouple Loose Fittings
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ATTACHMENT 4

PARTIAL LIST OF PERSONS CONTACTED

Adams, E. - Dosimetry Technician
Altic, Bill - Senior Instructor, Shift Training Advocate
Andreozzi, Vincent - 480 Vac Electrical System Engineer
Baumstark, J. - VP Engineering
Bishop, Dave - Work Week Manager
Blatt, Michael - External Affairs
Blichfeldt, C. - Maintenance
Brooks, Kevin - Operations
Brovarski, C. - Communications Manager
Browne, F. - Maintenance
Buletta, John - Watch Engineer
Burns, T. - Supervisor, Nuclear Environmental Manager
Burns, R. - Emergency Planning Analyst
Carpenter, S. - Response Team Maintenance Contact
Comax, Denis - Watch Engineer, Operations
Dahl, George - Fire Protection Engineer
Dean, Greg - Assistant Operations Manager
Dean, Roger - Senior Instructor, Shift Training Advocate
DeGasperis, Eddie - Nuclear Plant Operator
DiUglio, Anthony - Employee Concerns Program Manager
Dong, Ang - I & C Supervisor
Donnegan, M. - HP Manager
Dunleavy, C. - Administrative Officer, Orange County Office of Emergency Management
Durr, B. - Shift Manager,
Elam, T. - Outage Planning Supervisor
Entenberg, M. - Section Manager, Electrical Design and Facilities Engineering
Ferraro, T. - Sr. Emergency Planning Engineer
Finucan, Ken - Senior Quality Assurance Examiner
Freer, S. - Computer Applications
Gibb, J. - New York Emergency Management Agency
Ginsburg, Arthur - Chemistry Department
Goebel, Joseph - Lead Auditor - Quality Assurance
Gotchius, Ed - Manager of Safety Analysis
Greeley, D. - Asst. Director, Rockland County Office of Fire & Emergency Service
Greene, D. - Asst. Director, Orange County Office of Emergency Management
Griffith, Phil - PRA Supervisor
Gross, G. - Instrument Supervisor
Hale, J. - Senior Consultant
Horner, T. - Electrical Design Engineer
Hornyak, Michael - Corrective Action Group
Huestis, M. - Outage Manager
Inzirillo, F. - EP Manager
Jayaraman, Vadakkant - Engineering
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Klein, Tom - Electrical Design Technical Specialist
Langerfeld, R. - Senior Reactor Operator, Generation Support

Lasley, R. - Department Manager, System Performance
Lee, A. - Sr. Emergency Planning Consultant, OSSI
Libby, Earl - Senior Instructor
Lijoi, J. - Control Room Supervisor
MacKenzie, Bruce - Corrective Action Group
Mansell, Jon - Outage Coordinator
Marguglio, Ben - Quality Assurance Auditor
Margulio, B. - Quality Assurance Auditor
McCaffrey, T. - Electrical System Engineer
McKee, Tom - Test Engineer
Meek, Brian - EDG and Gas Turbine System Engineer
Miele, Michael - Radprotection and Chemistry
Miller, Mark - Operations
Murdock, John - Shift Manager
Murphy, L. - Director, Westchester County Office of Emergency Management
Murphy, Diedre - Nuclear Training Manager
Naku, Klaus - Inspection Response Team Member
Nichols, John - Operations Training Section Manager
Parker, D. - Maintenance Section Manager
Parry, J. - Project Manager
Pehush, J. - 50.54(f) Reviewer, Setpoint Group
Poplees, Frank - Chemistry Instructor
Porrier, Tom - Work Control Manager
Pries, D. - Maintenance
Rampolla, M. - Director, Putnam County Office of Emergency Management
Ready, Jim - Field Support Supervisor
Reynolds, Joseph - Corrective Action Group
Robinson, H. - Senior Electrical Design Engineer
Rogers, Mike - Shift Training Advocate, Computer Applications Liaison
Rohla, Ross - Operations
Rowland, J. - 50.54(f) Reviewer, Configuration Management Group
Rumold, Jerry - Field Support Supervisor
Russell, Pat - Corrective Action Group Manager
Santini, Phil - Watch Engineer
Shah, Dean - Engineering
Shalabi, Khalil - Work Process Manager
Shoen, P. - Shift Manager
Smith, Bill - Assistant Operations Manager for Planning
Smith, L. - Section Manager, Civil Design Engineering
Speedling, Paul - Fire Protection Specialist
Teague, Thomas - Chemistry Department
Toscano, Jim - Unit Coordinator
Townsend, Larry - Shift Manager, Operations
Tumicki, Michael - Corrective Action Group
Tuohy, J. - Department Manager, Design Engineering

Ventosa, John - Site Engineering
Villani, L. - Response Team Engineering Lead Contact
Von Staden, Pat - Assistant Operations Manager (Corrective Actions/Training Coordinator)

Persons Contacted (Cont.)

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Xing, Michael - PSA Contractor
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U.S. ATOMIC ENERGY COMMISSION

REGULATORY GUIDE

DIRECTORATE OF REGULATORY STANDARDS

REGULATORY GUIDE 1.78

ASSUMPTIONS FOR EVALUATING THE HABITABILITY OF A NUCLEAR POWER PLANT CONTROL ROOM DURING A POSTULATED HAZARDOUS CHEMICAL RELEASE

A. INTRODUCTION

Criterion 4, "Environmental and missile design bases," of Appendix A "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Licensing of Production and Utilization Facilities," requires, in part, that structures, systems, and components important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. Criterion 19, "Control room," requires that a control room be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions. Release of hazardous chemicals can potentially result in the control room becoming uninhabitable. This guide describes assumptions acceptable to the Regulatory staff to be used in assessing the habitability of the control room during and after a postulated external release of hazardous chemicals and describes criteria that are generally acceptable to the Regulatory staff for the protection of the control room operators. This guide does not consider the explosion or flammability hazard of these chemicals, which also must be addressed. The Advisory Committee on Reactor Safeguards has been consulted concerning this guide and has concurred in the regulatory position.

B. DISCUSSION

The control room of a nuclear power plant should be appropriately protected from hazardous chemicals that may be discharged as a result of equipment failures, operator errors, or events and conditions outside the control of the nuclear power plant.

At present, there is no one standard design evaluation method in use for evaluating the habitability of control rooms during the course of all postulated hazardous chemical releases.[1] However, the "Accidental-Episode Manual"[2] prepared for the Environmental Protection Agency (EPA) in April 1972 presents a method for the evaluation and estimation of the area affected by the release of hazardous chemicals as a function of source strength, type of chemical, distance from source, and meteorology. The "Accidental-Episode Manual" rates accident potentials from both mobile and stationary sources and identifies some hazardous chemicals that may be released. Human tolerance for hazardous chemicals should be considered in the design stage of nuclear facilities.

For hazardous chemicals shipped on routes near the nuclear power plant, the shipment frequencies specified for consideration in this guide (Regulatory Position 2) reflect the relative accident probabilities for common modes of transportation. A discussion of accident rates for various transportation modes can be found in Appendix A, "Analysis of Transportation Accidents," of WASH-1238.[3] Consideration is also given to the quantity of hazardous chemical shipped.

The purpose of this guide is to identify those chemicals which, if present in sufficient quantities, could result in the control room becoming uninhabitable. The general design considerations that are used in assessing

[1] A regulatory guide is being developed to describe specific design provisions and procedures that are acceptable to mitigate hazards to control room operators from an onsite chlorine release.

[2] Office of Air Programs, Publication APTD-1114. Copies may be obtained from National Technical Information Service, 5285 Port Royal Road, Springfield, Virginia 22151.

[3] WASH-1238, "Environmental Survey of Transportation of Radioactive Materials to and from Nuclear Power Plants" December 1972. Copies may be obtained from National Technical Information Service, 5285 Port Royal Road, Springfield, Virginia 22151.

the capability of the control room, as designed, to withstand hazardous chemical releases occurring either on the site or within the surrounding area are presented. Some of the chemicals specifically identified, such as helium and nitrogen, should generally not present a problem except when very large quantities are stored on the site. Asphyxiating chemicals such as these need not be considered unless a significant fraction of the control room air could be displaced as a result of their release.

Fire-fighting equipment used for fighting chemical and electrical fires should be considered as a potential source of hazardous chemicals.

This guide identifies chlorine as a potentially hazardous chemical. Chlorine is used in a majority of nuclear power plants for water treatment and is normally stored onsite as a liquified gas. A separate guide will be issued to describe the detailed design provisions which are considered adequate to protect control room operators from an onsite chlorine release.

C. REGULATORY POSITION

In evaluating the habitability of a nuclear power plant control room during a postulated hazardous chemical release the following assumptions should be made:

1. If major depots or storage tanks of hazardous chemicals such as the chemicals listed in Table C-1 of this guide are known or projected to be present within a five-mile radius of the reactor facility, these chemicals should be considered in the evaluation of control room habitability.[4] Whether a major depot or storage area constitutes a hazard is determined on the basis of the quantity of stored chemicals, the distance from the nuclear plant, the inleakage characteristics of the control room, and the applicable toxicity limits (see Regulatory Position 4 for definition). Table C-2 gives the criteria to be used in evaluating the hazards of chemicals to control rooms. A procedure for adjusting the quantities given in Table C-2 to appropriately account for the toxicity limit of a specific chemical, meteorology conditions of a particular site, and air exchange rate of a control room is presented in Appendix A of this guide.

Chemicals stored or situated at distances greater than five miles from the facility need not be considered because, if a release occurs at such a distance, atmospheric dispersion will dilute and disperse the incoming

[4] The list of chemicals given in Table C-1 is not all-inclusive but indicates the chemicals most commonly encountered. See also "Guide for Emergency Services for Hazardous Materials (1973)–Spills, Fires, Evacuation Areas" copies of which may be obtained from the U.S. Department of Transportation, Office of Hazardous Materials, Washington, D.C.

TABLE C-1

SOME HAZARDOUS CHEMICALS POTENTIALLY INVOLVED IN ACCIDENTAL RELEASES FROM STATIONARY AND MOBILE SOURCES[a]

Chemical	Toxicity Limit[b] ppm[c]	Toxicity Limit[b] mg/m³ [d]	Chemical	Toxicity Limit ppm	Toxicity Limit mg/m³
Acetaldehyde	200	360	Ethylene oxide	200	180
Acetone	2000	4800	Fluorine	2	4
Acrylonitrile	40	70	Formaldehyde	10	12
Anhydrous ammonia	100	70	Helium		asphyxiant
Aniline	10	38	Hydrogen cyanide	20	22
Benzene	50	160	Hydrogen sulfide	500	750
Butadiene	0.1%[e]	2200	Methanol	400	520
Butenes		asphyxiant	Nitrogen (compressed or liquified)		asphyxiant
Carbon dioxide	1.0%[e]	1840	Sodium oxide	–	2
Carbon monoxide	0.1%[e]	1100	Sulfur dioxide	5	26
Chlorine	15	45	Sulfuric acid	–	2
Ethyl chloride	10000	26000	Vinyl chloride	1000	2600
Ethyl ether	800	2400	Xylene	400	1740
Ethylene dichloride	100	400			

[a] This list is not all-inclusive but indicates the hazardous chemicals most commonly encountered.

[b] Adapted from Sax's "Dangerous Properties of Industrial Materials."

[c] Parts of vapor or gas per million parts of air by volume at 25°C and 760 torr (standard temperature and pressure).

[d] Approximate milligrams of particulate per cubic meter of air, at standard temperature and pressure, based on listed ppm values.

[e] Percent by volume.

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TABLE C-2

EXAMPLES OF WEIGHTS OF HAZARDOUS CHEMICALS THAT REQUIRE CONSIDERATION IN CONTROL ROOM EVALUATIONS (FOR A 50 mg/m^3 TOXICITY LIMIT AND STABLE METEOROLOGICAL CONDITIONS[a])

Distance From Control Room (miles)[b]	*Weight (1000 lb)*		
	Type A Control Room[c]	*Type B Control Room*	*Type C Control Room*
0.3 to 0.5	9	2.3	0.1
0.5 to 0.7	35	8.8	0.4
0.7 to 1.0	120	20	1.0
1 to 2	270	52	2.5
2 to 3	1300	280	13
3 to 4	3700	780	33
4 to 5	8800	1400	60

[a] For different toxicity limits as given in Table C-1 and different meteorological conditions, the weights should be proportionately scaled as described in Appendix A.

[b] All hazardous chemicals present in weights greater than 100 lb within 0.3 mile of the control room should be considered in a control room evaluation.

[c] Control room types (Appendix A illustrates the use of this table for other air exchange rates):

Type A – A "tight" control room having low leakage construction features and the capability of detecting at the fresh air intake those hazardous chemicals stored or transported near the site. Detection of the chemical and automatic isolation of the control room are assumed to have occurred. An air exchange rate of 0.015 per hour is assumed (0.015 of the control room air by volume is replaced with outside air in one hour). The control room volume is defined as the volume of the entire zone serviced by the control room ventilation system. The assumption that the air exchange rate is less than 0.06 per hour requires verification by field testing.

Type B – Same as Type A, but with an air exchange rate of 0.06 per hour. This value is typical of a control room with normal leakage construction features. The assumption that the air exchange rate is less than 0.06 per hour, requires verification by field testing.

Type C – A control room that has not been isolated, has no provision for detecting hazardous chemicals, and has an air exchange rate of 1.2 per hour.

plume to such a degree that there should be sufficient time for the control room operators to take appropriate action. In addition, the probability of a plume remaining within a given sector for a long period of time is quite small.

2. If hazardous chemicals such as those indicated in Table C-1 are known or projected to be frequently shipped by rail, water, or road routes within a five-mile radius of a nuclear power plant, estimates of these shipments should be considered in the evaluation of control room habitability. The weight limits of Table C-2 (adjusted for the appropriate toxicity limit, meteorology, and control room air exchange rate) apply also to frequently shipped quantities of hazardous chemicals. Shipments are defined as being frequent[3] if there are 10 per year for truck traffic, 30 per year for rail traffic, or 50 per year for barge traffic.[3] If the quantity, per shipment, of hazardous chemicals frequently shipped past a site is less than the adjusted quantity shown in Table C-2 for the control room type being evaluated, the shipments need not be considered in the analysis.

3. In the evaluation of control room habitability during normal operation, the release of any hazardous

[3] For explosive hazards, a lower number of shipments would be considered frequent since the effects of an explosion would be independent of wind direction.

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chemical to be stored on the nuclear plant site in a quantity greater than 100 lb should be considered. Any hazardous chemical stored onsite should be accompanied by instrumentation that will detect its escape, set off an alarm, and provide a readout in the control room.

4. The toxicity limits should be taken from appropriate authoritative sources such as those listed in the References section. For each chemical considered, the values of importance are the human detection threshold and the maximum concentration that can be tolerated for two minutes without physical incapacitation of an average human (i.e., severe coughing, eye burn, or severe skin irritation). The latter concentration is considered the "toxicity limit." Table C-1 gives the toxicity limits (in ppm by volume and mg/m^3) for the chemicals listed. Where these data are not available, a determination of the values to be used will be made on a case-by-case basis.

5. Two types of industrial accidents should be considered for each source of hazardous chemicals: maximum concentration chemical accidents and maximum concentration-duration chemical accidents.

a. For a maximum concentration accident, the quantity of the hazardous chemical to be considered is the instantaneous release of the total contents of one of the following: (1) the largest storage container falling within the guidelines of Table C-2 and located at a nearby stationary facility, (2) the largest shipping container (or for multiple containers of equal size, the failure of only one container unless the failure of that container could lead to successive failures) falling within the guidelines of Table C-2 and frequently transported near the site, or (3) the largest container stored onsite (normally the total release from this container unless the containers are interconnected in such a manner that a single failure could cause a release from several containers.)

For chemicals that are not gases at 100°F and normal atmospheric pressure but are liquids with vapor pressures in excess of 10 torr, consideration should be given to the rate of flashing and boiloff to determine the rate of release to the atmosphere and the appropriate time duration of the release.

The atmospheric diffusion model to be used in the evaluation should be the same as or similar to the model presented in Appendix B of this guide.

b. For a maximum concentration-duration accident, the continuous release of hazardous chemicals from the largest safety relief valve on a stationary, mobile, or onsite source falling within the guidelines of Table C-2 should be considered. Guidance on the atmospheric diffusion model is presented in Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors," and Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors."

6. The value of the atmospheric dilution factor between the release point and the control room that is used in the analysis should be that value that is exceeded only 5% of the time.

When boiloff or a slow leak is analyzed, the effects of density on vertical diffusion may be considered if adequately substantiated by reference to data from experiments. Density effect of heavier-than-air gases should not be considered for releases of a violent nature or for released material that becomes entrained in the turbulent air near buildings.

7. For both types of accidents described in Regulatory Position 5 above, the capability of closing the air ducts of the control room with dampers and thus isolating the control room should be considered in the evaluation of control room habitability. In particular, the time required to shut off or redirect the intake flow should be justified. The detection mechanism for each hazardous chemical should be considered. Human detection may be appropriate if the buildup of the hazardous chemical in the control room is at a slow rate due to slow air turnover. The air flows for infiltration, makeup, and recirculation should be considered for both normal and accident conditions. The volume of the control room and all other rooms that share the same ventilating air, during both normal conditions and accident conditions, should be considered. The time required for buildup of a hazardous chemical from the detection concentration to the toxicity limit should be considered.[6] Table C-3 of this guide contains a sample list of the chemical and control room data needed for the evaluation of control room habitability.

8. In the calculation of the rate of air infiltration (air leaking into the control room from ducts, doors, or other openings) with the control room isolated and not pressurized, use of the following assumptions is suggested:

a. A pressure differential of 1/8 inch water gauge across all leak paths.[7]

[6] The time from detection to incapacitation should be greater than two minutes. Two minutes is considered sufficient time for a trained operator to put a self-contained breathing apparatus into operation, if these are to be used.

[7] This pressure differential accounts for wind effects, thermal column effects, and barometric pressure changes. It does not account for pressure differences resulting from the operation of ventilation systems supplying zones adjacent to the control room. It should be adjusted appropriately when the ventilation system supplies zones adjacent to the control room.

TABLE C-3

TYPES OF CHEMICAL AND CONTROL ROOM DATA FOR HABITABILITY EVALUATION

CHEMICAL

1. Name of hazardous chemical.
2. Type of source (stationary, mobile, or onsite).
3. Human detection threshold, ppm.
4. Maximum allowable two-minute concentration (toxicity limit as defined in Regulatory Position 4, ppm and mg/m³).
5. Maximum quantity of hazardous chemical involved in incident.
6. Maximum continuous release rate of hazardous chemical.
7. Vapor pressure, torr, of hazardous chemical (at maximum ambient plant temperature).
8. Fraction of chemical flashed and rate of boiloff when spilling occurs.
9. Distance of source from control room, miles.
10. Five percentile meteorological dilution factor between release point and control room for instantaneous and continuous releases.

CONTROL ROOM

1. Volume of control room, including the volume of all other areas supplied by the control room emergency ventilation system, ft³.
2. Normal flow rates for volume defined above, cfm:[a]
   - unfiltered inleakage or makeup air,
   - filtered makeup air,
   - filtered recirculated air,
3. Emergency flow rates for volume defined above, cfm[a] (as in item 2. above).
4. Time required to isolate the control room, sec.

[a] "Filtered air" refers to the air filtered through filters whose removal capability for the particular chemical being considered has been established.

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b. The maximum design pressure differential for fresh air dampers on the suction side of recirculation fans.

9. When the makeup air flow rate required to pressurize the control room is calculated, a positive pressure differential of 1/4 inch water gauge should be assumed in the control room relative to the space surrounding the control room.

10. To account for the possible increase in air exchange due to ingress or egress, an additional 10 cfm of unfiltered air should be assumed for those control rooms without airlocks. This additional leakage should be assumed whether or not the control room is pressurized.

11. If credit is taken in the evaluation for the removal of hazardous chemicals by filtration or other means, the experimental basis for the dynamic removal capability of the removal system for the particular chemical being considered should be established.

12. Concurrent chemical release of container contents during an earthquake, tornado, or flood should be considered for chemical container facilities that are not designed to withstand these natural events. It may also be appropriate to consider release from a single onsite container or pipe coincident with the radiological consequences of a design basis loss-of-coolant accident, if the container facilities are not designed to withstand an earthquake.

13. If consideration of possible accidents for any hazardous chemical indicates that the applicable toxicity limits may be exceeded, self-contained breathing apparatus of at least one-half hour capacity or a tank source of air with manifold outlets and protective clothing, if required, should be provided for each operator in the control room. Additional air capacity with appropriate equipment should be provided if a chemical hazard can persist longer than one-half hour. For accidents of long duration, sufficient air for six hours (coupled with provisions for obtaining additional air within this time period) is adequate. Each operator should be taught to distinguish the smells of hazardous chemicals peculiar to the area. Instruction should include a periodic refresher course. Practice drills should be conducted to ensure that personnel can don breathing apparatus within two minutes.

14. Detection instrumentation, isolation systems, filtration equipment, air supply equipment, and protective clothing should meet the single-failure criterion. (In the case of self-contained breathing apparatus and protective clothing, this may be accomplished by supplying one extra unit for every three units required.)

15. Emergency procedures to be initiated in the event of a hazardous chemical release within or near the station should be written. These procedures should address both maximum concentration accidents and maximum concentration-duration accidents and should identify the most probable chemical releases at the station. Methods of detecting the event by station personnel, both during normal workday operation and during minimum staffing periods (late night and weekend shift staffing), should be discussed. Special instrumentation that has been provided for the detection of hazardous chemical releases should be described including sensitivity, action initiated by detecting instrument and level at which this action is initiated, and Technical Specification limitations on instrument availability. Criteria should be defined for the isolation of the control room, for the use of protective breathing apparatus or other protective measures, and for orderly shutdown or scram. Criteria and procedures for evacuating nonessential personnel from the station should also be defined.

Arrangement should be made with Federal, State, and local agencies or other cognizant organizations for the prompt notification of the nuclear power plant when accidents involving hazardous chemicals have occurred within five miles of the plant

REFERENCES

1. "Matheson Gas Data Book," Fourth Edition, The Matheson Company, Inc., East Rutherford, New Jersey (1966).

2. N. Irving Sax, "Dangerous Properties of Industrial Materials," Third Edition, Reinhold Book Corp., New York, New York (1968).

3. "Hygienic Guide Series," published by the American Industrial Hygiene Association, William E. McCormick, Executive Director, 66 South Willer Road, Akron, Ohio 44313.

4. "Toxic Substances List, 1973 Edition," U.S. Department of Health, Education, and Welfare, National Institute for Occupational Safety and Health, Rockville, Maryland 20852 (June, 1973). Prepared for NIOSH under contract by Tracor Jitco, Inc., 1300 East Gude Drive, Rockville, Maryland 20852.

5. "Threshold Limit Values for Chemical Substances and Physical Agents in the Workroom Environment," American Conference of Governmental Industrial Hygienists, Cincinnati, Ohio (1973).

APPENDIX A

PROCEDURE FOR CALCULATING WEIGHTS OF HAZARDOUS CHEMICALS NECESSITATING THEIR CONSIDERATION IN CONTROL ROOM EVALUATION

The weights presented in Table C-2 are based on the following assumptions:

1. A toxicity limit of 50 mg/m^3
2. Air exchange rates for the three control room types of 0.015, 0.06, and 1.2 per hour
3. Pasquill stability category F

These conditions are generally applicable to most of today's plants for a gas such as chlorine (toxicity limit of 45 mg/m^3). If the toxicity limit, air exchange rate, or meteorological conditions are significantly different from the assumptions used in Table C-2, simple corrections that result in only minor errors can be made.

Toxicity Limit

The weights presented in Table C-2 are directly proportional to the toxicity limit. If a particular chemical has a toxicity limit of 500 mg/m^3, the weights from the table (based on 50 mg/m^3) are increased by a factor of ten.

Air Exchange Rate

Table C-2 weights are inversely proportional to the air exchange rate. If a type C control room has an exchange rate of 2.4 per hour, the weights from the table (based on 1.2 per hour) are decreased by a factor of two. When adjustments of this type are made, the control room type (A, B, or C) that has an air exchange rate closest to that of the control room in question should be selected.

It should be noted that the use of an air exchange rate of less than 0.06 per hour for an isolated control room requires that the control room leakage rate be verified by periodic field testing.

For control rooms without automatic isolation capability, the weights given for Type C control rooms should be used, appropriately adjusted for the actual fresh air exchange rate. Weights for Type B control rooms should be used when the control room has automatic isolation. Weights for Type A control rooms, appropriately adjusted for the design isolated air exchange rate, should be used only when the control room has been designed specifically for low inleakage.

Pasquill Stability Category

The weights given in Table C-2 are based on stable atmospheric dispersion conditions equivalent to Pasquill Condition F. This represents the worst five percentile meteorology observed at the majority of nuclear plant sites and, for most cases, there will be no need to adjust the weights because of meteorology. If it is determined that the worst five percentile meteorology is better or worse than Condition F, the following adjustments should be made:

Five Percentile Dispersion Category	*Weight Multiplication Factor*
E	2.5
F	1.0
G	0.4

Appendix B provides additional discussion of atmospheric dispersion.

APPENDIX B

DIFFUSION CALCULATIONS FOR AN INSTANTANEOUS (PUFF) RELEASE

1. Diffusion Equation

The diffusion equation for an instantaneous (puff) groundlevel release with a finite initial volume is:[1]

$$\frac{\chi}{Q_I} = \left[7.87\left(\sigma_{x,y}^2 + \sigma_I^2\right)\left(\sigma_z^2 + \sigma_I^2\right)^{1/2}\right]^{-1}$$

$$\cdot \exp\left[-\frac{1}{2}\left(\frac{x^2}{\sigma_x^2 + \sigma_I^2} + \frac{y^2}{\sigma_y^2 + \sigma_I^2} + \frac{z^2}{\sigma_z^2 + \sigma_I^2}\right)\right]$$

where:

$\frac{\chi}{Q_I}$ = unit concentration at coordinates x, y, z from the center of the puff, m^{-3}

$\sigma_x, \sigma_y, \sigma_z$ = standard deviations of the gas concentration in the horizontal alongwind, horizontal crosswind, and vertical crosswind directions, respectively (assume $\sigma_x = \sigma_y$), m

$7.87 = 2^{1/2}\pi^{3/2}$

σ_I = initial standard deviation of the puff, m

$= \left[\frac{Q_I}{7.87\chi_o}\right]^{1/3}$ where Q_I is the puff release quantity, g, and χ_o is the density of the gas at standard conditions, g/m^3.

x, y, z = distance from the puff center in the horizontal alongwind, horizontal crosswind, and vertical crosswind directions, respectively, m.

[1] G.R. Yanshey, E.H. Markee, Jr., and A.P. Richter "Climatography of the National Reactor Testing Station," IDO-12048, January 1966. Copies may be obtained from National Technical Information Service, 5285 Port Royal Road, Springfield, Virginia 22151.

Windspeed does not enter into the determination of unit concentration per se, but does affect the time-integrated concentration since it determines cloud passage time. The variation of unit concentration at a specific stationary receptor location is determined by evaluating x in the exponential term in the above equation as follows:

$$x = D - ut$$

where D is the source-receptor distance, u is the windspeed, and t is the time after release.

2. Determination of Input Data

The following assumptions and methods should be applied when analyzing worst-case instantaneous source releases:

a. Select the appropriate stability category based on the worst five percentile meteorology observed at the site according to the ΔT method. Regulatory Guide 1.23 (Safety Guide 23), "Onsite Meteorological Programs," presents a classification of various atmospheric stability categories as a function of temperature change (ΔT) with height. Normally, this category will be Pasquill Condition F. In some cases, the worst case stability category may be either Pasquill Condition E or G. This occurs at sites having distinctly better or worse diffusion than is normally encountered. Figures 1 and 2 of this appendix include conditions E, F, and G and encompass the worst expected stability conditions at nearly all sites.

b. Determine the x, y, and z standard deviation values based on the Pasquill stability categories as presented in Figures 1 and 2.

c. Additional credit due to building wake or other dispersive phenomena may be allowed, depending on the properties of the released gas, the method of release, and the intervening topology or structures.

d. Windspeed should be selected to maximize the two-minute concentration within the control room.

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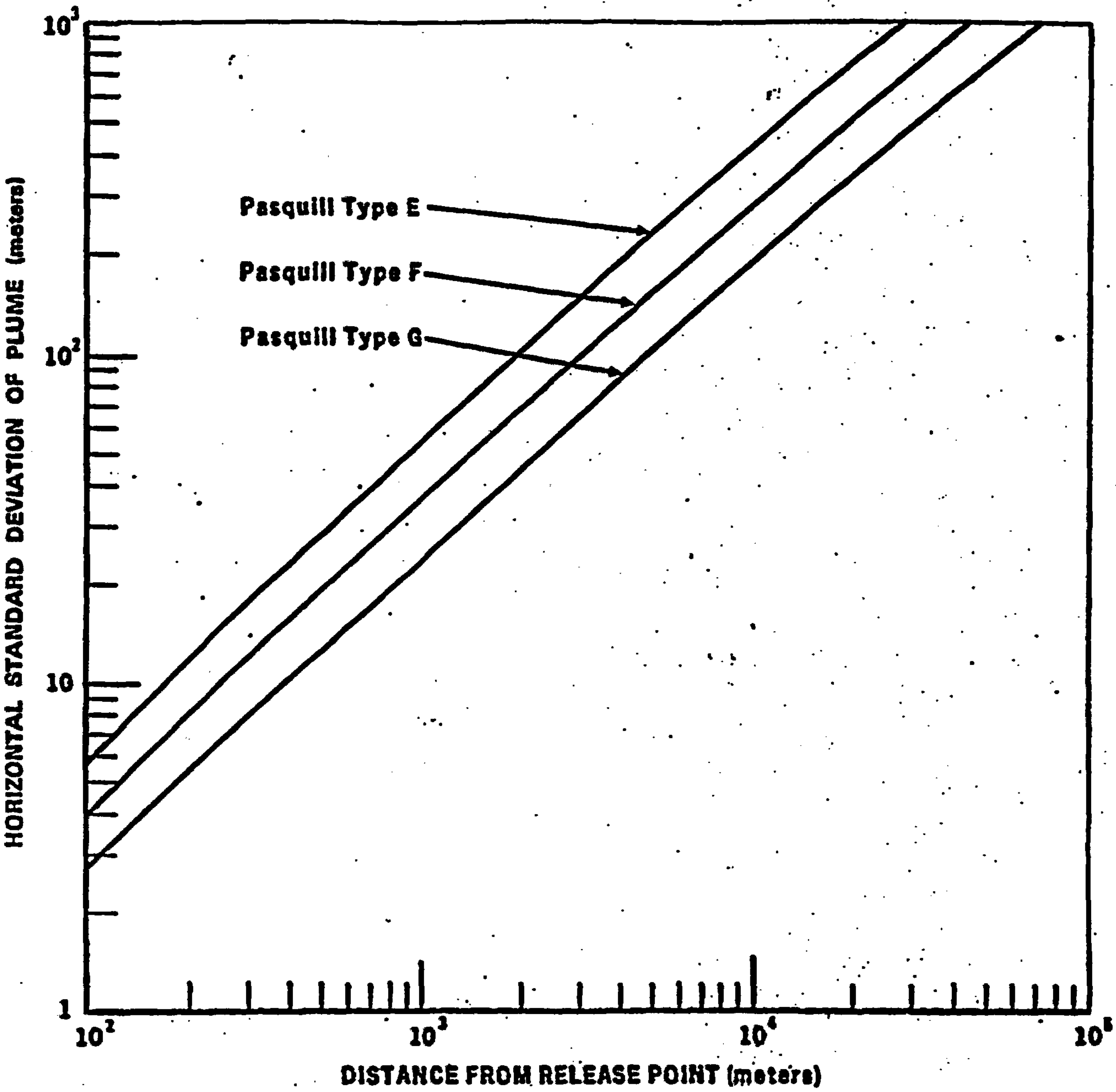

Figure 1. Horizontal Standard Deviation of Material in a Plume

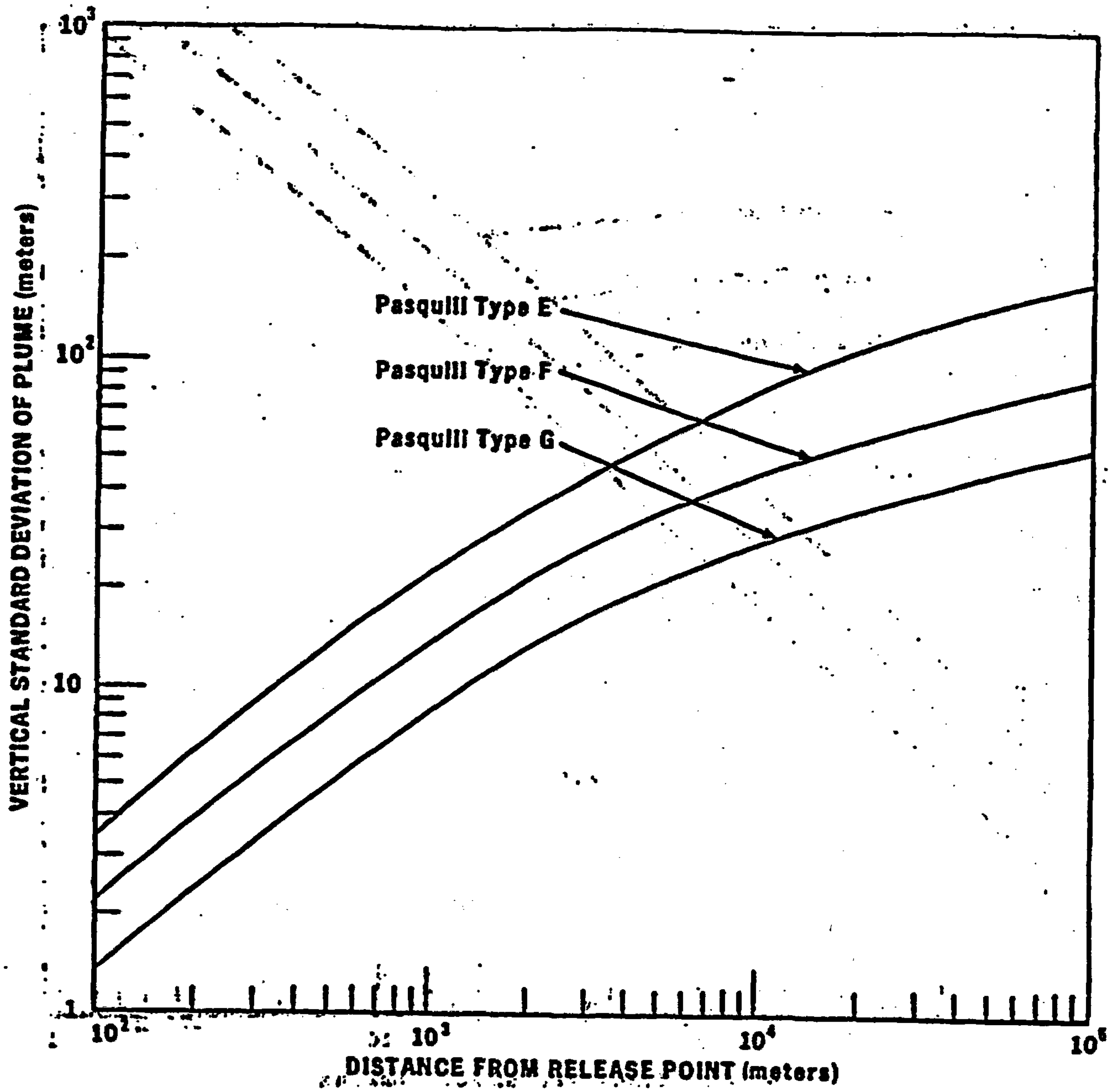

Figure 2. Vertical Standard Deviation of Material in a Plume

INSPECTION PROCEDURE 81110

OPERATIONAL SAFEGUARDS RESPONSE EVALUATION (OSRE)

PROGRAM APPLICABILITY: 2515

FUNCTIONAL AREA: PLANT SUPPORT (PLTSUP)

81110-01 INSPECTION OBJECTIVES

01.01 To evaluate a licensee's ability to respond to the external design basis threat[1] by focusing on (1) the interactions between a licensee's operations and security departments in establishing priorities for protecting equipment and (2) the protective strategies used.

01.02 To review the impact of security measures on safe plant operations.

81110-02 INSPECTION REQUIREMENTS

02.01 Management Overview of Protective Strategy

a. Meet with the appropriate segment of licensee management to review the licensee's fundamental strategy to protect against the design basis threat.

b. Ascertain armed response force manning levels, contingency equipment, and deployment positions.

c. Determine plant operations' participation in defining and validating the protective strategy.

d. Identify and discuss any perceived adverse impact on plant operations by security systems or procedures.

e. Review commitments to the licensee from the local law enforcement authority to determine response capabilities.

[1]OSRE Adversary Characteristics, which are detailed in a SAFEGUARDS INFORMATION document of the same name, dated August 29, 2000 and which is not publicly releaseable, are in Supplement A of this Inspection Procedure and also Inspection Procedure 71130.03.

02.02 Preliminary and On-Site Target Analysis

a. Preliminary Target Analysis

1. Conduct in-office target analysis of the plant design by using the Updated Final Safety Analysis Report (UFSAR), and other pertinent information.
2. Identify preliminary target sets.

b. On-Site Target Analysis

1. Review preliminary target sets with appropriate licensed senior reactor operator and/or design engineer to determine a realistic analysis approach.

2. Identify and resolve discrepancies to obtain a mutual agreement with the target sets.

02.03 Protected and Vital Area Tours

a. Protected Area Tour

1. Walk the protected area (PA) perimeter and assess potential routes of travel to target sets by an adversary.

2. Identify entry locations into the PA which are most likely to provide a challenge to the protective strategy.

b. Vital Area Tour. Conduct a tour of vital areas (VAs) to assess the physical location of and accessibility to target equipment, defensive positions, and defensive measures.

02.04 Table-Top Drills, Contingency Exercises, and Licensee Exercise Critiques

a. Table-Top Drills

1. Conduct table-top drills with appropriate security personnel to assess the number of responders, deployment positions, the deployment strategy, and relative response times.

2. Make initial evaluation of the licensee's protective strategy with respect to a variety of targets and challenges.

b. Contingency Exercises

1. Provide the licensee's mock adversary with target-set objectives, a profile of their would-be characteristic, a description of equipment at their disposal, and the point to enter the PA.

2. Observe the licensee conduct contingency exercises demonstrating its protective strategy.

3. Evaluate the effectiveness of the licensee's protective strategy relative to a timely response by a sufficient number of appropriately armed and protected responders.

4. Evaluate performance of exercise participants in simulating realistic conditions.

c. Licensee Exercise Critiques

1. Observe the licensee's critique of the contingency exercises to determine if weaknesses are identified and appropriately addressed.

2. Determine if lessons learned are normally incorporated into the protective strategy.

02.05 Deadly Force Analysis. Interview a sample of armed personnel to determine their understanding of the appropriate application of deadly force.

02.06 Tactical Training

a. Review the tactical training program to assess consistency with the protective strategy, as demonstrated during contingency exercises.

b. Review instructor certification and experience to determine appropriateness.

c. Assess availability of contingency weapons and equipment.

02.07 Firearms Training

a. Observe live fire at a firing range to assess the courses of fire in simulating the conditions likely to occur during an actual contingency at the site.

b. Evaluate effectiveness of range safety and assess security personnel's proficiency relative to weapons manipulation.

c. Assess the courses of fire relative to reinforcement of appropriate use of deadly force.

02.08 Safety/Safeguards Impact Review

a. Interview operations and security personnel to assess:

1. Impact of security measures and procedures on plant operations and personnel safety.

2. Coordination between operations and security during adverse conditions.

b. Conduct a walking tour of existing security measures to verify that:

   1. Both access to and egress from the PA and VAs would be prompt in an emergency condition or a situation that could lead to emergency conditions.

   2. Security practices and restrictions do not adversely impact the safe operation of the plant and personnel safety.

02.09 OSRE Team Meetings

a. Conduct team meetings to summarize and evaluate licensee activities.

b. Document conclusions and rationale.

c. Determine subjects to be discussed with licensee management.

02.10 Licensee Management and OSRE Team Meeting

a. Meet periodically with security management to discuss team findings.

b. Review and confirm scheduled activities.

c. Provide security management with selected exercise scenarios and mock adversary characteristics prior to the exercises.

02.11 Preliminary Exit Meeting and Exit Meeting

a. Preliminary Exit Meeting. Conduct a preliminary exit meeting with appropriate licensee management to review preliminary findings.

b. Exit Meeting. Conduct an exit meeting with appropriate licensee management.

81110-03 EVALUATION GUIDANCE

General Guidance

a. An OSRE is conducted at a power reactor to affirm a licensee's program to defend against the design basis threat (DBT) and to assess a licensee's access control system to ensure prompt access to vital equipment during emergency conditions or situations that could lead to emergency conditions. The Office of Nuclear Reactor Regulations (NRR) leads and usually initiates an OSRE.

b. The OSRE team shall consider a spectrum of external adversaries with varying characteristics. The lower range should consist of one dedicated individual with no special terrorist training, armed with a shotgun, rifle or handgun, a prybar to force-open doors, and dynamite to damage safety equipment. The highest level of the spectrum will include adversaries with the characteristics of the DBT as defined in 10 CFR 73.1(a).

c. The team should further assume that a significant radiological release would be the objective of an act of sabotage at a power reactor and should use prevention of significant core damage as an evaluation criterion. This criterion makes adversary success more difficult and more accurately reflects significant public health and safety concerns than would a criterion of prevention of damage to any piece of safety equipment.

Specific Guidance

During the entrance meeting with key licensee management, stress that there will be no clandestine testing and that safety is paramount. Explain that all testing will be conducted with the full awareness of the security manager. Further explain that in a few instances, an individual's awareness that a test is being conducted could negate the effectiveness of the test.

Inform the licencee that the "onsite security forces" are those specified and committed to in its physical security plan as the minimum number of armed responders always available to respond to a security contingency. This plan commitment could include those responders immediately available and those available at a later time. The number of responders participating in drills and exercises shall be limited to plan commitments.

The NRC may incorporate operator actions in evaluation of the success of the scenarios. This can be accomplished by a post-exercise analysis with operational solutions factored in, and/or with involvement of operational staff as active participants during the exercises, if a licensee chooses. This is not intended to require the involvement of the on-duty control room staff or the control room of the nuclear unit involved in the exercise.

Advise the licensee that, other than providing target sets and selecting exercise scenarios, the team will not play an active role in any exercise. To increase realism, the mock adversary force will be provided tours of the site only to the extent that a member of the public could tour the site. An NRC OSRE team member and one NRC contractor will participate in table-top drills. These individuals will then brief the mock adversary force on the point of attack and the intended target set.

Explain that enforcement action will be taken if a performance weakness is identified such that there is not high assurance the licensee has the capability to protect against the DBT. Any violation identified during an OSRE will be addressed in the OSRE report and enforcement action will be initiated consistent with agency enforcement policy. Less significant weaknesses will be identified in the OSRE report and a response from the licensee is required to address corrective actions planned. Regional inspectors will verify implementation of corrective measures during subsequent routine inspections.

Inform the licensee that exercises and firearm demonstrations will be video taped and that the videos are considered safeguards information and should be protected accordingly. Upon request, the team will provide a copy of the video tapes to the licensee.

03.01 Management Overview of Protective Strategy. Interview the security manager and staff to ascertain the quality of the security training, experience, and knowledge of trainers and frequency, relevancy, and depth of the training program. Verify data on type, number, and location of response equipment. Establish the number of security force members (SFMs) who will respond during the contingency exercises. This should be the minimum number that is normally available for and designated to respond to security contingencies. Interview appropriate operations personnel to assess operations' involvement with and input to the protective strategy.

Discuss the Local Law Enforcement Authority's (LLEA's) capabilities, timing and size of response, weapons, communications, and special capabilities such as Special Weapons and Tactics (SWAT) teams and hostage negotiators.

03.02 Preliminary and On-site Target Analysis

a. Preliminary Target Analysis. A preliminary target analysis should be conducted before the team arrives on site. The analysis should include a review of site-specific conditions and information from earlier reports and communications with the licensee, such as the plant's UFSAR and previous Regulatory Effectiveness Review (RER)/OSRE reports.

   A target set is a combination of equipment that would have to be disabled for an adversary to achieve core damage. Typical target sets are deduced from the minimum cut sets of a fault tree analysis with full credit given to operations' personnel training in routine and emergency procedures.

   In addition to the minimum cut sets, the team engineer may identify other damage-control resources that were not included in the licensee's operations procedures. These resources may also be added to the target sets. The target set does not assume any coincidental system failures, such as fire, flood, human error, or any equipment damage caused by a non-precipitated security event.

   The team engineer will identify several potential target sets that would likely lead to significant core damage. The team will select the target sets which will be used to evaluate the licensee's protective program.

b. On-site Target Analysis. The team engineer reviews the selected target sets with the licensee's operations/engineering staff: (1) to confirm the team's analysis, (2) to consider damage control features that could prevent radiological release, and (3) to determine whether additional targets should be added to any of the sets.

03.03 No inspection guidance provided.

03.04 Table-Top Drills, Contingency Exercises, and Licensee Exercise Critiques

a. Table-Top Drills. Conduct table-top, time-line drills. These drills should simulate overt external assaults. For each drill, interview a licensee contingency response team leader (RTL). The team identifies adversary characteristics at the beginning of each drill, typically as an assessment by the alarm station operator of a simulated perimeter alarm. Consider that the simulated assault might occur at a time advantageous to the adversary, such as during the night, on a weekend, or during foul weather.

A team member should play the adversary's role. As such, the would-be adversary should indicate entry point into the PA, movement toward safety equipment, and tactics employed. Have the RTL indicate how responding SFMs would be deployed to interdicting positions, based on the actual locations of the responding officers at the time of the drill.

If different outcomes are possible, their potential impact on the outcome of the drill should be analyzed. Drills should not be viewed as pass or fail, but should be used as tools to identify those elements that are critical to the licensee's achieving successful results in an actual contingency.

b. Contingency Exercises. Normally, the team observes a minimum of four licensee contingency exercises to evaluate the licensee's capabilities relative to interdicting the adversaries in a timely manner with a sufficient number of appropriately armed responders in protected positions. Assess the realism of actions by exercise participants (adversaries, responders, alarm station operators, operations personnel, controllers, etc.). Licensee exercise controllers should judge the outcome of all adversary-responder engagements.

The exercises are used to evaluate the licensee's efforts at establishing priorities for protection of equipment and the protective strategies for deployment of officers and equipment; individual and team tactical movement, command and control; communications in a contingency setting; defensive positions; and barriers. Observe and assess coordinated contingency response efforts by operations and security.

Observe the use of simulated/training weapons by armed responders during the exercises. Analyze and discuss instances with the licensee when simulated fire was directed at unknown or friendly personnel, or in a direction which could potentially cause injury or unnecessary damage. This analysis may include additional interviews with response personnel.

c. Licensee Exercise Critiques. At least one team member should observe the licensee's critique of each exercise. The entire team should meet after each day's exercises to conduct a team-only critique of the exercises and the licensee's critiques. Evaluate the licensee's utilization of exercises both as a training tool and as a means of self-auditing the protective strategy. Determine if the licensee uses this or any other type of exercise/drill to train response force personnel.

03.05 Deadly Force Analysis. Assess whether deadly force would be used unnecessarily in a situation that did not threaten the health and safety of an individual or the general public. Determine if deadly force would be appropriately applied if warranted. Interview SFMs who would be likely to encounter an armed intruder and be placed in a situation warranting deadly force. The SFM could be a patrol officer, tower officer, or member of the armed response.

The interviews shall be conducted one at a time in private. Have each SFM explain his or her understanding of the licensee's deadly force policy. Based on Information Notice 89-05, "Use of Deadly Force by Guards Protecting Nuclear Power Reactors Against Radiological Sabotage," and the licensee's policy, pose hypothetical deadly force situations to the SFMs and assess their responses. Identify instances where deadly force would have been applied inappropriately.

Discuss significant concerns with licensee management. Concerns should not be attributed to a specific SFM. To the extent possible, preserve the confidentiality of comments and views expressed by individual SFM.

03.06 Tactical Training. The team should interview instructors to determine the quality, frequency, relevancy, and depth of tactical training. Visit and evaluate any tactical training facility that the licensee may have. Evaluate contingency training relative to initial, refresher, frequency, length, and content. The team should interview tactical trainers and security officers and observe tactical demonstrations to determine if training appears sufficient in scope and frequency to assure a reliable capability.

Assess the response weapons and equipment available to the response force relative to the DBT, site-specific conditions, and engagement conditions explained during the table-top drills and as demonstrated during the contingency exercises. The team should also assess the deployment of the response weapons and equipment to identify any potential adverse impact on the protective strategy.

03.07 Firearms Training. The team should evaluate the appropriateness of the firearms training, the experience of the training staff, the facilities available, and the techniques and frequency of training employed to assure that contingency response personnel are capable of executing their assigned responsibilities.

The licensee is normally requested to demonstrate its tactical or stress courses of fire if it has any. Observe a sufficient number of SFMs demonstrate the courses of fire to assess their ability to manipulate the weapons safely and competently. The team should also assess the proficiency of the SFMs executing the chosen courses of fire, giving due consideration for the difficulty of the course.

Assess the relevancy of the courses of fire to on-site conditions, although not specifically required by the regulations. The most desirable courses are those that simulate and provide training in conditions that an SFM would encounter on site in an actual engagement with an armed intruder. The appropriateness of course content should be examined relative to the simulated weapon engagements that occur during the table-top drills and actual exercises. This comparison could be especially significant if the licensee had no site-specific weapons course and provided SFMs with only the training specified in Appendix B to 10 CFR 73.55.

The team should also determine whether the licensee trains SFMs in friend/foe target identification firing. The inclusion and quality of friend/foe target identification training should be considered in the assessment of SFMs' ability to exercise good judgment in the application of deadly force.

03.08 <u>Safety/Safeguards Impact Review</u>. The team engineer should interview control room operators, equipment operators, and key security personnel to identify any potential weaknesses relative to the licensee's implementation of 10 CFR 73.55(d)(7)(ii) and to assess coordination between operations and security personnel during emergency conditions or situations that would lead to emergency conditions. The team engineer should determine if operations personnel have any concerns relative to a potential adverse impact on plant and personnel safety by security measures or procedures. If a potential problem is identified, the team engineer should directly observe the problem and discuss it with an operations supervisor and the resident inspector.

The team engineer and licensee operations personnel should make a walking tour of existing security measures to assess licensee procedures relative to: rapid access to vital equipment by plant personnel; rapid entry into the PA by off-site medical and fire-fighting personnel; rapid escape from enclosed areas under adverse conditions; and to appropriate restrictions on plant-security radio transmissions to prevent interference with plant operations.

03.09 <u>OSRE Team Meetings</u>. Upon completion of the day's schedule, the team should meet to evaluate licensee performance collectively and formulate conclusions. A negative conclusion can be a finding or a minor observation, which may only need to be passed on to security personnel. A finding and its relative importance should be the consensus of the team. If non-team-member

NRC personnel are present, briefly summarize team procedures, methodology, previous findings, etc., as appropriate.

Determine subjects to discuss with the licensee at the next briefing, which may include, when appropriate, the proposed target sets for the next exercise. Confirm or revise the proposed schedule.

03.10 Licensee Management and OSRE Team Meeting. Typically, the licensee security manager will meet with the OSRE team each morning. Three things should be accomplished at this meeting: (1) a review and discussion of the previous day's findings by the team; (2) a review and confirmation of or modification to the schedule; and (3), as appropriate, a discussion of the exercises to include the target set, PA entry points, and adversary characteristics.

Confirm with the licensee any pending actions relative to a significant negative finding which greatly diminishes high assurance that the licensee has the capability to protect against the DBT. Inform the licensee that such a finding is a potential violation and will result in enforcement action being taken. Any violation identified during an OSRE will be addressed in the OSRE report and enforcement action will be initiated consistent with agency enforcement policy. Less significant weaknesses will be identified in the OSRE report and a response from the licensee is required to address corrective actions planned. Regional inspectors will verify implementation of corrective measures during subsequent routine inspections.

03.11 Preliminary Exit Meeting and Exit Meeting

a. Preliminary Exit Meeting. The team leader will meet with appropriate licensee management to present the preliminary findings. This meeting should allow for licensee input on the accuracy of the preliminary findings. The licensee will then be allowed a minimum of 2 weeks to assess all findings relative to operational initiatives. Inform the licensee that, with regard to a significant negative preliminary finding, an operational mitigation or solution will be considered prior to NRC determination of the significance of the vulnerability.

b. Exit Meeting

Meet with licensee management to present the findings. Confirm with the licensee any pending actions relative to a significant negative finding. Inform the licensee that such findings will either be resolved by NRC Headquarters or turned over to the respective region for resolution. This exit meeting may be conducted at the site, at NRR, at the regional office, or by telephone, depending on the nature of the findings.

81110-04 RESOURCE ESTIMATE

The average time to complete this procedure is 4 to 5 days by a team normally consisting of four NRC staff members and three NRC contractors. The time expended for this evaluation should be reported as direct inspection and is, therefore, fee recoverable.

81110-06 REFERENCES

NUREG-0992, "Report of the Committee to Review Safeguards Requirements at Power Reactors," May 1983.

IE Information Notice No. 83-36, "Impact of Security Practices of Safe Operations," June 9, 1983.

IE Information Notice 85-79, "Inadequate Communication Between Maintenance, Operations, and Security Personnel."

NRC Information Notice No. 89-05, "Use of Deadly Force by Guards Protecting Nuclear Power Reactors Against Radiological Sabotage," January 19, 1989.

Parts 73.1(a); 73.55(a); 73.55(b)(4)(i); 73.55(d)(7)(ii); 73.55(h)(1); and 73.55(h)(2) to Title 10 of the Code of Federal Regulations

Appendix B to 10 CFR 73.55

IMC 2901, "Team Inspections"

END

Attachments:

Supplement A, "OSRE Adversary Characteristics" Note: This document is not publicly releaseable.

December 17, 2001

The Honorable James M. Jeffords
United States Senate
Washington, D.C. 20510

Dear Senator Jeffords:

I am responding on behalf of the Commission to your letter of November 20, 2001, forwarding questions concerning the security of the Nation's commercial nuclear facilities. Although nuclear power plants are among the most hardened and secure civilian facilities in the United States, the recent attacks have focused attention on the need to review policies and practices related to safeguards and physical security measures for civilian nuclear facilities.

Immediately following the terrorist attacks on the World Trade Center and the Pentagon, the NRC advised nuclear power plant licensees to go to the highest level of security, and all promptly did so. With continued uncertainty about the possibility of additional terrorist activities, the Nation's nuclear power plants remain at the highest level of security and the NRC continues to monitor the situation. For the longer term, I, with the full support of the Commission, have directed the NRC staff to thoroughly reevaluate the NRC's safeguards and physical security programs. This reevaluation will be a top-to-bottom analysis involving all aspects of the Agency's safeguards and physical security programs.

Given the nature of the attacks on September 11, the identification of any necessary adjustments to the safeguards and physical security measures for civilian nuclear facilities must involve consultation and coordination with other U.S. national security organizations. The NRC is currently interacting with the Federal Bureau of Investigation, other intelligence and law enforcement agencies, and the Department of Defense to ensure any changes to the NRC's programs are informed by pertinent information from other relevant U.S. agencies.

Because the NRC's reevaluation is ongoing, the enclosed answers to your questions are founded on the information that is available at this time. If you have further comments or questions, please feel free to contact me.

Sincerely,

/RA/

Richard A. Meserve

Enclosure: Responses to Questions

Identical letter to:

The Honorable James M. Jeffords
United States Senate
Washington, D.C. 20510

The Honorable Hillary Rodham Clinton
United States Senate
Washington, D.C. 20510

The Honorable Jon S. Corzine
United States Senate
Washington, D.C. 20510

The Honorable Harry Reid
United States Senate
Washington, D.C. 20510

Enclosure 1

RESPONSE TO QUESTIONS

Question 1: *Immediately after the September 11 terrorist attack, the NRC recommended but did not require nuclear power plants to go to a higher level of security.*

a) *Could you please explain why the Commission did not require higher security.*
b) *How did the NRC confirm whether plants moved to a higher security level?*
c) *In broad terms, could you describe what steps this involves?*
d) *Are security guards working overtime to meet these requirements? If so, what steps are the NRC recommending to reduce possible fatigue effects from long periods of overtime?*
e) *Has the NRC recommended supplementing guard forces with National Guard troops?*

Answer:

1a. The NRC recommended on September 11, 2001, that licensees move to a higher security level. This recommendation was in the form of a Threat Advisory. A Threat Advisory provides a vehicle for communication between the NRC and its licensees when a rapid response is required. As discussed below, the Threat Advisory achieved the desired response and, consequently, the NRC did not find it necessary to issue orders. Had the Commission found it necessary to direct action by particular licensees it could have promptly issued individual orders to them.

1b. The licensees reported to the NRC that they had implemented the higher level of security as urged by the Threat Advisory. This was later verified by the NRC resident inspectors, who are stationed at each commercial nuclear plant site with an operating license. There have also been audits of the heightened security measures at all NRC licensed operating and decommissioning nuclear plants by NRC inspectors from the regional NRC offices.

1c. The heightened security stance generally included increased patrols, augmented security forces and capabilities, additional security posts, heightened coordination with law enforcement and military authorities, and additional limitations on access of personnel and vehicles to the site, among other measures. On October 6, the NRC issued a safeguards advisory delineating certain prompt and longer-term additional actions to strengthen licensee capability to respond to a terrorist attack at or beyond the design basis threat. Licensees are currently complying with this advisory.

1d. The security guards are working overtime to meet these requirements. Licensees are required by 10 CFR Part 26, Fitness for Duty Programs, to have in place a program which provides reasonable assurance that nuclear power plant personnel will perform their tasks in a reliable and trustworthy manner. Fatigue is one of the factors which is addressed under this program. NRC inspectors have closely scrutinized the security measures in place at the Nation's commercial nuclear power plants and have not identified a noticeable decline in the effectiveness of the security forces as a result of fatigue.

1e. On September 26, 2001, the Chairman sent a letter to the Governors of those States which have sensitive commercial nuclear facilities. The purpose of the letter was to explain the actions taken by the NRC and its licensees to augment security after September 11 and to note limitations on licensee capabilities to deal with beyond design basis threats. The letter noted that as the security situation unfolds, State resources might be needed to supplement licensees' capabilities. However, the Commission did not request such supplementation. The Commission believes that the individual Governors, working in consultation with their security advisors and federal law enforcement authorities, can best determine where to deploy National Guard assets to protect critical infrastructure.

<u>Question 2</u>: *Several months ago, the NRC approved the start of a pilot test program to replace the current security program.*

a) Does the NRC believe this is an appropriate time to test new security training programs?

b) Why wouldn't the NRC's resources be better utilized by improving the program already in place, the so-called Operational Safeguards Response Evaluation (OSRE) program, which has a strong NRC oversight component?

<u>Answer</u>:

2a,2b. At this time, the NRC believes it is inappropriate to conduct force-on-force exercises due to the conditions of heightened security. Therefore, force-on-force exercises have been temporarily suspended. As previously noted, a thorough review of the NRC's physical security and safeguards programs was initiated shortly after the September 11 attacks.

Before September 11, the Commission agreed to a one-year pilot of the Safeguards Performance Assessment (SPA) program. The intent of the SPA pilot was to determine if a more performance-based approach, making greater use of licensee resources while permitting more frequent NRC evaluation of force-on-force exercises, could be developed. During the conduct of the SPA pilot, the NRC would continue OSRE inspections at a rate of six per year, which would be combined with eight NRC-evaluated SPA inspections.

It is important to note that the frequency of NRC-evaluated exercises would increase from once every eight years under the OSRE program to triennially under SPA program. The performance of more frequent periodic drills and exercises under the SPA program could enhance our licensees' capabilities to protect against the design basis threat of radiological sabotage. Thus, the Commission approved a one-year trial of the SPA program, subject to close NRC oversight and evaluation. A final Commission decision regarding the method of conducting force-on-force testing would follow formal evaluation of the pilot program and the continuing OSRE program.

Question 3: *Media reports indicate nearly half the nuclear power plants failed their OSRE exercises.*

a) What are the biggest causes for this failure?
b) Is it a lack of training, a lack of equipment, and/or poor tactics?
c) What are steps the NRC is taking to improve the performance of licensees in these tests?
d) Does the NRC assess fines against licensees that fail these tests? If not, why not?

Answer:

3a,3b. A typical OSRE has several components, including table top drills leading to four force-on-force exercises in which the attacking force attempts to exploit any vulnerabilities the NRC security specialists identify in the plant's protective strategy. The attacking force is credited with detailed knowledge of the plant's lay-out, vulnerabilities and security force defense plans. The overall goal of the OSRE is to improve the efficacy of facility security by identification and correction of weaknesses.

In 37 of 81 OSRE's conducted between August 1991 and August 2001, the NRC identified weaknesses.[1] In those plants in which a weakness was found, the attacking force was typically able in one of four exercises to reach a target set and simulate destruction of that equipment. In general these weaknesses occurred due to deficiencies in the licensee's contingency response plan, in training, or in executing the plan. No one issue dominates the weaknesses noted.

It is agency policy for NRC licensees to address identified weaknesses immediately through the implementation of compensatory measures and, where appropriate, permanent corrective actions. The NRC believes that the program has served an important function by contributing to the identification of areas for improvement in the licensees' security programs. The tests are difficult because they are designed to exploit potential vulnerabilities revealed in the table top drills. They do not necessarily reflect the likelihood of success by a less informed attacking force.

3c. Licensees are required to correct deficiencies in their security programs, including deficiencies identified during OSRE force-on-force exercises. In addition to OSRE exercises, NRC inspectors routinely inspect licensee security programs as part of the baseline inspection program. NRC has the statutory tools necessary to ensure that any security deficiencies are corrected in a timely fashion.

---

[1]For the 15 OSREs conducted between April 2000 and August 2001, weaknesses were identified in 9 of 59 exercises or 15 percent of the time. Eighty-five percent of the time the attacking force was defeated.

3d. The NRC has never assessed fines against licensees for weaknesses uncovered in an OSRE. In 1988, in discussing a similar program carried out by the NRC at category 1 fuel cycle facilities (facilities that handle weapons-grade highly enriched uranium) the Commission stated: "The exercises would demonstrate the guard force state of readiness and test the effectiveness of delay mechanisms, alarm and communication systems, response times, deployment of response forces, firing skills (simulated), and tactical maneuvers. The results would be used to determine whether additional training or security system improvements are needed. The exercises are not intended to be viewed in terms of 'pass' or 'fail.'"

Since April 2000 when the Commission adopted its revised reactor oversight process, the NRC staff has applied its significance determination process to OSRE inspection results and has informed the public of the significance of weaknesses without divulging any details that might aid a terrorist. In January 2001, the Commission concluded that subsection (a) of 10 CFR 73.55 lacked the clarity necessary for consistent enforcement, and directed the staff to pursue rulemaking to clarify the provisions of 10 CFR 73.55 (a) and to refrain from enforcement action based on 10 CFR 73.55 (a) as a result of force-on-force exercises until further Commission direction is provided.

Question 4: *In the NRC's long-term budget forecasting, is the NRC budgeting for continued use of the OSRE program?*

Answer:

Yes, the NRC's long-term budget includes funds for at least six OSRE inspections per year, with additional funds which can be used either to support the OSRE program or the SPA program. The results of the SPA pilot (if conducted) will help determine the future direction of NRC's activities in this area.

Question 5: *To ensure that safety plans can adequately protect a nuclear facility, the NRC requires additional force-on-force exercises to verify the ability of security forces to implement the security plans. Does the NRC have a comparable program to ensure that emergency response plans can be successfully implemented in the event of an accident? If so, does this involve coordinated exercises with all local, state and federal emergency responders?*

Answer:

The NRC requires in 10 CFR 50.47, "Emergency plans", 10 CFR 50.54, "Conditions of licenses", and in 10 CFR 50 Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities", that commercial plant licensees have and maintain a comprehensive emergency response plan and that exercises be conducted at each site at least every 2 years to evaluate emergency response plans. This involves a coordinated exercise with State and local authorities having a role under the plan. Licensees routinely conduct more frequent drills to ensure their employees are familiar with their emergency response duties.

Question 6: *Does the NRC require state and local governments to develop evacuation plans to respond to a potential release from a nuclear power plant? If so, how often are these plans updated to reflect demographic changes around the plants? Are only the communities near the plants involved, or are communities that could be exposed to a contamination plume far from the plant considered?*

Answer:

The NRC requires in 10 CFR 50.47, and in Appendix E to 10 CFR 50, that emergency response plans include a range of protective actions for the plume exposure pathway emergency planning zone (EPZ), an area about 10 miles in radius, and the ingestion exposure pathway EPZ, an area about 50 miles in radius. The licensees' plans are also to include a time estimate for evacuation of the plume exposure pathway. Criteria for protective actions are further defined in NUREG-0654/FEMA-REP-1, Revision 1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants". These criteria include expectations that State and local organizations establish a capability for implementing protective measures. The State and local plans to implement protective measures for the plume exposure pathway are to include maps showing evacuation routes, evacuation areas, and relocation centers in host areas beyond the plume exposure EPZ. State plans are to also include protective measures to be used for the ingestion pathway EPZ, including methods for protecting the public from consumption of contaminated foodstuffs.

The size of the emergency planning zones represents a judgment on the extent of detailed planning which must be performed to assure an adequate response base. Detailed planning within the EPZ provides a substantial base for the expansion of response efforts in the event that this proved necessary. In accordance with requirements in Appendix E of 10 CFR 50, licensees are to have provisions that ensure emergency plans are kept up to date. Furthermore, 10 CFR 50.54(t) requires that all elements of the emergency preparedness program be reviewed at least once every 24 months. Accordingly, a consideration of changes in the demographics around the plant would be expected to be included in the licensees reviews conducted in accordance with these requirements.

The NRC does require that adequate emergency plans, including evacuation plans, be in place for each licensed nuclear power plant, and the NRC's determination of adequacy is based in part on findings made by the Federal Emergency Management Agency (FEMA). State and local governments maintain evacuation plans to respond to a potential release. These plans are usually exercised during the required biennial exercises. Typically the local governments which participate in the exercises fall within a ten mile radius of the nuclear facility. State emergency plans generally have provisions for extending protective actions as needed.

Question 7: *We understand that there may be as few as 10-12 guards, on average, at facilities in your jurisdiction. Is this true, and is this adequate in your opinion? Is there a federal requirement, applied consistently at all facilities in your jurisdiction, for a certain number of guards? For the background and training of these guards?*

Answer:

The site contingency response plan delineates the number of armed responders and guard force necessary to protect against the design basis threat, and the OSRE force-on-force exercises provide a test of the adequacy of those forces. Corrective actions identified via the OSRE program must be addressed. Section 73.55(h)(3) of 10 CFR requires a nominal force of ten guards, and no less than five, but each licensee has established site-specific commitments in its security plan based on the size and layout of the facility. The details of the site-specific commitments are Safeguards Information.

The security forces at nuclear facilities are well-trained, well paid, and have high retention rates. This is a sharp contrast to airport security before the recent improvements. The background and training of the guards is specified in 10 CFR Part 73 and Appendix B to 10 CFR Part 73. These provisions address such things as physical and mental requirements, authority of guards, the use of deadly force, tactics, site security systems, communication system operation, and weapons training, including demonstrating proficiency with weapons to be used by that guard.

In response to the terrorist attack of September 11, 2001, the NRC has initiated a thorough review of the safeguards and physical security programs. This effort will include input from the national security organizations, the Office of Homeland Security, the FBI, intelligence and law enforcement agencies, the Department of Defense and others to evaluate the level of threat to which civilian nuclear facilities must be able to respond. Based on this review, if the NRC determines that additional or revised safety or physical protection measures or requirements need to be taken, the NRC will take appropriate actions to implement those measures.

Question 8: *How many facilities in your jurisdiction are now protected by National Guard personnel? Are there any facilities that have refused the services of the Guard? If so, what reason did the licensees provide?*

Answer:

Approximately 12 of the 63 operating nuclear power plant sites are currently protected by National Guard personnel. One of ten decommissioned nuclear power plants (Haddam Neck in Connecticut) is currently protected by National Guard personnel. Although the NRC is not typically involved with communications between States and NRC-licensed facilities, the NRC is not aware of a situation where a licensee has refused a request by a State to place National Guard troops at its facility. A number of States have initiated a dialog with NRC licensees to discuss options for enhancing the physical security at their facilities. The NRC is aware that those discussions have involved consideration of the National Guard.

Question 9: *How are the civilian guard forces at facilities in your jurisdiction armed? Is there a federal requirement applied consistently at all facilities in your jurisdiction for armed personnel?*

Answer:

The weapons used by the guard forces are generally comparable to those weapons used by local law enforcement officers, although in some States there are significant limitations on weapons which private security forces can possess. 10 CFR 73.46 has requirements for the weapons (at a minimum) to be carried by guards and Tactical Response Team members.

The NRC has requested legislation which would allow nuclear facility guard forces to use weapons comparable to those available to the Department of Energy's private security forces, notwithstanding State law restrictions. The Commission has also sought legislation which would authorize the use of deadly force if necessary to protect the facility. Without federal legislation, there are State laws at some sites which limit the types of weapons permitted and the use of deadly force. We strongly urge prompt Congressional action on our legislative proposals.

Question 10: *What is the average number of guards at decommissioned facilities? How does the security at these facilities compare to active facilities? Do you think security at these facilities is adequate?*

Answer:

Specific details on guard forces are Safeguards Information. Currently there is no federal requirement for a certain number of guards at decommissioning reactors, but rather that the facility demonstrate the capacity to provide adequate protection. Decommissioning facilities typically have a much smaller area to protect the operating facilities and a smaller guard force. However, the guard force meets the same background and training requirements as at operating plants.

In response to the terrorist attack of September 11, 2001, the NRC has issued threat advisories identifying additional security measures at decommissioning facilities and has verified that these resources are in place. In addition, the NRC has begun a thorough review of the safeguards and physical security programs that will encompass the decommissioning facilities. This effort will include input from the national security organizations, the FBI, intelligence and law enforcement agencies, the Office of Homeland Security, the Department of Defense and others to evaluate the level of threat to which civilian nuclear facilities must be able to respond. Based on this review, if the NRC determines that additional or revised safety or physical protection measures or requirements need to be taken, the NRC will take appropriate actions to implement those measures.

Question 11: *Does the current design basis threat assume that something the size of a tractor-trailer truck may be utilized to carry explosives to a facility? Does it assume that water-borne threats to reactors located near rivers or oceans may occur?*

Answer:

The current design basis threat for radiological sabotage at nuclear power reactors includes a design basis vehicle bomb, with sensitive non-public details concerning vehicle weight, speed, and explosive charge size. The specific numbers for the design basis vehicle bomb were selected based on an analysis of relevant vehicle bombing attacks, including the type of vehicle used and the estimated size of the explosive charge, that occurred around the world over several decades. The design basis vehicle bomb represents a reasonable characterization of a vehicle bomb threat. The specific numbers are protected from public disclosure as Safeguards Information and can be provided under separate cover. The current design basis threat does not include a waterborne component. Our ongoing top-to-bottom reevaluation of the agency's safeguards and physical security programs include analyses of the design basis vehicle bomb and waterborne threats.

Question 12: *Based upon what you know now, do you think the design basis threat should be updated? Do you have the authority to perform the update now? If so, what is your time frame?*

Answer:

Yes, the design basis threat should be reviewed and updated and we have put in place a process for doing so. The staff has initiated a number of actions in response to the September 11 attacks. These include a reassessment of the threat environment to examine the design basis threat. Further, this review will be conducted in coordination with the Office of Homeland Security, Department of Energy, Federal Bureau of Investigation, and other Federal agencies, and could include identification of any necessary changes to the current design basis threats. An additional dimension to this current effort may be the identification of threats beyond the design basis threat, i.e., threats that our licensees may not be fully able to protect against without assistance from local, State or Federal authorities. The Commission has the authority and can direct staff to initiate rulemaking to formally revise 10 CFR 73.1 and modify the design basis threat as appropriate. It is anticipated that this effort will extend well into the year 2002.

Question 13: *New information has recently come to light regarding the vulnerability of nuclear power plants to attack by air. What are the measures the NRC is considering to protect against such threats?*

Answer:

The NRC has been in regular communication with other federal agencies, most specifically the Federal Aviation Administration and the Department of Defense, which have acted more than once to protect airspace above nuclear power plants. The Aviation and Transportation Security Act of 2001 will also provide additional protection against air attacks on all industrial facilities, both nuclear and non-nuclear. The NRC believes that the nation's efforts associated with protecting against terrorist attacks by air should be directed toward enhancing security at airports and within airplanes, and not toward seeking to defend all potential targets of such terrorism.

Question 14: *Since September 11, have there been any credible threats received against any nuclear power plants in the United States and if so, what measures were taken to protect against those threats?*

Answer:

There have been no credible threats against any nuclear power plants in the United States. One threat to a nuclear power plant, based on classified intelligence, was initially assessed as credible, requiring a timely response. NRC coordinated the response to this threat with the licensee, Federal Bureau of Investigation, the Department of Defense, and other Federal agencies. The response included licensee, state, and Federal measures being taken for an appropriate period of time. The threat subsequently was determined to be not credible.

Question 15: *What are the actions the NRC is taking to ensure that proper background checks have been conducted of all staff at all nuclear power plants across the country?*

Answer:

In order to be authorized for unescorted access at a nuclear power plant, an individual must undergo a background screening and investigation pursuant to 10 CFR 73.56, and such workers are subject to ongoing fitness-for-duty requirements. The screening criteria include: (1) a background investigation designed to identify past actions which are indicative of an individual's future reliability within a protected or vital area of a nuclear power reactor; (2) a psychological assessment designed to evaluate the possible impact of any noted psychological characteristics which may have a bearing on trustworthiness and reliability; and (3) behavioral observations, conducted by supervisors and management personnel, designed to detect individual behavioral changes which, if left unattended, could lead to acts detrimental to the public health and safety. In accordance with 10 CFR 73.56, the background investigation includes employment history, education history, criminal history, military service, and credit history, as well as a psychological evaluation, interview of developed references, and fitness-for-duty testing. Inspections are routinely conducted by the NRC to verify that licensees are complying with these requirements. The inspection results have assured the NRC that these requirements are being uniformly complied with, and proper background checks have been conducted. In addition, since September 11, 2001, the FBI has provided to the NRC frequently updated lists of individuals who may have ties or information related to terrorist activities. At the request of the FBI, the NRC provided these lists to the nuclear power plants, the nonpower reactor facilities, decommissioning plants, and selected fuel facilities to be checked against utility employment and visitor records. The Nuclear Energy Institute has also been provided the lists to be checked against a database of temporary nuclear utility workers. All results are being provided by the NRC to the FBI for resolution. To date, all potential matches have been resolved through the FBI.

December 31, 2001

Mr. Robert J. Barrett
Vice President, Operations
Entergy Nuclear Operations, Inc.
Indian Point Nuclear Generating Unit 3
295 Broadway, Suite 3
Post Office Box 308
Buchanan, NY 10511-0308

SUBJECT: INDIAN POINT 3 NUCLEAR POWER PLANT - NRC INSPECTION REPORT 50-286/01-09

Dear Mr. Barrett:

On November 17, 2001, the NRC completed an inspection at the Indian Point 3 nuclear power plant. The enclosed report presents the results of that inspection. The results were discussed on December 5, 2001, with you and other members of your staff.

The inspection was an examination of activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations, and with the conditions of your license. Within these areas, the inspection consisted of a selected examination of procedures and representative records, observations of activities, and interviews with personnel.

Based on the results of this inspection, the inspectors identified two issues of very low safety significance (Green), one of which involved a violation of NRC requirements. These issues involved a failure to conduct triennial hydrostatic tests on self-contained-breathing-apparatus (SCBA) air cylinders, and a failure to monitor for potential degradation of underground cable splices. Because of its very low safety significance and because it has been entered into your corrective action program, the NRC is treating the SCBA issue as a Non-cited Violation, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny this Non-cited Violation, you should provide a response with the basis for your denial within 30 days of the date of this inspection report to, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 2055-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 2055-0001; and the NRC Resident Inspector at the Indian Point Unit 3 Nuclear Power Plant.

Since September 11, 2001, Indian Point Nuclear Generating Unit 3 has assumed a heightened level of security based on a series of threat advisories issued by the NRC. Although the NRC is not aware of any specific threat against nuclear facilities, the heightened level of security was recommended for all nuclear power plants and is being maintained due to the uncertainty about the possibility of additional terrorist attacks. The steps recommended by the NRC include increased patrols, augmented security forces and capabilities, additional security posts, heightened coordination with local law enforcement and military authorities, and limited access of personnel and vehicles to the site.

The NRC continues to interact with the Intelligence Community and to communicate information to Entergy Nuclear Operations, Inc.. In addition, the NRC has monitored maintenance and other activities which could relate to the site's security posture.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of the NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm.html (the Public Electronic Reading Room).

Sincerely,

/RA by Brian E. Holian
Acting For/

G. Scott Barber, Acting Chief
Projects Branch 2
Division of Reactor Projects

Docket No.50-286
License No. DPR-64

Enclosure: Inspection Report No. 50-286/01-09

Attachment: Supplemental Information

cc w/encl: J. Yelverton, Chief Executive Officer
M. Kansler, Senior Vice President and CEO
J. DeRoy, General Manager - Operations
D. Pace, Vice President - Engineering
J. Knubel, Vice President Operations Support
F. Dacimo, Vice President - Operations, IP2
J. Kelly, Director - Licensing
C. D. Faison, Manager - Licensing
H. P. Salmon, Jr., Director of Oversight
J. Comiotes, Director, Nuclear Safety Assurance
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R. Albanese, Executive Chair, Four County Nuclear Safety Committee
S. Lousteau, Treasury Department, Entergy Services, Inc.
Chairman, Standing Committee on Energy, NYS Assembly
Chairman, Standing Committee on Environmental Conservation, NYS Assembly
Chairman, Committee on Corporations, Authorities, and Commissions
The Honorable Sandra Galef, NYS Assembly
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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No. 50-286

License No. DPR-64

Report No. 50-286/01-09

Licensee: Entergy Nuclear Northeast

Facility: Indian Point 3 Nuclear Power Plant

Location: 295 Broadway, Suite 3
Buchanan, NY 10511-0308

Dates: September 30 - November 17, 2001

Inspectors: P. Drysdale, Senior Resident Inspector
L. James, Resident Inspector
J. McFadden, Health Physicist
D. Silk, Senior Emergency Preparedness Inspector

Approved by: G. Scott Barber, Acting Chief
Projects Branch 2
Division of Reactor Projects

000642

SUMMARY OF FINDINGS

IR 05000286-01-09, on 09/30-11/17/2001, Entergy Nuclear Northeast, Indian Point 3 Nuclear Power Plant. Resident inspection report, radiation safety, emergency preparedness

The inspection was conducted by resident and regional inspectors. The inspection identified one Green Finding which was determined to be a Non-Cited Violation.

A. <u>Inspector Identified Findings</u>

Cornerstone: Mitigating Systems

- GREEN. The licensee did not monitor for water intrusion or degradation of underground splices in electrical cables associated with mitigating systems. This finding is greater than minor because, if left uncorrected, degraded splices could increase the risk of loss of electric or control power to a mitigating system, and could result in a plant transient.

  This issue was determined to be of very low safety significance using the NRC's safety determination process (SDP) because no degradation was observed, and no equipment failures or transients had resulted from cable splice degradation (Section 1R06).

Cornerstone: Emergency Preparedness

- GREEN. A Non-cited Violation of 10 CFR 20.1703(c)(4)(vii) for failure to conduct triennial hydrostatic tests on approximately 80 self-contained-breathing-apparatus (SCBA) air cylinders.

  This finding is greater than minor because, if left uncorrected, inadequately tested respiratory protection equipment could have been used by personnel in the event of an emergency. This finding is of very low safety significance because unqualified equipment was not actually used, all of the affected air cylinders displayed the proper air pressure indicating that the cylinders maintained the requisite integrity, and a sufficient supply in excess of requirements was available for use. (Section 2OS3)

B. <u>Licensee Identified Violations</u>

There were no licensee identified violations.

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000644

Report Details

SUMMARY OF PLANT STATUS

The Indian Point 3 Nuclear Power Plant remained at 100% power for the entire inspection period from September 30 through November 17, 2001. No significant equipment failures or events occurred during this time period.

1. REACTOR SAFETY
(Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, Emergency Preparedness)

1R04 Equipment Alignment

a. Inspection Scope (71111.04Q and 71111.04S)

On October 2, 2001, the inspectors completed a partial walkdown of the 32 and 33 EDG mechanical and electrical systems to verify the availability of that equipment when the 31 EDG was out of service for preventive maintenance. The inspectors used check-off list COL-EL-5, "Diesel Generators," and flow diagrams 9321-F-20293, -20303, -27223, and -30083 to confirm that the engine fuel oil systems, air start systems, cooling systems, and electrical systems were configured to permit automatic start and operation of the EDGs.

On November 8, 2001, the inspectors walked down accessible portions of the 31 and 33 safety injection (SI) pumps and flow trains. During this inspection, the 32 SI pump was out of service to perform corrective maintenance (WR 01-03420-01) to repair a pump casing plug leak. The inspectors reviewed the equipment configuration designated on protective tagout 01-1718 to assure the 32 pump was properly isolated from the SI system, and that its isolation did not affect the availability of the 31 and 33 pumps. The inspectors also used check-off list COL-SI-1, "Safety Injection System," and flow diagram 9327-F-27503 to verify to correct alignment of valves and control room switches associated with the pumps, and the status of control room alarms during the maintenance on the 32 pump.

On November 13, 2001, the inspectors completed a full walkdown of accessible portions of the component cooling water (CCW) system to verify the correct equipment alignment for system operability at full power operations. The inspectors reviewed the following documents:

- Check-off list COL-CC-1, "Component Cooling System"
- System operating procedure SOP-CC-001B, "Component Cooling System Operation"
- Off-Normal operating procedure ONOP-CC-1, "Loss of Component Cooling"
- Design Basis Document IP3-DBD-308, "Component Cooling Water System"
- Technical Specifications and Bases Sections 3.7.8, "Component Cooling Water (CCW) System"
- Final Safety Analysis Report Section 9.3, "Auxiliary Coolant Systems"

The inspectors also reviewed outstanding maintenance activities, open work requests, outstanding corrective action program deficiencies, temporary modifications, and operator work-arounds associated with the CCW system and interviewed the system engineer and non-licensed operators regarding the operations of the CCW system.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope (71111.05Q)

The inspectors conducted fire protection tours in the fire zones listed below 1) to observe if the licensee had been controlling transient combustibles in accordance with fire protection procedure FP-9 "Control of Combustibles," 2) to verify that the licensee had been controlling ignition sources in accordance with FP-8, "Controlling of Ignition Sources," 3) to verify that the licensee had provided the fire protection equipment as specified in Pre-Fire Plans listed below; and 4) to assess the general material condition of the fire protection equipment and fire protection barriers.

- On October 22, 2001, the inspectors toured the 35 foot elevation of the turbine building under the main output generator. This area was affected by a small leak from the generator hydrogen coolers and required temporary ventilation to keep a high concentration from accumulating (Pre-Fire Plan 43, "General Floor Plan - Turbine Building 36ft-9in").
- On October 26, 2001, the inspectors toured the 480 Volt switchgear room on the 15 foot elevation of the control building to assess the condition of fire protection equipment and convective air pathways in the room, and to identify the potential existence of transient combustible materials (Pre-Fire Plan 25, "480V Switchgear Room - Control Building").
- On October 29, 2001, the inspectors toured the Fuel Storage Building to evaluate the existence of potential fire hazards (Pre-Fire Plan 20, "Fuel Storage Bay Area - Fuel Storage Building").

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

a. Inspection Scope (71111.06)

The inspectors reviewed FSAR Section 16.1 that described flood protection measures designed into the plant and flood mitigation equipment available to plant operators in the event of an internal flooding event caused from the potential break of a large pipe in the circulating water, condensate water, fire water, or city water systems.

The inspectors reviewed the licensee's proceduralized actions designated for response to an internal flooding event as described in operations directive OD-8, "Guidelines for Severe Weather," off-normal procedure ONOP-RW-3, "Plant Flooding," alarm response procedure ARP-7, "Panel SDF - Turbine Recorder," and maintenance procedure MET-002-GEN, "Location of Sandbags in Flood Warning Conditions."

The inspectors toured areas important to safety inside the plant to assess the condition of equipment intended to mitigate the consequences of internal flooding such as elevated berms, floor drains, and breakaway flood gates. Areas toured included the turbine building, the primary auxiliary building (PAB), and the auxiliary feedwater (AFW) pump room,

The inspectors reviewed the work request (WR 00-01619-00) and test procedure (3PT-R22) for the most recent functional test of the flood detection instruments (float switches LC-1240S and LC-1241S) on the 5 ft elevation of the turbine building (condenser trench). The float switches were installed to detect an 18 inch water level in the condenser trench, and to signal an alarm in the control room following a large condensate or circulating water system failure that could threaten the 6.9 KV switchgear on the 15 foot elevation of the turbine building. The test was satisfactorily performed on April 11, 2001, and is repeated every refueling outage. The inspectors also inspected the general areas around the condenser water boxes to verify that flood water flow paths were not obstructed and would direct water to the 5 foot elevation away from the 6.9 KV switchgear.

The inspectors noted that the licensee had identified several recent problems with floor drain blockage on the 15 foot elevation of the control building. The inspectors reviewed the DERs written over the past 12 months that were key worded for "flooding" and "drains," as follows:

DER 01-04168, Rust Discharged from Drains in 15ft Control Building"
DER 01-04067, Control Building Floor Drains - Debris Blown Out"
DER 01-03320, "Floor Drain Plugged in 480 Volt Switchgear Room" on the 15 foot elevation of the control building
DER 01-02163, "TCV-1103 Leaky Flange Rework"

The inspectors toured external areas of the site to observe areas of water accumulation, storm drainage paths, and areas where potential blockage could occur. The inspectors also observed the locations of storm drains and their proximity to underground cable vaults. Electrical engineering personnel provided the inspectors with a general schematic diagram of all buried cable routes on site, and also provided the manufacturer's design specifications and qualifications for buried power cables, control cables, and cable splices for service in a ground water environment.

b. Findings

During tours of external plant areas, the inspectors requested that the licensee remove four manhole covers to assess the general condition of underground cables and splices, and to inspect for possible accumulation of water in the manhole pits. The inspectors observed that a large amount of water had accumulated inside two manhole pits near the service water intake structure. These pits contained numerous cable bundles that supplied power to service water pumps and strainers, circulating water pumps, traveling screens, and screenwash pumps. All of the cables in these pits were completely submerged in water. It was not possible to identify each of the cables visually, but some of them had visible splices that were completely submerged. The inspectors also observed that a manhole pit near the main output transformers contained a large number of cable bundles that were completely submerged. Many of these cables had splices, but they also could not be identified visually. The licensee had stated that buried safety-related cables on site did not have any splices; however, there were no records on site which verified that the existing splices in systems important for mitigating a plant transient (e.g., circulating water pumps) were actually installed in accordance with the manufacturer's specifications.

The licensee subsequently pumped all the water out of the opened manhole pits and performed a visual inspection of the cables and splices for degradation. No degradation was observed; however, the licensee initiated DER 01-04270 to investigate the source of the water, and to specifically identify which cables contained splices. The licensee stated that no recent equipment performance problems were evident that could be attributed to underground cable or splice degradation.

The inspectors noted that the licensee did not have a planned periodic activity to inspect manhole pits for water accumulation, and did not have a preventive maintenance activity to inspect underground cables and splices for degradation. At the end of the inspection period, the licensee was developing a plan to initiate those activities and to perform an extent-of-condition review on buried cables and splices for other equipment important to safety such as the Appendix R diesel generator.

The lack of inspection records which could confirm that existing splices were installed in accordance with the manufacturer's specifications for service in a ground water environment; and the lack of a preventive maintenance activity to inspect for potential degradation in buried cable splices represents a Green finding. The issue is more than minor because, if left uncorrected, degraded underground cable splices could increase the risk of ground water intrusion, and the risk of subsequent electric or control power failures in equipment important to safety. Such conditions could degrade the ability of mitigating systems to perform their functions, or result in an electrical fault that could cause a plant transient.

1R12 Maintenance Rule Implementation

a. Inspection Scope (71111.12)

The inspectors reviewed the following systems and performance issues to assess the effectiveness of the maintenance program. Using 10CFR50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," and Regulatory Guide 1.1.60, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," the inspectors verified that the licensee was implementing their maintenance program in accordance with NRC regulations and guidelines; properly scoping of the system within the maintenance rule; proper classification of failures of structures, systems, and components (SSCs); proper classification of SSCs into 10 CFR 50.65 (a)(1) and (a)(2) status; and appropriate performance criteria for (a)(2) systems or the improvement plan for (a)(1) systems.

- 34 Reactor Coolant Pump Seal Outlet Temperature Indicator (TI-125) Fluctuating (Problem Identification (PID) 03393)

- Wide Range Temperature Element for Cold Leg Loop 34 TE-443B had been drifting (PIDs 03391, 03924, and 03930)

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessment and Emergent Work

a. Inspection Scope (71111.13)

The inspectors reviewed the maintenance risk assessments and corrective maintenance work packages for the following emergent work, and discussed the deficient conditions with cognizant personnel (system engineers, maintenance technicians, and work planners). The inspectors evaluated the licensee's revisions to the daily plant risk profile (i.e., changes to the conditional core damage probability) and changes to the scheduled sequence of preplanned activities resulting from the emergent work:

- Unplanned loaded run of the 32 EDG: On November 13, 2001, the licensee ran the 32 EDG to troubleshoot an air leak in the engine inlet valve lubricating system. This emergent work would have coincided with a scheduled surveillance of the containment fan cooler units; however, the licensee delayed the fan cooler work so that the two activities would not be performed concurrently.

- Extent-of-condition review for a loose nut on the 31 EDG oil pump: On October 3, 2001, the 31 EDG was out of service for planned maintenance (diagnostic testing) when a loose lock nut was identified on the engine-driven oil pump shaft that could have resulted in an engine failure (DER 01-03780). Given the potential significance of this problem, the licensee considered sequentially removing each of the other diesels from service to perform an immediate inspection of the corresponding locking nuts. However, the licensee performed a risk analysis for having two EDGs inoperable with the plant at 100% power. The analysis indicated the conditional core damage probability (CDP) would exceed the nominal CDP and the licensee's risk threshold (E-6). The licensee concluded that ongoing maintenance on the 31 EDG must be completed prior to removing the other diesels from service for an inspection. Following return of the 31 EDG to service, the licensee inspected the oil pumps on the 32 and 33 EDGs, and the Appendix R diesel for a potential common cause failure associated with the oil pump shaft nuts. The investigations revealed that the corresponding nuts on the other three engines were properly torqued.

b. Findings.

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope (71111.15)

The inspectors reviewed various DERs on degraded or non-conforming conditions that raised questions on equipment operability. The inspectors reviewed the resulting operability determinations (ODs) for technical adequacy, whether or not continued operability was warranted, and to what extent other system degradations adversely impacted the affected system or compensatory actions. The following DERs, calculations, and ODs were evaluated:

- OD 01-039 Operability of the 33 EDG with Leaking Fuel Pump Gaskets: Approximately 1 liter/minute leakage of fuel oil through the gaskets on the pump's "banjo bolts" occurred on initial engine start. The leakage stopped when the engine warmed up, and all of the leakage was retained within a retention tank. The leakage did not affect operation of the engine before and after it was loaded.

- OD 01-040 Pinhole Leak between Containment Wall and Service Water Valve SWN-44-3: An auxiliary operator discovered a one drop per minute pinhole leak on the service water outlet line from the 33 containment fan cooler unit. The licensee performed non-destructive testing in the area of the leak and performed a pipe stress analysis to determine operability of the pipe. The results concluded that flaw was minor and did not affect operability. The licensee also used criteria contained in ASME Code Case N-513 to determine that the leak was minor and could be repaired during a future outage.

- OD 01-041 EDG 32 Starting Air System Leakage During Engine Operation: The air inlet valve lubricator was isolated to perform troubleshooting. The lubrication system was not vital to engine operation and did not affect EDG operability.

- OD 01-042 EDG 31, 32, & 33 Air Consumption: The total air consumption of the inlet valve lubrication system could deplete the air start accumulator during engine operation within 13.1 hours after a loss of offsite power and subsequent loss of the air compressor. Without air makeup, the EDG ventilation system outlet dampers would eventually fail closed, and the EDG engine could overheat. The licensee concluded that 13.1 hours was sufficient time to block open the dampers to prevent overheating in the EDG room, and initiated a temporary procedure change (TPC 01-0545) to alarm response procedure ARP-019, "Panel Local - Diesel Generators," requiring operators to block open exhaust louvers if air to the louvers or the room temperature control panel is lost.

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing

a. Inspection Scope (71111.19)

The inspectors reviewed post-maintenance test (PMT) procedures and associated testing activities to assess whether 1) the effect of testing in the plant had been adequately addressed by control room personnel, 2) testing was adequate for maintenance performed, 3) acceptance criteria were clear and adequately demonstrated operational readiness consistent with design and licensing documents, 4) test instrumentation had current calibrations, range, and accuracy for the application, and 5) test equipment was removed following testing. The following surveillance activities were evaluated:

- WR 00-04796-00: Preventive Maintenance performed on 32 Safety Injection Pump Motor

  On October 1, 2001, the inspectors reviewed the PMT documentation for the preventive maintenance (PM) performed on 32 safety injection (SI) pump motor. The inspectors verified that the PMT demonstrated functional capability of the 32 SI pump as delineated in the design basis document IP3-DBD-306, "Safety Injection System," and in Technical Specification 3.5.2, Emergency Core Cooling Systems - Operating.

- WR 01-00366-01: Post Maintenance Test for Boric Acid Transfer Pump Preventive Maintenance

  On October 11, 2001, the inspectors reviewed the PMT documentation for the PM inspection performed on the 32 boric acid transfer pump (BATP) under WR 01-00366-00. The inspectors verified that the PMT demonstrated functional capability of 32 BATP as delineated in the Final Safety Analysis Report section 9.2, Chemical and Control Volume System, and in Technical Requirements Manual 3.1.C, Boration Systems.

- WR 01-00455-01: Post-Maintenance Test following the 2-Year Preventive Maintenance on the 32 EDG

  The test was performed on October 24, 2001, and the results demonstrated satisfactory completion of the maintenance performed. No significant issues resulted from the PMT; however, the compressor for the engine air start system cycled approximately every 10 minutes between 295 - 285 psig during the test (DER 01-04036). The licensee subsequently conducted troubleshooting to evaluate this problem. The leakage occurred through the engine inlet valve lubrication system, which was isolated and had no impact on engine operation.

- WR 01-03420-01: Post-Maintenance Test following Corrective Maintenance on the 32 Safety Injection Pump

  On November 08, 2001, a plug on the 32 safety injection pump casing developed a leak of 20 ml/min (approximately 0.32 gallons/hour). This represented approximately two-thirds of the total emergency core cooling system (ECCS) leakage outside containment at that time. The licensee initiated DER 01-04202 to document the deficiency, performed corrective maintenance to remove the casing plug and to replace its seal, and then ran the pump to test for leakage. Although the seal continued to leak a very small amount, the leakage was reduced to approximately 0.1 ml/min when the pump was running. This leakage did not affect operability of the pump and significantly reduced ECCS leakage outside containment. The licensee intends to investigate replacement of the seal with an alternate design, or to install a vent valve to replace the plug and seal completely.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope (71111.22)

The inspectors observed portions of the following surveillance tests and reviewed the surveillance test procedures to assess whether 1) the test preconditioned the component(s), 2) the effect of testing was adequately addressed in the control room, 3) the acceptance criteria demonstrated operational readiness consistent with design calculations and licensing documents, 4) the test equipment range and accuracy was adequate with proper calibration, 5) the test was performed in the proper sequence, and 6) the test equipment was removed following testing.

- 3PT-Q22 "Residual Heat Removal System Valves" (October 22, 2001)
- 3PT-M62 "480V Undervoltage/Degraded Grid Protection System Functional" (October 17, 2001)
- 3PT-Q120C "33 ABFP (Motor Driven) Surveillance and IST" (October 25, 2001)

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope (71111.23A)

The inspectors reviewed the work package for one temporary modification (TM)

- TM 01-04455-02: Isolation of 32 EDG Inlet Air Valve lubrication system. Based upon the design and purpose of the inlet valve lubrication system, and the ability of the engine to function without it, isolation of the system did not affect operability. The temporary modification permitted the engine air start system to maintain a constant pressure without losses when the engine was operating.

b. Findings

There were no findings identified during this inspection.

Emergency Preparedness (EP)

1EP4 Emergency Action Level and Emergency Plan Changes

a. Inspection Scope

The inspectors conducted an in-office review of licensee submitted changes for the emergency plan-related documents listed below to determine if the changes decreased the effectiveness of the plan. A thorough review was conducted of documents related to the risk significant planning standards (RSPS), such as classifications, notifications and protective action recommendations. A cursory review was conducted for non-RSPS documents. The submitted and reviewed documents were as follows:

Emergency Plan

Section 1, Definitions/Acronyms, Rev 31
Section 2, Scope and Applicability, Rev 31

Section 3,	Overview of the IP-3 Emergency Plan Procedures, Rev 30
Section 4,	Emergency Conditions, Rev 37
Section 5,	Organizational Control of Emergencies, Rev 34
Section 6,	Emergency Measures, Rev 33
Section 7,	Emergency Facilities and Equipment, Rev 33
Section 8,	Maintaining Emergency Preparedness, Rev 35
Section 9,	Recovery, Rev 3
Appendix A,	Letters of Agreement, Rev 0
Appendix B,	Time-Dose-Distance Plots, Rev 29
Appendix C,	Evaluation of Core Degradation Using Containment Accident Monitors, Rev 29
Appendix D,	Listing of Immediate Action Guidelines and Implementing Procedures, Rev 31
Appendix E,	NUREG-0654, Rev. 1; IP-3 Emergency Plan Cross Reference, Rev 30
Appendix F,	Stored Emergency Plan Equipment and Supplies, Rev 30
Appendix G,	Evacuation Plan for Westchester, Rockland, Orange, and Putnam Counties, Rev 29

Implementing Procedures

IP-1038,	Offsite Emergency Notification, Rev 26
IP-1059,	Air Raid Alert, Rev 7
IP-1076,	Roster Notification Methods, Rev 25

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation

a. Inspection Scope (71114.06)

On October 31, 2001, the licensee conducted an "off year" annual emergency preparedness exercise. The inspectors observed the drill from the onsite simulator facility to evaluate the adequacy of drill performance, and attended the post-drill critique to assess the licensee's identification of weaknesses and deficiencies. The drill involved simulation of an intense weather storm at the Indian Point Station that caused significant damage to the onsite electrical distribution systems and challenged operators' abilities to restore vital electrical power. The scenario also included a simulated fire at the Appendix R diesel generator, and a simulated search and rescue for an individual that was unaccounted for.

The evolution of the drill scenario required simulated classifications for an Alert and a Site Area Emergency, required progressive activation of the Technical Support Center (TSC) the Operations Support Center (OSC), the Alternate Emergency Operations Facility (AEOF), and the Joint News Center (JNC).

The inspector observed the progression of events during the simulated scenario and witnessed operators make the appropriate emergency classifications and notifications.

However, the inspectors also noted that the scenario did not generally progress as planned in that the operators declared a general emergency when it appeared that vital electrical power could not be restored. As planned, the simulated emergency was not expected to go beyond a site area emergency, since restoration of a vital electrical bus was expected in less than one half hour. However, the exercise controllers did not provide sufficient information to the operators to indicate that restoration of a vital electrical bus would be possible.

The inspector attended the post-exercise critique and noted that the licensee had identified inadequate controls on the part of the exercise controllers. The critique was comprehensive and provided a detailed review of other deficiencies, also identified by the licensee, as documented in the following DERs:

DER 01-04111, Failure to Control Exercise
DER 01-04112, Scenario Did Not Provide Enough Details to the ERO
DER 01-04113, I&C and Ops Personnel did not Adhere to the Accountability Process
DER 01-04114, JNC Objectives not Met.
DER 01-04120, Weaknesses in the Simulator Crew.

With the exception of issues related to staffing, and untimely notifications from the Joint News Center, all of the objectives for the scenario were achieved. However, the licensee developed a series of actions to address the deficiencies noted prior to the next scheduled exercise.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety (OS)

2OS1 Access Control To Radiologically Significant Areas

a. Inspection Scope (71121.01)

The inspectors reviewed the effectiveness of access controls to radiologically significant areas.

The inspectors toured the radiologically-controlled-areas (RCAs) including: various elevations of the primary auxiliary and RAMS buildings and outside areas within the protected area and the health physics (HP) access control point. During these walkdowns, the inspectors observed and verified the appropriateness of the radiological safety controls in place for active radiological work permits (RWPs). Also, the inspectors reviewed the locking, posting, barricading, and labeling, as appropriate; of radiation and high radiation areas, contamination areas, and radioactive material areas. The status of locked High Radiation Areas was also reviewed. The inspectors also observed activities at the main RCA access control point to verify compliance with requirements for RCA entry and exit, wearing of record dosimetry, and issuance and use of alarming electronic radiation dosimeters. The inspectors evaluated the effectiveness of a pre-job radiation safety briefing for a containment entry at power on October 24, 2001

The inspection included a review of the following RWPs, procedures, records, and documents to evaluate the adequacy of controls for access to radiologically-controlled areas including the expected response to electronic radiation dosimeter alarms and controls for radioactive contamination outside the main RCA:

RWP 01-009,	Assessments in the RCA, Rev. 2
RWP 01-028,	Containment Entry-Reactor Critical, Rev. 1
RWP 01-011,	Health Physics Calibration-Routine, Rev. 1
RE-ADM-1-5,	RES Assessment, Rev. 11
RE-REA-4-1,	Radiation Work Permit (RWP), Rev. 16
RE-ADM-4-2,	Radiation Work Permit (RWP) Computer System, Rev. 5
RE-ACC-5-2,	Instructions to Control Point Personnel, Rev. 13
RE-INS-7UG-4,	Use of Merlin-Gerin DMC-100 Dosimeters, Rev. 6
RE-REA-4-6,	Containment Entry at Power or Initially After Shutdown, Rev. 13
RE-SUR-6-1,	Radiation Surveys, Postings, and Assessment, Rev. 12
RE-SUR-6-2,	Contamination Surveys, Postings, and Assessment, Rev. 10
RE-SUR-6-6,	Health Physics Periodic Task Scheduling, Rev. 13
RE-ADM-1-21,	Mixed Waste Management Program, Rev. 1
RE-EP-13-11,	Hazardous Waste Inspections, Rev. 6
RE-ADM-1-22,	Site Soil Characterization, Rev. 0
ICP-DD-01,	Work Package Planning, Rev. 11
RE-CON-3-4,	Release of Material from the Rad. Controlled Area, Rev. 10
RES-SD-05,	Rad. Control of Volumetric Materials Leaving the Site, Rev. 1
RE-CCI-037,	Analysis of volumetric material for free release, Rev. 3

IP3 Radiological and Environmental Services Department, Annual Self-Assessment Report, July 2000 to July 2001, dated July 13, 2001

IP3 Radiological and Environmental Services Department, Focused Self-Assessment Report, Contamination and Radioactive Material Control, dated October 15, 2001

Indian Point 3 nuclide mix evaluation report - 1998, TID-99-002, Rev. 0
Mixed waste weekly inspection log sheet records for October 3, 10, 17, and 24, 2001.

Letter titled "Implementation of 10 CFR 50.75(g)" (Reporting and recordkeeping for decommissioning planning), dated December 22, 1997 (IP-RES-97-256).

Decommissioning Planning: Soil Characterization and Remediation, TID 95-002, Rev. 0

10 CFR 50.75(g) Status Report for IP3, dated November 8, 1999.

The review of the above-cited documents and activities was against criteria contained in: Title 10 of the Code of Federal Regulations (CFR) Parts 20.1201 (Occupational dose limits for adults), 20.1204 (Determination of internal exposure), 20.1208 (Dose equivalent to an embryo/fetus), Subpart F (Surveys and monitoring), 20.1601 (Control of access to high radiation areas), Subpart H (Respiratory protection and controls to restrict internal exposures in restricted areas), 20.1902 (Posting requirements), site Technical Specification 6.12 (High Radiation Area), and site procedures (identified above in this section).

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Control

a. Inspection Scope (71121.02)

The inspectors reviewed the effectiveness of ALARA (As Low As is Reasonably Achievable) planning and control.

The inspectors reviewed the following procedures, records, and documents for regulatory compliance and for adequacy of control of radiation exposure:

Radiation Protection Plan (AP-7), Rev. 24
Operation of Portable Ventilation System (RE-REA-4-10) Vent Carts, Rev. 6
ALARA Report for Refueling Outage No. 11
ALARA Committee Meeting Minutes for August 28, 2001
Second Quarter Review of Station ALARA Program
Third Quarter Review of Station ALARA Program
IP3 Outage Dose Reduction Project Action Plan IRES-APL-01-003, October 2001

The inspectors also reviewed the dose estimates versus actual exposures incurred and the post-job ALARA reviews for the following RWPs used during refueling outage no. 11 for regulatory compliance, and for the adequacy of the planning to minimize radiation exposure:

RWP 01-221, I&C Support for Reactor Head Work
RWP 01-233, Work on Reactor Coolant Pump (RCP) Motors
RWP 01-235, Removal and Replacement of No. 34 RCP Rotating Element
RWP 01-244, Secondary Side Steam Generator Sludge Lancing and Bundle Flush
RWP 01-260, Non-regenerative Heat Exchanger Gasket Replacement

The inspectors' review was against criteria contained in 10 CFR 20.1101 (Radiation protection programs), 10 CFR 20.1701 (Use of process or other engineering controls) and site procedures identified above.

b. Findings

No findings of significance were identified.

2OS3 Radiation Monitoring Instrumentation

a. Inspection Scope (71121.03)

The inspectors reviewed the program for health physics instrumentation and for installed radiation monitoring instrumentation to determine the accuracy and operability of the instrumentation. Also reviewed was the program to provide self-contained breathing apparatus (SCBA) to occupational workers.

During plant tours, the inspectors reviewed field instrumentation utilized by health physics technicians and plant workers to measure radioactivity and radiation levels, including portable field survey instruments, hand-held contamination frisking instruments, and continuous air monitors. The inspectors conducted a review of the instruments observed in the toured areas, specifically verification of current calibration, of appropriate source checks, and of proper function. The inspectors evaluated the calibration records for Eberline RO-2/-2A Survey Meters, dated September 10-20, 2001; and the following procedures for regulatory compliance and adequacy.

RE-ADM-1-10,	Portable Instrumentation Calibration Protocols, Rev. 6
RE-INS-7CA-1,	Calibration of RO-2 and RO-2A Ion Chambers, Rev. 5
RE-INS-7CA-12,	Calibration of the Eberline 6112 Teletector, Rev. 6
RE-INS-7CA-14,	Calibration of Ludlum 177 and Eberline RM-14 Friskers, Rev. 6
RE-INS-7CA-17,	Frisker Probe Efficiency Check, Rev. 8
RE-INS-7CB-1,	Calibration of Portable Air Samplers, Rev. 8
RE-INS-7CC-8,	Calibration of the Eberline AMS-4 Using Windows, Rev. 4
RE-INS-7CD-7,	Calibration and Use of the MGP Telepole, Rev. 0
RE-INS-7UC-8,	Use of the Eberline AMS-4 Continuous Air Monitor, Rev. 2
RE-INS-7UA-1,	Use of Portable Ion Chamber Instrumentation, Rev. 7
RE-INS-7UA-2,	Use of Portable GM Survey Instrumentation, Rev. 7

The inspectors identified and noted the condition and operability of selected installed area and process radiation monitors, and any accessible local response information on those monitors. The inspectors also interviewed the system engineer for the installed radiation monitoring system and reviewed for compliance and adequacy the following procedure and calibration record for an installed process radiation monitor.

RE-INS-7UD-6, Plant Radiation Monitors (ARMs/PRMs), Rev. 7
Process Radiation Monitor R11 Calibration Record dated July 30, 2001

The inspectors reviewed the adequacy of the program to provide SCBA for entering and working in areas of unknown radiological conditions. The inspection included a review of the status and surveillance records of bottled-breathing-air-cylinder stations, of SCBA air bottles, and of SCBA with air bottles attached and staged and ready for use in the plant. The inspectors also reviewed the status of training and qualification in the use of SCBA for operations and security personnel. The following procedures and documents were examined in the course of this review for regulatory compliance and adequacy.

AP-7.2,	Respiratory Protection Program, Rev. 4
EP-ADM-05,	Emergency Plan Equipment Inventory Administrative Procedure, Rev. 10
FP-13,	Inspection and Testing of Self-contained Breathing Apparatus, Rev. 10

Respirator Qualification Status Records for Operations and Security Personnel; dated October 24, 2001

The inspectors' review was against criteria contained in 10 CFR 20.1501, 10 CFR 20 Subpart H, site Technical Specifications, and site procedures.

b. Findings

A Green non-cited violation was identified involving failure to conduct triennial hydrostatic tests on approximately 80 self-contained-breathing-apparatus (SCBA) air cylinders.

The inspectors identified this issue when comparing the periodic checks performed on bottled-breathing-air-cylinder stations in accordance with procedure EP-ADM-05, "Emergency Plan Equipment Inventory Administrative Procedure," to the periodic checks performed on SCBA air bottles in accordance with procedure FP-13, "Inspection and Testing of Self-contained Breathing Apparatus." EP-ADM-05 included a check on the currency of the cylinder hydrostatic test date while the latter procedure did not. The inspectors identified this fact and inquired how the licensee ensured that SCBA air cylinders were within their hydrostatic test frequency. During a subsequent sampling inspection of SCBA air cylinders, the licensee identified that several cylinders were outside their hydrostatic test frequency. In determining the extent of the condition, the licensee subsequently identified a total of approximately 80 SCBA air cylinders which were overdue for hydrostatic testing, some by more than a year. This number represented approximately 25 percent of the total number of SCBA air cylinders available for use. The licensee removed all affected cylinders from service and replaced them with qualified cylinders.

10 CFR 50.54(q) requires that emergency plans meeting the standards in 50.47(b), be followed and maintained in effect. 10 CFR 50.47(b)(8) requires that the onsite emergency response plans provide and maintain adequate emergency equipment to support the emergency response. Section 7.4.2 A, of the Emergency Plan provides for protective equipment specifically designated for emergency use including respiratory protective devices. Emergency Plan implementing procedure IP-1070, Rev. 31,

specified that the Emergency Planning Coordinator or designee ensure that adequate supplies and equipment are specified in procedure EP-ADM-05. Procedure EP-ADM-05, provided for conduct of checks of bottled-breathing-air-cylinder stations while procedure FP-13 provided for inspection of SCBA bottles. Neither procedure ensured that the periodic hydrostatic test date for SCBA cylinders was checked to ensure that SCBA cylinders used for emergency response do not have expired tests and are qualified for use. 10 CFR 20.1703(c)(4)(vii) requires that a respiratory protection program be implemented and maintained that includes written procedures regarding storage, issuance, maintenance, repair, testing, and quality assurance of respiratory protection equipment. The failure to implement adequate quality assurance and maintenance procedures to ensure that respiratory protection equipment is properly tested and maintained is contrary to 10 CFR 50.54 (q) and 10 CFR 20.1703(c)(4)(vii).

The licensee uses SCBAs as emergency response equipment. Consequently, this finding was evaluated under the Emergency Preparedness Significance Determination Process. This finding is greater than minor because, if left uncorrected, inadequately tested and maintained respiratory protection equipment could have been used by personnel in the event of an emergency. This finding was determined to be of very low safety significance (GREEN), because, although it involved a failure to maintain emergency response equipment in accordance with regulatory requirements, it did not result in the licensee's failure to meet a planning standard since unqualified equipment was not actually used, all of the affected air cylinders displayed the proper air pressure indicating that the cylinders maintained the requisite integrity, and a sufficient supply (in excess of requirements) was available for use. The licensee placed this issue into its corrective action system as DER 01-04041.

The inspectors identified that because this violation of 10 CFR 20.1703 (C)(4)(vii) is of very low safety significance and because it is in the licensee's corrective action process (DER 01-03577), this violation was being treated as a Non-Cited violation consistent with Section VI.A of the NRC Enforcement Policy issued May 1, 2000 (65FR25368). **(NCV 50-286/01-09-01)**

4. OTHER ACTIVITIES (OA)

4OA1 Performance Indicator Verification

a. Inspection Scope (71151)

Reactor Coolant System Specific Activity

The inspectors reviewed the licensee's sample and analysis data used to report the reactor coolant system (RCS) specific activity performance indicator for the first and second quarters 2001 against the applicable criteria specified in Nuclear Energy Institute (NEI) 99-02, Regulatory Assessment Performance Indicator Guideline, Rev 1. In addition to record reviews, the inspectors observed a chemistry technician obtain an RCS sample on November 13, 2001.

Occupation Exposure Control Effectiveness

The inspectors selectively examined records used by the licensee to identify occurrences involving high radiation areas, very high radiation areas, and unplanned personnel exposures for the time period from mid-May 2001 to late October September 2001 against the applicable criteria specified in Nuclear Energy Institute (NEI) 99-02, Regulatory Assessment Performance Indicator Guideline, Rev 1, to verify that all conditions that met the NEI criteria were recognized and identified as Performance Indicators. The reviewed records included corrective action program records (Deviation/Event Reports) and issues captured by procedure RE-UOE-14-4, "Radiological Event Classification and Investigation." In conjunction with the reviews documented in previous inspection reports, this examination covered the intervening period back to late March 2001.

b. Findings

No findings of significance were identified.

4OA6 Meetings

Exit Meeting Summary

On December 5, 2001, the inspectors presented the inspection results to Mr. R. Barrett and other Entergy staff members who acknowledged the inspection results presented. The inspectors asked Entergy personnel whether any materials evaluated during the inspection were considered proprietary. No proprietary information was identified.

ATTACHMENT 1

SUPPLEMENTAL INFORMATION

a. Key Points of Contact

K. Baumbach	Site Surveillance Coordinator
R. Barrett	Vice President, Operations - IP3
J. Barry	Sr. Radiological Engineer
R. Burroni	I&C Manager
R. Cavalieri	Outage and Planning Manager
J. Comiotes	Director, Safety Assurance
E. Danko	Licensing Engineer
J. DeRoy	General Manager of Plant Operations
R. Deschamps	Radiation Protection Manager/RES Dept. Manager
J. Donnelly	Licensing Manager
M. Gillman	Operations Manager
D. Gray	Senior Radiological Engineer-Environmental
B. Kyler	ALARA Specialist
R. LaVera	Sr. Radiological Engineer
J. LePere	Waste Management General Supervisor
F. Mitchell	HP General Supervisor
J. Perrotta	Quality Assurance Manager
K. Peters	Corrective Actions and Assessment Manager
T. Phillips	Waste Management Supervisor
C. Putnam	System Engineer, Radiation Monitoring Systems
E. Reagan	Waste Management Operator
R. Rodino	Radiological Engineer
J. Russell	Special Projects Manager
S. Sandike	Plant Chemistry
M. Smith	Director, IP-3 Engineering
R. Solano	HP Supervisor
R. Tagliamonte	Waste Management Supervisor
S. Van Buren	Fire Program Administrator
A. Vitale	Maintenance Manager
J. Wheeler	Training Manager

b. List of Items Opened, Closed, and Discussed

Opened

50-286/01-09-01	NCV	Failure to hydrostatically test self-contained-breathing-apparatus air cylinders. (Section 2OS3)

Closed

None

Opened/Closed

None

c. List of Acronyms

ABFP	auxiliary boiler feedwater pump
AEOF	Alternate Emergency Operations Facility
ALARA	As Low As Reasonably Achievable
ARP	alarm response procedure
ASME	American Society of Mechanical Engineers
CFR	Code of Federal Regulations
COL	check-off list
CCW	component cooling water
DBD	design basis document
DER	Deviation/Event Report
EDG	emergency diesel generator
EP	Emergency Preparedness
ERO	Emergency Response Organization
FP	fire protection
FSAR	Final Safety Analysis Report
HP	health physics
HRA	High Radiation Area
JNC	Joint News Center
LHRA	Locked High Radiation Area
IR	inspection report
ml	milli-liter
NCV	non-cited violation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
OD	operability determination
ONOP	off-normal operating procedure
OS	occupational radiation safety
OSC	Operations Support Center
PI	performance indicator
PID	problem identification
PMT	post-maintenance test
RCA	radiological controlled area
RCP	reactor coolant pump
RCS	reactor coolant system
RSPS	Risk Significant Planning Standard
RWP	radiation work permit
SCBA	self-contained breathing apparatus
SI	safety injection
SOP	system operating procedure
SSCs	structures, systems, and components
TS	Technical Specifications
TSC	Technical Support Center
WR	work request

Review Process for 10 CFR 2.206 Petitions

Directive 8.11

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Review Process for 10 CFR 2.206 Petitions Directive 8.11

Policy
(8.11–01)

It is the policy of the U.S. Nuclear Regulatory Commission under Section 2.206 of Title 10 of the *Code of Federal Regulations* (10 CFR 2.206) to provide members of the public with the means to request action to enforce NRC requirements. The Commission may deny or grant a request for enforcement action, in whole or in part, and may take action that satisfies the safety concerns raised by the requester, even though it is not necessarily an enforcement action. Requests that raise health and safety and other issues without requesting enforcement action will be reviewed by means other than the 10 CFR 2.206 process.

Objectives
(8.11–02)

- To provide the public with a means to bring to the NRC's attention potential health and safety issues requiring NRC enforcement action. (021)
- To ensure the public health and safety through the prompt and thorough evaluation of any potential safety problem addressed by a petition filed under 10 CFR 2.206. (022)
- To provide for appropriate participation by the petitioners and the public in NRC's decision-making activities related to the 10 CFR 2.206 petition process. (023)
- To ensure effective communication with the petitioner on the status of the petition, including providing relevant documents and notification of NRC and licensee interactions on the petition. (024)

Organizational Responsibilities and Delegations of Authority
(8.11–03)

Executive Director for Operations (EDO)
(031)

Receives and assigns action for all petitions filed under 10 CFR 2.206.

Director, Office of the Chief Information Officer (OCIO)
(032)

Provides hardware, software, and communication services support of the NRC Home Page for making information publicly available on the status of the petitions.

Office of the General Counsel (OGC)
(033)

- Provides legal review and advice on 10 CFR 2.206 petitions and director's decisions upon specific request from the staff in special cases or where the petition raises legal issues. (a)
- Gives legal advice to the EDO, office directors, and staff on relevant 2.206 matters. (b)

Office Directors (or Designees)
(034)

- Have overall responsibility for assigned petitions. (a)
- Approve or deny a petitioner's request for immediate action. (b)
- Sign all acknowledgment letters and director's decisions. (c)
- Determine whether criteria for a meeting with the petitioner and licensee are met, and notify the Commission, through the EDO, once a determination is made that a 2.206 petition meets the criteria for a meeting. (d)

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Office Directors (or Designees)
(034) (continued)

- Provide up-to-date information for the monthly status report on all assigned petitions, including the total number of staff hours expended on each open petition; provide this information to the agency coordinator who, in turn, ensures that the information is made publicly available in the Public Document Room and on the NRC Home Page. (e)
- Appoint a petition review board chairperson. (f)
- Designate a petition manager for each petition. (g)
- Concur, as appropriate, in each extension request from the petition manager and forward the extension request to the Office of the EDO (OEDO) for approval. (h)
- Promptly notify the Office of Investigations (OI) of any allegations of suspected wrongdoing by a licensee, or the Office of the Inspector General (OIG) of suspected wrongdoing by an NRC staff person or NRC contractor, that are contained in the petitions they may receive. (i)
- Obtain review and concurrence from the Office of Enforcement for proposed director's decisions that involve potential enforcement implications. (j)
- Ensure that the director's decision and the supporting evaluation of the petition adequately reflects information presented at any meetings with the petitioner, to the extent that such information was useful. (k)

Regional Administrators
(035)

- Refer any 2.206 petitions they may receive to the EDO. (a)
- Promptly notify OI of any allegations of suspected wrongdoing by a licensee, or OIG of suspected wrongdoing by an NRC staff person or NRC contractor, that are contained in the petitions they may receive. (b)
- As needed, provide support and information for the preparation of an acknowledgment letter and/or a director's decision on a 2.206 petition. (c)
- Make the petition manager aware of information that is received or that is the subject of any correspondence relating to a pending petition. (d)

2.206 Petition Review Board Chairperson
(Each program office has a board chairperson, generally an SES manager.)
(036)

- Chairs petition review board meetings. (a)
- Ensures appropriate review of all new petitions in a timely manner. (b)
- Ensures appropriate documentation of petition review board meetings. (c)
- Chairs periodic meetings with the petition managers to discuss the status of open petitions and to provide guidance for timely issue resolution. (d)

Director, Division of Licensing Project Management, Office of Nuclear Reactor Regulation (NRR)
(037)

Appoints the Agency 2.206 Coordinator, NRR, who prepares monthly reports to the EDO on petition status, age, and resource expenditures for the signature of the Associate Director for Project Licensing and Technical Analysis.

Applicability
(8.11–04)

The policy and guidance in this directive and handbook apply to all NRC employees.

Handbook
(8.11–05)

Handbook 8.11 details the procedures for staff review and disposition of petitions submitted under Section 2.206.

Definitions
(8.11–06)

A 10 CFR 2.206 Petition. A written request filed by any person to institute a proceeding to modify, suspend, or revoke a license, or for any other enforcement action that may be proper and that meets the criteria for review under 10 CFR 2.206 (see Part II of Handbook 8.11).

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Definitions
(8.11–06) (continued)

A 10 CFR 2.206 Petition Meeting. A meeting open to the public and held by NRC staff to provide an opportunity to the petitioner and licensee to supply information to assist NRC staff in the evaluation of petitions that raise new, significant safety issues, as defined in Part II(D)(3)(a) of Handbook 8.11, or that provide new information or approaches for the evaluation of significant safety issues previously evaluated.

References
(8.11–07)

Code of Federal Regulations—

10 CFR 2.206, "Requests for Action Under this Subpart."

10 CFR 2.790, "Public Inspections, Exemptions, Requests for Withholding."

Nuclear Regulatory Commission—

Enforcement Manual, "General Statement of Policy and Procedure for NRC Enforcement Actions," Office of Enforcement, NUREG-1600.

Investigative Procedures Manual, Office of Investigations, revised August 1996.

Management Directive (MD) 3.5, "Public Attendance at Certain Meetings Involving the NRC Staff."

— MD 8.8, "Management of Allegations."

— MD 12.6, "NRC Sensitive Unclassified Information Security Program."

Memorandum of Understanding Between the NRC and the Department of Justice, December 12, 1988.

"Nuclear Regulatory Commission Issuances," published quarterly as NUREG-0750.

Review Process for 10 CFR 2.206 Petitions

Handbook 8.11

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Part I
Initial Staff Actions

Introduction (A)

Title 10 of the *Code of Federal Regulations*, Section 2.206 (1)

This section of the regulations has been a part of the Commission's regulatory framework since the Commission was established in 1975. Section 2.206 permits any person to file a petition to request that the Commission institute a proceeding to take enforcement action. (a)

The petition must request that a license be modified, suspended, or revoked, or that other appropriate enforcement action be taken and must provide sufficient facts that constitute the bases for taking the particular action. (b)

Section 2.206 provides a procedure that allows any person to file a request to institute a proceeding for enforcement action and requires that the petition be submitted in writing and provide sufficient grounds for taking the proposed action. Do not treat general opposition to nuclear power or a general assertion of a safety problem, without supporting facts, as a formal petition under 10 CFR 2.206. Treat general requests as routine correspondence. (c)

NRC's Receipt of a Petition (2)

After NRC receives a petition, it is assigned to the director of the appropriate office for evaluation and response. The official response is a written decision of the office director that addresses the issues raised in the petition. The director's decision can grant, partially grant, or deny the petition. The Commission may, on its own initiative, review the director's decision (to determine if the director has abused his or her discretion), but no petition or other request for Commission review of the director's decision will be entertained by the Commission.

Introduction (A) (continued)

NRC Home Page (3)

The NRC Home Page provides the up-to-date status of pending 2.206 petitions, director's decisions issued, and notices of meetings. The NRC external home page is accessible via the World Wide Web, and documents may be found at *http://www.nrc.gov/NRC/PUBLIC/2206/index.html.* Director's decisions are published in NRC Issuances (NUREG-0750).

Assignment of Staff Action and 2.206 Petition Review Board (B)

Office of the Executive Director for Operations (OEDO) (1)

The OEDO assigns the petition to the appropriate office for action. The original incoming is sent to the office and a copy of the petition is sent to the Office of the General Counsel (OGC).

Agency 2.206 Coordinator, Office of Nuclear Reactor Regulation (NRR) (2)

The Agency 2.206 Coordinator, NRR (appointed by the Director, Division of Licensing Project Management), receives copies of all 2.206 petitions from OEDO and prepares the 2.206 periodic status report.

Assigned Office (3)

The office director of the assigned office designates a petition manager and an office petition review board chairperson for each petition. The petition manager drafts the acknowledgment letter and *Federal Register* notice (see Exhibits 1 and 2 of this handbook). The petition manager ensures that the petition is placed in the public document room after it is determined that the petition does not contain allegations or sensitive information. A petition review board meets within 3 weeks of receipt of the petition. Each assigned office conducts at least one review board meeting for each petition. The petition review board consists of—(a)

- A petition review board chairperson (SES manager or above) (i)
- A petition manager (ii)
- Cognizant technical review branch chief(s), as necessary (iii)
- An Office of Enforcement (OE) or Office of Investigations (OI) representative, as needed (iv)

In addition, OGC normally will participate. (b)

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Assignment of Staff Action and 2.206 Petition Review Board (B) (continued)

Assigned Office (3) (continued)

The purpose of the petition review board meeting is to—(c)

- Determine whether the petitioner's request meets the criteria defined in 10 CFR 2.206 (see Part II(A) of this handbook) (i)
- Determine whether the petition meets the criteria for a meeting with the petitioner and licensee (see Part II(C) of this handbook) (ii)
- Promptly address any request for immediate action (iii)
- Address the possibility of issuing a partial director's decision (iv)
- Draft a schedule for responding to the petitioner so that a commitment is made by management and the technical review staff to respond to the petition in a timely manner (see Part IV(A) of this handbook) (v)
- Determine whether the petition is sufficiently complex that additional review board meetings should be scheduled to ensure that suitable progress is being made (vi)

The appointed petition review board chairperson for each office—(d)

- Chairs and coordinates 2.206 petition review board meetings for the assigned office (i)
- Ensures the 2.206 petition review board meetings are documented (ii)

Assigned Office Action (C)

Office Director (1)

The assigned office director signs and issues the acknowledgment letter and the *Federal Register* notice. This action should be completed by the date specified by OEDO for the action. (a)

The office director, or designee, ensures that the appropriate licensee is sent a copy of the acknowledgment letter and a copy of the incoming request at the same time as the petitioner. If appropriate, the licensee will be requested to provide a response to the NRC on the issues specified in the petition, usually within 30 days. (b)

Assigned Office Action (C) (continued)

Office Director (1) (continued)

When an unannounced technical inspection or an OI investigation is involved, the office director makes the decision to release information to the licensee in a manner to ensure that the staff does not release information that would indicate to the licensee or the public that an unannounced inspection or investigation will be undertaken or information that would undermine the inspection or investigation. (c)

The office director carefully considers any potential conflict or loss of objectivity that might result from assigning the same staff who were previously involved with the issue that gave rise to the petition. (d)

Petition Manager (2)

The petition manager—(a)

- Briefs the petition review board on the petitioner's request(s), any background information, the need for an independent technical review, and a proposed plan for resolution, including target completion dates (i)
- Promptly advises the licensee of the petition, sends the licensee a copy of the petition, and places the petition and all subsequent related correspondence in the Public Document Room. (ii)
- Drafts the acknowledgment letter and *Federal Register* notice, serves as the NRC point of contact with the petitioner, provides updates to the periodic 2.206 status report to the Executive Director for Operations (EDO), and monitors the progress of any OI investigation and related enforcement actions (iii)
- Prepares the director's decision on the petition for the office director's consideration, including coordination with the appropriate staff supporting the review (iv)
- Ensures appropriate documentation of all 10 CFR 2.206 petition determinations, including the determination on whether a meeting is offered (v)

The petition manager ensures that a copy of this management directive is included with the acknowledgment letter. The acknowledgment letter also should include the name and telephone number of the petition manager and identify the technical staff organizational units that will participate in the review. (b)

Assigned Office Action (C) (continued)

Petition Manager (2) (continued)

The acknowledgment letter, as well as the transmittal letter for the director's decision or partial director's decision, should acknowledge the petitioner's efforts in bringing issues to the staff's attention. (c)

If appropriate, the decision transmittal letter should acknowledge that the petitioner identified valid issues and should specify the corrective actions that have been or will be taken to address these issues, notwithstanding that some or all of the petitioner's specific requests for action have not been granted. (d)

The petition manager places the petitioner on distribution for all relevant NRC correspondence to the licensee to ensure that the petitioner receives copies of all NRC correspondence with the licensee pertaining to the petition. If there is a service list(s) add the petitioner to the list(s) for all headquarters and regional documents on the affected dockets. Remove the petitioner's name from distribution and/or the service list(s) 90 days after issuance of the director's decision. The petition manager sends licensee-prepared documents submitted to the NRC that are relevant to the petition to the petitioner for the same duration as staff-generated documents. If the licensee is asked to respond, the petition manager advises the licensee that the NRC intends to place the licensee's response in the Public Document Room and provide the response to the petitioner. (e)

Unless necessary for NRC's proper evaluation of the petition, the licensee should avoid using proprietary or personal privacy information that requires protection from public disclosure. If such information is necessary to properly respond to the petition, the petition manager ensures the information is protected in accordance with 10 CFR 2.790. (f)

The petition manager also ensures that the petitioner is placed on distribution for other NRC correspondence relating to the issues raised in the petition, including relevant generic letters or bulletins that are issued during the pendency of the NRC's consideration of the petition. This does not include NRC correspondence or documentation related to an OI investigation, which will not be released outside NRC without the approval of the Director, OI. (g)

Assigned Office Action (C) (continued)

Petition Manager (2) (continued)

Before the petition review board meeting, the petition manager informs the petitioner that the 2.206 petition process is a public process in which the petition and all the information in it will be made public. If the petitioner requests anonymity and that the petition not be made public, advise the petitioner that, because of its public nature, the 2.206 process cannot provide protection of the petitioner's identity. In such cases, advise the petitioner that the matter will be handled as an allegation and that the petitioner should withdraw the petition in writing. During this telephone contact, offer the petitioner an opportunity to have one representative give a presentation to the petition review board. The petitioner (or representative) may participate in person or by teleconference on a recorded line and only for the purpose of explaining the requested actions, their bases, and answering staff questions. The presentation will be limited to about a half hour and will be transcribed. Treat the transcription as a supplement to the petition and send a copy of the transcription to the petitioner and to the same distribution as the original petition. (h)

If the petition contains a request for immediate enforcement action by the NRC, such as a request for immediate suspension of facility operation until final action is taken on the request, the acknowledgment letter must respond to the immediate action requested. If the immediate action is denied, the staff must explain the basis for the denial in the acknowledgment letter. If the staff plans to take an action that is contrary to an immediate action requested in the petition before issuing the acknowledgment letter (such as permitting restart of a facility when the petitioner has requested that restart not be permitted), the petition manager must promptly notify the petitioner by telephone of the pending staff action. The petitioner will not be advised of any wrongdoing investigation being conducted by OI. (i)

In cases where the staff identifies certain issues in a petition that it believes are more appropriately addressed using the allegation process, the petition manager advises the petitioner of this staff view during the initial telephone contact and suggests to the petitioner that he or she withdraw those issues from the petition with the understanding that they will be addressed through the allegation process. (j)

Assigned Office Action (C) (continued)

Petition Manager (2) (continued)

All telephone contacts with the petitioner will be documented by a memorandum to file, which becomes part of the petition file. (k)

OGC Staff Attorney (3)

OGC normally participates in the petition review board meetings for the 2.206 petition and provides legal review and advice on 10 CFR 2.206 petitions and director's decisions upon specific request from the staff in special cases or where the petition raises legal issues. OGC may be assigned as the responsible office for the review, if appropriate.

Reporting Requirements and Updating the Status of Petitions on the NRC Home Page (D)

On a monthly basis, the Agency 2.206 Coordinator, NRR, will contact all petition managers reminding them to prepare a status report on 2.206 petitions in their office. This report will be made available in the PDR and placed on the NRC Home Page. The petition managers should electronically mail the status report for each open petition, with the exception of sensitive information as described below, to PETITION. The Agency 2.206 Coordinator combines all the status reports, including staff performance metrics for petitions processed under 10 CFR 2.206 for the current year, in a monthly report to the EDO from the Associate Director, Project Licensing and Technical Analysis, and provides a copy of the report to the Web operator for placement on the NRC Home Page. (1)

If the information on the status of the petition is sensitive information that may need to be protected from disclosure (e.g., safeguards or facility security information, proprietary or confidential commercial information, information relating to an ongoing investigation of wrongdoing or enforcement actions under development, or information about referral of matters to the Department of Justice), the petition manager and Agency 2.206 Coordinator should ensure that this information is protected from disclosure. Sensitive information should be handled in accordance with Management Directive 12.6, "NRC Sensitive Unclassified Information Security Program." (2)

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Part II

Criteria for Petition Evaluation

Use the criteria discussed in this part for determining whether a petition should be considered under 10 CFR 2.206, if similar petitions should be consolidated, and if a public meeting should be offered.

Criteria for Reviewing Petitions Under 10 CFR 2.206 (A)

Review a petition under the requirements of 10 CFR 2.206 if the request meets all of the following criteria: (1)

- The petition contains a request for enforcement action: either requesting that NRC impose requirements by order; or issue an order modifying, suspending, or revoking a license; or issue a notice of violation, with or without a proposed civil penalty. (a)
- The enforcement action requested and the facts that constitute the bases for taking the particular action are specified. The petitioner must provide some element of support beyond the bare allegation. The supporting facts must be credible and sufficient to warrant further inquiry. (b)
- Acceptance for review under 10 CFR 2.206 will not result in circumventing an available proceeding in which the petitioner is or could be a party. (c)

If a petition meets the criteria but does not specifically cite 10 CFR 2.206, the petition manager will attempt to contact the petitioner by telephone to determine if the individual wants the request processed pursuant to 10 CFR 2.206. If the petition is unclear or appears to be marginal in meeting the criteria for review, the petition manager will encourage and facilitate a presentation to the petition review board by the petitioner so that the concerns can be clarified. (2)

Criteria for Rejecting Petitions Under 10 CFR 2.206 (B)

Do not review a petition under 10 CFR 2.206, whether specifically cited or not, under the following circumstances: (1)

- The incoming correspondence does not ask for an enforcement action or fails to provide sufficient facts to support the petition but simply alleges wrongdoing, violations of NRC regulations, or existence of safety concerns. The request cannot be simply a general statement of opposition to nuclear power or a general assertion without supporting facts (e.g., the quality assurance at the facility is inadequate). These assertions will be treated as allegations and referred for appropriate action in accordance with Management Directive (MD) 8.8, "Management of Allegations." (a)
- The petitioner raises issues that already have been the subject of NRC staff review and evaluation either on the cited facility, other plant facilities, or on a generic basis, for which a resolution has been achieved, the issues have been dispositioned, and the resolution is applicable to the facility in question. (b)
- The request is to reconsider or reopen a previous enforcement action (including a decision not to initiate an enforcement action) or a director's decision and will not be treated as a 2.206 petition unless it presents significant new information. (c)
- The request is to deny a license application or amendment. This type of request should initially be addressed in the context of the relevant licensing action, not under 10 CFR 2.206. (d)

If a petitioner's request does not meet the criteria for consideration under 10 CFR 2.206, a letter will be sent to the petitioner explaining why the request is not being reviewed under 10 CFR 2.206 (see Exhibit 3). (2)

Criteria for Consolidating Petitions (C)

All requests submitted by different individuals will, as a general practice, be treated and evaluated separately. When two or more petitions request the same action, specify the same bases, provide adequate supporting information, and are submitted at about the same time, the petition review board considers the benefits of consolidating the petitions against the potential of diluting the importance of any petition and recommends whether or not consolidation is appropriate. The assigned office director determines whether or not to consolidate the petitions.

Criteria for Meetings (D)

For petitions meeting the criteria specified in this section, the staff offers the petitioner an opportunity for a meeting. A meeting, which is a resource for the staff in evaluating the petition, also affords the petitioner and the licensee an opportunity for enhanced involvement in the Commission's decision-making process. (1)

A meeting is not automatically granted and will not be offered simply at the petitioner's request. If the staff offers the petitioner the opportunity for a meeting, the petitioner then has the option to accept or reject the offer. If the petitioner rejects the offer, a meeting will not be conducted and the petition review will continue. If the petitioner accepts the offer of a meeting, the licensee will be invited to participate in the meeting. (2)

The staff uses the following criteria to determine if an opportunity for a meeting is to be offered to the petitioner. Either one of the two elements listed below must be met. (3)

- The petition raises the potential for a significant safety issue. For nuclear reactors and nuclear material licensees, a significant safety issue is an issue that could lead to a significant exposure, could cause significant core damage, or could otherwise result in a significant reduction of protection of public health and safety. The information is considered "new" if one the following applies: (a)
  - The petition presents a significant safety issue not previously evaluated by the staff. (i)
  - The petition presents significant new information on a significant safety issue previously evaluated. (ii)
  - The petition presents a new approach for evaluating a significant safety issue previously evaluated and, on preliminary assessment, the new approach appears to have merit and to warrant reevaluation of the issue. (iii)
- The petition alleges violations of NRC requirements involving a significant safety issue for which new information or a new approach has been provided, and it presents reasonable supporting facts that tend to establish that the violation occurred. (b)

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Criteria for Meetings (D) (continued)

A meeting will not be held if to do so will compromise "sensitive" information that may need to be protected from disclosure, such as safeguards or facility security information, proprietary or confidential commercial information, or information relating to an ongoing investigation of wrongdoing. The petition manager ensures that a meeting will not compromise the protection of this information before offering the petitioner the opportunity for a meeting. A meeting also will not be held simply because the petitioner claims to have additional information and will not present it in any other forum. (4)

Part III

Procedures for Conducting a 10 CFR 2.206 Petition Meeting

After the staff determines that a petition meets the criteria for a meeting, set forth in Part II (D) of this handbook, and the petitioner accepts the offer of a meeting, the petition manager contacts the petitioner to schedule a mutually agreeable date for the meeting. The petition manager also requests the licensee to participate in the meeting to present its position and coordinates the schedules and dates with the licensee. The meeting must be scheduled so as not to adversely impact the established petition review schedule.

Meeting Location (A)

Meetings normally will be held at NRC headquarters in Rockville, Maryland, with provisions for participation by telephone or video link. If justified by special circumstances, the staff may hold the meeting at some location other than NRC headquarters.

Notice of Meeting (B)

Provisions for a meeting notice will be made in accordance with agency policy. The NRC petition manager will ensure that a copy of the meeting notice is placed on the NRC Home Page, that the scheduled meeting is included in the Public Meeting Notice System, that the Office of Public Affairs is notified of the meeting, and that the meeting notice is communicated to the petitioner. (1)

All meetings are transcribed, and the transcripts are publicly available. (2)

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Meeting Chairperson (C)

The meeting is chaired by the NRC office director responsible for addressing the petition, or by his or her designee. (1)

The purpose of the meeting is to obtain additional information from the petitioner and the licensee for NRC staff use in evaluating the petition. It is not a forum for the staff to offer any preliminary decisions on the evaluation of the petition. The chairperson has final authority to determine the conduct of the meeting. Members of the public may attend as observers. (2)

Meeting Format (D)

The meeting chairperson provides a brief summary of the 2.206 process, the purpose of the meeting, and the petition. Following the opening statement—(1)

- The petitioner is allowed a reasonable amount of time (approximately 30 minutes) to articulate the basis for the petition. (a)
- NRC staff have an opportunity to ask the petitioner questions for purposes of clarification. (b)
- The licensee is then allowed a reasonable amount of time (approximately 30 minutes) to address the issues raised in the petition. (c)
- NRC staff have an opportunity to ask the licensee questions for purposes of clarification. (d)

Part IV
Further Staff Actions

General (A)

Schedule (1)

The assigned office holds a petition review board meeting on the submitted 2.206 petition within 3 weeks of receipt of the petition. The review board helps determine the appropriate schedule as well as how best to respond to the petitioner's concerns. (a)

The goal is to issue the director's decision, or partial director's decision, within 120 days from the date of issuance of the acknowledgment letter. The Office of the Executive Director for Operations (OEDO) tracks the target date, and any change of the date requires approval by the OEDO. Enforcement actions that are prerequisites to a director's decision must be expedited and completed in time to meet the the 120-day goal. Investigations by the Office of Investigations (OI) should be expedited to the extent practicable. However, the goal of issuing a full, or partial, director's decision within 120 days after issuing the acknowledgment letter applies only to petitions whose review schedules are within the staff's control. If issues in a petition are the subject of an extended OI investigation, or a referral to the Department of Justice (DOJ), or if NRC decides to await a Department of Labor (DOL) decision, a partial director's decision is issued within 120 days, and the 120-day goal is not applied to the remainder of the petition. When more time is needed (e.g., when issues in a petition are the subject of an extended OI investigation, or a referral to DOJ, or if NRC decides to await a DOL decision), the assigned office director determines the need for an extension of the schedule and requests the extension from the OEDO. (b)

General (A) (continued)

Schedule (1) (continued)

If the director's decision cannot be issued in 120 days, the petition manager promptly contacts the petitioner explaining the reason(s) for the delay and maintains a record of such contact. If the delay results from an ongoing OI investigation, the petition manager contacts the Director, OI, to obtain approval for citing the OI investigation as the reason for the delay. (c)

If there is alleged wrongdoing on the part of licensees, their contractors, or their vendors, immediately notify OI. If there is alleged wrongdoing involving an NRC employee, NRC contractors, or NRC vendors, immediately notify the Office of the Inspector General (OIG). (d)

Petition Review Board Actions (2)

The petition review board ensures that an appropriate petition review process is followed. This includes recommending whether or not: (a)

- The submittal qualifies as a 2.206 petition. (i)
- The petitioner should be offered or informed of an alternative process (e.g., consideration of issues as allegations, consideration of issues in a pending license proceeding, or conduct of an inspection). (ii)
- The petition should be consolidated with another petition. (iii)
- A public meeting should be offered. (iv)
- Referral to OI or OIG is appropriate. (v)
- There is a need for additional review board meetings. (vi)
- There is a need for the Office of the General Counsel (OGC) to participate in the review. (vii)
- An adequate review schedule and technical review participation have been established. (viii)
- Any petitioner's request for immediate action should be granted or denied. (ix)
- The licensee should be requested to respond to the petition. (x)
- A partial director's decision should be issued. (xi)

General (A) (continued)

Petition Manager Actions (3)

The petition manager drafts the acknowledgment letter and *Federal Register* notice and coordinates all information required from the professional staff within his or her organization and other organizations and from OI if a wrongdoing issue is under consideration. The petition manager also advises his or her management of the need for OGC review and advice regarding a petition in special cases. An Associate Director of the Office of Nuclear Regulation (NRR), a Division Director in the Office of Nuclear Material Safety and Safeguards (NMSS), or the Director of the Office of Enforcement(OE) makes a request for OGC involvement to the OGC special counsel assigned to 2.206 matters. (a)

The petition manager ensures that the petitioner is notified at least every 60 days of the status of the petition, or more frequently if significant actions occur. The petition manager makes the bimonthly status reports by telephone and should not leave a message on a voice mail message system unless repeated efforts to contact the petitioner are unsuccessful. The petition manager keeps up-to-date on the status of the petition so that reasonable detail can be provided with the status reports. However, the status report to the petitioner will not indicate—(b)

- An ongoing OI investigation, unless approved by the Director, OI (i)
- The referral of the matter to DOJ (ii)
- Enforcement action under consideration (iii)

The petition manager also will make the following telephone contacts with the petitioner: (c)

- Within 1 week after receipt of the petition and before the petition review board meeting, contact the petitioner to explain the public nature of the 2.206 petition process. During this contact, offer the petitioner an opportunity to have one representative give a presentation to the petition review board. The petitioner (or representative) may participate in person or by teleconference on a recorded line and only for the purpose of explaining the requested actions, their bases, and answering staff questions. The presentation will be limited to about a half hour and will be transcribed. Treat the transcription as a supplement to the petition and send a copy of the transcription to the petitioner and to the same distribution as the original petition. (i)

General (A) (continued)

Petition Manager Actions (3) (continued)

- After the petition review board meets, and before issuance of the acknowledgment letter, inform the petitioner as to whether or not the petition qualifies as a 2.206, disposition of any requests for immediate action, how the review will proceed, and that an acknowledgment letter is coming. (ii)
- Before dispatching the director's decision (or partial decision), inform the petitioner of the imminent issuance of the decision and the substance of the decision. (iii)
- When the director's decision has been signed, promptly send a copy electronically or by fax, if possible, to the petitioner. (iv)

Director's Decision (B)

The staff normally prepare a partial director's decision when some of the issues associated with the 2.206 petition are resolved in advance of other issues and if significant schedule delays are anticipated before resolution of the entire petition. If a wrongdoing investigation is being conducted in relation to the petition, the staff consider the results of the OI investigation, if available, in completing the action on the petition. (1)

Management Directive 8.8, "Management of Allegations," provides agency policy with regard to notifying OI of wrongdoing matters, as well as initiating, prioritizing, and terminating investigations. The petition manager should become familiar with the current version of this directive and follow the policy outlined therein when dealing with issues requiring OI investigations. (2)

All information related to an OI wrongdoing investigation, or even the fact that an investigation is being conducted, will receive limited distribution within NRC and will not be released outside NRC without the approval of the Director, OI. Within NRC, access to this information is limited to those having a need-to-know. Regarding a 2.206 petition, the assigned office director, or his designee, maintains copies of any documents required and ensures that no copies of documents related to an OI investigation are placed in the docket file, the agency's document management system, or the Public Document Room (PDR), without the approval of the Director, OI. (3)

Director's Decision (B) (continued)

The petition manager submits the completed draft decision to his or her management for review. After management's review, the petition manager incorporates any proposed revisions in the decision. If the decision is based on or references a completed OI investigation, OI must concur in the accuracy and characterization of the OI findings and conclusions that are used in the decision. (4)

If appropriate, the petition manager obtains OE management's review of and concurrence in the draft director's decision for potential enforcement implications. (5)

Granting the Petition (C)

Upon granting the petition, in whole or in part, the petition manager prepares a "Director's Decision Under 10 CFR 2.206" for the office director's signature. The decision explains the bases upon which the petition has been granted and identifies the actions that NRC staff have taken or will take to grant all or that portion of the petition. The Commission may grant a request for enforcement action, in whole or in part, and also may take action to satisfy the safety concerns raised by the petition, although such action is not necessarily an enforcement action. A petition is characterized as being granted in part when NRC did not grant the action as asked but took other action to address the underlying safety problem. If the petition is granted in full, the director's decision explains the bases for granting the petition and states that the Commission's action resulting from the director's decision is outlined in the Commission's order or other appropriate communication. (1)

If the petition is granted by issuing an order, the petition manager prepares a letter to transmit the order to the licensee. He or she prepares another letter to explain to the petitioner that the petition has been granted and encloses a copy of the order. Copies of the director's decision and *Federal Register* notice to be sent to the licensee and individuals on the service list(s) are dispatched simultaneously with the petitioner's copy. (2)

Denying the Petition (D)

Upon denial of the petition, in whole or in part, the petition manager prepares a "Director's Decision Under 10 CFR 2.206" for the office director's signature. The decision explains the bases for the denial and discusses all matters raised by the petitioner in support of the request. If appropriate, the decision transmittal letter acknowledges that

Denying the Petition (D) (continued)

the petitioner identified valid issues and specifies the corrective actions that have been or will be taken to address these issues, notwithstanding that some of all of the petitioner's specific requests for action have not been granted. The office director sends a letter to the petitioner transmitting the director's decision, along with a *Federal Register* notice explaining that the request has been denied. (1)

If an OI investigation is completed either before granting or denying the petition, the petition manager contacts OI and OE to coordinate NRC's actions when the wrongdoing matter has been referred to DOJ. It may be necessary to withhold action on the petition in keeping with the memorandum of understanding with DOJ. (2)

Issuance of Director's Decision (E)

A decision under 10 CFR 2.206 consists of a letter to the petitioner, the director's decision, and the *Federal Register* notice. The petition manager or administrative staff contacts the Office of the Secretary (SECY) to obtain a director's decision number (i.e., DD-YEAR-00). A director's decision number is assigned to each director's decision in numerical sequence. This number is typed on the letter to the petitioner, the director's decision, and the *Federal Register* notice. Note that the director's decision itself is not published in the *Federal Register*; only the notice of its availability, containing the substance of the decision, is published (see Exhibit 4). (1)

The assigned office director signs the *Federal Register* notice. After the notice is signed, it is forwarded to the Rules and Directives Branch, Office of Administration (ADM/DAS/RDB), for transmittal to the Office of the Federal Register for publication. (2)

Distribution (F)

The administrative staff of the assigned office reviews the 10 CFR 2.206 package before it is dispatched and determines appropriate distribution. The administrative staff also performs the following actions on the day the director's decision is issued: (1)

- Telephones the Rulemakings and Adjudications Staff, SECY, to advise the staff that the director's decision has been issued. (a)
- Immediately hand-carries the listed material to the following offices (in the case of the petitioner, promptly dispatch the copies.): (b)

Distribution (F) (continued)

- Rulemakings and Adjudications Staff, SECY (i)
  - Five copies of the director's decision (*a*)
  - Two courtesy copies of the entire decision package including the distribution and service lists. Ensure that documents referenced in the decision are publicly available in the NRC Public Document Room (*b*)
  - Two copies of the incoming petition and any supplement(s) (*c*)
- Petitioner (ii)
  - Signed original letter (*a*)
  - Signed director's decision (*b*)
  - A copy of the *Federal Register* notice (*c*)
- Chief, Rules and Directives Branch (iii)
  - Original signed *Federal Register* notice (*a*)
  - Five paper copies of the notice (*b*)

Promptly fulfill these requirements because the Commission has 25 calendar days from the date of the decision to determine whether or not the director's decision should be reviewed. (2)

Although 2.206 actions are controlled as green tickets, use the following guidelines when distributing copies internally and externally: (3)

- Attach the original 2.206 petition and any enclosure(s) to the Docket or Central File copy of the first response (acknowledgment letter). Issue copies to the appropriate licensees and individuals on the docket service list(s). (a)
- When action on a 2.206 petition is completed, the petition manager should ensure that all publicly releasable documentation is placed in the PDR and the agency document control system. (b)
- The distribution list should include appropriate individuals and offices as determined by the assigned office. (c)

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Followup Actions (G)

The administrative staff of the assigned office completes the following actions within 2 working days of issuance of the director's decision:

- Provide one paper copy of the director's decision to the OGC special counsel assigned to 2.206 matters. (1)
- Copy the final version of the director's decision onto a diskette in WordPerfect. Send this diskette and two paper copies of the signed director's decision to the NRC Issuances (NRCI) Project Officer, Electronic Publishing Section (EPS), Publishing Services Branch (PSB), Office of the Chief Information Officer (OCIO). (2)
- When writing opinions, footnotes, or partial information (such as errata) on the diskette, identify the opinion, the director's decision number, and the month of issuance at the beginning of the diskette. Clearly identified information on the diskettes will help to avoid administrative delays and improve the technical production schedule for proofreading, editing, and composing the documents. (3)
- Electronically mail a signed, dated, and numbered copy of the director's decision to NRCWEB for the NRC Home Page. (4)
- Electronically prepare a headnote, which is a summary of the petition consisting of no more than two paragraphs describing what the petition requested and how the director's decision resolved or closed out the petition. Electronically send the headnote to the PSB, OCIO, for monthly publication in the NRC Issuances, NUREG-0750. The headnotes should reach PSB before the 5th day of the month following the issuance of the director's decision. (5)

Commission Actions (H)

SECY informs the Commission of the availability of the director's decision. The Commission, at its discretion, may determine to review the director's decision within 25 days of the date of the decision and may direct the staff to take some other action than that in the director's decision. If the Commission does not act on the director's decision within 25 days, the director's decision becomes the final agency action and a SECY letter is sent to the petitioner informing the petitioner that the Commission has taken no further action on the petition.

Exhibit 1

Sample Acknowledgment Letter

[Petitioner's Name]
[Petitioner's Address]

Dear Mr. :

Your petition dated [insert date] and addressed to the [insert addressee] has been referred to me pursuant to 10 CFR 2.206 of the Commission's regulations. You request [state petitioner's requests]. As the basis for your request, you state that [insert basis for request]. I would like to express my sincere appreciation for your effort in bringing these matters to my attention.

Your request to [insert request for immediate action] at [insert facility name] is [granted or denied] because [staff to provide explanation].

As provided by Section 2.206, we will take action on your request within a reasonable time. I have assigned [first and last name of petition manager] to be the petition manager for your petition. Mr. [last name of petition manager] can be reached at [301–415–extension of petition manager] Your petition is being reviewed by [organizational units] within the Office of [name of appropriate Office]. If necessary, add: I have referred to the NRC Office of the Inspector General (OIG) those allegations of NRC wrongdoing contained in your petition. I have enclosed for your information a copy of the notice that is being filed with the Office of the *Federal Register* for publication. I have also enclosed for your information a copy of Management Directive 8.11 on the public petition process.

Sincerely,

[Office Director]

Enclosures: *Federal Register* Notice
Management Directive 8.11 re: Petition Process

cc: [Licensee (w/copy of incoming 2.206 request) & Service List]

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Exhibit 2

[7590–01–P]

Sample *Federal Register* Notice

U.S. NUCLEAR REGULATORY COMMISSION

Docket No(s).

License No(s).

[Name of Licensee]

RECEIPT OF REQUEST FOR ACTION UNDER 10 CFR 2.206

Notice is hereby given that by petition dated [insert date], [insert petitioner's name] (petitioner) has requested that the NRC take action with regard to [insert facility or licensee name]. The petitioner requests [state petitioner's requests].

As the basis for this request, the petitioner states that [state petitioner's basis for request].

The request is being treated pursuant to 10 CFR 2.206 of the Commission's regulations. The request has been referred to the Director of the Office of [insert action office]. As provided by Section 2.206, appropriate action will be taken on this petition within a reasonable time. [If necessary, add] By letter dated ________, the Director (granted or denied) petitioner's request for [insert request for immediate action] at [insert facility/licensee name]. A copy of the petition is available for inspection at the Commission's Public Document Room at 2120 L Street, NW. (Lower Level), Washington, DC 20555-0001.

FOR THE NUCLEAR REGULATORY COMMISSION

[Office Director]

Dated at Rockville, Maryland

This ________ day of ______________, 1999.

Exhibit 3

Sample One Step Acknowledgment / Denial Letter

[Insert petitioner's name & address]

Dear [insert petitioner's name]:

In a letter dated [insert date], to [OEDO/or addressee, NRC], signed by you and submitted pursuant to 10 CFR 2.206, you requested that the NRC order the [insert facility or licensee name] to be immediately shut down and remain shut down until either (1) all of the failed fuel assemblies are removed from the reactor core, or (2) the plant's design and licensing bases are properly updated to reflect continued operation with failed fuel assemblies. Attached to the petition was a copy of a report dated April 2, 1998, titled "Potential Nuclear Safety Hazard – Reactor Operation With Failed Fuel Cladding."

The attached report, asserts that existing design and licensing requirements for nuclear power plants preclude their operation with known fuel cladding leakage. The report recommends that the NRC take steps to prohibit nuclear power plants from operating with fuel cladding damage and specifically recommends that plants be shut down when fuel leakage is detected. The report also recommends that safety evaluations be included in plant licensing bases that consider the effects of operating with leaking fuel to justify operation under such circumstances.

Your petition stated that, because [insert facility or licensee name] was operating with known fuel damage, it is possible that significantly more radioactive material would be released to the reactor coolant system during a transient or accident than during steady-state operation; therefore, the design-basis accident analysis does not bound operation with known fuel cladding failures. In addition, the petition stated that the licensee appeared to be violating its licensing basis for worker radiation protection under the as low as is reasonably achievable (ALARA) program because industry experience has demonstrated that reactor operation with failed fuel cladding increases radiation exposure for plant workers.

The NRC has been observing the licensee's response to this issue since the licensee first received indication on March 25, 1999, of a potential leaking fuel rod on Unit 1. The licensee reviewed radiochemistry data that indicated the integrity of the fuel cladding had been compromised. Subsequent analysis revealed an increase in the dose-equivalent iodine that remained significantly below the limit allowed by technical specifications. After locating the leaking fuel assembly, the licensee suppressed the flux around the bundle by fully inserting three adjacent control rods. The staff finds the licensee's actions timely and appropriate.

Exhibit 3 (continued)

As you noted in your petition, you have previously submitted petitions on the [insert facility or licensee name] nuclear plant(s) after learning that these plants were operating with known fuel leakage. These petitions also based the requested actions on your report of April 2, 1998. The NRC responded to these petitions by a director's decision dated April 18, 1999, which is provided as an enclosure to this letter. In its decision, the staff presented its evaluation of the report which addressed the generic safety concerns for plants operating with known fuel cladding leakage. The staff concluded that operation with a limited amount of leaking fuel is within a plant's licensing basis and, in itself, does not violate ALARA-related regulations. We have compared the staff's evaluation in that director's decision against the plant-specific situation at [insert facility or licensee name] and have determined that the generic conclusions are applicable.

We have reviewed your letter of April 5, 1999, and find that the issues raised in the petition have been addressed in the director's decision dated April 18, 1999. The petition does not raise any significant new information about safety issues which were adequately addressed in the director's decision issued before and, therefore, does not meet the criteria for consideration under 10 CFR 2.206.

Thank you for bringing these issues to the NRC. I trust that this letter and the enclosed director's decision are responsive to your concerns.

Sincerely,

[Insert Division Director's Name]
[Office of [insert Division's Name]

Docket Nos. [50-, 50-]

Enclosure: Director's Decision 99-08

cc w/encl: See next page

Exhibit 4

[7590–01–P]

Sample *Federal Register* Notice for Director's Decision

U.S. NUCLEAR REGULATORY COMMISSION

Docket No(s).

License No(s).

[Name of Licensee]

NOTICE OF ISSUANCE OF DIRECTOR'S DECISION UNDER 10 CFR 2.206

Notice is hereby given that the Director, [name of office], has issued a director's decision with regard to a petition dated [insert date], filed by [insert petitioner's name], hereinafter referred to as the "petitioner." The petition concerns the operation of the [insert facility or licensee name].

The petition requested that [insert facility or licensee name] should be [insert request for enforcement action]. [If necessary, add] The petitioner also requested that a public hearing be held to discuss this matter in the Washington, DC, area.

As the basis for the [insert date] request, the petitioner raised concerns stemming from [insert petitioner's supporting basis for the request]. The [insert petitioner's name] considers such operation to be potentially unsafe and to be in violation of Federal regulations. In the petition, a number of references to [insert references] were cited that the petitioner believes prohibit operation of the facility with [insert the cause for the requested enforcement action].

The petition of [insert date] raises concerns originating from [insert summary information on more bases/rationale/discussion and supporting facts used in the disposition of the petition and the development of the director's decision].

Exhibit 4 (continued)

On [insert date], the NRC conducted a meeting regarding [insert facility or licensee name]. The meeting gave the petitioner, the licensee, and the public an opportunity to provide additional information and to clarify issues raised in the petition.

The Director of the Office of [name of office] has determined that the request(s), to require [insert facility or licensee name] to be [insert request for enforcement action], be [granted/denied]. The reasons for this decision are explained in the director's decision pursuant to 10 CFR 2.206 [Insert DD No.], the complete text of which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW. (Lower Level), Washington, DC 20555-0001, and at the local public document rooms located at the [insert the local public document room information for the licensee]. The director's decision is available via the NRC Home Page on the World Wide Web at the following address: *http://www.nrc.gov/NRC/PUBLIC/2206/index.html.*

A copy of the director's decision will be filed with the Secretary of the Commission for the Commission's review in accordance with 10 CFR 2.206 of the Commission's regulations. As provided for by this regulation, the director's decision will constitute the final action of the Commission 25 days after the date of the decision, unless the Commission, on its own motion, institutes a review of the director's decision in that time.

Dated at Rockville, Maryland, this [insert date] day of [insert month, year].

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed By

[Insert Office Director's Name]
Office of [insert Office Name]

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US NRC-OADM
BRANCH CHIEF
RULES AND DIRECTIVES BR
2WFN-6059
WASHINGTON   DC   20555

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Recd
11/18/99

To: Adensam/Clifford

ORIGINAL DUE DT: 01/18/00 TICKET NO: 019990277

FROM:
Barbara Grattan

DOC DT: 11/08/99
NRR RCVD DATE: 11/17/99

TO:

Shirley Ann Jackson

FOR SIGNATURE OF : ** YEL **

DESC:

Enclosed Certified Copy of Resolution #994 --
Supports Closure of the Entire Millstone Facility

ROUTING:

Collins/Zimmermn
ADIP
Sheron
NRR Mailroom

ASSIGNED TO: DLPM

CONTACT: Zwolinski

SPECIAL INSTRUCTIONS OR REMARKS:

000702

Evelyn Williams 415-1964

OFFICE OF THE SECRETARY
CORRESPONDENCE CONTROL TICKET

PAPER NUMBER: CRC-99-0936 LOGGING DATE: Nov 12 99

ACTION OFFICE: EDO/OGC

AUTHOR: BARBARA GRATTAN
AFFILIATION: NEW YORK

ADDRESSEE: DR. JACKSON

LETTER DATE: Nov 8 99 FILE CODE: ID&R 5 MILLSTONE

SUBJECT: ENCLOSED CERTIFIED COPY OF RESOLUTION #994 -- SUPPORTS CLOSURE OF THE ENTORE MILLSTONE FACILITY

ACTION: Appropriate

DISTRIBUTION: CHAIRMAN, COMRS,

SPECIAL HANDLING: NONE

CONSTITUENT:

NOTES:

DATE DUE:

SIGNATURE: DATE SIGNED:
AFFILIATION:

OGC-99- 004753

LEXSEE 59 fed. reg. 38889

FEDERAL REGISTER

VOL. 59, No. 146

Rules and Regulations

NUCLEAR REGULATORY COMMISSION (NRC)

10 CFR Part 73

RIN 3150-AE81

Protection Against Malevolent Use of Vehicles at Nuclear Power Plants

59 FR 38889

DATE: Monday, August 01, 1994

ACTION: Final rule.

To view the next page, type .np* TRANSMIT.
To view a specific page, transmit p* and the page number, e.g. p*1

[*38889]

SUMMARY: The Nuclear Regulatory Commission (NRC) is amending its physical protection regulations for operating nuclear power reactors. The amendments modify the design basis threat for radiological sabotage to include use of a land vehicle by adversaries for transporting personnel and their hand-carried equipment to the proximity of vital areas and to include a land vehicle bomb. The amendments also require reactor licensees to install vehicle control measures, including vehicle barrier systems, to protect against the malevolent use of a land vehicle. The Commission believes this action is prudent based on an evaluation of an intrusion incident at the Three Mile Island (TMI) nuclear power station and a bombing of the World Trade Center. The objective of this final rule is to enhance reactor safety by protecting against the use of a vehicle to gain unauthorized proximity to vital areas. Further, the amendments will enhance reactor safety by protecting vital equipment from damage by detonation of a large explosive charge at the point of vehicle denial.

EFFECTIVE DATE: August 31, 1994.

FOR FURTHER INFORMATION CONTACT: Phillip F. McKee, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC, telephone (301) 504-2933.

SUPPLEMENTARY INFORMATION:

Background

On November 4, 1993 *(58 FR 58804)*, the Commission published a proposed rule in the Federal Register for public comment that presented amendments to the physical protection requirements for operating commercial nuclear power reactors. The amendments proposed to modify the design basis threat for radiological sabotage to include use of a land vehicle by adversaries for transporting personnel, hand-carried equipment, and/or explosives. A total of 35 letters of public comment were received from respondents representing more than 160 individual comments. Comments received in association with a public meeting conducted by the NRC on May 10, 1993, on this same topic have also been analyzed as part of this final rulemaking. An additional 11 comments were received as a result of the meeting, representing an additional 38 individual comments. Written comments received from the Advisory Committee on Reactor Safeguards (ACRS) and public comments made at a February 10, 1994, meeting of the ACRS are also addressed under the following analysis. Copies of the public comments received on this proposed rule are available for inspection and copying for a fee at the NRC Public Document Room, 2120 L Street NW (Lower Level), Washington, DC.

Public Comment Analysis

General

Public comment on the rule was received from 25 licensees that operate commercial nuclear power reactors; two industry groups, the Nuclear Management and Resources Council (NUMARC) and the Nuclear Utility Backfitting and Reform Group (NUBARG); two public citizens and one citizen's group, Ohio Citizen's for Responsible Energy; two advocacy groups, the Nuclear Control Institute (NCI) and the Committee to Bridge the Gap; one State nuclear safety agency; and two vendors.

Additional comments were received as a result of an NRC-sponsored public meeting of May 10, 1993. Comments were received from eight private citizens (the letter from one enclosed a petition signed by 40 individuals); two utilities; and one public interest group, Ohio Citizens for Responsible Energy. The proposed rule indicated that comments regarding malevolent use of vehicles submitted in association with the meeting would be treated under this final rule and that duplicate comments need not be submitted. Many of these respondents recommended strengthening the design basis threat to cover the maximum credible threat and increasing the number of security force members at power reactor sites as the best method to counter a terrorist vehicle bomb attack. The aforementioned petition, submitted to the Chairman of the NRC, indicated, among other things, that Congress should strengthen safeguards at nuclear facilities and should legislate the use of Federal guards at NRC-licensed sites. Comments received from 2 utilities that operate commercial nuclear power reactors either indicated support for the then-developing NUMARC comments or were similar to comments received on the proposed rule.

A variety of general comments were received on the proposed rule and supporting documentation. Several strongly supported the rulemaking as proposed and expressed the view that rulemaking on this topic was the proper, proactive approach. A number of comments strongly supported a belief that vehicle intrusion and vehicle bomb threats exist. These comments refer to the Three Mile Island intrusion event and the World Trade Center bombing event as evidence of these threats. The NCI commented that the rule was long overdue. Some of those that supported the rule offered more detailed comments proposing further expansion of the design basis threat and placing more rigid controls on licensee actions to implement the rule.

NUMARC provided detailed comments on behalf of the industry. Fourteen utilities confirmed their support or agreement with NUMARC's comments. NUMARC commented that industry believes that it is important to deter unauthorized land vehicle penetration challenges to a licensee's protected area and that industry recognizes that facilities must be able to shut down safely in the unlikely event of the detonation of an explosive device outside the protected area. NUMARC considers these actions to be prudent for the protection of its employees, investment, and public confidence. NUMARC commented that because the NRC (as expressed in the proposed rule) [*38890] and NUMARC agree in principle, the issue should be addressed in an integrated manner using a reasonable and realistic approach without imposing unnecessary conservatism. The details of NUMARC's comments identified areas where they considered the proposed rule took too conservative an approach. NUMARC also expressed general concerns about the backfit justification for the rule and the schedule for implementation.

NUBARG, whose members include 15 nuclear utilities, provided comments that generally challenge the backfitting and regulatory analyses based on their concerns that the analyses did not provide a sufficient quantified basis for finding the requisite "substantial increase" in safety under the NRC's backfitting rule. Two of the comment letters provided by utilities confirmed their support or agreement with NUBARG's comments.

Several comments expressed the view that the proposed rule could not be substantiated based on the current threat. As support for this position, comments referred to conclusions reached by the NRC in denial of a 1991 petition for rulemaking to require licensees to protect against truck bombs. Other comments indicated that two isolated events (the Three Mile Island intrusion event and World Trade Center bombing) did not justify rulemaking, particularly in light of the fact that the Federal Bureau of Investigation (FBI), by their account, does not support the position that the threat of malevolent use of vehicles has increased and the NRC position is that no actual vehicle bomb threat against power reactors exists.

Several comments opposed the proposed rule because they considered that it did not provide a substantial increase in protection of public health and safety or common defense and security at a justifiable cost. Other comments indicated that the rule was extreme and unnecessarily burdensome with little if any safety benefit and that contingency plans for vehicle bombs currently in place adequately addressed the threat of malevolent use of vehicles.

The NRC staff presented the proposed rulemaking package to the Security Subcommittee of the Advisory Committee on Reactor Safeguards (ACRS) on November 3, 1993, and the full committee on November 4, 1993. The full committee was briefed on December 10, 1993, in a closed session, by the Director, Office of Nuclear Material Safety and Safeguards. Following these briefings, ACRS's December 10, 1993 letter to the Chairman raised concerns about the rulemaking, particularly the justification for the rule, the lack of a quantitative risk assessment to support it, and the expedited nature of the rulemaking. A minority of four members of the ACRS expressed a view that the proposed rule represents a prudent and effective step toward enhancing public health and safety. On February 10, 1994, the ACRS heard presentations on the rulemaking from the NUMARC, the NCI, one public citizen, and the NRC staff members. On April 7, 1994, the staff briefed the ACRS in a closed session regarding additional, quantitative evaluations that supported this rulemaking. Issues raised by the ACRS in their December 10, 1993, letter are encompassed by issues raised by the public and are addressed in the following responses.

Like the ACRS, NUMARC, NUBARG, and numerous utilities expressed concern that the safety benefit was not adequately justified or quantified. They challenged the validity of the regulatory and backfit analyses because of lack of quantification of the threat. They contended that the analyses contain no quantified risk data or safety goal evaluation to support the conclusion that the proposed regulations result in a substantial increase in public health and safety. Another comment, while acknowledging the potential difficulty in quantification of the threat, stated that the analyses were no more than "conclusionary" and fall short of demonstrating the requisite substantial increase in radiological safety.

The Commission notes that the use of probabilistic risk assessment (PRA) as a tool for estimating risk is sound when based on results from demonstrable, repeatable events and test data-for example, establishing the probability of failure and the mean time to failure for aircraft wing root structures due to metal fatigue or for valve failures due to water hammer or corrosion, etc. The NRC has examined the use of PRA to predict sabotage as an initiating event and concluded that to do so would not be credible or valid because terrorist attacks, by their very nature, may not be quantified. Past attempts to apply PRA techniques to acts of sabotage have resulted in similar findings. For example, in 1978, NUREG/CR-0400, the "Risk Assessment Review Group Report to the U.S. Nuclear Regulatory Commission" stated, "it was recognized that the probability of sabotage of a nuclear power plant cannot be estimated with any confidence." For this same reason, according to this report, consideration of risk of sabotage was deliberately omitted in the Reactor Safety Study (WASH-1400).

In the "Policy Statement on Safety Goals for the Operation of Nuclear Power Plants" published on March 14, 1983 *(48 FR 10772)*, the Commission stated:

The possible effects of sabotage or diversion of nuclear materials is not presently included in the safety goal. At present there is no basis on which to provide a measure of the risk of these matters. It is the Commission's intention that everything that is needed shall be done to keep such risks at their present, very low, level; and it is our expectation that efforts on this point will continue to be successful. With these exceptions it is our intent that the risk from all various initiating mechanisms be taken into account to the best of the capability of the current evaluation techniques.

In the 1983 Indian Point licensing hearings, the NRC staff testified that PRA is unable to predict the probability of sabotage as an initiating event. Also, in a June 11, 1991, petition to institute an individual plant examination program for threats beyond the design basis, the NCI stated a position similar to the NRC's by recognizing that PRA-type methods cannot be used to analyze for core damage frequency since one cannot quantify the likelihood of a terrorist attack.

The Commission continues to believe that arbitrary selection of numbers to "quantify" threat probability without demonstrable, actual, supporting event data would yield misleading results at best. Knowledgeable terrorism analysts recognize the danger and are unwilling to quantify the risk. Over the past several years, a number of National Intelligence Estimates have been produced addressing the likelihood of nuclear terrorism. The analyses and conclusions are not presented in terms of quantified probability but recognize the unpredictable nature of terrorist activity in terms of likelihood. The NRC continues to believe that, although in many cases considerations of probabilities can provide insights into the relative risk of an event, in some cases it is not possible, with current knowledge and methods, to usefully quantify the probability of a specific vulnerability threat.

The NRC notes that, although not quantified, its regulatory analysis recognizes the importance of the perception of the likelihood of an attempt to create radiological sabotage in assessing whether to redefine adequate protection. The NRC's assessment that there is no indication of an actual vehicle threat against the domestic commercial nuclear industry was an important consideration in concluding that neither the Three Mile Island intrusion nor the World Trade [*38891] Center bombing demonstrated a need to redefine adequate protection.

The NRC does not agree that quantifying the probability of an actual attack is necessary to a judgment of a substantial increase in overall protection of the public health and safety (a less stringent test of the justification for a rule change). Inherent in the NRC's current regulations is a policy decision that the threat, although not quantified, is likely in a range that warrants protection against a violent external assault as a matter of prudence.

The potential threat posed by malevolent use of vehicles as part of a violent external assault and the need to protect against it have been the subject of detailed consideration and reconsideration by the Commission for more than fifteen years. The original requirements for physical security at power reactor sites proposed in the mid-1970s included a requirement for barriers to prevent ready access to vital areas by ground vehicles. The Commission decided not to include the requirement at that time.

The Commission reexamined the vehicle issue in great detail in the 1980s. In 1986, the Commission concluded that, even though perimeter chain link fences would not prevent vehicle intrusion, the requirement for prompt response by guards armed with shoulder-fired weapons would limit actions of intruders. In reconsidering the risk from use of a vehicle to gain proximity to vital areas, the NRC's regulatory analysis does not suggest that the likelihood of a violent external assault has increased. Rather, the staff focussed its regulatory analysis on whether a vehicle could provide an advantage to an adversary with the characteristics of the design basis threat.

The NRC assessed lessons learned from the TMI intrusion and concluded that a vehicle could provide advantages to an adversary not previously considered. In SECY-86-101, "Design Basis Threat-Options for Consideration," March 31, 1986, the NRC concluded that, even though perimeter chain link fences would not prevent vehicle intrusion, the requirement for prompt response by guards armed with shoulder-fired weapons would limit actions of intruders. Accordingly, in 1986, the NRC concluded that the installation of vehicle barriers might not constitute a substantial overall increase in the protection of public health and safety. More recently, the NRC has analyzed the capability of existing licensee security measures to protect against a violent external assault that includes a vehicle as a mode of transportation. These new analyses support the NRC's conclusions in the regulatory analysis for the proposed rulemaking. The NRC believes that the vehicle intrusion issue alone warrants the installation of vehicle barriers at nuclear power plants.

In the 1980s, the NRC also consulted with other Federal agencies, including the National Security Council, regarding the use of vehicle bombs in the Middle East and their possible impact on the domestic threat situation. In June 1988, the NRC decided that it would not be necessary to change the design basis threat for radiological sabotage (10 CFR 73.1(a)(1)) nor to require long-range planning by power reactor licensees for permanent protection against land vehicle bombs. However, as a matter of prudence, it directed development of NRC and licensee contingency plans for dealing with a possible land vehicle bomb threat to power reactors, should one arise.

On June 11, 1991 *(56 FR 26782)*, the Commission denied a petition for rulemaking to revise the design basis threat to include explosive-laden vehicles (PRN-73-9). In denying that petition, the NRC noted that the decision was based, in part, on the fact that only one truck bomb attack (1970) had occurred in the United States; there had been no other vehicle bomb attacks in the Western Hemisphere; there had been none outside areas of civil unrest; and there had been none directed against a nuclear activity. The vehicle bomb attack on the World Trade Center represented a significant change to the domestic threat environment that changed many of the points used in denying the petition and eroded the basis for concluding that vehicle bombs could be excluded from any consideration of the domestic threat environment. For the first time in the United States, a conspiracy with ties to Middle East extremists clearly demonstrated the capability and motivation to organize, plan, and successfully conduct a major vehicle bomb attack. Regardless of the motivations or connections of the conspirators, it is significant that the bombing was organized within the United States and implemented with materials obtained on the open market in the United States. Accordingly, the Commission believes that the threat characterized in the final rule is appropriate.

As a result of the World Trade Center bombing, the NRC believes that the construction of a vehicle bomb is more likely to develop without advance indications. The NRC does not believe that it can quantify the likelihood of vehicle bomb attack. However, it has performed a conditional probabilistic risk analysis for an existing power reactor site, assuming an attempt to damage a nuclear power plant with a design basis vehicle bomb placed at locations within the protected area that would create the greatest risk to public health and safety. The analysis indicated that the contribution to core damage frequency could be high.

Barriers installed to protect against vehicle intrusion into protected areas would also protect, to varying degrees, against vehicle bombs. The NRC believes that adjusting the location of barriers where necessary to ensure a capability of protecting vital equipment against a design basis vehicle bomb would provide an additional, substantial increase in the overall protection of the public health and safety. Further, the NRC believes that the incremental costs to licensees to analyze the degree of protection against a vehicle bomb and to make adjustments in vehicle control measures in limited cases are justified, particularly considering the provisions in the rule allowing licensees to propose alternative measures if a site-specific analysis indicates that the costs of fully meeting the rule's design goals and criteria are not justified by the added protection that would be provided. The NRC's additional deterministic evaluations and limited probabilistic assessments have supported the NRC's earlier findings that protecting against vehicle intrusion and a vehicle bomb would substantially increase the overall protection of public health and safety. The NRC has updated the regulatory analysis to include these evaluations.

Additional issues raised and the NRC response to these issues are provided in the sections listed below that follow:

I. Threat Considerations

A. Coupling Vehicle Intrusion and Vehicle Bomb Threat

B. Characteristics of Design Basis Vehicle/Explosive

C. "Margin of Prudence"

D. Design Basis Threat Re-Evaluation

E. Applicability of 10 CFR 50.13

F. "Threat" or "Alert" Program

II. Regulatory and Backfit Analyses

A. Redundant Engineered Safeguards Systems

B. Peer Review of Analyses

C. Clarification

III. Rule Implementation

A. Schedule

B. NRC Review and Approval of Submittals

C. Vehicle Barriers

D. Passive Vehicle Barriers

E. Active Vehicle Barriers

F. Alternative Measures to Protect Against Explosives

IV. NRC Inspection

V. Miscellaneous

A. Research Reactors [*38892]

B. Independent Spent Fuel Storage Installations

C. Office of Management and Budget Supporting Statement

I. Threat Considerations

A. Coupling Vehicle Intrusion and Vehicle Bomb Threat

Comment. NUMARC and several utilities commented that the proposed rule unnecessarily linked vehicle intrusion with a vehicle bomb. NUMARC commented that the proposed rule contemplates that the intruding vehicle would be fully loaded with personnel, equipment, and a large explosive device. NUMARC also commented that any considerations of a vehicle bomb should be for a stationary vehicle. NUMARC stated that coupling the vehicle intrusion event and vehicle bomb event added unnecessary conservatism. For example, to protect against a moving vehicle, bomb barriers would, in some cases, need to be more substantial to stop penetration of vehicle. NUMARC proposed that the revised design basis threat should include either a land vehicle intrusion or a detonation of explosives outside the protected area, but not a combination of the two. Along this same line, one comment expressed the opinion that the proposed language implies the need to protect against a vehicle used for transport, not for breaching a barrier or for use as a truck bomb.

Another comment expressed a concern that a major defect in the rule is the lack of the assumption that the adversary could blast away a fence if a licensee were to choose to use, for example, cabling in the fence as the means to stop a vehicle. The respondent proposed that any barrier should be a heavy mass which would be resistant to destruction.

Response. The Commission agrees with the NUMARC comment that the proposed rule could be read to imply that licensees would be required to provide protection against an intrusion by adversaries using a vehicle for transportation coincident with a vehicle bomb. This was not the intent and the rule wording has been revised to clarify this point. Commission deliberations on the rule have considered use of the vehicle as transportation for an adversary and a vehicle bomb as separate threats to be protected against. Any coupling of adversary tactics associated with the rule was intended to allow for more efficient and cost effective protection against either a vehicle intrusion to gain rapid access to vital areas, as a single act, or against a vehicle bomb.

Meeting the requirements of the final rule will result in substantial protection from a vehicle bomb whether it is moving or stationary. The NRC's regulatory analysis indicated that, because of the short distances between vital areas and portions of some protected area boundaries, protection against a vehicle at those boundaries would be inconsistent with NUMARC's stated goal of being able to safely shut down a plant following the detonation of an explosive device outside the protected area.

Regarding the comment that the rule should include the assumption that adversaries may use devices to destroy less substantial barriers and then gain access, the Commission does not agree that this assumption should be included in the rule. The NRC assessment of the threat environment does not support this assumption. Further, use of such a technique by an adversary would tend to diminish one of the major advantages of use of a vehicle-the element of surprise.

B. Characteristics of Design Basis Vehicle/Explosive

Comment. NUMARC provided a detailed proposal for characteristics of a design basis vehicle that could be used to attempt penetration of a nuclear power plant protected area and a design basis bomb that could be used in an attempt to damage plant equipment. Other comments indicated that vehicle speed should take into consideration terrain and seasonal conditions and that the proposed vehicle explosive device size was excessive and not justified by historical experience, particularly that in the United States.

Response. The Commission notes that it has relied on analogous historical data when enumerating the attributes of a design basis threat because there has never been a terrorist attack on an NRC-licensed power reactor facility or a credible threat of an attack. This was the methodology used in formulating the original design basis threat statements in the late 1970s, and it was used in defining the proposed design basis vehicle threat. The design basis vehicle was defined after examining several hundred actual vehicle bombing attacks occurring worldwide during approximately the past decade. Historical data indicates that vehicle bombs, similar to the design basis vehicle, have been used in the past and their use can reasonably be expected to continue to occur in the future. The Commission has made some changes in the detailed characteristics of the design basis vehicle. The revised characteristics will require licensees to provide substantial protection against a moving vehicle bomb. In addition, the NRC's implementation guidance discusses how the design of barrier systems can account for site-specific limits on the speed that a vehicle could attain because of factors such as terrain.

Comment. One comment expressed confusion over reference to the design basis vehicle as a "4-wheel drive vehicle" in that this could imply that non 4-wheel drive vehicles would not have to be protected against. The comment recommended that the final rule language be changed to require protection against all land vehicles.

Response. The Commission disagrees that the term "4-wheel drive vehicle" needs clarification. It reasons that protection against intrusion by a 4-wheel drive vehicle encompasses protection against a land vehicle with less than 4-wheel drive.

Comment. Other comments noted that the regulatory language should be changed to remove reference to equipment and explosives capable of being hand-carried, as opposed to that which the vehicle could carry.

Response. As stated previously, this issue is being clarified by a revision of the design basis threat statement to separate the threat of intrusion versus vehicle bomb. In an intrusion event, the vehicle is obviously capable of transporting the equipment and explosives proposed to be hand-carried by an adversary. While the vehicle could carry more equipment than can be carried by the persons being transported, it is unlikely that this additional equipment would be of use to the adversaries. The vehicle is essentially a means of transport for the adversaries, and it is unlikely that once adversaries have left the vehicle they would be able to return to obtain additional equipment or explosives.

Comment. One utility provided specific questions regarding several assumptions associated with the vehicle bomb. These included whether:

The vehicle is under control by adversaries up to the point of detonation;

The vehicle bomb automatically detonates when the adversary loses control of the vehicle or after a pre-defined time period;

The vehicle is used in combination with a secondary external event, e.g., loss of offsite power; and,

Point of detonation, i.e., crash point or at a later point as vehicle rolls towards a facility.

Response. With respect to a vehicle bomb, for analysis purposes the device would be considered to detonate at the point where the vehicle impacted the [*38893] vehicle barrier system. Whether adversaries still have control of the vehicle or whether the detonation of the device is delayed should have little impact on the analysis of the effect of the explosive blast. Because the barrier system is intended to protect against vehicles gaining proximity to vital areas, the barrier system should not allow a vehicle to fully penetrate it and continue to roll towards a facility.

With respect to a secondary external event, power reactor licensees must protect against all capabilities and attributes described by the design basis threat for radiological sabotage. This would not include protection against other natural events, such as damage from a hurricane, coincident with a sabotage threat. However, with respect to loss of off-site power, licensees should consider its loss, if vital equipment is assumed damaged, in their analysis of the effects of a vehicle bomb. This consideration is compatible with the basic premise that equipment not designated and protected as vital is vulnerable to damage and is not available.

C. "Margin of Prudence"

Comment. NUMARC and several utilities commented on NRC's use of the term "margin of prudence" as the basis for support of the proposed rulemaking. NUMARC commented that it is inappropriate to use such an undefined concept as a basis for rulemaking. These comments indicated that NRC expansion into matters of prudence is unwarranted and would result in expansion of the NRC's sphere of regulatory influence beyond plant safety.

Response. Use of the term "margin of prudence" must be put in perspective as used by the NRC in this rulemaking. The NRC requires an established level of security at nuclear power reactor sites as a provision against possible security contingencies that might arise. The NRC has concluded that a satisfactory level of security is one that is designed and implemented to protect against a hypothetical threat (design basis threat) that contains certain adversary attributes. These attributes have been selected based on Commission analyses of actual terrorist attributes and on judgment. The term "margin of prudence" was used in recent Commission deliberations to suggest that the World Trade Center bombing and the Three Mile Island intrusion had caused a change in the domestic threat environment or in the NRC's understanding of the sabotage threat that was not satisfactorily addressed by the existing design basis threat. Further, the term was used to suggest that a modification of the design basis threat was necessary to reestablish a level of security commensurate with the nature of security contingencies that might arise. Its use was illustrative only of the relationship between an actual threat and the hypothetical design basis threat and the change in that relationship caused by the World Trade Center and Three Mile Island events. The NRC intended no wider or expanded use of the term.

D. Design Basis Threat Re-Evaluation

Comment. NUMARC and several utilities commented that the revision to the design basis threat to address malevolent use of vehicles should be addressed in an integrated manner so that rulemaking on this topic would not be impacted after completion of an ongoing, more comprehensive review of the design basis threat. Other comments expressed concerns about deficiencies in the design basis threat that need to be addressed. Deficiencies identified by these comments included: protection against more than one insider, protection against a larger number of external attackers, capability of attackers to operate as more than one team, and use of aquatic vehicles. One comment was made that ongoing considerations for reductions in the insider requirements should be part of the overall reconsideration of the design basis threat.

Response. The Commission notes that use of a vehicle by adversaries was addressed under Phase I of a re-evaluation of the design basis threat which the NRC began in the Spring of 1993. This phase of the re-evaluation has been completed. Other attributes associated with the design basis threat, such as those characterized in comments on the proposed rule, have been reviewed and considered as part of Phase II of the re-evaluation. NRC staff recommendations on this part of the re-evaluation were provided to the Commission in a classified paper on March 15, 1994.

E. Applicability of 10 CFR 50.13

Comment. NUMARC, NUBARG, and several utilities stated that the proposed change in the design basis threat to include malevolent use of a vehicle amounts to escalation of the threat to efforts by an enemy of the United States. The comments contended that the proposed changes to the design basis threat are, therefore, in conflict with 10 CFR 50.13, which specifies that licensees are not required to provide for design features to protect against attacks and destructive acts by an enemy of the United States. One comment recommended that NRC should re-evaluate the design basis threat assumption to now include foreign enemies of the United States.

Response. In 10 CFR 50.13, which was promulgated on September 26, 1967 *(32 FR 13445),* the regulations provide that applicants for construction permits, operating licenses, or amendments thereto, need not provide for design features or other measures to protect against the attacks or destructive acts, including sabotage, by an enemy of the United States. The issue raised in a contested application for a power reactor construction permit, which led to the promulgation of 10 CFR 50.13, was whether the reactor should be constructed to withstand a missile attack from Cuba. There is a significant difference in the practicality of defending against a missile attack and constructing a vehicle barrier at a safe standoff distance from vital areas.

The statement of considerations for 10 CFR 50.13 makes it clear that the scope of that regulation is to relieve applicants of the need to provide protective measures that are the assigned responsibility of the nation's defense establishment. The Atomic Energy Commission recognized that it was not practical for the licensees of civilian nuclear power reactors to provide design features that could protect against the full range of the modern arsenal of weapons. The statement concluded with the observation that assessing whether another nation would use force against a nuclear power plant was speculative in the extreme and, in any case, would involve the use of sensitive information regarding both the capabilities of the United States' defense establishment and diplomatic relations.

The new rule, with its addition to the design basis threat and added performance requirements, is in response to a clearly demonstrated domestic capability for acts of extreme violence directed at civilian structures. The participation or sponsorship of a foreign state in the use of an explosives-laden vehicle is not necessary. The vehicle, explosives, and know-how are all readily available in a purely domestic context. It is simply not the case that a vehicle bomb attack on a nuclear power plant would almost certainly represent an attack by an enemy of the United States, within the meaning of that phrase in 10 CFR 50.13.

Further, characterizing the threat as "para-military" adds little to the understanding of the intent of 10 CFR 50.13. "Para-military" suggests an armed, trained group acting outside of a legally constituted military [*38894] organization. In that sense, the design basis threat prior to this amendment already described a "para-military" group. "Para-military" groups of entirely domestic origin exist. Accordingly, the amended regulation and supporting analyses need not address 10 CFR 50.13, either on the grounds that a vehicle bomb attack is an attack by an enemy of the United States or the action of a "paramilitary" group. That regulation is irrelevant to the present rulemaking.

The implication of the comments regarding 10 CFR 50.13 is that the simple addition of a vehicle bomb to the design basis threat should shift the function of providing physical security for nuclear power plants from the licensee to the Federal Government. The respondents present no real evidence or persuasive arguments for such a radical change in the regulatory environment.

F. "Threat" or "Alert" Program

Comment. One comment suggested that the NRC develop and implement a "threat or alert" program similar to the Department of Defense's Defense Condition "DEFCON" program. It was recommended that, under such a program, the NRC would immediately notify the industry when information is received from the intelligence community of an impending security alert and provide a recommended level of action. Licensees, in turn, would be required to develop security response plans based on NRC-established threat levels.

Response. The Commission believes that its current Information Assessment Team approach for notifying licensees of significant events has been effective in disseminating and coordinating such information. The Information Assessment Team (IAT) assesses in a timely manner reported threats to NRC-licensed facilities, materials, and activities to determine credibility and make recommendations to NRC management. The IAT is composed of experienced Headquarter's and Regional staff who are on-call 24 hours a day and bring a variety of expertise to the assessment process, such as reactor systems, site specific information, and liaison with other Federal agencies, including close coordination with the Department of Energy on threat advisories to the utility industry and NRC licensees. The IAT was established in 1976, and since that time has supported NRC decision makers responding to a range of threats, from bomb threats against reactors to times of international tension during Operation Desert Shield and Storm. For example, coordinated threat advisories related to the latter were issued by the IAT on August 24, 1990, January 9, 1991, and April 2, 1991. However, the NRC does not believe that the IAT is an adequate alternative to vehicles barriers at nuclear power plants.

II. Regulatory and Backfit Analysis

A. Redundant Engineered Safeguards Systems

Comment. One comment indicated that the proposed rule did not adequately take into consideration the existing engineered safeguards systems installed at nuclear power plants. The comment was made that unauthorized access and possible damage to any one vital area does not necessarily prevent the safe shut down of the nuclear reactor.

Response. The Commission agrees that consideration should be given to engineered safeguards systems and believes that flexibility has been built into the rule to allow for consideration of such existing systems. The redundancy and diversity of existing engineered safeguards systems was considered in the NRC analysis of the capability of existing licensee security measures to protect against a violent external assault that includes a vehicle as a mode of transportation. Specific plant equipment layout can be a factor in protective considerations against a vehicle bomb. Equipment that is redundant or provides backup to equipment assumed to be damaged by a vehicle bomb may be considered in the analysis for determining whether protective measures established to protect against vehicle intrusion fully meet the design goals and criteria for protection against a land vehicle bomb.

B. Peer Review of Analysis

Comment. One comment recommended that any research results, risk analyses, cost calculations and other work by the NRC should be subject to peer review.

Response. The NRC believes that its work is subject to various types of review and, in a sense, is subject to peer review. Portions of the risk analyses were conducted by groups with appropriate expertise, including threat assessment, physical security system performance evaluation, critical target set analysis, safety system inspections, probabilistic risk analysis, vehicle barrier design, and vehicle bomb analysis. In addition, the types of efforts mentioned by the comment are often the subject of multiple office review within the NRC. Several technical review groups, both within and external to the NRC, provide further consideration of NRC staff work. Finally, with respect to rulemaking, analyses are the subject of public comment.

C. Clarification

Comment. One comment noted that the wording associated with the backfit analysis in the proposed Federal Register notice did not precisely coincide with that found under 10 CFR 50.109 (a)(3).

Response. The Commission notes that the wording in the notice is wording that is used for most NRC rules that are subject to backfitting. The Commission considers that this wording is consistent with the requirement cited.

III. Rule Implementation

A. Schedule

Comment. A large number of comments were received on the schedules associated with the proposed rule. Some indicated that the proposed schedule to submit a summary description of the barrier system and results of vehicle bomb comparison within 90 days was not long enough. One comment was received supporting the proposed schedule. Those commenting that the schedule was too tight expressed concern that 90 days did not provide sufficient time to perform a thorough design analysis, particularly if alternative measures were to be proposed. NUMARC, and several other respondents, recommended that licensees be provided 180 days after issuance of the rule to provide a summary description of the barrier system.

A number of comments were also received stating that the proposed schedule to confirm implementation within 360 days after issuance of the rule was not long enough. Those commenting that the schedule for completion of installation was too tight expressed concern that the schedule did not adequately account for material procurement and availability, outage schedules, and weather circumstances. NUMARC and several other respondents recommended that licensees be provided 18 months after issuance of the rule to complete installation of measures to meet the rule. A few comments were received that recommended that implementation schedules be established on a case-by-case basis.

Response. The Commission agrees that an extension to the schedule is reasonable based on the fact that this is a new program for power reactor sites, that there may be some difficulty in procurement of active vehicle barrier systems, and that possible deleterious [*38895] effects on scheduling may result from the weather or planned outages. Accordingly, the time period for submission of the summary required by 10 CFR 73.55(c)(9)(i) is extended from 90 to 180 days from the effective date of the rule. The implementation period required under 10 CFR 73.55 (c)(9)(ii) is extended from 360 days to 18 months from the rule's effective date.

B. NRC Review and Approval of Submittals

Comment. Three comments recommended that the NRC should review and approve all licensee submittals, including the summary description of the proposed measures to protect against vehicle intrusion, the results of the vehicle bomb comparison, and, for applicable licensees, alternative measures to protect against an explosive device.

Response. The NRC believes that approval of all summaries submitted under 10 CFR 73.55(c)(9)(i) would unnecessarily delay expeditious implementation of this rule. All licensees are required to amend their physical security plans to commit to the implementation and use of the vehicle barrier system described by the regulations. These commitments are fully inspectable and enforceable by the NRC. The NRC would review and approve the limited number of requests expected to use alternative measures that might not fully meet the design goals and criteria for protection against a vehicle bomb. The final rule has been changed to clarify that proposals for alternative measures be submitted in accordance with the provisions of 10 CFR 50.90.

C. Vehicle Barriers

Comment. NUMARC and several other respondents expressed concern that barrier systems would be required to be "nuclear grade" and that this would unnecessarily escalate costs. Another comment expressed the opinion that, instead of licensees certifying to the NRC that vehicle barriers meet requirements, they be able to choose barriers from some pre-approved list. NUMARC commented that design and certification needed to utilize existing technology and barrier device test results, or costs would unnecessarily escalate. NUMARC also requested that the discussion in the Regulatory Guide be expanded to describe flexibility available to licensees in designing and installing barriers.

Response. The NRC is unaware of any requirement for "nuclear grade equipment" and notes that the expression does not appear in the proposed rule or supporting guidance. The NRC agrees with the industry comment that commercially available materials suffice for the construction of the vehicle barrier if the barrier is capable of countering the design basis vehicle threat. As suggested by many respondents, the NRC recommends that affected licensees take advantage of available information on vehicle barrier testing, much of which has been conducted by Federal laboratories and agencies.

With respect to the use of "pre-approved barriers," the Commission believes that most vendors of commercial vehicle barrier systems know what the "stopping powers" of their barriers are. Licensees should use this as a resource in determining what barrier can counter the attributes of the Commission's design basis vehicle most cost effectively. In addition, the NRC has provided information on performance levels of several types of barriers to affected licensees. The Commission agrees with the NUMARC comment concerning expansion of the discussion on the flexibility of designing and installing barriers in the regulatory guide supporting the rule. The regulatory guide now reflects this.

Comment. NUMARC expressed the view that compensatory measures, not explicitly addressed in the proposed rule or regulatory guide, for maintenance or repair of barriers should be determined by the licensee. Another comment stated that compensatory measures required if a barrier is temporarily inoperable, as with maintenance, need to be addressed at an early stage.

Response. The NRC anticipates that vehicle barriers, particularly passive barriers, will infrequently become non-functional once installed. For those infrequent cases, any compensatory measures should take into consideration the type and cause of the problem and the time the barrier will be non-functional. For example, for short term problems with active or passive barriers, compensatory measures would not be expected to be extensive. In cases where barriers are non-functional for longer periods, compensatory measures may include placement of heavy vehicular equipment, concrete highway median barriers arranged in a serpentine fashion, installation of strands of airplane arresting wires, or the positioning of an officer armed with a high power contingency weapon may be appropriate. The regulatory guide issued in support of this rulemaking has been revised to include guidance regarding compensatory measures.

D. Passive Vehicle Barriers

Comment. One comment was directed at the guidance that specified measures should be established to periodically verify the integrity of passive barriers outside the protected area. It was commented that passive barriers by their nature (ditches, berms, concrete filled embedded poles, etc.) do not require inspection, or if so, the period for inspecting should be on the order of several years. If licensees were to install a unique passive barrier that should need periodic inspection, it should be addressed on a case-by-case basis.

Response. The Commission agrees that the components of many passive barrier systems do not need to be inspected on a weekly or monthly basis due to the nature of their construction. Observations by routine security patrols should be sufficient to detect any degradation in the barrier. Some types of barriers may be more susceptible to deterioration, damage, or tampering and therefore should be subject to more frequent observation by security patrols or, in some cases, periodic inspection. Given the large variation in components of passive barriers, the Commission considers it appropriate to provide licensees with flexibility on how to assure the continued integrity of barrier components. If the barrier system is damaged, the Commission expects that such damage would be identified in a reasonable period and actions would be taken promptly to repair the damage.

E. Active Vehicle Barriers

Comment. Two comments were received requesting that the wording in the proposed regulatory guidance clarify that only one active barrier is needed to deny access. Also, one utility commented that the provision in the regulatory guide that specified vehicles and their operators be authorized for entry before being permitted access inside the vehicle barrier system would preclude their current practice of searching the vehicle after entry inside the active barrier.

Response. The NRC agrees with these comments and the guidance in the regulatory guide supporting the rule has been changed.

Comment. Another comment recommended that specific kinetic energy be identified for use in design of active barriers with documented performance satisfying specific energy requirements because this approach would help avoid costly independent testing to demonstrate performance.

Response. Guidance previously forwarded to licensees, designated as Safeguards Information, defines the [*38896] kinetic energy associated with the design basis vehicle. As previously stated, the NRC has provided information to affected licensees on performance levels of several types of barriers to help avoid costly independent testing.

F. Alternative Measures to Protect Against Explosives

Comment. One comment objected to the rule's provisions that would allow some licensees to provide only "substantial protection" and not equivalent protection to fully meet the Commission's design goals and criteria for protection against a vehicle bomb. One comment indicated that the NRC should not be considering costs in determining the acceptability of alternative measures because costs should not be considered relative to enforcing adequate protection. NUMARC commented that it was reasonable for licensees to have the option to propose alternative measures for Commission review when the design goals and criteria for protection against a vehicle bomb cannot be met without a significant resource burden.

Response. The NRC's regulatory analysis concluded that neither the Three Mile Island or World Trade Center events demonstrated a need to redefine adequate protection. The NRC's basis for the backfit being implemented by this rulemaking was a determination that it would result in a substantial increase in protection of the public health and safety. Paragraph 50.109(a)(3) of Title 10, Code of Federal Regulations, authorizes such a backfit only if the costs of implementation are justified in view of the increased protection. The NRC concluded that the estimated costs for all licensees to provide barriers to protect against vehicle intrusion were justified. However, at some sites, the location of barriers to protect against vehicle intrusion could provide substantial protection against a vehicle bomb without fully meeting the NRC's design goals and criteria for protection against an explosive device. For these licensees, the incremental costs for placing barriers further from vital areas or for providing additional protective measures to fully meet the design goal and criteria may not be justified by the incremental protection beyond the substantial level.

Comment. NUMARC objected to the provision that licensees proposing alternative measures must compare their costs with the costs of measures needed to fully meet the design goals and criteria for protection against a vehicle bomb and must provide an assessment supporting a finding that the additional costs are not justified by the added protection that would be provided. NUMARC asserted that the NRC was requiring licensees to perform analyses beyond what the NRC staff has done in support of the proposed rule.

NUBARG similarly asserted that the NRC was requiring licensees to prove that alternative measures substantially increase safety, which is unfair. NUBARG asserts that this requires licensees to perform a backfit analysis on why they should not install a proposed modification (one that would fully meet the design goals and criteria) and that this runs counter to the backfit principle of the NRC providing the analysis.

Several respondents stated that they understood that the rule and regulatory guidance specified that those licensees proposing alternative measures would need to submit to the NRC a quantitative analysis to justify that the cost of plant specific measures are not justified by the added protection afforded. The comments indicated that, based on this understanding, such a task would be difficult, if not impossible.

A public interest group expressed the opinion that contingency planning as part of alternative measures is unacceptable when compared to a permanent vehicle control system.

Response. The optional licensee analysis provided for in the revised regulations is intended to be similar in approach to that performed by the NRC in the development of the regulatory analysis for the rulemaking. The Commission recognizes the difficulties with respect to quantification of the protection provided (see general discussion) and would expect licensees to provide a more deterministic analysis in comparing the relative protection provided by alternative measures taken by the licensee that don't fully meet the Commission design goal and criteria for protection against a vehicle bomb. The Commission did not intend to require its licensees to do more of an analysis or a different type of analysis than that performed by the NRC. The quantitative aspects of the analysis required by the regulation only apply to cost considerations, particularly the comparison of costs needed to fully meet the Commission's design goals and criteria for protection against a vehicle bomb with the cost of alternative measures.

The comment that contingency planning would be an unacceptable alternative to permanent vehicle barriers does not recognize the provision in the rule that specifies that all licensees are required to establish a vehicle barrier system to protect against use of a land vehicle as a means of transportation to gain unauthorized proximity to vital areas. Licensees may not substitute contingency plans for vehicle barriers. Rather, contingency plans were identified as one possible option for licensees (those few where it may be practical for them to propose alternative measures to protect against explosives) to supplement protection provided by the licensee's vehicle barrier system for protection against a vehicle bomb.

IV. NRC Inspection

Comment. One comment indicated that the NRC should establish procedures to assure licensee compliance with the rule.

Response. The NRC plans to inspect licensee implementation of the rule as part of the ongoing reactor inspection program. Most likely the inspection will be accomplished using a temporary inspection procedure, which is planned to be prepared after publication of the rule but before the required implementation date.

As previously stated, all affected licensees are required to amend their physical security plans in response to this rule. All commitments in physical security plans are fully inspectable and enforceable by the NRC.

V. Miscellaneous

A. Research Reactors

Comment. One comment recommended that, in light of the upcoming 1996 Olympics, all reactor fuel, heavy water, and kilocuries of Co and Cs be removed immediately from the Georgia Tech campus.

Response. While research reactors do not fall within the scope of this rulemaking, the Commission notes that its threat assessment activities are performed on a continuing basis, in close liaison with the intelligence community. Should the level of domestic threat change at any time, appropriate action will be taken by the NRC. Specifically, the Atlanta Field Office of the FBI has established liaison with all Federal agencies in Georgia, including the NRC, relative to the Olympics. The FBI is the lead law enforcement agency in charge of the Olympics and, to date, has not indicated that there is any threat to NRC-licensed facilities or materials relative to the Olympics.

B. Independent Spent Fuel Storage Installations

Comment. NUMARC commented that independent spent fuel storage installations (ISFSIs) should be clearly exempted from the rule. [*38897]

Response. The NRC did not intend for ISFSIs to be subject to this regulation because of the lower consequences associated with storage of irradiated fuel removed from a power reactor core, particularly since spent fuel stored at ISFSIs must be aged for at least one year. The NRC is currently preparing a proposed rule to clarify physical protection requirements for ISFSIs. The lessons learned from the TMI intrusion will be considered in that rulemaking. In addition, the NRC is attempting to quantify the consequences of a vehicle bomb detonated in the vicinity of an ISFSI. The results of this study will assist in making a determination as to whether vehicle bomb protection is needed at ISFSIs. In the interim, the staff believes that the inherent nature of the fuel, along with the degree of protection provided by the approved storage means for spent fuel, provides adequate protection against a vehicle bomb.

C. Office of Management and Budget Supporting Statement

Comment. One comment identified that the NRC-estimated financial burden to licensees did not include capital costs for modifications.

Response. The NRC notes that the financial burden cited by the comment was derived from the Office of Management and Budget Supporting Statement, required under the Paperwork Reduction Act. This statement deals solely with the licensee recordkeeping and reporting burden resulting from the new rule, i.e., the paperwork burden. Actual construction costs are considered in the regulatory analysis that supports the rule.

Summary of Changes Made to Rule

The following changes have been made as a result of public comment analysis:

1. The design basis threat statement for radiological sabotage has been clarified to separate the threat of a land vehicle used for intrusion with that of a land vehicle used as a vehicle bomb.

2. ISFSIs have been specifically exempted from the rule.

3. Clarification of what is meant by "the Commission's design goals and criteria" has been added to the regulatory text.

4. The appropriate means for submitting alternative measures has been clarified under 10 CFR 73.55(c)(9)(i) by adding the phrase "in accordance with 10 CFR 50.90."

5. Summary and implementation schedules have been revised-from 90 to 180 days for summary submittals, and from 360 to 540 days (18 months) for completion of implementation. Both time periods are from the effective date of the rule which is 1 month from the date of publication in the Federal Register.

Availability of Supporting Guidance

Two guidance documents are being developed by the NRC in support of this rule and are expected to be distributed to affected licensees before the effective date of the rule. These documents are: (1) Regulatory Guide 5.68, "Protection Against Malevolent Use of Vehicles at Nuclear Power Plants" and (2) NUREG/CR 6190, "Protection Against Malevolent Use of Vehicles at Nuclear Power Plants."

Regulatory Guide 5.68 will be available for inspection and copying for a fee at the Commission's Public Document Room, 2120 L Street, NW. (Lower Level), Washington, DC. Copies of issued guides may be purchased from the Government Printing Office at the current GPO price. Information on current GPO prices may be obtained by contacting the Superintendent of Documents, U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20013-2171. Issued guides may also be purchased from the National Technical Information Service on a standing order basis. Details on this service may be obtained by writing NTIS, 5825 Port Royal Road, Springfield, VA 22161.

Copies of NUREG/CR-6190 may be purchased from the Superintendent of Documents, U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20013-7082. Copies also will be available from the National Technical Information Service, 5285 Port Royal Road, Springfield, VA 22161. A copy also will be available for inspection and copying for a fee in the NRC Public Document Room, 2120 L Street, NW. (Lower Level), Washington, DC.

Electronic Submittals

Required paperwork may be submitted, in addition to an original paper copy, in electronic format on a DOS-formatted (IBM compatible) 5.25 or 3.5 inch computer diskette. Text files should be provided in WordPerfect format or unformatted ASCII code. The format and version should be identified on the diskette's external label.

Finding of No Significant Environmental Impact: Availability

The Commission has determined under the National Environmental Policy Act of 1969, as amended, and the Commission's regulations in Subpart A of 10 CFR Part 51, that this rule is not a major Federal action significantly affecting the quality of the human environment and, therefore, an environmental impact statement is not required. The rule involves installation of vehicle barriers at operating power reactor sites and an evaluation of these barriers by the licensee to determine whether they provide acceptable protection against a land vehicle bomb under design goals and criteria established by the Commission.

Implementation of these amendments will not involve release of or exposure to radioactivity from the site. Construction activities associated with passive vehicle barriers will involve some earth movement, either for excavation or development of berms, and possible destruction of trees and shrubbery. Since most active vehicle barriers are hydraulically operated, there may on occasion be leakage of this fluid to the environment. The activities required to implement these amendments involve no significant environmental impact.

The environmental assessment and finding of no significant impact on which this determination is based are available for inspection at the NRC Public Document Room, 2120 L Street NW. (Lower Level), Washington, DC. Single copies of the environmental assessment and finding of no significant impact are available from: Carrie Brown, U.S. Nuclear Regulatory Commission, Washington, DC, telephone (301) 504-2382.

Paperwork Reduction Act Statement

This final rule amends information collection requirements that are subject to the Paperwork Reduction Act of 1980 (*44 U.S.C. 3501* et seq.). These requirements were approved by the Office of Management and Budget approval number 3150-0002.

The public reporting burden for this collection of information is estimated to average 500 hours per response, including the time for reviewing instructions, searching existing data sources, gathering and maintaining the data needed, and completing and reviewing the collection of information. Send comments regarding this burden estimate or any other aspect of this collection of information, including suggestions for reducing the burden, to the Information and Records Management Branch (MNBB-7714), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-3019, (3150-0002), Office of Management and Budget, Washington, DC 20503. [*38898]

Regulatory Analysis

The Commission has prepared a regulatory analysis on this regulation. The analysis examines the costs and benefits of the alternatives considered by the Commission. Interested persons may examine a copy of the regulatory analysis at the NRC Public Document Room, 2120 L Street NW. (Lower Level), Washington, DC. Single copies of the analysis may be obtained from Robert J. Dube, U.S. Nuclear Regulatory Commission, Washington, DC 20555, telephone (301) 504-2912.

Regulatory Flexibility Certification

As required by the Regulatory Flexibility Act of 1980, *5 U.S.C. 605*(b), the Commission certifies that this final rule does not have a significant economic impact on a substantial number of small entities. The rule affects only licensees authorized to operate a nuclear power reactor. The utilities that operate these nuclear power reactors do not fall within the scope of the definition "small entities" as given in the Regulatory Flexibility Act or the Small Business Size Standards promulgated in regulations issued by the Small Business Administration (13 CFR Part 121).

Backfit Analysis

As required by 10 CFR 50.109, the Commission has completed a backfit analysis for the final rule. The Commission has determined, based on this analysis, that backfitting to comply with the requirements of this final rule provides a substantial increase in protection to public health and safety or the common defense and security at a cost which is justified by the substantial increase. The backfit analysis on which this determination is based reads as follows.

I. Statement of the specific objectives that the proposed action is designated to achieve.

To publish a rule in response to direction from the Commission in a staff requirements memorandum dated June 29, 1993. The Commissioners' decision to proceed with expedited rulemaking was the result of two events. On February 7, 1993, there was a forced vehicle entry into the protected area (PA) at Three Mile Island (TMI) Unit 1. On February 25, 1993, a van bomb, containing between 500 and 1,500 pounds of TNT equivalent, was detonated at the World Trade Center in New York City.

In its subsequent review of the threat environment, the NRC staff concluded that there is no indication of an actual vehicle threat against the domestic commercial nuclear industry. Nonetheless, in light of the vehicle intrusion at TMI and the World Trade Center vehicle bombing, the NRC staff concluded that a vehicle intrusion or bomb threat to a nuclear power plant could develop without warning in the future. The objective of the rulemaking is to enhance reactor safety by maintaining a prudent margin between what is the current threat estimate (low) and the design basis threat for radiological sabotage specified in 10 CFR 73.1(a) (higher).

II. General description of the activity that would be required by the licensee or applicant in order to complete the proposed action.

The rule requires each licensee authorized to operate a nuclear power plant to establish vehicle control measures to protect against the use of a design basis land vehicle as a means of transportation to gain unauthorized proximity to vital areas. This provides two benefits. First, it enhances a licensee's ability to interdict an adversary attempting to use a vehicle as an aid to reach critical safety equipment. Second, it provides protection against a land vehicle bomb.

The rule requires licensees to evaluate the effectiveness of their vehicle control measures with respect to the protection they provide against a land vehicle bomb. Licensees are required to confirm to the Commission that the vehicle control measures to protect against vehicle intrusion, alone or in combination with additional measures, fully meet the Commission's design goals and criteria for protection against a vehicle bomb. Licensees that can show that the additional costs for measures required to fully meet the Commission's design goals and criteria for protection against a vehicle bomb are not justified by the added protection that would be provided have the option to propose alternative measures to the Commission. These licensees will not be relieved of the requirement to protect the facility against vehicle intrusion.

Licensees that propose alternative measures are required to describe the level of protection that these measures would provide against a land vehicle bomb and compare the costs of the alternative measures with the costs of measures necessary to fully meet the criteria. The NRC will approve the alternative measures if the measures provide substantial protection against a land vehicle bomb and if the licensee demonstrates by an analysis, using the essential elements of the criteria in 10 CFR 50.109, that the costs of fully meeting measures needed to protect against a vehicle bomb are not justified by the added protection provided.

III. Potential change in the risk to the public from the accidental offsite release of radioactive material.

The potential change in the risk to the public from the accidental offsite release of radioactive material is discussed in detail in pages 4 through 7 and 10 through 14 of the regulatory analysis that supports the rulemaking. Failure to protect against attempted radiological sabotage could result in reactor core damage and large radiological releases. Based on its assessment, the NRC concludes that amending its regulations to protect against malevolent use of a vehicle against a nuclear power plant provides a substantial increase in overall protection of the public health and safety.

In summary, the TMI event demonstrated some aspects regarding use of a vehicle by a potential adversary that could present some challenges not previously considered by staff and licensees. The NRC considers that providing vehicle intrusion protection provides substantial enhancement against such a threat. Enhancements to protect against the vehicle intrusion threat also provide, to varying degrees dependent on site characteristics, enhancement for protection against vehicle bombs.

The World Trade Center event demonstrated a capability within the United States to construct a truck bomb undetected. This recently demonstrated capability indicates that although a vehicle bomb attack at a nuclear power plant is not reasonably to be expected, it is somewhat more likely to develop without advance indications than the NRC previously believed. Therefore, the NRC considers that providing permanently installed vehicle bomb protection provides substantial enhancement against such a threat.

IV. Potential impact on radiological exposure of facility employees and other onsite workers.

By enhancing protection against the malevolent use of a vehicle, the rule decreases the potential for radiological exposure of facility employees and other onsite workers. Although the threat of a determined, violent attack at a nuclear power plant is considered to be low, the rule also decreases the risk that onsite workers could be injured by weapons fire or an explosion.

V. Installation and continuing costs associated with the action, including the cost of facility downtime or the cost of construction delay.

Estimates of installation costs are discussed in detail on pages 7 through 10 and 14 of the regulatory analysis. [*38899] Ranges in cost estimates for three vehicle types illustrate the strong influence of vehicle characteristics. In addition, site-specific characteristics influence costs, including the need at some sites to extend the vehicle exclusion area beyond portions of the current PA boundary or providing a more substantial passive barrier.

The NRC staff estimates that about 80 to 90 percent of the sites will provide safe standoff distances against a vehicle bomb by providing a vehicle barrier in proximity to the present PA boundary. For these sites, cost estimates range from $ 290K for protecting the smallest protected area against a passenger vehicle to $ 2,955K for protecting the largest protected area against a large truck. (The characteristics of the design basis vehicle used to establish protection goals are described in a Safeguards Information document provided separately to affected licensees.) For the remaining 10 to 20 percent of the sites, cost estimates range from $ 440K to $ 3,655K.

An important consideration in assessing costs for the 10 to 20 percent of the sites that may have to protect beyond the existing protected areas is that the only definitive requirement for all licensees is that they provide measures to protect against the use of a land vehicle as a means of transportation to gain proximity to vital areas and that they assess any incremental measures, if necessary, to meet the design goal for a land vehicle bomb. The NRC will accept alternative measures if the measures provide substantial protection against a land vehicle bomb and if the licensee demonstrates by an analysis, using the essential elements of the criteria in 10 CFR 50.109, that the costs of fully meeting measures needed to protect against a vehicle bomb are not justified by the added protection provided.

Continuing costs to maintain barriers should be small. Implementation of the rule will not require facility downtime or construction delay.

VI. The potential safety impact of changes in plant or operational complexity, including the relationship to proposed and existing regulatory requirements and NRC staff positions.

There should be no adverse safety impact from the rule. Construction of barriers will be near or beyond existing protected area perimeters and should not delay authorized access to the protected area.

VII. The estimated resource burden on the NRC associated with the action and the availability of such resources.

There should be no new resource burden on the NRC. There will be no NRC staff licensing review of licensees' vehicle control measures before implementation. Licensees will be required to retain their analyses on site for NRC staff review during routine inspections. Inspection of the approximately 67 total sites for explosive protection will be about 1 FTE. Reviewing licensee proposals for alternative measures and 10 CFR 50.109 type analyses will require approximately 1 FTE and 40K of technical assistance from the United States Army Corps of Engineers.

VIII. The potential impact of differences in facility type, design, or age on the relevancy and practicality of the proposed action.

The action is relevant for all nuclear power reactors. The action should also be practical at most sites. If a barrier stopped a vehicle at the PA perimeter with little or no further penetration, about 90 percent of the sites would provide significant protection against the design basis vehicle bomb.

In those cases where licensees determine additional security measures may be needed to protect safe shutdown capability, the rule permits licensees to either implement the additional security measures or develop alternative protection strategies. The licensee may propose alternative measures if the measures provide substantial protection against a land vehicle bomb and if they demonstrate by an analysis, using the essential elements of the criteria in 10 CFR 50.109, that the costs of fully meeting measures needed to protect against a vehicle bomb are not justified by the added protection provided. The NRC staff will review licensee's alternative proposals and make an acceptability determination. The Commission will be notified of such NRC staff action.

NRC staff's analysis also indicates that there is a high likelihood that all sites will be capable of achieving and maintaining safe shutdown if a design basis bomb were detonated at any land accessible location of a nuclear power plant outside of the owner controlled area.

IX. Whether the proposed action is interim or final, and if interim, the justification for imposing the proposed action on an interim basis.

The action is to promulgate a final rule. The rulemaking does not involve interim actions.

List of Subjects in 10 CFR Part 73

. Criminal penalties, Hazardous materials transportation, Nuclear materials, Nuclear power plants and reactors, Reporting and recordkeeping requirements, Security measures.

For the reasons set out in the preamble and under the authority of the Atomic Energy Act, as amended, and *5 U.S.C. 552* and 553, the NRC is adopting the following amendments to 10 CFR Part 73.

PART 73-PHYSICAL PROTECTION OF PLANTS AND MATERIALS

1. The authority citation for Part 73 continues to read as follows:

Authority: Secs. 53, 161, 68 Stat. 930, 948, as amended, sec.147, 94 Stat. 780 *(42 U.S.C. 2073,* 2167, 2201); sec. 201, as amended, 204, 88 Stat. 1242, as amended, 1245 *(42 U.S.C. 5841,* 5844).

Section 73.1 also issued under secs. 135, 141, Pub. L. 97-425, 96 Stat. 2232, 2241, *(42 U.S.C. 10155,* 10161). Section 73.37(f) also issued under sec. 301, Pub. L. 96-295, 94 Stat. 789 *(42 U.S.C. 5841* note). Section 73.57 is issued under sec. 606, Pub. L. 99-399, 100 Stat. 876 *(42 U.S.C. 2169).*

2. In § 73.1, the introductory text of paragraph (a) and the text of (a)(1)(ii) are revised and new paragraphs (a)(1)(i)(E) and (a)(1)(iii) are added to read as follow:

§ 73.1 -- Purpose and scope.

(a) *Purpose.* This part prescribes requirements for the establishment and maintenance of a physical protection system which will have capabilities for the protection of special nuclear material at fixed sites and in transit and of plants in which special nuclear material is used. The following design basis threats, where referenced in ensuing sections of this part, shall be used to design safeguards systems to protect against acts of radiological sabotage and to prevent the theft of special nuclear material. Licensees subject to the provisions of § 72.182, § 72.212, § 73.20, § 73.50, and § 73.60 are exempt from § 73.1(a)(1)(i)(E) and § 73.1(a)(1)(iii).

(1) * * *

(i) * * *

(E) A four-wheel drive land vehicle used for transporting personnel and their hand-carried equipment to the proximity of vital areas, and

(ii) An internal threat of an insider, including an employee (in any position), and

(iii) A four-wheel drive land vehicle bomb.

* * * * *

3. In § 73.21, a new paragraph (b)(1)(xiii) is added to read as follows:

§ 73.21 -- Requirements for the protection of safeguards information.

* * * * *

(b) * * * [*38900]

(1) * * *

(xiii) Information required by the Commission pursuant to 10 CFR 73.55 (c) (8) and (9).

* * * * *

4. In § 73.55, new paragraphs (c) (7), (8), (9), and (10) are added to read as follow:

§ 73.55 -- Requirements for physical protection of licensed activities in nuclear power reactors against radiological sabotage.

* * * * *

(c) * * *

(7) Vehicle control measures, including vehicle barrier systems, must be established to protect against use of a land vehicle, as specified by the Commission, as a means of transportation to gain unauthorized proximity to vital areas.

(8) Each licensee shall compare the vehicle control measures established in accordance with 10 CFR 73.55 (c)(7) to the Commission's design goals (i.e., to protect equipment, systems, devices, or material, the failure of which could directly or indirectly endanger public health and safety by exposure to radiation) and criteria for protection against a land vehicle bomb. Each licensee shall either:

(i) Confirm to the Commission that the vehicle control measures meet the design goals and criteria specified; or

(ii) Propose alternative measures, in addition to the measures established in accordance with 10 CFR 73.55 (c)(7), describe the level of protection that these measures would provide against a land vehicle bomb, and compare the costs of the alternative measures with the costs of measures necessary to fully meet the design goals and criteria. The Commission will approve the proposed alternative measures if they provide substantial protection against a land vehicle bomb, and it is determined by an analysis, using the essential elements of 10 CFR 50.109, that the costs of fully meeting the design goals and criteria are not justified by the added protection that would be provided.

(9) Each licensee authorized to operate a nuclear power reactor shall:

(i) By February 28, 1995 submit to the Commission a summary description of the proposed vehicle control measures as required by 10 CFR 73.55 (c)(7) and the results of the vehicle bomb comparison as required by 10 CFR 73.55 (c)(8). For licensees who choose to propose alternative measures as provided for in 10 CFR 73.55 (c)(8), the proposal must be submitted in accordance with 10 CFR 50.90 and include the analysis and justification for the proposed alternatives.

(ii) By February 29, 1996 fully implement the required vehicle control measures, including site-specific alternative measures as approved by the Commission.

(iii) Protect as Safeguards Information, information required by the Commission pursuant to 10 CFR 73.55(c) (8) and (9).

(iv) Retain, in accordance with 10 CFR 73.70, all comparisons and analyses prepared pursuant to 10 CFR 73.55 (c) (7) and (8).

(10) Each applicant for a license to operate a nuclear power reactor pursuant to 10 CFR 50.21(b) or 10 CFR 50.22, whose application was submitted prior to August 31, 1994, shall incorporate the required vehicle control program into the site Physical Security Plan and implement it by the date of receipt of the operating license.

* * * * *

Dated at Rockville, Maryland, this 26th day of July 1994.

For the Nuclear Regulatory Commission.

John C. Hoyle,

Acting Secretary of the Commission.

[FR Doc. 94-18638 Filed 7-29-94; 8:45 am]

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Rules and Regulations

NUCLEAR REGULATORY COMMISSION (NRC)

10 CFR Parts 60, 72, 73, 74, and 75

RIN 3150-AF32

Physical Protection for Spent Nuclear Fuel and High-Level Radioactive Waste

63 FR 26955

DATE: Friday, May 15, 1998

ACTION: Final rule.

To view the next page, type .np* TRANSMIT.
To view a specific page, transmit p* and the page number, e.g. p*1

[*26955]

SUMMARY: The Nuclear Regulatory Commission is amending its regulations to clarify physical protection requirements for spent nuclear fuel and high-level radioactive waste stored at independent spent fuel storage installations (ISFSIs), monitored-retrievable storage (MRS) installations, and geologic repository operations areas (GROAs). These amendments codify standards for protecting spent fuel at the various storage sites licensed under the Commission's regulations.

EFFECTIVE DATE: November 12, 1998.

FOR FURTHER INFORMATION CONTACT: Priscilla A. Dwyer, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, telephone (301) 415-8110, e-mail PAD@NRC.GOV.

SUPPLEMENTARY INFORMATION:

I. Background

On August 15, 1995 *(60 FR 42079)*, the Commission published for public comment a proposed rule that would clarify its regulations on the physical protection of spent nuclear fuel and high-level radioactive waste. The proposed regulation would have applied to spent fuel and high-level radioactive waste stored at ISFSIs, power reactors that have permanently ceased reactor operations, MRS installations, and the GROA. The proposed rule stated that the requirements for physically protecting this type of material lacked clarity in defining which regulations were to be applied at these sites. This resulted in a non-cohesive regulatory base. The proposed rule would provide a set of performance-based requirements, consistent with current programs that are currently licensed and implemented at sites under a unified policy for physical protection.

The proposed rule also indicated that the Commission was studying the need for specific protection against the malevolent use of a vehicle at sites affected by the rule (this is discussed further under the "Protection Goal" heading). The rule also proposed a conforming amendment to 10 CFR Part 60-to require material control and accounting (MC&A) measures at the GROA that would be identical to that required of ISFSIs under Part 72. The proposed rule added a provision under 10 CFR Part 75 to clarify that if GROAs are subject to International Atomic Energy Agency (IAEA) safeguards, then NRC's nuclear material accounting and control regulations for implementing the "Agreement between the United States and the IAEA for the Application of Safeguards in the United States" apply. Finally, the Commission requested specific comment on five questions regarding impacts of the proposed regulation on licensees.

II. Summary and Analysis of Public Comments

The proposed rule was subject to a 90-day public comment period which ended on November 13, 1995. Twenty letters of comment were received. Sources for these comments included a nuclear industry group [the Nuclear Energy Institute (NEI)]; one national laboratory; fifteen utilities involved in nuclear activities; two Federal agencies [the Environmental Protection Agency (EPA) and the Department of Energy (DOE)]; and one citizen's group. Twelve letters of comment explicitly endorsed, either in total or in part, the views expressed by the NEI. Four letters of comment, in part, supported the general objectives of the proposed rulemaking. Correspondence received from EPA indicated no comment. The comments have been grouped under the following general topics:

1. Protection Goal.

2. Basis for Requirements.

3. Required Level of Physical Protection.

4. Backfit and Regulatory Analysis.

5. Rule Language Specifics.

6. GROAs.

7. Staff-Generated Amendments.

8. Summary of Responses to Commission's Specific Questions.

1. Protection Goal

Comment. Commenters noted that, although it was appropriate that a protection goal for spent fuel and high-level radioactive waste be defined, the protection goal needed to be less stringent than the codified design basis threat for radiological sabotage. It was further stated that a 10 CFR Part 100 release, the unofficial criterion for determining radiological sabotage of power reactors, would be extremely difficult to realize with respect to spent fuel and high-level radioactive waste. The citizen's group commented that any protection goal developed for spent fuel should also counter the malevolent use of an airborne vehicle.

Response. The NRC agrees that the establishment of a protection goal should be the first step in the development of any physical protection standards. One issue that may have caused confusion in the proposed rule is that the assumptions for determining "radiological sabotage" differ between Part 72, "Licensing Requirements for the

Independent Storage of Spent Fuel and High-Level Radioactive Waste," and Part 73, "Physical Protection of Plants and Material." The differing assumptions are appropriate because "radiological sabotage," as used under Part 73, applies to a power reactor and implies the unofficial criterion of a Part 100 release for power reactors. "Radiological sabotage" as used under Part 72 applies to the storage of spent fuel and high-level radioactive waste and is based on the consequences of a design basis accident as defined under Part 72. Although the same term is used under both 10 CFR Parts; it is based on different assumptions and results in different levels of required protection. The Commission agrees that this is confusing and that "radiological sabotage," as used for operating reactors, is not an appropriate protection level for spent fuel and high-level radioactive waste. The Commission concludes that the protection goal is best characterized by the phrase: "protection against the loss of control of the facility that could be sufficient to cause radiation exposure exceeding the dose as described in 10 [*26956] CFR 72.106." The final rule has been modified accordingly.

With regard to protection against the malevolent use of a land-based vehicle, NRC has determined, based on the opinions of expert study and a peer review of findings, that there is no compelling justification for requiring a vehicle barrier as perimeter protection for spent fuel and high-level radioactive waste stored under a Part 60 or Part 72 license. Inclusion of an airborne vehicle was assessed for possible inclusion into the protection goal for this rule. However, protection against this type of threat has not yet been determined appropriate at sites with greater potential consequences than spent fuel storage installations. Therefore, this type of requirement is not included within the protection goal for this final rule.

2. Basis for Requirements

Comment. Commenters frequently questioned the need for tying Part 72 requirements to Part 73. The commenters assumed that by involving Part 73 in the rulemaking, it was implied that the level of physical protection normally attributed to power reactors was being required. Phraseology used in the proposed requirements, such as using the term "protected area," (PA) tended to further foster this impression.

Response. The Commission disagrees that placing requirements under Part 73 implies any association with the physical protection requirements for power reactors. It is noted that Part 73 provides, in one consolidated Part, all of the requirements for those facilities needing physical protection. This is one reason why an explicit requirement for the protection of spent fuel and high-level radioactive waste is being added to Part 73. Part 73 includes more stringent requirements for power reactor and Category I fuel cycle facilities and much less stringent requrements for the protection of Category III facilities. With regard to use of the term "protected area," the Commission has determined that the term is correctly used in review of its definition under 10 CFR 73.2. Nonetheless, the Commission has reviewed the physical protection terminology found in the final rule to ensure that it does not imply a different level of physical protection than intended.

3. Level of Physical Protection Needed

Comment. Some commenters expressed the opinion that the level of physical protection described by the proposed amendments was unnecessary and overly burdensome. The industry group noted that what was truly needed was a level of physical protection comparable to "enhanced industrial security." Cited examples of this type of protection were: use of suitable fencing, locked access points, sufficient illumination, and periodic security patrols. Other commenters questioned the need for some of the redundancy that was included in the proposed rule. One citizen's group believed that physical protection measures should be more stringent than those described in the proposed rule.

Response. The Commission believes that the appropriate level of physical protection for spent fuel and high-level radioactive waste lies somewhere between industrial-grade security and the level that is required at operating power reactors. The Commission also notes that the nature of spent fuel and of its storage mechanisms offers unique advantages in protecting the material. This factor, along with revised consequence considerations, leads the Commission to conclude that physical protection at sites where spent fuel and high-level radioactive waste are stored under a 10 CFR Part 60 or 72 license can be more flexibly applied than previously proposed. Accordingly, the final rule has been revised to minimize redundancy and add flexibility. Specific changes are outlined in Section III, "Summary of Specific Changes Made to the Proposed Rule as a Result of Public Comment."

4. Backfit and Regulatory Analysis

Comment. NEI and a few licensees commented that the proposed regulation imposes a generic backfit as defined under 10 CFR 50.109 and 72.62. The NRC asserted in the proposed rule that the amendments merely codified and standardized physical protection measures that, through license amendment, were already in place at existing sites. Hence, it was concluded that no backfit was involved. Commenters further stated that, in terms of backfit requirements, the cost to implement the proposed rule was not justified based on the potential increase in protection that the rule would afford public health and safety.

Other commenters specifically responded to the Regulatory Analysis that accompanied the rule. These commenters expressed concern that certain provisions of the regulatory analysis could turn into de facto requirements.

Additionally, it was recommended that affected sites should be "grandfathered" under any final rulemaking. Accordingly, these sites would not be required to meet the provisions of the new physical protection rule because an adequate level of physical protection was already in place at the site, based on an NRC-approved physical protection plan.

Response. Under the proposed rule, the Commission stated that the backfit rule in 10 CFR 50.109 did not apply because the amendments did not impose any additional requirements on Part 50 licensees. Furthermore, the Commission notes that all references to Part 50 licensees are deleted in the final rule.

The Commission further stated that the backfitting requirements in 10 CFR 72.62 did not apply because the proposed amendments neither imposed nor modified procedures or organizations of ISFSIs licensed under Part 72. The Commission considers these statements true based on their assessment of the proposed regulation and its intended implementation. However, on further review, the backfit rule in 10 CFR 72.62 may be applicable to one facility which has only one isolation zone exterior to the perimeter barrier. The NRC staff has identified alternative measures currently in place that provide an equivalent level of physical protection. The staff does not intend to require this facility to establish an interior isolation zone. Thus, no backfit occurs due to the new rule. Because 10 CFR 72.62 does not cover reporting and recordkeeping requirements, the inclusion of 10 CFR 73.51 in 73.71 event reporting is not a backfit.

With respect to grandfathering existing sites, the Commission believes that implementation of this final rule at these sites presents no undue burden to affected licensees and provides a minimum level of physical protection to adequately protect the public health and safety. Accordingly, there is no need for a grandfathering provision and no change has been made in the final rule in response to this comment. The Commission notes that the Regulatory Analysis for the final rule has been revised to reflect changes made in response to public comment and to eliminate ambiguities.

5. Rule Language Specifics

Comment. A variety of comments were received regarding specific rule terminology. The suggestion was made that the term "protected area" be revised to "ISFSI controlled access area."

Response. As indicated previously in this notice, the use of the term "protected area," is consistent with its definition in 10 CFR 73.2. Furthermore, because it is the Commission's position [*26957] that a site where spent fuel and high-level radioactive waste is stored be surrounded by a fence, it is not considered adequate to call the enclosure a controlled access area (CAA). Under 10 CFR 73.2, the definition of a CAA requires only a demarcation of the area, not a fence.

Comment. Another commenter supported the Commission position that operating power reactor licensees that store spent fuel under a general license should have the option of using the physical protection measures of either 10 CFR 72.212(b)(5) or the proposed 10 CFR 73.51. The commenter also questioned whether the requirements of 10 CFR 72.182, 72.184, and 72.186 apply to a general license, in addition to Subpart K. A related question requested clarification on how general license holders were to notify NRC regarding which option they would exercise.

Response. The Commission notes that a licensee having a Part 50 license does not fall within the scope of the final rule. The Commission believes it is premature to bring these licensees under the provisions of the final rule because continued protection for spent fuel in storage pools at Part 50 sites is currently under study by the NRC.

Comment. One commenter requested clarification on the specific exclusion of an exemption for ISFSIs from the malevolent use of a vehicle threat within the design basis threat. The commenter indicated that it was not readily apparent and also a cumbersome process to determine the current exempt status of an ISFSI under present regulations.

Response. The Commission agrees and has revised the text of the rule to exclude reference to the design basis threat described under 10 CFR 73.1.

Comment. One commenter questioned whether the proposed rule would apply to a permanently shutdown power plant where spent fuel is stored and the plant is operating with a Part 50 possession-only license.

Response. A facility with a Part 50 license is not subject to the provisions of the final rule. This revision to the final rule has been made because the Commission believes it is premature to include these licensees within the scope of the rule because continued protection for spent fuel in storage pools at Part 50 sites is currently under study by the NRC.

Comment. A commenter requested clarification on the need for back-up power for physical protection-related equipment.

Response. The Commission believes that affected licensees should not be vulnerable to loss of offsite power. Thus, it is necessary for licensees to assure either continuous operation of required physical protection equipment during power failure or to demonstrate the ability to provide immediate compensation for such failures.

Comment. Required illumination levels, assessment techniques, required frequency of physical protection patrols, and searches before entry to the PA were all subjects of comment. A commenter suggested that illumination be provided only during periods of assessment and that the entire PA need not be illuminated to a level of 0.2 footcandle.

Response. The Commission agrees that illumination to a 0.2 footcandle level represents a large operating cost and may be difficult to achieve, given cask structure. This provision has been amended to more clearly indicate that, while illumination should be maintained during all periods of darkness, only an adequate level of illumination is required within the PA for the detection assessment means used. In addition, required performance capabilities regarding detection are clarified in the final rule by specifying the use of active intrusion detection equipment, as opposed to passive systems.

Comment. Some commenters noted that the frequency of patrols should coincide with watchmens' duty shift lengths, as opposed to once every eight hours as recommended in the proposed rule.

Response. The Commission does not agree that the frequency of patrols should coincide with duty shift lengths. However, the Commission agrees that some flexibility can be provided. Accordingly, this provision of the final rule is revised to require daily random patrols, only.

Comment. Licensees cited the burden of maintaining expensive and delicate explosives detection equipment to meet the proposed requirement for explosives searches conducted before entry to the PA.

Response. The Commission agrees. To clarify this issue, the Commission has revised the proposed rule to require only a visual search for explosives. Because pedestrian and vehicular traffic is not expected to be high volume at facilities affected by the rule, this type of search is not considered an undue burden to affected licensees. Furthermore, the amount of explosives that may cause a radiological release is not easily concealed.

Comment. Other commenters noted redundant records retention requirements in 10 CFR 72.180 and 10 CFR 73.51(c).

Response. This concern has been corrected in the final rule.

Comment. One commenter noted an apparent contradiction in the proposed regulation regarding use of deadly force in the protection of an ISFSI. The commenter had been advised by NRC staff that use of deadly force was not expected of members of the security organization at ISFSIs. The commenter reasoned that this was not consistent with the requirement to protect against radiological sabotage under the proposed rule.

Response. The issue involving the use of the term radiological sabotage has been resolved as discussed previously. Further, the Commission never intended that onsite physical protection personnel at an ISFSI would provide a response to a safeguards event other than calling for assistance from local law enforcement or other designated response force unless their timely response could not be ensured. The Commission also notes that 10 CFR 73.51 only calls for unarmed watchmen, not armed guards.

Comment. Commenters believe that the requirements for redundant alarm monitoring stations and specified staffing levels for the primary alarm station are overly burdensome and unnecessary.

Response. The Commission agrees that the requirement for redundant alarm stations is excessive. Regarding alarm monitoring, this provision is revised in the final rule to require, in the redundant location, only a summary indication that an alarm has been generated. This location need not necessarily be located onsite and could, for example, be a simple readout in a continually-staffed local law enforcement agency office. This is contingent on the assurance that communications with the local law enforcement agency or the designated response force can be maintained. Regarding required staffing levels of the primary alarm station, the Commission has deleted the specific requirement that the physical protection organization be comprised of at least two watchmen from the final rule. This deletion is contingent on the Commission's expectation that a human presence be maintained in the primary alarm station at all times. To achieve this, the Commission clarifies its position that the primary alarm station must be located within the PA, be bullet-resisting, and be configured such that activities within the station are not visible from outside the PA. The intent of these measures is to ensure that a single act cannot destroy the capability of an onsite watchman to call for [*26958] assistance. The final rule has been modified accordingly.

Comment. Finally, concerning the actual terminology and format of the proposed rule, commenters expressed support for its performance-based nature but rejected the set of provisions under 10 CFR 73.51(d) as being overly prescriptive.

Response. The Commission responds that the proposed regulation found in 10 CFR 73.51(d) is needed to provide additional clarity in meeting the performance capabilities in 10 CFR 73.51(b) and notes that many of the physical protection measures described under 10 CFR 73.51(d) are relaxed in the final rule and are less prescriptive in a number of cases.

6. GROA

Comment. Two comments were received from DOE on the amendments to Part 60 dealing with the geologic repository. The first commenter requested that it be emphasized in the "Statement of Considerations" for the final rule that the requirement for physical protection of GROAs be applicable only during their operational phases and not after closure.

Response. The Commission agrees with this observation and has clarified the exemption in the final rule to specifically exempt GROAs from the requirements of 10 CFR 73.51 after permanent closures.

Comment. The second commenter requested clarification on apparent conflicts in Part 60, "Disposal of High-Level Radioactive Waste in Geologic Repositories," regarding the level of detail required of physical protection plans during the different phases of the certification process.

Response. The Commission notes that NUREG 1619, "Standard Review Plan for Physical Protection Plans for the Independent Storage of Spent Fuel and High-Level Radioactive Waste," to be issued concurrently with the effective date of the final rule, will contain guidance in this area.

7. NRC Staff-Generated Amendments

Subsequent to publication of the proposed rule, a technical issue arose involving the cooling time of spent fuel as it relates to the degree of physical protection needed. Because a response to this issue continues to evolve within the NRC, the Commission believes it would be inappropriate to apply the provisions of the final rule at this time to a licensee holding a 10 CFR Part 50 license. Hence, licensees holding a 10 CFR Part 50 license are not within the scope of the final rule. Further, review indicated that there was some confusion pertaining to MC&A requirements for ISFSIs. Specifically, the NRC staff asked if ISFSIs were exempt from the requirements of 10 CFR 74.51 and, if not, why not. Specific MC&A requirements for ISFSIs are found under Part 72. After consideration of the issue, for clarification, the NRC staff has included an amendment to 10 CFR Part 74 that specifically exempts ISFSIs from 10 CFR 74.51 in the final rule.

8. Summary of Responses to Commission's Specific Questions

Question 1. Would the proposed amendments impose any significant additional costs for safeguards of currently stored spent nuclear fuel beyond what is now incurred for that purpose?

Summary of Responses. Five responses from nuclear utilities specifically addressed this issue. All indicated that the amendments, as proposed, would significantly increase costs. Manpower-intensive measures, such as the requirement to maintain a minimum of two watchmen per shift, were most often cited as creating an undue burden. One licensee estimated costs of $ 1 to $ 2 million to implement, and a continuing cost increase of 30-50 percent, annually, to physical protection operations.

NRC Response. Licensees holding a 10 CFR Part 50 license are no longer within the scope of this rule. The final rule has been revised to minimize redundancy and add flexibility to its implementation. There should be no significant increase in cost to current licensees.

Question 2. Is there reason to expect the costs to future licensees to differ substantially from those of current licensees?

Summary of Responses. Four responses from nuclear utilities specifically addressed this issue. Three utilities cited both higher current and annual operating costs. One utility noted that, to the extent that current licensees have been required to commit to the practices recommended in the proposed rule in initial licensing, there is no anticipated difference in cost.

NRC Response. Licensees holding a 10 CFR Part 50 license are no longer within the scope of this rule. The final rule has been amended to be more consistent with physical protection implemented at sites with currently approved physical protection plans. Hence, there should be no significant increase in costs to future licensees.

Question 3. Are the cost estimates in Table III of the Draft Regulatory Analysis representative of current industry experience? Are there significant costs that have not been included in the table?

Summary of Responses. Three responses from nuclear utilities specifically addressed this issue. One respondent indicated that the cost estimates in Table III of the "Draft Regulatory Analysis" are sufficiently broad to address industry experience. However, the inclusion of a continual surveillance system is not covered and the respondent suggested that it should be a separate line item. Another respondent indicated that the cost estimates appear to be comprehensive except they do not include construction and maintenance of physical protection office space, a records retention area, and alarm station(s).

NRC Response. The "Regulatory Analysis" has been revised to reflect public comment to include any omissions or changes made to the final rule.

Question 4. Are the costs justified by the benefits that would be afforded by the proposed amendments? Are there alternatives that would afford essentially the same benefits but be more cost effective?

Summary of Responses. Three responses from nuclear utilities specifically addressed this issue. All three indicated that the costs were not justified by the benefits derived from the proposed rule. One respondent stated that the individual measures of 10 CFR 73.51(d) have merit, but, when taken in aggregate, they are not necessary to protect public health and safety. This respondent further stated that redundancy in the proposed rule was not needed and the rulemaking should give affected licensees latitude in selecting and justifying the means of physical protection. Alternatives that were suggested involved the deletion of specific provisions of the proposed rule and also the restructuring of the rule so as to not group all ISFSIs under one set of physical protection criteria.

NRC Response. The Commission has revised the requirements of the proposed rule to eliminate unnecessary redundancies, add flexibility in implementation, and reduce manpower-intensive measures while maintaining an adequate level of physical protection.

Question 5. Are the proposed amendments to 10 CFR 73.51 appropriate for an MRS or geologic repository operated by DOE?

Summary of Response. NEI was the only respondent to this issue. NEI noted that NRC should be mindful of the evolving nature of MRS installations and the geologic repository in the development of physical protection regulations for these sites. **[*26959]**

NRC Response. NRC staff continues to work closely with DOE staff in the development of the certification process for MRS installations and the GROA.

III. Summary of Specific Changes Made to the Proposed Rule as a Result of Public Comment

Major changes made to the proposed rule include:

(1) The incorporation of a protection goal, and

(2) Regarding required levels of physical protection, redundancies have been reduced, flexibility added, and manpower-for example-

. Regarding alarm monitoring, the redundant alarm station need only provide a summary indication at a continually staffed location;

. Redundant records retention has been eliminated;

. The required staffing level for the security organization has been eliminated and required siting and configuration of the primary alarm station clarified;

. Hand-held equipment searches for explosives are replaced with visual searches; and

. Illumination levels need only permit adequate assessment of the PA according to the assessment means used. Detection equipment must be active in nature.

As discussed previously, the final rule does not apply to a licensee holding a 10 CFR Part 50 license.

A section-by-section comparison of the proposed and final rules follows.

Part 60--Disposal of High-Level Radioactive Wastes in Geologic Repositories

1. Section 60.21, Content of application. This section is unchanged from the proposed rule.

2. Section 60.31, Construction authorization. This section is unchanged from the proposed rule.

3. Section 60.41, Standards for issuance of a license. This section is unchanged from the proposed rule.

4. Section 60.78, Material control and accounting records and reports. This section is unchanged from the proposed rule.

Part 72--Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste

5. Section 72.24, Contents of application: Technical information. This section is unchanged from the proposed rule. The term "radiological sabotage" is based on Part 72 assumptions and not a Part 100 radiological release.

6. Section 72.180, Physical security plan. This section is unchanged from the proposed rule except for changing the title to Physical Protection Plan to be consistent with 10 CFR Part 73.

7. Section 72.212, Conditions of general license issued under § 72.210. Revisions to this section have been deleted in their entirety.

Part 73--Physical Protection of Plants and Materials

8. Section 73.1, Purpose and Scope. Paragraph (b)(6) is unchanged from the proposed rule.

9. Section 73.50, Requirements for physical protection of licensed activities. This section remains unchanged from the proposed rule.

10. Section 73.51, Requirements for the physical protection of stored spent nuclear fuel and high-level radioactive waste. Paragraph (a), Applicability, has been revised to more precisely define the type of material affected by the rule and to eliminate 10 CFR Part 50 licensees from the provisions of the rule.

Paragraph (b)(3), General Performance Objectives, has been revised to read: "The physical protection system must be designed to protect against loss of control of the facility that could be sufficient to cause radiation exposure exceeding the dose as described in 10 CFR 72.106." This revised statement describes a more appropriate protection goal that is consistent with Part 72. It also allows for a physical protection system less stringent than required to protect against radiological sabotage at operating power reactors.

The introductory text of paragraph (d) has been revised to more clearly indicate the Commission's intent that alternative measures may also be available for meeting the provisions of (d). For example, several questions arose during final rule development as to whether the use of a hardened and protected alarm station sited at an adjacent operating power reactor would meet the intent of paragraph (d)(3) to have a hardened alarm station within the PA of the ISFSI. Staff considers this to be an acceptable alternative measure for meeting this provision of the final rule.

In paragraph (d)(1), the last sentence has been deleted because it is no longer necessary due to the revision cited in the previous paragraph above.

Paragraph (d)(2) has been revised to read: "Illumination must be sufficient to permit adequate assessment of unauthorized penetrations of or activities within the protected area." This revision has been made to permit flexibility in illumination levels.

Paragraph (d)(3) has been revised to read: "The perimeter of the protected area must be subject to continual surveillance and be protected by an active intrusion alarm system that is capable of detecting penetration through the isolation zone and that is monitored in a continually staffed primary alarm station located within the protected area, and in one additional continually staffed location to ensure that a single act cannot destroy the capability of the onsite watchman to call for assistance. The primary alarm station must be located within the protected area; have bullet-resisting walls, doors, ceiling, and floor; and the interior of the station must not be visible from outside the protected area. A timely means for assessment must also be provided. Regarding alarm monitoring, the redundant location need only provide a summary indication that an alarm has been generated." This clarifies the Commission's position that the necessary level of protection should ensure that a single act cannot destroy the capability of the onsite watchman to call for assistance.

Paragraph (d)(4) has been revised to reduce the frequency of patrol from "not less than once every 8 hours" to "daily random patrols" with additional discussion provided in guidance issued to support the rule.

Paragraph (d)(5) has been revised to read: "A security organization with written procedures must be established. The security organization must include sufficient personnel per shift to provide for monitoring of detection systems and the conduct of surveillance, assessment, access control, and communications to assure adequate response. Members of the security organization must be trained, equipped, qualified and requalified to perform assigned job duties in accordance with Appendix B to Part 73, I.A, (1) (a) and (b); B(1)(a); and the applicable portions of II." This change eliminates a required staffing level and describes qualification and training levels for watchmen, only, as the primary members of the security organization.

Paragraph (d)(6) has been changed to require "timely" response from the designated response forces. If timely response cannot be provided, additional protective measures may be required, to include use of armed guards.

Paragraph (d)(7) has been deleted.

Paragraph (d)(8) has been redesignated as paragraph (d)(7) and revised to read as follows: "A personnel identification system and a controlled lock system must be established and maintained to limit access to authorized individuals." This eliminates the unnecessary coupling of the identification system with the system [*26960] used for key and lock control as requested by commenters.

Paragraph (d)(9) has been deleted. If a person is authorized access to the PA, properly identified, and subject to search, there is no need for the individual to be escorted.

Paragraph (d)(10) has been redesignated as paragraph (d)(8). Regarding communications, the term "security organization" has been revised to "onsite security force members" to more precisely define communication channels.

Paragraph (d)(11) has been redesignated as paragraph (d)(9) and revised to read as follows: "All individuals, vehicles and hand-carried packages entering the protected area must be checked for proper authorization and visually searched for explosives before entry." This is permissible because the amount of explosives needed to cause a radiological release is not easily concealable.

Paragraph (d)(12) has been redesignated as paragraph (d)(10). The text of this paragraph is unchanged from the proposed rule.

Paragraph (d)(13) has been redesignated as paragraph (d)(11) and revised to read as follows: "All detection systems, surveillance/ assessment systems, and supporting subsystems including illumination systems must be tamper-indicating with line supervision and be maintained in operable condition. Timely compensatory measures must be taken after discovery of inoperability to assure that the effectiveness of the physical protection system is not reduced."

Paragraph (d)(14) has been redesignated as paragraph (d)(12) and remains unchanged from the proposed rule.

Paragraph (d)(15) has been redesignated as paragraph (d)(13). This provision has been added to assure that duplication of records under § 72.180 is not required. Paragraph (d)(13)(ii) has been revised to read as follows: "Screening records of members of the security organization." Finally, the log of patrols must contain all patrols, not just routine patrols.

Paragraph (e) has been revised for clarity.

11. Section 73.71, Reporting of safeguards events, remains unchanged from the proposed rule.

Part 74--Material Control and Accounting of Special Nuclear Material

12. In Section 74.51, Nuclear material control and accounting for special nuclear material, paragraph (a) has been revised to read as follows: "General performance objectives. Each licensee who is authorized to possess five or more formula kilograms of strategic special nuclear material (SSNM) and to use such material at any site, other than a nuclear reactor licensed pursuant to Part 50 of this chapter, an irradiated fuel reprocessing plant, an operation involved with waste disposal, or an independent spent fuel storage facility licensed pursuant to Part 72 of this chapter, shall establish, implement, and maintain a Commission approved material control and accounting (MC&A) system that will achieve the following objectives: * * *" This paragraph specifically exempts Part 72 ISFSIs from the requirements of 10 CFR 74.51.

Part 75--Safeguards on Nuclear Material--Implementation of US/IAEA Agreement

13. Section 75.4, Definitions, remains unchanged from the proposed rule.

Criminal Penalties

NRC notes that these final amendments are issued under Sections 161b and i of the Atomic Energy Act of 1954, as amended. Therefore, violation of these regulations may subject a person to criminal sanctions under section 223 of the Atomic Energy Act.

Environmental Impact: Categorical Exclusion

The Commission has determined that this final rule is the type of action described as a categorical exclusion in 10 CFR 51.22(c)(3)(i) and (iii). Therefore, neither an environmental impact statement nor an environmental assessment has been prepared for this final rule.

Paperwork Reduction Act Statement

This final rule amends information collection requirements that are subject to the Paperwork Reduction Act of 1995 *(44 U.S.C. 3501 et seq.)*. These requirements were approved by the Office of Management and Budget (OMB), approval numbers 3150-0002, 3150-0055, 3150-0123, and 3150-0132.

Public Protection Notification

If an information collection does not display a currently valid OMB control number, the NRC may not conduct and a person is not required to respond to, the information collection.

Regulatory Analysis

The Commission has prepared a "Final Regulatory Analysis" for this final rule. The final analysis examines the benefits and alternatives considered by the Commission. The "Final Regulatory Analysis" is available for inspection in the NRC Public Document room, 2120 L Street NW (Lower Level), Washington DC. Single copies of the analysis may be obtained from Priscilla A. Dwyer, Division of Fuel Cycle Safety and Safeguards, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001. The "Final Regulatory Analysis" is available for viewing and downloading from the NRC's rulemaking bulletin board.

Regulatory Flexibility Certification

As required by the Regulatory Flexibility Act, *5 U.S.C. 605*(b), the Commission certifies that this rule does not have a significant economic impact on a substantial number of small entities. The final rule affects operators of ISFSIs and DOE as the operator of the MRS and GROA. The affected licensees do not fall within the scope of the definition of "small entities" set forth in Section 601(3) of the Regulatory Flexibility Act, or the NRC's size standards (10 CFR 2.810).

Small Business Regulatory Enforcement Fairness Act

In accordance with the Small Business Regulatory Enforcement Fairness Act of 1996, NRC has determined that this action is not a "major rule" and has verified this determination with the Office of Information and Regulatory Affairs of OMB.

Backfit Analysis

The Commission has determined that the backfit rule in 10 CFR 50.109 does not apply because this final rule does not impose new requirements on existing 10 CFR part 50 licensees. The backfit rule in 10 CFR 72.62 may be applicable to one facility which has only one isolation zone exterior to the perimeter barrier. However, the NRC staff has identified alternative measures currently in place that provide an equivalent level of physical protection. The staff does not intend to require this facility to establish an interior isolation zone. Thus, no backfit occurs due to the new rule. Because 10 CFR 72.62 does not cover reporting and recordkeeping requirements, the inclusion of 10 CFR 73.51 in 10 CFR 73.71 event reporting is not a backfit. Finally, the transfer of spent fuel from a reactor, licensed under 10 CFR part 50 and subject to 10 CFR 73.55 physical protection requirements, to an ISFSI licensed under 10 CFR part 72, and its associated physical protection provisions (e.g., 10 CFR 73.51) is not a backfit. A new license under 10 CFR art 72 is a matter of compliance with regulations. In all [*26961] cases, transition from 10 CFR 73.55 to 73.51 is a relaxation of requirements and not a backfit.

List of Subjects

10 CFR Part 60

Criminal penalties, High-level waste, Nuclear power plants and reactors, Nuclear materials, Reporting and recordkeeping requirements, Waste treatment and disposal.

10 CFR Part 72

Manpower training programs, Nuclear materials, Occupational safety and health, Reporting and recordkeeping requirements, Security measures, Spent fuel.

10 CFR Part 73

Criminal penalties, Hazardous materials transportation, Export, Import, Nuclear materials, Nuclear power plants and reactors, Reporting and recordkeeping requirements, Security measures.

10 CFR Part 74

Accounting, Criminal penalties, Hazardous materials transportation, Material control and accounting, Nuclear materials, Packaging and containers, Radiation protection, Reporting and recordkeeping requirements, Scientific equipment, Special nuclear material.

10 CFR Part 75

Criminal penalties, Intergovernmental relations, Nuclear materials, Nuclear power plants and reactors, Reporting and recordkeeping requirements, Security measures.

For the reasons set out in the preamble and under the authority of the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974, as amended, and *5 U.S.C. 552* and 553 the NRC is adopting the following amendments to 10 CFR parts 60, 72, 73, 74, and 75.

PART 60--DISPOSAL OF HIGH-LEVEL RADIOACTIVE WASTES IN GEOLOGIC REPOSITORIES

1. The authority citation for part 60 continues to read as follows:

Authority: Secs. 51, 53, 62, 63, 65, 81, 161, 182, 183, 68 Stat. 929, 930, 932, 933, 935, 948, 953, 954, as amended *(42 U.S.C. 2071,* 2073, 2092, 2093, 2095, 2111, 2201, 2232, 2233); secs. 202, 206, 88 Stat. 1244, 1246 *(42 U.S.C. 5842,* 5846); secs. 10 and 14, Pub. L. 95-601, 92 Stat. 2951 *(42 U.S.C. 2021a* and 5851); sec. 102, Pub. L. 91-190, 83 Stat. 853 *(42 U.S.C. 4332);* secs. 114, 121, Pub. L. 97-425, 96 Stat. 2213g, 2228, as amended *(42 U.S.C. 10134,* 10141) and Pub. L. 102-486, sec 2902, 106 Stat. 3123 *(42 U.S.C. 5851).*

2. In § 60.21, paragraphs (b)(3), (b)(4), and (c)(10) are revised to read as follows:

§ 60.21 -- Content of application.

* * * * *

(b) * * *

(3) A detailed plan to provide physical protection of high-level radioactive waste in accordance with § 73.51 of this chapter. This plan must include the design for physical protection, the licensee's safeguards contingency plan, and security organization personnel training and qualification plan. The plan must list tests, inspections, audits, and other means to be used to demonstrate compliance with such requirements.

(4) A description of the program to meet the requirements of § 60.78.

* * * * *

(c) * * *

(10) A description of the program to be used to maintain the records described in § § 60.71 and 60.72.

* * * * *

3. In § 60.31, paragraph (b) is revised to read as follows:

§ 60.31 -- Construction authorization.

* * * * *

(b) *Common defense and security.* That there is reasonable assurance that the activities proposed in the application will not be inimical to the common defense and security.

* * * * *

4. In § 60.41, paragraph (c) is revised to read as follows:

§ 60.41 -- Standards for issuance of license.

* * * * *

(c) The issuance of the license will not be inimical to the common defense and security and will not constitute an unreasonable risk to the health and safety of the public.

* * * * *

5. A new § 60.78 is added to read as follows:

§ 60.78 – Material control and accounting records and reports.

DOE shall implement a program of material control and accounting (and accidental criticality reporting) that is the same as that specified in § § 72.72, 72.74, 72.76, and 72.78 of this chapter.

PART 72–LICENSING REQUIREMENTS FOR THE INDEPENDENT STORAGE OF SPENT NUCLEAR FUEL AND HIGH-LEVEL RADIOACTIVE WASTE

6. The authority citation for part 72 continues to read as follows:

Authority: Secs. 51, 53, 57, 62, 63, 65, 69, 81, 161, 182, 183, 184, 186, 187, 189, 68 Stat. 929, 930, 932, 933, 934, 935, 948, 953, 954, 955, as amended, sec. 234, 83 Stat. 444, as amended *(42 U.S.C. 2071,* 2073, 2077, 2092, 2093, 2095, 2099, 2111, 2201, 2232, 2233, 2234, 2236, 2237, 2238, 2282); sec. 274, Pub. L. 86-373, 73 Stat. 688, as amended *(42 U.S.C. 2021);* sec. 201, as amended, 202, 206, 88 Stat. 1242, as amended, 1244, 1246 *(42 U.S.C. 5841,* 5842, 5846); Pub. L. 95-601, sec. 10, 92 Stat. 2951 as amended by Pub. L. 102-486, sec. 7902, 106 Stat. 3123 *(42 U.S.C. 5851);* sec. 102, Pub. L. 91-190, 83 Stat. 853 *(42 U.S.C. 4332);* secs. 131, 132, 133, 135, 137, 141, Pub. L. 97-425, 96 Stat. 2229, 2230, 2232, 2241, sec. 148, Pub. L. 100-203, 101 Stat. 1330-235 *(42 U.S.C. 10151,* 10152, 10153, 10155, 10157, 10161, 10168).

Section 72.44(g) also issued under secs. 142(b) and 148(c), (d), Pub. L. 100-203, 101 Stat. 1330-232, 1330-236 *(42 U.S.C. 10162*(b), 10168 (c), (d)). Section 72.46 also issued under sec. 189, 68 Stat. 955 *(42 U.S.C. 2239);* sec. 134, Pub. L. 97-425, 96 Stat. 2230 *(42 U.S.C. 10154).* Section 72.96(d) also issued under sec. 145(g), Pub. L. 100-203, 101 Stat. 1330-235 *(42 U.S.C. 10165*(g)). Subpart J also issued under secs. 2(2), 2(15), 2(19), 117(a), 141(h), Pub. L. 97-425, 96 Stat. 2202, 2203, 2204, 2222, 2224 *(42 U.S.C. 10101,* 10137(a), 10161(h)). Subparts K and L are also issued under sec. 133, 98 Stat. 2230 *(42 U.S.C. 10153)* and sec. 218(a), 96 Stat. 2252 *(42 U.S.C. 10198).*

7. In § 72.24, paragraph (o) is revised to read as follows:

§ 72.24 – Contents of application; Technical information.

* * * * *

(o) A description of the detailed security measures for physical protection, including design features and the plans required by subpart H. For an application from DOE for an ISFSI or MRS, DOE will provide a description of the physical protection plan for protection against radiological sabotage as required by subpart H.

* * * * *

8. Section 72.180 is revised to read as follows:

§ 72.180 – Physical protection plan.

The licensee shall establish, maintain, and follow a detailed plan for physical protection as described in § 73.51 of this chapter. The licensee shall retain a copy of the current plan as a record until the Commission terminates the license for which the procedures were developed and, if any portion of the plan is superseded, retain the superseded material for 3 years after each change or until termination of the license. The plan must describe how the applicant will meet the requirements of § 73.51 of this chapter and provide physical protection during on-site transportation [*26962] to and from the proposed ISFSI or MRS and include within the plan the design for physical protection, the licensee's safeguards contingency plan, and the security organization personnel training and qualification plan. The plan must list tests, inspections, audits, and other means to be used to demonstrate compliance with such requirements.

PART 73–PHYSICAL PROTECTION OF PLANTS AND MATERIALS

9. The authority citation for part 73 continues to read as follows:

Authority: Secs. 53, 161, 68 Stat. 930, 948, as amended, sec. 147, 94 Stat. 780 *(42 U.S.C. 2073,* 2167, 2201); sec. 201, as amended, 204, 88 Stat. 1242, as amended, 1245, sec. 1701, 106 Stat. 2951, 2952, 2953 *(42 U.S.C. 5841,* 5844, 2297f).

Section 73.1 also issued under secs. 135, 141, Pub. L. 97-425, 96 Stat. 2232, 2241 *(42 U.S.C, 10155,* 10161). Section 73.37(f) also issued under sec. 301, Pub. L. 96-295, 94 Stat. 789 *(42 U.S.C. 5841* note). Section 73.57 is issued under sec. 606, Pub. L. 99-399, 100 Stat. 876 *(42 U.S.C. 2169).*

10. In § 73.1, paragraph (b)(6) is revised to read as follows:

§ 73.1 – Purpose and scope.

* * * * *

(b) * * *

(6) This part prescribes requirements for the physical protection of spent nuclear fuel and high-level radioactive waste stored in either an independent spent fuel storage installation (ISFSI) or a monitored retrievable storage (MRS) installation licensed under part 72 of this chapter, or stored at the geologic repository operations area licensed under part 60 of this chapter.

* * * * *

11. The introductory text of § 73.50 is revised to read as follows:

§ 73.50 – Requirements for physical protection for licensed activities.

Each licensee who is not subject to § 73.51, but who possesses, uses, or stores formula quantities of strategic special nuclear material that are not readily separable from other radioactive material and which have total external radiation dose rates in excess of 100 rems per hour at a distance of 3 feet from any accessible surfaces without intervening shielding other than at a nuclear reactor facility licensed pursuant to part 50 of this chapter, shall comply with the following:

* * * * *

12. A new § 73.51 is added to read as follows:

§ 73.51 – Requirements for the physical protection of stored spent nuclear fuel and high-level radioactive waste.

(a) *Applicability.* Notwithstanding the provisions of § § 73.20, 73.50, or 73.67, the physical protection requirements of this section apply to each licensee that stores spent nuclear fuel and high-level radioactive waste pursuant to paragraphs (a)(1)(i), (ii), and (2) of this section. This includes-

(1) Spent nuclear fuel and high-level radioactive waste stored under a specific license issued pursuant to part 72 of this chapter:

(i) At an independent spent fuel storage installation (ISFSI) or

(ii) At a monitored retrievable storage (MRS) installation; or

(2) Spent nuclear fuel and high-level radioactive waste at a geologic repository operations area (GROA) licensed pursuant to part 60 of this chapter;

(b) *General performance objectives.* (1) Each licensee subject to this section shall establish and maintain a physical protection system with the objective of providing high assurance that activities involving spent nuclear fuel and high-level radioactive waste do not constitute an unreasonable risk to public health and safety.

(2) To meet the general objective of paragraph (b)(1) of this section, each licensee subject to this section shall meet the following performance capabilities.

(i) Store spent nuclear fuel and high-level radioactive waste only within a protected area;

(ii) Grant access to the protected area only to individuals who are authorized to enter the protected area;

(iii) Detect and assess unauthorized penetration of, or activities within, the protected area;

(iv) Provide timely communication to a designated response force whenever necessary; and

(v) Manage the physical protection organization in a manner that maintains its effectiveness.

(3) The physical protection system must be designed to protect against loss of control of the facility that could be sufficient to cause a radiation exposure exceeding the dose as described in § 72.106 of this chapter.

(c) *Plan retention.* Each licensee subject to this section shall retain a copy of the effective physical protection plan as a record for 3 years or until termination of the license for which procedures were developed.

(d) *Physical protection systems, components, and procedures.* A licensee shall comply with the following provisions as methods acceptable to NRC for meeting the performance capabilities of § 73.51(b)(2). The Commission may, on a specific basis and upon request or on its own initiative, authorize other alternative measures for the protection of spent fuel and high-level radioactive waste subject to the requirements of this section, if after evaluation of the specific alternative measures, it finds reasonable assurance of compliance with the performance capabilities of paragraph (b)(2) of this section.

(1) Spent nuclear fuel and high-level radioactive waste must be stored only within a protected area so that access to this material requires passage through or penetration of two physical barriers, one barrier at the perimeter of the protected area and one barrier offering substantial penetration resistance. The physical barrier at the perimeter of the protected area must be as defined in § 73.2. Isolation zones, typically 20 feet wide each, on both sides of this barrier, must be provided to facilitate assessment. The barrier offering substantial resistance to penetration may be provided by an approved storage cask or building walls such as those of a reactor or fuel storage building.

(2) Illumination must be sufficient to permit adequate assessment of unauthorized penetrations of or activities within the protected area.

(3) The perimeter of the protected area must be subject to continual surveillance and be protected by an active intrusion alarm system which is capable of detecting penetrations through the isolation zone and that is monitored in a continually staffed primary alarm station and in one additional continually staffed location. The primary alarm station must be located within the protected area; have bullet-resisting walls, doors, ceiling, and floor; and the interior of the station must not be visible from outside the protected area. A timely means for assessment of alarms must also be provided. Regarding alarm monitoring, the redundant location need only provide a summary indication that an alarm has been generated.

(4) The protected area must be monitored by daily random patrols.

(5) A security organization with written procedures must be established. The security organization must include sufficient personnel per shift to provide for monitoring of detection systems and the conduct of surveillance, assessment, access control, and communications to assure adequate response. Members of the security organization must be trained, equipped, qualified, and requalified to perform assigned job duties in accordance with appendix B to [*26963] part 73, sections I.A, (1) (a) and (b), B(1)(a), and the applicable portions of II.

(6) Documented liaison with a designated response force or local law enforcement agency (LLEA) must be established to permit timely response to unauthorized penetration or activities.

(7) A personnel identification system and a controlled lock system must be established and maintained to limit access to authorized individuals.

(8) Redundant communications capability must be provided between onsite security force members and designated response force or LLEA.

(9) All individuals, vehicles, and hand-carried packages entering the protected area must be checked for proper authorization and visually searched for explosives before entry.

(10) Written response procedures must be established and maintained for addressing unauthorized penetration of, or activities within, the protected area including Category 5, "Procedures," of appendix C to part 73. The licensee shall retain a copy of response procedures as a record for 3 years or until termination of the license for which the procedures were developed. Copies of superseded material must be retained for 3 years after each change or until termination of the license.

(11) All detection systems, surveillance/assessment systems, and supporting subsystems, including illumination systems, must be tamper-indicating with line supervision and be maintained in operable condition. Timely compensatory measures must be taken after discovery of inoperability, to assure that the effectiveness of the security system is not reduced.

(12) The physical protection program must be reviewed once every 24 months by individuals independent of both physical protection program management and personnel who have direct responsibility for implementation of the physical protection program. The physical protection program review must include an evaluation of the effectiveness of the physical protection system and a verification of the liaison established with the designated response force or LLEA.

(13) The following documentation must be retained as a record for 3 years after the record is made or until termination of the license. Duplicate records to those required under § 72.180 of part 72 and § 73.71 of this part need not be retained under the requirements of this section:

(i) A log of individuals granted access to the protected area;

(ii) Screening records of members of the security organization;

(iii) A log of all patrols;

(iv) A record of each alarm received, identifying the type of alarm, location, date and time when received, and disposition of the alarm; and

(v) The physical protection program review reports.

(e) A licensee that operates a GROA is exempt from the requirements of this section for that GROA after permanent closure of the GROA.

13. In § 73.71, paragraphs (b)(1) and (c) are revised to read as follows:

§ 73.71 -- Reporting of safeguards events.

* * * * *

(b)(1) Each licensee subject to the provisions of § § 73.20, 73.37, 73.50, 73.51, 73.55, 73.60, or 73.67 shall notify the NRC Operations Center within 1 hour of discovery of the safeguards events described in paragraph I(a)(1) of appendix G to this part. Licensees subject to the provisions of § § 73.20, 73.37, 73.50, 73.51, 73.55, 73.60, or each licensee possessing strategic special nuclear material and subject to § 73.67(d) shall notify the NRC Operations Center within 1 hour after discovery of the safeguards events described in paragraphs I(a)(2), (a)(3), (b), and (c) of appendix G to this part. Licensees subject to the provisions of § § 73.20, 73.37, 73.50, 73.51, 73.55, or 73.60 shall notify the NRC Operations Center within 1 hour after discovery of the safeguards events described in paragraph I(d) of appendix G to this part.

* * * * *

(c) Each licensee subject to the provisions of § § 73.20, 73.37, 73.50, 73.51, 73.55, 73.60, or each licensee possessing SSNM and subject to the provisions of § 73.67(d) shall maintain a current log and record the safeguards events described in paragraphs II (a) and (b) of appendix G to this part within 24 hours of discovery by a licensee employee or member of the licensee's contract security organization. The licensee shall retain the log of events recorded under this section as a record for 3 years after the last entry is made in each log or until termination of the license.

* * * * *

PART 74--MATERIAL CONTROL AND ACCOUNTING OF SPECIAL NUCLEAR MATERIAL

14. The authority citation for part 74 continues to read as follows:

Authority: Secs. 53, 57, 161, 182, 183, 68 Stat. 930, 932, 948, 953, 954, as amended, sec. 234, 83 Stat. 444, as amended *(42 U.S.C. 2073,* 2077, 2201, 2232, 2233, 2282, 2297f); secs. 201, as amended 202, 206, 88 Stat. 1242, as amended, 1244, 1246 *(42 U.S.C. 5841,* 5842, 5846).

15. In § 74.51, the introductory text of paragraph (a) is revised to read as follows:

§ 74.51 – Nuclear material control and accounting for special nuclear material.

(a) *General performance objectives.* Each licensee who is authorized to possess five or more formula kilograms of strategic special nuclear material (SSNM) and to use such material at any site, other than a nuclear reactor licensed pursuant to part 50 of this chapter, an irradiated fuel reprocessing plant, an operation involved with waste disposal, or an independent spent fuel storage facility licensed pursuant to part 72 of this chapter shall establish, implement, and maintain a Commission-approved material control and accounting (MC&A) system that will achieve the following objectives:

* * * * *

PART 75–SAFEGUARDS ON NUCLEAR MATERIAL–IMPLEMENTATION OF US/IAEA AGREEMENT

16. The authority citation for part 75 continues to read as follows:

Authority: Secs. 53, 63, 103, 104, 122, 161, 68 Stat. 930, 932, 936, 937, 939, 948, as amended *(42 U.S.C. 2073,* 2093, 2133, 2134, 2152, 2201); sec. 201, 88 Stat. 1242, as amended *(42 U.S.C. 5841).*

Section 75.4 also issued under secs. 135, 141, Pub. L. 97-425, 96 Stat. 2232, 2241 *(42 U.S.C. 10155,* 10161).

17. In § 75.4, paragraph (k)(5) is revised to read as follows:

§ 75.4 – Definitions.

* * * * *

(k) * * *

(5) Any location where the possession of more than 1 effective kilogram of nuclear material is licensed pursuant to parts 40, 60, or 70 of this chapter, or pursuant to an agreement state license.

* * * * *

Dated at Rockville, Maryland, this 11th day of May, 1998.

For the Nuclear Regulatory Commission.

John C. Hoyle,

Secretary of the Commission.

[FR Doc. 98-12978 Filed 5-14-98; 8:45 am]

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Standard Review Plan
for the Review of Safety Analysis Reports for Nuclear Power Plants

LWR Edition

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

June 1987

BOOK 1

NUREG-0800
(Formerly NUREG-75/087)

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

3.5.1.6 AIRCRAFT HAZARDS

REVIEW RESPONSIBILITIES

Primary - Siting Analysis Branch (SAB)

Secondary - None

I. AREAS OF REVIEW

The staff reviews the applicant's assessment of aircraft hazards. The purpose of the review is to assure that the risks due to aircraft hazards are sufficiently low. Probabilistic considerations may be used to demonstrate that aircraft hazards need not be a design basis concern. Otherwise, design basis aircraft identification is made and the applicant's plant design is evaluated to assure that it is protected against the potential effects of aircraft impacts and fires.

The SAB reviews the applicant's assessment of aircraft hazards to the plant and determines whether or not they should be incorporated into the plant design basis. If the aircraft hazards are incorporated into the plant design basis, the SAB identifies and describes the design basis aircraft in terms of aircraft weight, speed, and other appropriate characteristics.

On request by SAB, the following branches with primary review responsibility will review specific aspects of aircraft hazards:

1. The Structural Engineering Branch (SEB), in the area of missile effects (SRP Section 3.5.3), with respect to aircraft impacts,

2. The Chemical Engineering Branch (CMEB), in the area of fire protection (SRP Section 9.5.1), with respect to aircraft fires, and

3. The Auxiliary Systems Branch (ASB), in the area of structures, systems, and components (SSC) important to safety (SRP Section 3.5.2), with respect to protection requirements against aircraft crashes.

For those areas of review identified above as being part of the primary responsibility of other branches, the acceptance criteria necessary for the review and the methods of their application are contained in the referenced SRP sections of the corresponding primary branches.

5. The Applied Statistics Branch (ASB/MPA) will provide technical review support with respect to aircraft accident statisics.

II. ACCEPTANCE CRITERIA

SAB acceptance criteria are based on meeting the relevant requirements of one of the following sets of regulations:

1. 10 CFR Part 100, §100.10 as it relates to indicating that the site location, in conjunction with other considerations (such as plant design, construction, and operation), should insure a low risk of public exposure. This requirement is met if the probability of aircraft accidents resulting in radiological consequences greater than 10 CFR Part 100 exposure guidelines is less than about 10^{-7} per year (see SRP Section 2.2.3). The probability is considered to be less than about 10^{-7} per year by inspection if the distances from the plant meet all the requirements listed below:

   (a) The plant-to-airport distance D is between 5 and 10 statute miles, and the projected annual number of operations is less than 500 D^2, or the plant-to-airport distance D is greater than 10 statute miles, and the projected annual number of operations is less than 1000 D^2,

   (b) The plant is at least 5 statute miles from the edge of military training routes, including low-level training routes, except for those associated with a usage greater than 1000 flights per year, or where activities (such as practice bombing) may create an unusual stress situation,

   (c) The plant is at least 2 statute miles beyond the nearest edge of a federal airway, holding pattern, or approach pattern.

If the above proximity criteria are not met, or if sufficiently hazardous military activities are identified (see item b above), a detailed review of aircraft hazards must be performed. Aircraft accidents which could lead to radiological consequences in excess of the exposure guidelines of 10 CFR Part 100 with a probability of occurrence greater than about 10^{-7} per year should be considered in the design of the plant. If the results of the review do not support a finding that the risk due to aircraft activities is acceptably low, then the design basis acceptance criteria outlined in Item II.2 below applies.

2. General Design Criterion (GDC) 4 of 10 CFR Part 50 (Ref. 13), Appendix A, requires that structures, systems, and components (SSC) important to safety be appropriately protected against the effects of missiles that may result from events and conditions outside the nuclear power unit. GDC 3 of 10 CFR Part 50, Appendix A, requires that SSC important to safety be appropriately protected against the effects of fires. The plant meets the relevant requirements of GDC 3 and GDC 4, and is considered appropriately protected against design basis aircraft impacts (Ref. 6) and fires (Ref. 3) if the SSC important to safety are capable of withstanding the effects of the

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postulated aircraft impacts and fires without loss of safe shutdown capability, and without causing a release of radioactivity which would exceed 10 CFR Part 100 dose guidelines.

The safety-related SSC to be considered with respect to the above acceptance criteria include those described in the Appendix to Regulatory Guide 1.117, "Structures, Systems, and Components of Light-Water-Cooled Reactors to be Protected Against Tornadoes." Other safety-related SSC, which may not be included in Regulatory Guide 1.117, will be considered on a case-by-case basis in accordance with the acceptance criteria of the appropriate branches having primary responsibility for their protection.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes aspects of the areas covered by this SRP section as may be appropriate for a particular case. The judgment on areas to be given attention and emphasis in the review is based on a inspection of the material presented to see whether it is similar to that recently reviewed on other plants and whether items of special safety significant are involved.

The staff's review of the aircraft hazard assessment consists of the following steps:

1. Aviation Uses. Data desribing aviation uses in the airspace near the proposed site, including airports and and their approach paths, federal airways, Federal Aviation Administration (FAA) restricted areas, and military uses is obtained from Section 2.2.1-2.2.2 of the SAR. For many cases, no detailed analysis need be made as the probability can be judged adequately low based on a comparison with analyses previously performed (Refs. 5, 7, 8, 9 and 10). In general, civilian and military maps should be examined to verify that all aviation facilities of interest have been considered. In the process, the reviewer should develop an independent assessment of the aircraft hazards. Communications with agencies responsible for aircraft operations and the evaluation of aircraft operational data may be utilized.

2. Airways. For situations where federal airways or aviation corridors pass through the vicinity of the site, the probability per year of an aircraft crashing into the plant (P_{FA}) should be estimated. This probability will depend on a number of factors such as the altitude and frequency of the flights, the width of the corridor, and the corresponding distribution of past accidents.

   One way of calculating P_{FA} is by using the following expression:

   $$P_{FA} = C \times N \times A/w$$

   where:

   C = inflight crash rate per mile for aircraft using airway,

   w = width of airway (plus twice the distance from the airway edge to the site when the site is outside the airway) in miles,

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N = number of flights per year along the airway, and

A = effective area of plant in square miles.

This gives a conservative upper bound on aircraft impact probability if care is taken in using values for the individual factors that are meaningful and conservative. For commercial aircraft a value of $C = 4 \times 10^{-10}$ (Ref. 11) per aircraft mile has been used. For heavily traveled corridors (greater than 100 flights per day), a more detailed analysis may be required to obtain a proper value for this factor.

3. Civilian and Military Airports and Heli-Ports (Refs. 2, 4, and 14). The probability of an aircraft crashing into the site should be estimated for cases where one or more of the conditions in Item II.1 of the Acceptance Criteria are not met.

The probability per year of an aircraft crashing into the site for these cases (P_A) may be calculated by using the following expression:

$$P_A = \sum_{i=1}^{L} \sum_{j=1}^{M} C_j N_{ij} A_j$$

where:

M = number of different types of aircraft using the airport,
L = number of flight trajectories affecting the site,
C_j = probability per square mile of a crash per aircraft movement, for the jth aircraft,
N_{ij} = number (per year) of movements by the jth aircraft along the ith flight path, and
A_j = effective plant area (in square miles) for the jth aircraft.

The manner of interpreting the individual factors in the above equation may vary on a case-by-case basis because of the specific conditions of each case or because of changes in aircraft accident statistics.

Values for C_j currently being used are taken from the data summarized in the following table:

Distance From End of Runway (miles)	Probability ($\times 10^8$) of a Fatal Crash per Square Mile per Aircraft Movement			
	U.S. Air Carrier[1]	General Aviation[2]	USN/USMC[1]	USAF[1]
0-1	16.7	84	8.3	5.7
1-2	4.0	15	1.1	2.3
2-3	0.96	6.2	0.33	1.1
3-4	0.68	3.8	0.31	0.42
4-5	0.27	1.2	0.20	0.40
5-6	0	NA[3]	NA	NA
6-7	0	NA	NA	NA
7-8	0	NA	NA	NA
8-9	0.14	NA	NA	NA
9-10	0.12	NA	NA	NA

[1]Reference 2.
[2]Reference 4.
[3]NA indicates that data was not available for this distance.

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4. Designated Airspaces. For designated airspaces involving military or civilian usage, a detailed quantitative modeling of all operations should be verified. The results of the model should be the total probability (C) of an aircraft crash per unit area and time in the vicinity of the proposed site.

The probability per year of a potentially damaging crash at the site due to operations at the facility under consideration (P_M) is then given for this case by the following expression:

$$P_M = C \times A$$

where:

C = total probability of an aircraft crash per square mile per year in the vicinity of the site due to the airports being considered, and

A = effective area of one unit of the plant in square miles.

Where estimated risks due to military aircraft activity are found to be unacceptably high, suitable airspace or airway relocation should be implemented. Past experience has been that military authorities have been responsive to modification of military operations and relocation of training routes in close proximity to nuclear power plant sites. (Ref. 12)

5. Holding Patterns. Holding patterns are race track shaped courses at specified altitudes, associated with one or more radio-navigational facilities, where aircraft can "circle" while awaiting clearance to execute an approach to a landing at an airport or to continue along an airway. Holding patterns which are sufficiently distant from the plant need not be considered (See subsection II above). Otherwise, traffic in the holding pattern should be converted into equivalent aircraft passages taking into account the characteristics, including orientation with respect to the plant, of the holding pattern. The information in Item III.2 above should be used in this evaluation.

6. The total aircraft hazard probability at the site equals the sum of the individual probabilities obtained in the preceding steps.

7. The effective plant areas used in the calculations should include the following:

   a. A shadow area of the plant elevation upon the horizontal plane based on the assumed crash angle for the different kinds of aircraft and failure modes.

   b. A skid area around the plant as determined by the characteristics of the aircraft under consideration. Artificial berms or any other man-made and natural barriers should be taken into account in calculating this area.

   c. The areas of those safety-related SSC which are susceptible to impact or fire damage as a result of aircraft crashes.

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IV. EVALUATION FINDINGS

The reviewer drafts an introductory paragraph for the evaluation findings describing the procedure used in evaluating the aircraft hazards with respect to the safety-related SSC. The reviewer verifies that the site location is acceptable and meets the requirements of 10 CFR Part 100, §100.10.

The basis for the above findings may be strictly in terms of the probabilities associated with potential aircraft crashes onsite. If the aircraft crash statistics applicable to the onsite facilities are such that SRP Section 2.2.3 criteria are met without explicit consideration of plant design features, then conclusions of the following type should be included in the staff's safety evaluation report:

> The staff concludes that the operation of the _______ plant in the vicinity of _________ does not present an undue risk to the health and safety of the public and meets the relevant requirements of 10 CFR Part 100, §100.10. This conclusion is based on the staff's independent verification of the applicant's assessment of aircraft hazards at the site that resulted in a probability less than about 10^{-7} per year for an accident having radiological consequences worse than the exposure guidelines of 10 CFR Part 100.
>
> In addition, plant sites reviewed in the past which had equivalent aircraft traffic in equal or closer proximity were, after careful examination, found to present no undue risk to the safe operation of those plants. Based upon this experience, in the staff's judgment, no undue risk is present from aircraft hazard at the plant site now under consideration.

In the event that the staff evaluation of the aircraft hazards does not support the above basis, i.e., if SRP Section 2.2.3 criteria are not met, then the basis for acceptance is derived from applying GDC 3 and GDC 4 criteria. If the protection against aircraft impacts and fires is such that the plant safety-related SSC meet GDC 3 and GDC 4 criteria, then 10 CFR Part 100 requirements are considered to be met and conclusion of the following type may be included in the staff's safety evaluation report:

> The staff concludes that the operation of the _______ plant in the vicinity of _______ does not present an undue risk to the health and safety of the public due to aircraft hazards and meets the relevant requirements of General Design Criteria 3 and 4. This conclusion is based on the staff having independently verified the applicant's assessment of aircraft hazards, including aircraft fires and impacts, at the site and that if the appropriate safety-related structures, systems, and components are designed to withstand the aircraft selected as the design basis aircraft, the probability of an aircraft strike causing radiological consequences in excess of the exposure guidelines of 10 CFR Part 100 is less than about 10^{-7} per year.

V. IMPLEMENTATION

The following is intended to provide guidance to applicants and licensees regarding the NRC staff's plans for using this SRP section.

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Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, and method described herein will be used by the staff in its evaluation of conformance with Commission regulations.

Implementation schedules for conformance to parts of the method discussed herein are contained in the referenced regulatory guides and NUREG.

VI. REFERENCES

1. 10 CFR Part 100, "Reactor Site Criteria."

2. D. G. Eisenhut, "Reactor Siting in the Vicinity of Airfields." Paper presented at the American Nuclear Society Annual Meeting, June 1973.

3. I. I. Pinkel, "Appraisal of Fire Effects from Aircraft Crash at Zion Power Reactor Facility," July 17, 1972 (Docket No. 50-295).

4. D. G. Eisenhut, "Testimony on Zion/Waukegan Airport Interaction" (Docket No. 50-295).

5. USAEC Regulatory Staff, "Safety Evaluation Report," Appendix A, "Probability of an Aircraft Crash at the Shoreham Site" (Docket No. 50-322).

6. "Addendum to the Safety Evaluation by the Division of Reactor Licensing, USAEC, in the Matter of Metropolitan Edison Company (Three Mile Island Nuclear Station Unit 1, Dauphin County, Pennsylvania)," April 26, 1968 (Docket No. 50-289).

7. Letter to Honorable J. R. Schlesinger from S. H. Bush, Chairman, Advisory Committe on Reactor Safeguards, "Report on Rome Point Nuclear Generating Station," November 18, 1971 (Project No. 455).

8. Letter to Mr. Joseph L. Williams, Portland General Electric Company, from R. C DeYoung (in reference to Mr. Williams' letter of May 7, 1973), November 23, 1973 (Project No. 485).

9. "Aircraft Considerations-Preapplication Site Review by the Directorate of Liensing, USAEC, in the Matter of Portland General Electric Company, Boardman Nuclear Plant, Boardman, Oregon," October 12, 1973 (Project No. 485).

10. Letter to Mr. J. H. Campbell, Consumers Power Company, from Col. James M. Campbell, Dep. Chief, Strategic Division, Directorate of Operations, U.S. Air Force, May 19, 1971 (Docket No. 50-155).

11. H. E. P. Krug, "Testimony on Aircraft Operations in Response to a Question from the Board" (Docket Nos. 50-275 and 50-323).

12. Letter to Mr. J. H. Campbell, Consumers Power Company, from Col. James M. Campbell, Dep. Chief, Strategic Division, Directorate of Operations, U.S. Air Force, May 19, 1971 (Docket No. 50-155).

13. 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."

14. NUREG-0533, "Aircraft Impact Risk Assessment Data Base for Assessment of Fixed Wing Air Carrier Impact Risk in the Vicinity of Airports."

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NUREG-0654
FEMA-REP-1
Rev. 1

Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants

U.S. Nuclear Regulatory Commission

UNITED STATES NUCLEAR REGULATORY COMMISSION

Federal Emergency Management Agency

000749

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Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants

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U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Federal Emergency Management Agency
Washington D.C. 20472

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FOREWORD

The purpose of this guidance and upgraded acceptance criteria is to provide a basis for NRC licensees, State and local governments to develop radiological emergency plans and improve emergency preparedness. The guidance is the product of the joint FEMA/NRC Steering Committee established to coordinate the agencies' work in emergency preparedness associated with nuclear power plants. The interim version of this document was published in January 1980, and subjected to public comment under Federal Register Notice 44 FR 9768 of February 13, 1980. Based upon the comments received, meetings with the Interorganizational Advisory Committee (made up of State and local representatives) and later at a September 1980 Workshop sponsored by FEMA for State officials, the final version was prepared for publication. The principal changes in the document consist of clarification of intent and accommodation of many of the unique situations which arise in State/local/utility interfaces. Therefore, plans prepared using the interim guidance should not require substantial revision. This document is consistent with NRC and FEMA regulations and supersedes other previous guidance and criteria published by FEMA and NRC on this subject. It will be used by reviewers in determining the adequacy of State, local and nuclear power plant licensee emergency plans and preparedness.

October 1980

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000752

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ACKNOWLEDGEMENT

The Steering Committee acknowledges with thanks the contributions of the Interorganizational Advisory Committee on Radiological Emergency Planning and Preparedness (IOAC) of the Conference of (State) Radiation Control Program Directors. Members of this Committee also include representatives of the National Emergency Management Association and the U. S. Civil Defense Council. The Steering Committee also acknowledges with thanks the contributions made by other Federal agencies, State and local governmental organizations, members of the public, and the nuclear industry.

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U. S. Nuclear Regulatory Commission - Federal Emergency Management Agency

CRITERIA FOR PREPARATION AND EVALUATION OF RADIOLOGICAL EMERGENCY RESPONSE PLANS AND PREPAREDNESS IN SUPPORT OF NUCLEAR POWER PLANTS

I. INTRODUCTION

A. Purpose

The purpose of this document is to provide a common reference and guidance source for:

1. State and local governments and nuclear facility operators in the develoment of radiological emergency response plans and preparedness in support of nuclear power plants.

2. Federal Emergency Management Agency (FEMA), Nuclear Regulatory Commission (NRC), and other Federal agency personnel engaged in the review of State, local government and licensee plans and preparedness.

3. The Federal Emergency Management Agency, the Nuclear Regulatory Commission and other Federal agencies in the development of the National Radiological Emergency Preparedness Plan.

B. Background

The NRC and FEMA staff have prepared this document as part of their responsibilities under the Atomic Energy Act, as amended, and the President's Statement of December 7, 1979, with the accompanying

B. Background (continued)

Fact Sheet. These responsibilities include development and promulgation of guidance to nuclear facility operators, States and local governments, in cooperation with other Federal agencies, for the preparation of radiological emergency response plans and assessing the adequacy of such plans.1/

This guidance is classified as final guidance. The interim version of this guidance, published in January, 1980, was commented upon by interested parties during the formal public comment period solicited by the Federal Register Notice 44 FR 9768 of February 13, 1980. Additionally, comments received on "Draft Emergency Action Level Guidelines", (September 1979), NUREG-0610 solicited by Federal Register Notice 44 FR 55446 of September 26, 1979 were also considered in the revision to Appendix 1 of the criteria document. A separate document has been prepared by NRC and FEMA which lists the comments received and which indicates the NRC and FEMA response to these comments. FEMA, NRC, and other involved Federal agencies intend to use the guidance contained in this document in their individual and joint reviews of State and local government radiological emergency response plans and preparedness, and of the plans and preparedness of NRC facility licensees. The NRC Final Rule on Emergency Planning

1/ In light of the President's Statement of December 7, 1979, the agency responsibilities assigned on January 24, 1973 by the Office of Emergency Preparedness, (and later reassigned on December 24, 1975 by the Federal Preparedness Agency/GSA) are being revised and will be promulgated in the near future by FEMA.

B. Background (continued)

(45 FR 55402) of August 19, 1980 has an effective date of November 3, 1980. This document is supportive of the NRC Final Rule and is referenced therein. This document is also supportive of the proposed FEMA Rule concerning the review and approval of State and local radiological emergency plans and preparedness, which at this writing is in the process of revision as a result of comments received during the public comment period.

NRC has now established a schedule for the implementation of the "Minimum Staffing Requirements for NRC Licensees for Nuclear Power Plant Emergencies" set forth in Table B-1, (see II.B.5), and for Appendix 2, "Meteorological Criteria for Emergency Preparedness at Operating Nuclear Power Plants" (see Annex to Appendix 2).

C. Scope

This document is concerned with accidents at fixed commercial nuclear power reactors which might have impact on public health and safety.[2/]

2/ Many of the planning elements contained in this guide may be useful for planners in the vicinity of test and research reactors, fuel processing plants, or other facilities using or producing large quantities of radioactive material. None of the numerical values in this document need be used for planning at such facilities. Similarly, while some planning elements presented here may apply to transportation accidents involving radioactive material, such accidents have unique characteristics which warrant separate guidance. These accidents are not specifically covered in this document and will be the subject of future guidance.

C. Scope (continued)

The guidance intended for use by NRC licensees and operators of commercial nuclear power reactors is based upon several existing documents familiar to such operators: first, NRC Regulatory Guide 1.101 (March 1977); second, NRC's letters of October 10, 1979 and November 29, 1979 to its power reactor licensees; third, NRC's final rule including the revised Appendix E to 10 CFR Part 50 and fourth, NRC's NUREG-0610, "Draft Emergency Action Level Guidelines for Nuclear Power Plants," September 1979, the revised version of which is Appendix 1 to this document.

The guidance intended for use by State and local governments has been drawn in large part from existing documents already familiar to planners: first, the NRC Guide and Checklist for the Development and Evaluation of State and Local Government Radiological Emergency Response Plans in Support of Fixed Nuclear Facilities, NUREG 75/111 (1974) and its Supplement No. 1 (March 1977); and second, guidance on the planning basis contained in the Report of the NRC/EPA Task Force on Emergency Planning, NUREG-0396, EPA 520/1-78-016 (December 1978). The Guide and Checklist, its supplement and the NRC/EPA Task Force Report, were subjected to very broad State and local government reviews prior to publication, in both draft and final form. NRC specifically endorsed the guidance contained in each of these documents. NRC's formal policy statement on the Emergency

C. Scope (continued)

Planning Zone concept was published in the Federal Register of October 23, 1979, (44 FR 61123). EPA's endorsement of the Emergency Planning Zone concept was published in the Federal Register of January 15, 1980 (45 FR 2893). This document supersedes NUREG 75/111 and Regulatory Guide 1.101. As in the January, 1980 version of this document, FEMA formally endorses this guidance concerning Emergency Planning Zones and urges its immediate use by States and local governments and by NRC licensed nuclear power plant operators. Also included in this document are some obvious lessons learned during and after the accident at Three Mile Island. The criteria put added emphasis on the following elements: Notification Methods and Procedures, Emergency Communications, Public Education and Information, Emergency Facilities and Equipment, Accident Assessment, and Exercises and Drills. FEMA and NRC regard all of the planning standards identified and contained herein as essential for an adequate radiological emergency plan.

D. Planning Basis

1. Background

The NRC/EPA Task Force Report on Emergency Planning, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants, NUREG-0396, EPA 520/1-78-016" provides a planning basis

D. Planning Basis (continued)

for offsite emergency preparedness efforts considered necessary and prudent for large power reactor facilities. The NRC's policy statement of October 23, 1979 (44 FR 61123), directs the NRC staff to incorporate the guidance in the report into emergency preparedness documents. Additionally, the guidance in the NRC/EPA Task Force Report on Emergency Planning is now reflected in the NRC Final Rule on Emergency Planning. FEMA has also concluded that the guidance in NUREG-0396 should be used as the planning basis for emergency preparedness around nuclear power facilities.

The overall objective of emergency response plans is to provide dose savings (and in some cases immediate life saving) for a spectrum of accidents that could produce offsite doses in excess of Protective Action Guides (PAGs).[3/,4/] No single specific accident sequence should be isolated as the one for which to plan because each accident could have different consequences, both in nature and degree. Further, the range of possible selection for a planning basis is very large, starting with a zero point of requiring no planning at all because significant offsite radiological accident consequences are unlikely to occur,

3/ Manual of Protective Action Guides and Protective Actions for Nuclear Incidents, EPA-520/1-75-001, September 1975, U. S. Environmental Protection Agency.

4/ Accidental Radioactive Contamination of Human Food and Animal Feeds, U. S. Department of Health, Education and Welfare (now U. S. Department of Health and Human Services), 43 FR 58790 of December 15, 1978.

D. Planning Basis (continued)

to planning for the worst possible accident, regardless of its extremely low likelihood. The NRC/EPA Task Force did not attempt to define a single accident sequence or even a limited number of sequences. Rather, it identified the bounds of the parameters for which planning is recommended, based upon knowledge of the potential consequences, timing, and release characteristics of a spectrum of accidents. Although the selected planning basis is independent of specific accident sequences, a number of accident descriptions were considered in the development of the guidance, including the core melt accident release categories of the Reactor Safety Study.

The most important guidance in the Report for planning officials is the definition of the area over which planning for predetermined actions should be carried out.

Information on the time frames of accidents is also important. The time between the initial recognition at the nuclear facility that a serious accident is in progress and the beginning of the radioactive release to the surrounding environment is critical in determining the type of protective actions which are feasible. Knowledge of the potential duration of release and the time

D. Planning Basis (continued)

available before exposures are expected several miles offsite is important in determining what specific instructions can be given to the public.

A knowledge of kinds of radioactive materials potentially released is necessary to decide the characteristics of monitoring instrumentation, to develop tools for estimating projected doses, and to identify the most important exposure pathways.

The need for specification of areas for the major exposure pathways is evident. The location of the population for whom protective measures may be needed, responsible authorities who would carry out protective actions and the means of communication to these authorities and to the population are all dependent on the characteristics of the planning areas. Emergency preparedness should be related to two predominant exposure pathways. They are:

a. Plume exposure pathway -- The principal exposure sources from this pathway are: (a) whole body external exposure to gamma radiation from the plume and from deposited material; and (b) inhalation exposure from the passing radioactive plume. The duration of the release leading to potential exposure could range from one-half hour to

D. Planning Basis (continued)

days. For the plume exposure pathway, shelter and/or evacuation would likely be the principal immediate protective actions to be recommended for the general public. When evacuation is chosen as the preferred protective measure, initial evacuation of a 360° area around the facility is desirable out to a distance of about two to five miles although initial efforts would, of course, be in the general downwind direction. This concept is indicated in Figure 1. The precise boundaries of such evacuations and sectors evacuated at extended downwind distances would be largely determined by political boundaries and would not fit the precise pattern of Figure 1. The possible administration of the thyroid blocking agent, potassium iodide, should also be considered.[5/] The U. S. Department of Health and Human Services (DHHS) is preparing guidance on the potassium iodide issue which will be considered by NRC and FEMA. The ability to best reduce potential exposure under the specific conditions during the course of an accident should determine the appropriate response.

b. Ingestion exposure pathway -- The principal exposure from this pathway would be from ingestion of contaminated water or foods such as milk, fresh vegetables or aquatic foodstuffs.

5/ Potassium Iodide as a Thyroid-Blocking Agent in a Radiation Emergency, U. S. Department of Health, Education and Welfare (now U. S. Department of Health and Human Services), 43 FR 58798 of December 15, 1978.

D. Planning Basis (continued)

The duration of potential exposure could range in length from hours to months. For the ingestion exposure pathway, the planning effort involves the identification of major exposure pathways from contaminated food and water and the associated control and interdiction points and methods. The ingestion pathway exposures in general would represent a longer term problem, although some early protective actions to minimize subsequent contamination of milk or other supplies should be initiated (e.g., remove cows from pasture and put them on stored feed).

Separate guidance is provided for these two exposure pathways, although emergency plans for a particular site will include elements common to assessing or taking protective actions for both pathways.

2. Emergency Planning Zones

With regard to the area over which planning efforts should be carried out, "Emergency Planning Zones" (EPZs) about each nuclear facility must be defined both for the short term "plume exposure pathway" and for the longer term "ingestion exposure pathways." The Emergency Planning Zone concept is illustrated in Figure 1. EPZs are defined as the areas for which planning is needed to assure that prompt and effective actions can be taken to protect the public in the event of an accident. The criteria in NUREG-0396 are to be applied by the response organizations in these

D. Planning Basis (continued)

zones as applicable. The NRC/EPA Task Force Report on Emergency Planning (NUREG-0396, EPA 520/1-78-016) anticipates that State, rather than local, response organizations will be principally responsible for the planning associated with the ingestion exposure pathway.

The choice of the size of the Emergency Planning Zones represents a judgment on the extent of detailed planning which must be performed to assure an adequate response base. In a particular emergency, protective actions might well be restricted to a small part of the planning zones. On the other hand, for the worst possible accidents, protective actions would need to be taken outside the planning zones.

The Task Force selected a radius of about 10 miles for the plume exposure pathway and a radius of about 50 miles for the ingestion exposure pathway, as shown in Figure 1 and in Table 1.[6/] Although the radius for the EPZ implies a circular area, the actual shape would depend upon the characteristics of a particular site.

6/ These radii are applicable to light water nuclear power plants, rated at 250 MWt or greater. The FEMA/NRC Steering Committee has concluded that small water cooled power reactors (less than 250 MWt) and the Fort St. Vrain gas cooled reactor may use a plume exposure emergency planning zone of about 5 miles in radius and an ingestion pathway emergency planning zone of about 30 miles in radius. In addition, the requirements for the alerting and notification system (Appendix 3) will be scaled on a case-by-case basis. This conclusion is based on the lower potential hazard from these facilities (lower radionuclide inventory and longer times to release significant amounts of activity for many accident scenarios). The radionuclides considered in planning should be the same as recommended in NUREG-0396/EPA-520/1-78-016.

D. Planning Basis (continued)

The size (about 10 miles radius) of the plume exposure EPZ was based primarily on the following considerations:

a. projected doses from the traditional design basis accidents would not exceed Protective Action Guide levels outside the zone;

b. projected doses from most core melt sequences would not exceed Protective Action Guide levels outside the zone;

c. for the worst core melt sequences, immediate life threatening doses would generally not occur outside the zone;

d. detailed planning within 10 miles would provide a substantial base for expansion of response efforts in the event that this proved necessary.

The NRC/EPA Task Force concluded that it would be unlikely that any protective actions for the plume exposure pathway would be required beyond the plume exposure EPZ. Also, the plume exposure EPZ is of sufficient size for actions within this zone to provide for substantial reduction in early severe health effects (injuries or deaths) in the event of a worst case core melt accident.

The size of the ingestion exposure EPZ (about 50 miles in radius, which also includes the 10-mile radius plume exposure EPZ) was selected because:

D. Planning Basis (continued)

a. the downwind range within which contamination will generally not exceed the Protective Action Guides is limited to about 50 miles from a power plant because of wind shifts during the release and travel periods;

b. there may be conversion of atmospheric iodine (i.e., iodine suspended in the atmosphere for long time periods) to chemical forms which do not readily enter the ingestion pathway;

c. much of any particulate material in a radioactive plume would have been deposited on the ground within about 50 miles from the facility; and

d. the likelihood of exceeding ingestion pathway protective action guide levels at 50 miles is comparable to the likelihood of exceeding plume exposure pathway protective action guide levels at 10 miles.

3. Time Factors Associated with Releases

The range of times between the onset of accident conditions and the start of a major release is of the order of one-half hour to several hours. The subsequent time period over which radioactive material may be expected to be released is of the order of one-half hour (short-term release) to a few days (continuous release). Table 2 summarizes the guidance on the time of the release, which

D. Planning Basis (continued)

has been used in developing the criteria for notification capabilities in Part II. (Other reasons for requiring prompt notification capabilities include faster moderate releases for which protective actions are desirable and the need for substantial lead times to carry out certain protective measures, such as evacuation, when this is indicated by plant conditions.)

4. Radiological Characteristics of Releases

Planners will need information on the characteristics of potential radioactivity releases in order to specify the characteristics of monitoring instrumentation,[7/] develop decisional aids to estimate projected doses, and identify critical exposure modes.

For atmospheric releases from nuclear power facilities, three dominant exposure modes have been identified: (a) whole body (bone marrow) exposure from external gamma radiation and from ingestion of radioactive material; (b) thyroid exposure from inhalation or ingestion of radioiodines; and (c) exposure of other organs (e.g., lung) from inhalation or ingestion of radioactive materials. Any of these exposure modes could dominate (i.e., result in the largest exposures) depending upon the relative quantities of various isotopes released.

7/ An interagency Task Force on Emergency Instrumentation (offsite) is now preparing guidance on offsite radiation measurement systems, accident assessment techniques, and the type and quantity of instruments needed for the various exposure pathways. Federal agencies represented on the Instrumentation Task Force include FEMA, NRC, EPA, HEW, and DOE.

D. Planning Basis (continued)

Radioactive materials produced in the operation of nuclear reactors include fission products, transuranics and activation products generated by neutron exposure of the structural and other materials within and immediately around the reactor core. The fission products consist of a very large number of different kinds of isotopes (nuclides), almost all of which are initially radioactive. The amounts of these fission products and their potential for escape from their normal places of confinement represent the dominant potential for consequences to the public. Radioactive fission products exist in a variety of physical and chemical forms of varied volatility. Virtually all activation products and transuranics exist as non-volatile solids. The characteristics of these materials show quite clearly that the potential for releases to the environment decreases dramatically in this order: (a) gaseous materials; (b) volatile solids, and (c) non-volatile solids. For this reason, guidance for source terms representing hypothetical fission product activity within a nuclear power plant containment structure emphasizes the development of plans relating to the release of noble gases and/or volatiles such as iodine. Consideration of particulate materials, however, should not be completely neglected. For example, capability to determine the presence or absence of key particulate radionuclides will be needed to identify requirements for additional resources. Table 3 provides a list of dominant radionuclides for each exposure pathway.

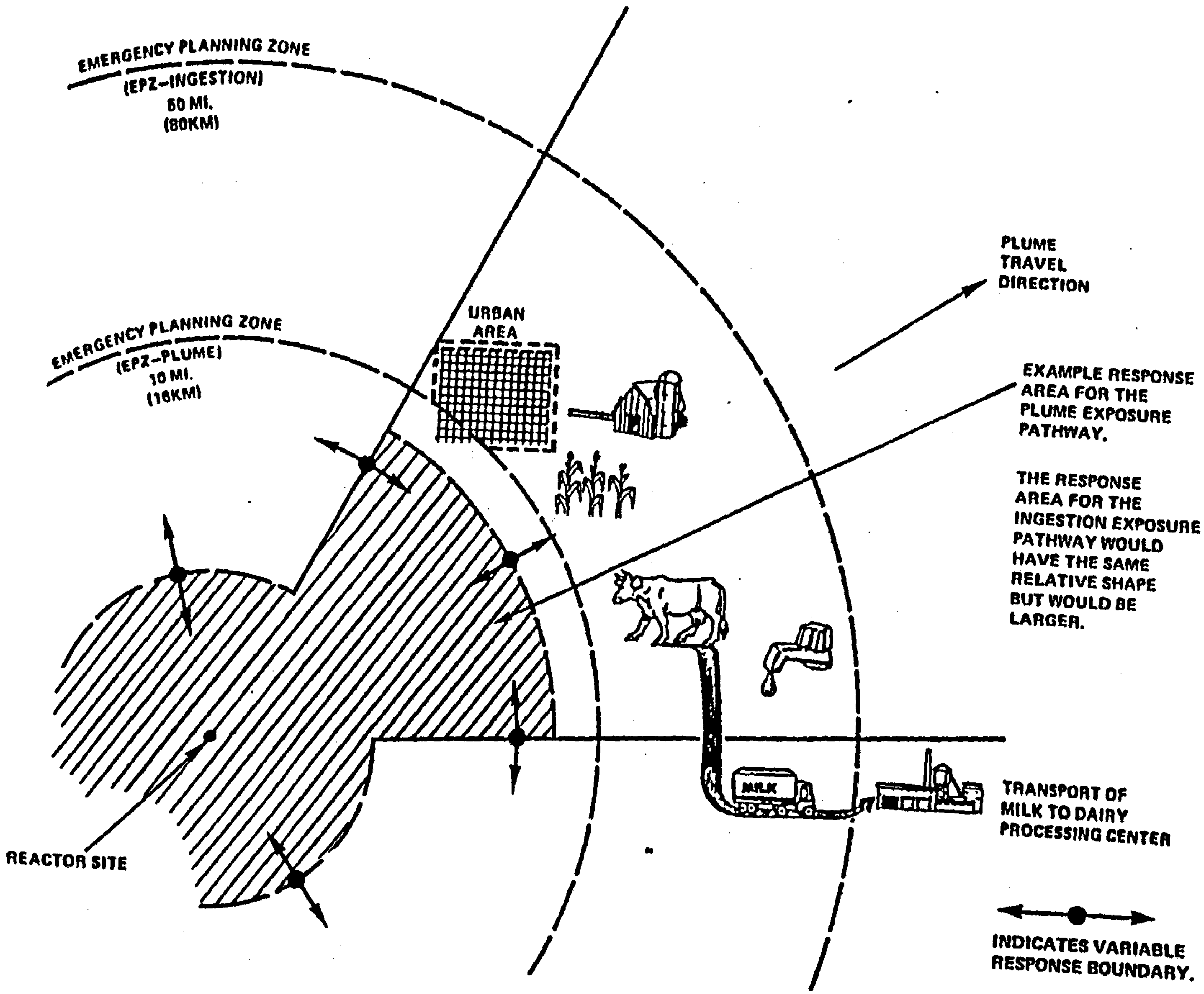

Figure 1 Concept of Emergency Planning Zones

TABLE 1

GUIDANCE ON SIZE OF THE EMERGENCY PLANNING ZONE

Accident Phase	Critical Organ and Exposure Pathway	EPZ Radius
Plume Exposure Pathway	Whole Body (external)	about 10 mile radius*
	Thyroid (inhalation)	
	Other organs (inhalation)	
Ingestion Pathway	Thyroid, whole body, bone marrow (ingestion)	about 50 mile radius**

* Judgment should be used in adopting this distance based upon considerations of local conditions such as demography, topography, land characteristics, access routes, and local jurisdictional boundaries.

**Processing plants for milk produced within the EPZ should be included in emergency response plans regardless of their location.

TABLE 2

GUIDANCE ON INITIATION AND DURATION OF RELEASE

Time from the initiating event to start of atmospheric release	0.5 hours to one day
Time period over which radioactive material may be continuously released	0.5 hours to several days
Time at which major portion of release may occur	0.5 hours to 1 day after start of release
Travel time for release to exposure point (time after release)	5 miles -- 0.5 to 2 hours 10 miles - 1 to 4 hours

Table 3

RADIONUCLIDES WITH SIGNIFICANT CONTRIBUTION TO DOMINANT EXPOSURE MODES

Radionuclides with Significant Contribution to Thyroid Exposure

Radionuclide	Half Life (days)
I-131	8.05
I-132	0.0958
I-133	0.875
I-134	0.0366
I-135	0.280
Te-132	3.25

Radionuclides with Significant Contribution to Whole Body Exposure

Radionuclide	Half Life (days)
I-131	8.05
Te-132	3.25
Xe-133	5.28
I-133	0.875
Xe-135	0.384
I-135	0.280
Cs-134	750
Kr-88	0.117
Cs-137	11,000

Radionuclides with Significant Contribution to Lung Exposure* (Lung only controlling when thyroid dose is reduced by iodine blocking or there is a long delay prior to releases).

Radionuclide	Half Life (days)
I-131	8.05
I-132	0.0958
I-133	0.875
I-134	0.0366
I-135	0.280
Cs-134	750
Kr-88	0.117
Cs-137	11,000
Ru-106	365
Te-132	3.25
Ce-144	284

*Derived from the more probable Reactor Safety Study core melt categories and from postulated design basis accident releases.

E. Contiguous-Jurisdiction Governmental Emergency Planning

The concept of Emergency Planning Zones (EPZs) necessarily implies mutually supportive emergency planning and preparedness arrangements by several levels of government: Federal, State and local governments, including counties, townships and even villages. For the purposes of this document, it is not necessary to outline the varied governmental and jurisdictional situations that can and do exist throughout the United States, nor is it necessary to describe in detail the varied emergency planning and preparedness mechanisms that can be developed among these governmental entities.

It would be useful to offer several generally representative governmental-jurisdictional situations relating to the Emergency Planning Zone concept. There are obvious permutations and combinations of these situations, but these are examples of what is desirable in terms of cross-jurisdictional emergency planning. The important point is that integrated emergency planning will benefit all of the communities within the Emergency Planning Zones.

Example No. 1 Local Government Jurisdictions Within the Plume Exposure Pathway (10 miles) Emergency Planning Zone

A variety of local government jurisdictions may be found within the 10-mile plume exposure pathway Emergency Planning Zone (EPZ). In some situations several county-level governments and municipal or township

E. <u>Contiguous-Jurisdiction Governmental Emergency Planning</u> (continued)

governments will have jurisdictional authority within the EPZ and these separate governmental entities will control their own emergency response organizations and resources. In multi-jurisdictional situations like this, an integrated multi-county level emergency response plan is preferable. The response organizations and resources of municipal or township governments can be integrated -- by mutual agreement -- into the overall multi-county emergency response plan.

In other situations, a municipal or township government might have a larger emergency response organization than its parent county. Under these circumstances, the municipality or township government might be mutually designated the "lead" emergency planning and response organization, incorporating the resources available to the county in the overall emergency plan.

Local government plans and response mechanisms are particularly important for the 10-mile EPZ. This is because relatively shorter times may be available to implement immediate protective measures associated with the plume exposure pathway (sheltering, thyroid blocking, evacuation), as opposed to the generally

E. Contiguous-Jurisdiction Governmental Emergency Planning (continued)

longer times available for implementing protective measures for the ingestion exposure pathway. State government resources may be too far away from the involved local jurisdictions to be of much immediate help for a plume exposure problem in the early hours of an accident. Local government emergency plans should be made a part of the State emergency plan.

Example No. 2 Local Government Within the Plume Exposure Pathway (10-mile) Emergency Planning Zone Whose Boundaries Are Also a State Boundary

This situation will normally be found where the nuclear facility is situated on a river which forms a boundary between States and local governments. In this case, the fact that a State boundary is now involved within the EPZ makes it necessary to have contiguous State emergency planning within the EPZ, involving cooperative planning at a higher level of government. This should not preclude cooperative planning between adjacent counties, municipalities or townships located in different States.

Example No. 3 State vs. Local Government Emergency Planning Within the Ingestion Exposure Pathway (50-mile) Emergency Planning Zone

The 50-mile EPZ for the ingestion (agricultural products consumption) exposure pathway may encompass one or

E. Contiguous-Jurisdiction Governmental Emergency Planning (continued)

several States, as well as many local government, municipal or township jurisdictions. Planning for the implementing of protective measures associated with the ingestion exposure pathway is best handled by the State governments, with support from local governments, particularly at the county level, with backup from the Federal Government. This is because the involved areas could be quite large, crossing many jurisdictional boundaries and involving the use of relatively sophisticated radiological analysis equipment generally found only at State and Federal Government levels. Further, the time available to implement protective measures associated with the ingestion exposure pathway is generally greater than the time available to implement protective measures associated with the plume exposure pathway. The State, with support from the Federal Government, should be able to respond quickly enough to implement any desirable protective measures for the ingestion exposure pathway.

E. <u>Contiguous-Jurisdiction Governmental Emergency Planning</u> (continued)

Example No. 4 <u>State and Local Government Jurisdictions Near An International Boundary</u>

At present, the only U. S. situations involving emergency planning considerations across an international boundary involve Canada. Both the U. S. and Canada have nuclear facilities near their common borders. Mutual emergency planning with Canada is desirable and the NRC and FEMA are pursuing this matter through appropriate channels.

F. <u>Integrated Guidance and Criteria</u>

NRC and FEMA have deliberately consolidated in this document guidance intended for use by State and local governments and that intended to guide the emergency planning and preparedness activities of NRC licensees because of a shared belief that an integrated approach to the development of response plans to radiological hazards is most likely to provide the best protection of the health and safety of the public. NRC and FEMA recognize that plans of licensees, State and local governments should not be developed in a vacuum or in isolation from one another. Should an accident occur, the public can be best protected when the response by all parties is fully integrated. Each party involved must have a clear understanding of what the overall level of preparedness must be and what role it will play in the event of

F. <u>Integrated Guidance and Criteria</u> (continued)

a nuclear accident. This understanding can be achieved best if there is an integrated development and evaluation of plans. There must also be an acceptance by the parties and a clear recognition of the responsibility they share for safeguarding public health and safety.

Although the guidance indicates that the criteria are appliable to one or more specific organizations, the intention throughout has been to provide for an adequate state of emergency preparedness around the facility. If weaknesses in one organization are identified, but compensated for in another organization, the reviewers can still find that an adequate state of emergency preparedness exists.

This consolidated guidance should also allow the parties to recognize and understand each other's capabilities, responsibilities and obligations. The guidance makes clear which party has responsibility for which essential element. In many cases, the NRC licensee, the State and the local governments are all called upon to produce material for the same essential element. The consolidated guidance will allow reviewers to do a more thorough analysis and to probe the relationship of one plan with another. This document has been designed to assist reviewers in their work.

G. <u>Funding and Technical Assistance</u>

While funding and technical assistance are not addressed in this document, it is a subject which must be discussed between the individual nuclear utilities and the involved State and local governments who must prepare emergency plans to support the nuclear facilities. The nuclear utility may have an incentive based on its own self interest as well as its responsibility to provide electric power, to assist in providing manpower, items of equipment, or other resources that the State and local governments may need but are themselves unable to provide. The Federal Regional Assistance Committees, now under the chairmanship of FEMA, will play an increasing role in the development of these plans. Training programs for State and local officials formerly sponsored by NRC and now sponsored by FEMA will continue without interruption.

H. <u>Nuclear Facility Licensee Response Organization</u>

NRC and FEMA agree that the licensees of nuclear facilities have a primary responsibility for planning and implementing emergency measures <u>within</u> their site boundaries. These emergency measures include corrective actions at the site and protective measures and aid for persons onsite. Since facility licensees cannot do this alone, it is a necessary part of the facility emergency planning to make advance arrangements with State and local organizations for special emergency assistance such as ambulance, medical, hospital, fire and police services.

H. Nuclear Facility Licensee Response Organization (continued)

An additional emergency activity for which facility licensees have primary responsibility is accident assessment. This includes prompt action to evaluate any potential risk to the public health and safety, both onsite and offsite, and timely recommendations to State and local governments concerning protective measures. In some situations, there could be a need for protective measures within short time intervals -- a half-hour or perhaps even less -- after determination that a hazard exists. For this reason, licensee emergency planners must recognize the importance of prompt accident assessment at the source. The criteria in this document reflect the identification and classification of accidents and the notification of offsite agencies by the facility licensee consistent with NRC rules as set forth in Appendix 1.

Emphasis on inplant identification of potential hazards is a change from the previous emphasis in many licensee response plans on measurement of actual levels of radioactivity before notifications of offsite organizations are made and actions to protect the public recommended.

Because of the potential need to take immediate action offsite in the event of a significant radiological accident, notifications to appropriate offsite response organizations (State or States and local government organizations) must go directly from the facility licensee. The response organizations which receive these notifications should

H. Nuclear Facility Licensee Response Organization (continued)

have the authority and capability to take immediate predetermined actions based on recommendations from the facility licensee. These actions could include prompt notification of the public in the offsite area, followed by advisories to the public in certain areas to stay inside (take shelter) or, if appropriate, evacuate to predetermined relocation or host areas. State agencies, which are likely to have greater radioprotective resources than local agencies, would bring their resources to bear and make decisions with regard to whether the recommended protective measures are adequate.

In the longer time frame, substantial corporate and private sector organization resources should also supplement the initial response of the nuclear facility licensee. A facility licensee organization is therefore required to have a "recovery organization" similar to the one recommended by the Atomic Industrial Forum, which can use and absorb Federal and private support which in all likelihood will be available following any radiological accident.

I. Federal Response

The Department of Energy's current Radiological Assistance Program (RAP), the Federal Interagency Radiological Assistance Plan (IRAP), other radiological emergency assistance plans, and DOE's National Laboratories capabilities as well as those of the U. S. Environmental Protection

I. Federal Response (continued)

Agency and the Department of Health and Human Services and other Federal capability, are being incorporated in a Federal Radiological Monitoring and Assessment Plan. Response plans should contain provisions for integration of this important Federal assistance.

The facility licensee must make provisions for an NRC presence onsite following an accident and for supplying information to and receiving advisories from NRC regional or headquarters operations centers. In addition, the plan should provide for communication between State authorities, NRC and FEMA.

The interrelationships of the Federal agencies and their roles during a radiological emergency will be defined in a National Radiological Emergency Preparedness Plan now being developed by FEMA, and in an NRC agency plan. These plans will be compatible with State, local and licensee plans developed using the "Planning Standards" of this guidance and criteria document.

J. Form and Content of Plans

The criteria in this document are organized under the topic headings of NUREG-75/111 (the principal previous NRC guidance to State and local response organizations) wherever possible. That format may be followed by planners.

J. <u>Form and Content of Plans</u> (continued)

The guidance does not specify a single format for emergency response plans but it is important that the means by which all criteria are met be clearly set forth in the plans. All plans should contain a table of contents, and a cross-reference to the criteria contained in this document is also needed. Applicable supporting and reference documents and tables may be incorporated by reference, and appendices should be used whenever necessary. The plans should be kept as concise as possible. The average plan should consist of perhaps hundreds of pages, not thousands. The plans should make clear what is to be done in an emergency, how it is to be done and by whom.

In addition to addressing the substance of all criteria, the plans must, of course, define the facility or facilities and area to which the plans apply. The plans should include definitions of any terms that are unique to the facility under consideration or are given connotations that differ from normally accepted usage.

Findings by FEMA and NRC with regard to the adequacy of emergency preparedness will be related to the capability of the facility licensee, State and local response organizations, to respond in a coordinated manner to emergencies at or related to particular nuclear facilities.

J. <u>Form and Content of Plans</u> (continued)

A continued state of readiness must be maintained by all organizations. Periodic reviews by FEMA and NRC will verify the capability of response organizations to implement various aspects of the response plans. This will include observation of exercises and certain drills by NRC, FEMA and other Federal agencies participating in the Regional Assistance Committees.

II. Planning Standards and Evaluation Criteria

A. Assignment of Responsibility (Organization Control)

Planning Standard

Primary responsibilities for emergency response by the nuclear facility licensee, and by State and local organizations within the Emergency Planning Zones have been assigned, the emergency responsibilities of the various supporting organizations have been specifically established, and each principal response organization has staff to respond and to augment its initial response on a continuous basis.

Evaluation Criteria	Applicability and Cross Reference to Plans		
	Licensee	State	Local
1.a. Each plan shall identify the State, local, Federal and private sector organizations (including utilities), that are intended to be part of the overall response organization for Emergency Planning Zones. (See Appendix 5).	X	X	X
b. Each organization and suborganization having an operational role shall specify its concept of operations, and its relationship to the total effort.	X	X	X
c. Each plan shall illustrate these interrelationships in a block diagram.	X	X	X
d. Each organization shall identify a specific individual by title who shall be in charge of the emergency response.	X	X	X
e. Each organization shall provide for 24-hour per day emergency response, including 24-hour per day manning of communications links.	X	X	X

A. Assignment of Responsibility (Organization Control) (continued)

Evaluation Criteria	Applicability and Cross Reference to Plans		
	Licensee	State	Local
2.a. Each organization shall specify the functions and responsibilities for major elements and key individuals by title, of emergency response, including the following: Command and Control, Alerting and Notification, Communications, Public Information, Accident Assessment, Public Health and Sanitation, Social Services, Fire and Rescue, Traffic Control, Emergency Medical Services, Law Enforcement, Transportation, Protective Response (including authority to request Federal assistance and to initiate other protective actions), and Radiological Exposure Control. The description of these functions shall include a clear and concise summary such as a table of primary and support responsibilities using the agency as one axis, and the function as the other. (See Section B for licensee).		X	X
b. Each plan shall contain (by reference to specific acts, codes or statutes) the legal basis for such authorities.		X	X
3. Each plan shall include written agreements referring to the concept of operations developed between Federal, State, and local agencies and other support organizations having an emergency response role within the Emergency Planning Zones. The agreements shall identify the emergency measures to be provided and the mutually acceptable criteria for their implementation, and specify the arrangements for exchange of information. These agreements may be provided in an appendix to the plan or the plan itself may contain descriptions of these matters and a signature page in the plan may serve to verify the agreements. The signature page format is appropriate for organizations where response functions are covered by laws, regulations or executive orders where separate written agreements are not necessary.	X	X	X

A. Assignment of Responsibility (Organization Control) (continued)

Evaluation Criteria	Applicability and Cross Reference to Plans		
	Licensee	State	Local
4. Each principal organization shall be capable of continuous (24-hour) operations for a protracted period. The individual in the principal organization who will be responsible for assuring continuity of resources (technical, administrative, and material) shall be specified by title.	X	X	X

B. Onsite Emergency Organization

Planning Standard

On-shift facility licensee responsibilities for emergency response are unambiguously defined, adequate staffing to provide initial facility accident response in key functional areas is maintained at all times, timely augmentation of response capabilities is available, and the interfaces among various onsite response activities and offsite support and response activities are specified.

Evaluation Criteria	Applicability and Cross Reference to Plans		
	Licensee	State	Local
1. Each licensee shall specify the onsite emergency organization of plant staff personnel for all shifts and its relation to the responsibilities and duties of the normal staff complement.	X		
2. Each licensee shall designate an individual as emergency coordinator who shall be on shift at all times and who shall have the authority and responsibility to immediately and unilaterally initiate any emergency actions, including providing protective action recommendations to authorities responsible for implementing offsite emergency measures.	X		
3. Each licensee shall identify a line of succession for the emergency coordinator position and identify the specific conditions for higher level utility officials assuming this function.	X		

B. Onsite Emergency Organization (continued)

Evaluation Criteria	Applicability and Cross Reference to Plans		
	Licensee	State	Local
4. Each licensee shall establish the functional responsibilities assigned to the emergency coordinator and shall clearly specify which responsibilities may not be delegated to other elements of the emergency organization. Among the responsibilities which may not be delegated shall be the decision to notify and to recommend protective actions to authorities responsible for offsite emergency measures.	X		
5. Each licensee shall specify the positions or title and major tasks to be performed by the persons to be assigned to the functional areas of emergency activity. For emergency situations, specific assignments shall be made for all shifts and for plant staff members, both onsite and away from the site. These assignments shall cover the emergency functions in Table B-1 entitled, "Minimum Staffing Requirements for Nuclear Power Plant Emergencies." The minimum on-shift staffing levels shall be as indicated in Table B-1. The licensee must be able to augment on-shift capabilities within a short period after declaration of an emergency. This capability shall be as indicated in Table B-1. The implementation schedule for licensed operators, auxiliary operators and the shift technical advisor on shift shall be as specified in the July 31, 1980 letter to all power reactor licensees. Any deficiencies in the other staffing requirements of Table B-1 must be capable of augmentation within 30 minutes by September 1, 1981, and such deficiencies must be fully removed by July 1, 1982.	X		

B. Onsite Emergency Organization (continued)

Evaluation Criteria	Applicability and Cross Reference to Plans: Licensee	State	Local
6. Each licensee shall specify the interfaces between and among the onsite functional areas of emergency activity, licensee headquarters support, local services support, and State and local government response organization. This shall be illustrated in a block diagram and shall include the onsite technical support center and the operational support (assembly) center and the licensee's near-site Emergency Operations Facility (EOF).	X		
7. Each licensee shall specify the corporate management, administrative, and technical support personnel who will augment the plant staff as specified in the table entitled "Minimum Staffing Requirements for Nuclear Power Plant Emergencies," (Table B-1) and in the following areas:	X		
a. logistics support for emergency personnel, e.g., transportation, communications, temporary quarters, food and water, sanitary facilities in the field, and special equipment and supplies procurement;	X		
b. technical support for planning and reentry/recovery operations;	X		
c. management level interface with governmental authorities; and	X		
d. release of information to news media during an emergency (coordinated with governmental authorities).	X		
8. Each licensee shall specify the contractor and private organizations who may be requested to provide technical assistance to and augmentation of the emergency organization.	X		

Table B-1

MINIMUM STAFFING REQUIREMENTS FOR NRC LICENSEES FOR NUCLEAR POWER PLANT EMERGENCIES (See B.5.)

Major Functional Area	Location	Major Tasks	Position Title or Expertise	On Shift*	Capability for Additions 30 min	60 min
Plant Operations and Assessment of Operational Aspects			Shift Supervisor (SRO)	1	--	--
			Shift Foreman (SRO)	1	--	--
			Control Room Operators	2	--	--
			Auxiliary Operators	2	--	--
Emergency Direction and Control (Emergency Coordinator)***			Shift Technical Advisor, Shift Supervisor or designated facility manager	1**	--	--
Notification/ Communication****		Notify licensee, State local and Federal personnel & maintain communication		1	1	2
Radiological Accident Assessment and Support of Operational Accident Assessment		Emergency Operations Facility (EOF) Director	Senior Manager	--	--	1
		Offsite Dose Assessment	Senior Health Physics (HP) Expertise		1	--
		Offsite Surveys		--	2	2
		Onsite (out-of-plant)		--	1	1
		In-plant surveys	HP Technicians	1	1	1
		Chemistry/Radio-chemistry	Rad/Chem Technicians	1	--	1
Plant System Engineering, Repair and Corrective Actions		Technical Support	Shift Technical Advisor	1	--	--
			Core/Thermal Hydraulics	--	1	
			Electrical	--	--	1
			Mechanical	--	--	1
		Repair and Corrective Actions	Mechanical Maintenance/	1**	--	1
			Rad Waste Operator			1
			Electrical Maintenance/	1**	1	1
			Instrument and Control (I&C) Technician	--	1	--

Table B-1 (contd)

Major Functional Area	Major Tasks	Position Title or Expertise	On Shift*	Capability for Additions 30 min	60 min
Protective Actions (In-Plant)	Radiation Protection: a. Access Control b. HP Coverage for repair, corrective actions, search and rescue first-aid & firefighting c. Personnel monitoring d. Dosimetry	HP Technicians	2**	2	2
Firefighting	--	--	Fire Brigade per Technical Specifications	Local Support	
Rescue Operations and First-Aid	--	--	2**	Local Support	
Site Access Control and Personnel Accountability	Security, firefighting communications, personnel accountability	Security Personnel	All per Security plan		
		Total	10	11	15

Notes:

* For each unaffected nuclear unit in operation, maintain at least one shift foreman, one control room operator and one auxiliary operator except that units sharing a control room may share a shift foreman if all functions are covered.

** May be provided by shift personnel assigned other functions.

*** Overall direction of facility response to be assumed by EOF director when all centers are fully manned. Director of minute-to-minute facility operations remains with senior manager in technical support center or control room.

**** May be performed by engineering aide to shift supervisor.

B. Onsite Emergency Organization (continued)

Evaluation Criteria	Applicability and Cross Reference to Plans: Licensee	State	Local
9. Each licensee shall identify the services to be provided by local agencies for handling emergencies, e.g., police, ambulance, medical, hospital, and fire-fighting organizations shall be specified. The licensee shall provide for transportation and treatment of injured personnel who may also be contaminated. Copies of the arrangements and agreements reached with contractor, private, and local support agencies shall be appended to the plan. The agreements shall delineate the authorities, responsibilities, and limits on the actions of the contractor, private organization, and local services support groups.	X		

C. Emergency Response Support and Resources

Planning Standard

Arrangements for requesting and effectively using assistance resources have been made, arrangements to accommodate State and local staff at the licensee's near-site Emergency Operations Facility have been made, and other organizations capable of augmenting the planned response have been identified.

Evaluation Criteria	Applicability and Cross Reference to Plans		
	Licensee	State	Local
1. The Federal government maintains in-depth capability to assist licensees, States and local governments through the Federal Radiological Monitoring and Assessment Plan (formerly Radiological Assistance Plan (RAP) and Interagency Radiological Assistance Plan (IRAP). Each State and licensee shall make provisions for incorporating the Federal response capability into its operation plan, including the following:			
a. specific persons by title authorized to request Federal assistance; see A.1.d., A.2.a.	X	X	
b. specific Federal resources expected, including expected times of arrival at specific nuclear facility sites; and	X	X	
c. specific licensee, State and local resources available to support the Federal response, e.g., air fields, command posts, telephone lines, radio frequencies and telecommunications centers.	X	X	X

C. Emergency Response Support and Resources (continued)

Evaluation Criteria	Applicability and Cross Reference to Plans		
	Licensee	State	Local
2.a. Each principal offsite organization may dispatch representatives to the licensee's near-site Emergency Operations Facility. (State technical analysis representatives at the nearsite EOF are preferred.)		X	X
b. The licensee shall prepare for the dispatch of a representative to principal offsite governmental emergency operations centers.	X		
3. Each organization shall identify radiological laboratories and their general capabilities and expected availability to provide radiological monitoring and analyses services which can be used in an emergency.	X	X	
4. Each organization shall identify nuclear and other facilities, organizations or individuals which can be relied upon in an emergency to provide assistance. Such assistance shall be identified and supported by appropriate letters of agreement.	X	X	X

D. Emergency Classification System

Planning Standard

A standard emergency classification and action level scheme, the bases of which include facility system and effluent parameters, is in use by the nuclear facility licensee, and State and local response plans call for reliance on information provided by facility licensees for determinations of minimum initial offsite response measures.

Evaluation Criteria	Applicability and Cross Reference to Plans		
	Licensee	State	Local
1. An emergency classification and emergency action level scheme as set forth in Appendix 1 must be established by the licensee. The specific instruments, parameters or equipment status shall be shown for establishing each emergency class, in the in-plant emergency procedures. The plan shall identify the parameter values and equipment status for each emergency class.	X		
2. The initiating conditions shall include the example conditions found in Appendix 1 and all postulated accidents in the Final Safety Analysis Report (FSAR) for the nuclear facility.	X		
3. Each State and local organization shall establish an emergency classification and emergency action level scheme consistent with that established by the facility licensee.		X	X
4. Each State and local organization should have procedures in place that provide for emergency actions to be taken which are consistent with the emergency actions recommended by the nuclear facility licensee, taking into account local offsite conditions that exist at the time of the emergency.		X	X

E. Notification Methods and Procedures

Planning Standard

Procedures have been established for notification, by the licensee of State and local response organizations and for notification of emergency personnel by all response organizations; the content of initial and followup messages to response organizations and the public has been established; and means to provide early notification and clear instruction to the populace within the plume exposure pathway Emergency Planning Zone have been established.

Evaluation Criteria	Applicability and Cross Reference to Plans		
	Licensee	State	Local
1. Each organization shall establish procedures which describe mutually agreeable bases for notification of response organizations consistent with the emergency classification and action level scheme set forth in Appendix 1. These procedures shall include means for verification of messages. The specific details of verification need not be included in the plan.	X	X	X
2. Each organization shall establish procedures for alerting, notifying, and mobilizing emergency response personnel.	X	X	X
3. The licensee in conjunction with State and local organizations shall establish the contents of the initial emergency messages to be sent from the plant. These measures shall contain information about the class of emergency, whether a release is taking place, potentially affected population and areas, and whether protective measures may be necessary.	X		

E. Notification Methods and Procedures (continued)

Evaluation Criteria	Applicability and Cross Reference to Plans		
	Licensee	State	Local
4. Each licensee shall make provisions for followup messages from the facility to offsite authorities which shall contain the following information if it is known and appropriate:	X		
a. location of incident and name and telephone number (or communications channel identification) of caller;	X		
b. date/time of incident;	X		
c. class of emergency;	X		
d. type of actual or projected release (airborne, waterborne, surface spill), and estimated duration/impact times;	X		
e. estimate of quantity of radioactive material released or being released and the points and height of releases;	X		
f. chemical and physical form of released material, including estimates of the relative quantities and concentration of noble gases, iodines and particulates;	X		
g. meteorological conditions at appropriate levels (wind speed, direction (to and from), indicator of stability, precipitation, if any);	X		
h. actual or projected dose rates at site boundary; projected integrated dose at site boundary;	X		
i. projected dose rates and integrated dose at the projected peak and at 2, 5 and 10 miles, including sector(s) affected;	X		

E. Notification Methods and Procedures (continued)

Evaluation Criteria	Applicability and Cross Reference to Plans		
	Licensee	State	Local
j. estimate of any surface radioactive contamination inplant, onsite or offsite;	X		
k. licensee emergency response actions underway;	X		
l. recommended emergency actions, including protective measures;	X		
m. request for any needed onsite support by offsite organizations; and	X		
n. prognosis for worsening or termination of event based on plant information.	X		
5. State and local government organizations shall establish a system for disseminating to the public appropriate information contained in initial and followup messages received from the licensee including the appropriate notification to appropriate broadcast media, e.g., the Emergency Broadcast System (EBS).		X	X
6. Each organization shall establish administrative and physical means, and the time required for notifying and providing prompt instructions to the public within the plume exposure pathway Emergency Planning Zone. (See Appendix 3.) It shall be the licensee's responsibility to demonstrate that such means exist, regardless of who implements this requirement. It shall be the responsibility of the State and local governments to activate such a system.	X	X	X

E. Notification Methods and Procedures (continued)

Evaluation Criteria	Applicability and Cross Reference to Plans		
	Licensee	State	Local
7. Each organization shall provide written messages intended for the public, consistent with the licensee's classification scheme. In particular, draft messages to the public giving instructions with regard to specific protective actions to be taken by occupants of affected areas shall be prepared and included as part of the State and local plans. Such messages should include the appropriate aspects of sheltering, ad hoc respiratory protection, e.g., handkerchief over mouth, thyroid blocking or evacuation. The role of the licensee is to provide supporting information for the messages. For ad hoc respiratory protection see "Respiratory Protective Devices Manual" American Industrial Hygiene Association, 1963 pp. 123-126.	X	X	X

F. Emergency Communications

Planning Standard

Provisions exist for prompt communications among principal response organizations to emergency personnel and to the public.

Evaluation Criteria	Applicability and Cross Reference to Plans		
	Licensee	State	Local
1. The communication plans for emergencies shall include organizational titles and alternates for both ends of the communication links. Each organization shall establish reliable primary and backup means of communication for licensees, local, and State response organizations. Such systems should be selected to be compatible with one another. Each plan shall include:			
a. provision for 24-hour per day notification to and activation of the State/local emergency response network; and at a minimum, a telephone link and alternate, including 24-hour per day manning of communications links that initiate emergency response actions.	X	X	X
b. provision for communications with continguous State/local governments within the Emergency Planning Zones;	X	X	X
c. provision for communications as needed with Federal emergency response organizations;	X	X	X
d. provision for communications between the nuclear facility and the licensee's near-site Emergency Operations Facility, State and local emergency operations centers, and radiological monitoring teams;	X	X	X
e. provision for alerting or activating emergency personnel in each response organization; and	X	X	X

F. <u>Emergency Communications</u> (continued)

Evaluation Criteria	Applicability and Cross Reference to Plans: Licensee	State	Local
f. provision for communication by the licensee with NRC headquarters and NRC Regional Office Emergency Operations Centers and the licensee's near-site Emergency Operations Facility and radiological monitoring team assembly area.	X		
2. Each organization shall ensure that a coordinated communication link for fixed and mobile medical support facilities exists.	X	X	X
3. Each organization shall conduct periodic testing of the entire emergency communications system (see evaluation criteria H.10, N.2.a and Appendix 3).	X	X	X

G. Public Education and Information

Planning Standard

Information is made available to the public on a periodic basis on how they will be notified and what their initial actions should be in an emergency (e.g., listening to a local broadcast station and remaining indoors), the principal points of contact with the news media for dissemination of information during an emergency (including the physical location or locations) are established in advance, and procedures for coordinated dissemination of information to the public are established.

Evaluation Criteria	Applicability and Cross Reference to Plans		
	Licensee	State	Local
1. Each organization shall provide a coordinated periodic (at least annually) dissemination of information to the public regarding how they will be notified and what their actions should be in an emergency. This information shall include, but not necessarily be limited to:	X	X	X

a. educational information on radiation;
b. contact for additional information;
c. protective measures, e.g., evacuation routes and relocation centers, sheltering, respiratory protection, radioprotective drugs; and
d. special needs of the handicapped.

Means for accomplishing this dissemination may include, but are not necessarily limited to: information in the telephone book; periodic information in utility bills; posting in public areas; and publications distributed on an annual basis.

G. Public Education and Information (continued)

Evaluation Criteria	Applicability and Cross Reference to Plans		
	Licensee	State	Local
2. The public information program shall provide the permanent and transient adult population within the plume exposure EPZ an adequate opportunity to become aware of the information annually. The programs should include provision for written material that is likely to be available in a residence during an emergency. Updated information shall be disseminated at least annually. Signs or other measures (e.g., decals, posted notices or other means, placed in hotels, motels, gasoline stations and phone booths) shall also be used to disseminate to any transient population within the plume exposure pathway EPZ appropriate information that would be helpful if an emergency or accident occurs. Such notices should refer the transient to the telephone directory or other source of local emergency information and guide the visitor to appropriate radio and television frequencies.	X	X	X
3.a. Each principal organization shall designate the points of contact and physical locations for use by news media during an emergency.	X	X	X
b. Each licensee shall provide space which may be used for a limited number of the news media at the nearsite Emergency Operations Facility.	X		
4.a. Each principal organization shall designate a spokesperson who should have access to all necessary information.	X	X	X
b. Each organization shall establish arrangements for timely exchange of information among designated spokespersons.	X	X	X
c. Each organization shall establish coordinated arrangements for dealing with rumors.	X	X	X

G. Public Education and Information (continued)

Evaluation Criteria	Applicability and Cross Reference to Plans		
	Licensee	State	Local
5. Each organization shall conduct coordinated programs at least annually to acquaint news media with the emergency plans, information concerning radiation, and points of contact for release of public information in an emergency.	X	X	X

H. Emergency Facilities and Equipment

Planning Standard

Adequate emergency facilities and equipment to support the emergency response are provided and maintained.

Evaluation Criteria	Applicability and Cross Reference to Plans		
	Licensee	State	Local
1. Each licensee shall establish a Technical Support Center and an onsite operations support center (assembly area) in accordance with NUREG-0696, Revision 1.	X		
2. Each licensee shall establish an Emergency Operations Facility from which evaluation and coordination of all licensee activities related to an emergency is to be carried out and from which the licensee shall provide information to Federal, State and local authorities responding to radiological emergencies in accordance with NUREG-0696, Revision 1.	X		
3. Each organization shall establish an emergency operations center for use in directing and controlling response functions.		X	X
4. Each organization shall provide for timely activation and staffing of the facilities and centers described in the plan.	X	X	X

H. Emergency Facilities and Equipment (continued)

Evaluation Criteria	Applicability and Cross Reference to Plans: Licensee	State	Local
5. Each licensee shall identify and establish onsite monitoring systems that are to be used to initiate emergency measures in accordance with Appendix 1, as well as those to be used for conducting assessment.	X		
The equipment shall include:			
a. geophysical phenomena monitors, (e.g., meteorological, hydrologic, seismic);	X		
b. radiological monitors, (e.g., process, area, emergency, effluent, wound and portable monitors and sampling equipment);	X		
c. process monitors, (e.g., reactor coolant system pressure and temperature, containment pressure and temperature, liquid levels, flow rates, status or lineup of equipment components); and	X		
d. fire and combustion products detectors.	X		
6. Each licensee shall make provision to acquire data from or for emergency access to offsite monitoring and analysis equipment including:			
a. geophysical phenomena monitors, (e.g., meteorological, hydrologic, seismic);	X		
b. radiological monitors including ratemeters and sampling devices. Dosimetry shall be provided and shall meet, as a minimum, the NRC Radiological Assessment Branch Technical Position for the Environmental Radiological Monitoring Program; and	X		
c. laboratory facilities, fixed or mobile.	X		

H. Emergency Facilities and Equipment (continued)

Evaluation Criteria	Applicability and Cross Reference to Plans: Licensee	State	Local
7. Each organization, where appropriate, shall provide for offsite radiological monitoring equipment in the vicinity of the nuclear facility.	X	X	X
8. Each licensee shall provide meteorological instrumentation and procedures which satisfy the criteria in Appendix 2, and provisions to obtain representative current meteorological information from other sources.	X		
9. Each licensee shall provide for an onsite operations support center (assembly area) which shall have adequate capacity, and supplies, including, for example, respiratory protection, protective clothing, portable lighting, portable radiation monitoring equipment, cameras and communications equipment for personnel present in the assembly area.	X		
10. Each organization shall make provisions to inspect, inventory and operationally check emergency equipment/instruments at least once each calendar quarter and after each use. There shall be sufficient reserves of instruments/equipment to replace those which are removed from emergency kits for calibration or repair. Calibration of equipment shall be at intervals recommended by the supplier of the equipment.	X	X	X
11. Each plan shall, in an appendix, include identification of emergency kits by general category (protective equipment, communications equipment, radiological monitoring equipment and emergency supplies).	X	X	X

H. Emergency Facilities and Equipment (continued)

Evaluation Criteria	Applicability and Cross Reference to Plans		
	Licensee	State	Local
12. Each organization shall establish a central point (preferably associated with the licensee's near-site Emergency Operations Facility), for the receipt and analysis of all field monitoring data and coordination of sample media.	X	X	X

I. Accident Assessment

Planning Standard

Adequate methods, systems and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition are in use.

Evaluation Criteria	Applicability and Cross Reference to Plans		
	Licensee	State	Local
1. Each licensee shall identify plant system and effluent parameter values characteristic of a spectrum of off-normal conditions and accidents, and shall identify the plant parameter values or other information which correspond to the example initiating conditions of Appendix 1. Such parameter values and the corresponding emergency class shall be included in the appropriate facility emergency procedures. Facility emergency procedures shall specify the kinds of instruments being used and their capabilities.	X		
2. Onsite capability and resources to provide initial values and continuing assessment throughout the course of an accident shall include post-accident sampling capability, radiation and effluent monitors, in-plant iodine instrumentation, and containment radiation monitoring in accordance with NUREG-0578, as elaborated in the NRC letter to all power reactor licensees dated October 30, 1979.	X		
3. Each licensee shall establish methods and techniques to be used for determining:			
a. the source term of releases of radioactive material within plant systems. An example is the relationship between the containment radiation monitor(s) reading(s) and radioactive material available for release from containment.	X		

I. Accident Assessment (continued)

Evaluation Criteria	Applicability and Cross Reference to Plans: Licensee	State	Local
b. the magnitude of the release of radioactive materials based on plant system parameters and effluent monitors.	X		
4. Each licensee shall establish the relationship between effluent monitor readings and onsite and offsite exposures and contamination for various meteorological conditions.	X		
5. Each licensee shall have the capability of acquiring and evaluating meteorological information sufficient to meet the criteria of Appendix 2. There shall be provisions for access to meteorological information by at least the nearsite Emergency Operations Facility, the Technical Support Center, the Control Room and an offsite NRC center. The licensee shall make available to the State suitable meteorological data processing interconnections which will permit independent analysis by the State, of facility generated data in those States with the resources to effectively use this information.	X		
6. Each licensee shall establish the methodology for determining the release rate/projected doses if the instrumentation used for assessment are offscale or inoperable.	X		
7. Each organization shall describe the capability and resources for field monitoring within the plume exposure Emergency Planning Zone which are an intrinsic part of the concept of operations for the facility.	X	X	X

I. Accident Assessment (continued)

Evaluation Criteria	Applicability and Cross Reference to Plans: Licensee	State	Local
8. Each organization, where appropriate, shall provide methods, equipment and expertise to make rapid assessments of the actual or potential magnitude and locations of any radiological hazards through liquid or gaseous release pathways. This shall include activation, notification means, field team composition, transportation, communication, monitoring equipment and estimated deployment times.	X	X	X
9. Each organization shall have a capability to detect and measure radioiodine concentrations in air in the plume exposure EPZ as low as 10^{-7} uCi/cc (microcuries per cubic centimeter) under field conditions. Interference from the presence of noble gas and background radiation shall not decrease the stated minimum detectable activity.	X	X	
10. Each organization shall establish means for relating the various measured parameters (e.g., contamination levels, water and air activity levels) to dose rates for key isotopes (i.e., those given in Table 3, page 18) and gross radioactivity measurements. Provisions shall be made for estimating integrated dose from the projected and actual dose rates and for comparing these estimates with the protective action guides. The detailed provisions shall be described in separate procedures.	X	X	
11. Arrangements to locate and track the airborne radioactive plume shall be made, using either or both Federal and State resources.		X	

J. Protective Response

Planning Standard

A range of protective actions have been developed for the plume exposure pathway EPZ for emergency workers and the public. Guidelines for the choice of protective actions during an emergency, consistent with Federal guidance, are developed and in place, and protective actions for the ingestion exposure pathway EPZ appropriate to the locale have been developed.

Evaluation Criteria	Applicability and Cross Reference to Plans		
	Licensee	State	Local
1. Each licensee shall establish the means and time required to warn or advise onsite individuals and individuals who may be in areas controlled by the operator, including:			
a. Employees not having emergency assignments;	X		
b. Visitors;	X		
c. Contractor and construction personnel; and	X		
d. Other persons who may be in the public access areas on or passing through the site or within the owner controlled area.	X		
2. Each licensee shall make provisions for evacuation routes and transportation for onsite individuals to some suitable offsite location, including alternatives for inclement weather, high traffic density and specific radiological conditions.	X	X	X
3. Each licensee shall provide for radiological monitoring of people evacuated from the site.	X		

J. Protective Response (continued)

Evaluation Criteria	Applicability and Cross Reference to Plans		
	Licensee	State	Local
4. Each licensee shall provide for the evacuation of onsite non-essential personnel in the event of a Site or General Emergency and shall provide a decontamination capability at or near the monitoring point specified in J.3.	X		
5. Each licensee shall provide for a capability to account for all individuals onsite at the time of the emergency and ascertain the names of missing individuals within 30 minutes of the start of an emergency and account for all onsite individuals continuously thereafter.	X		
6. Each licensee shall, for individuals remaining or arriving onsite during the emergency, make provisions for:			
a. Individual respiratory protection;	X		
b. Use of protective clothing; and	X		
c. Use of radioprotective drugs, (e.g., individual thyroid protection).	X		
7. Each licensee shall establish a mechanism for recommending protective actions to the appropriate State and local authorities. These shall include Emergency Action Levels corresponding to projected dose to the population-at-risk, in accordance with Appendix 1 and with the recommendations set forth in Tables 2.1 and 2.2 of the Manual of Protective Action Guides and Protective Actions for Nuclear Incidents (EPA-520/1-75-001). As specified in Appendix 1, prompt notification shall be made directly to the offsite authorities responsible for implementing protective measures within the plume exposure pathway Emergency Planning Zone.	X		

J. <u>Protective Response</u> (continued)

Evaluation Criteria	Applicability and Cross Reference to Plans		
	Licensee	State	Local
8. Each licensee's plan shall contain time estimates for evacuation within the plume exposure EPZ. These shall be in accordance with Appendix 4.	X		
9. Each State and local organization shall establish a capability for implementing protective measures based upon protective action guides and other criteria. This shall be consistent with the recommendations of EPA regarding exposure resulting from passage of radioactive airborne plumes, (EPA-520/1-75-001) and with those of DHEW (DHHS)/FDA regarding radioactive contamination of human food and animal feeds as published in the Federal Register of December 15, 1978 (43 FR 58790).		X	X
10. The organization's plans to implement protective measures for the plume exposure pathway shall include:			
a. Maps showing evacuation routes, evacuation areas, preselected radiological sampling and monitoring points, relocation centers in host areas, and shelter areas; (identification of radiological sampling and monitoring points shall include the designators in Table J-1 or an equivalent uniform system described in the plan);	X	X	X
b. Maps showing population distribution around the nuclear facility. This shall be by evacuation areas (licensees shall also present the information in a sector format);	X	X	X
c. Means for notifying all segments of the transient and resident population;	X	X	X
d. Means for protecting those persons whose mobility may be impaired due to such factors as institutional or other confinement;		X	X

TABLE J-1

SECTOR AND ZONE DESIGNATORS FOR RADIOLOGICAL SAMPLING AND MONITORING POINTS WITHIN EMERGENCY PLANNING ZONES

SECTOR NOMENCLATURE				ZONE NOMENCLATURE	
CENTERLINE OF SECTOR IN DEGREES TRUE NORTH FROM FACILITY	22 1/2° SECTOR			MILES FROM FACILITY	ZONE
0 & 360	*A		N	0-1	1
22 1/2	B		NNE	1-2	2
45	C		NE	2-3	3
67 1/2	D		ENE	3-4	4
90	E		E	4-5	5
112	F		ESE	5-6	6
135	G		SE	6-7	7
157	H	or	SSE	7-8	8
180	J		S	8-9	9
202 1/2	K		SSW	9-10	10
225	L		SW	10-15	15
247 1/2	M		WSW	15-20	20
270	N		W	20-25	25
292 1/2	P		WNW	25-30	30
315	Q		NW	30-35	35
337 1/2	R		NNW	35-40	40
				40-45	45
				45-50	50

AREA SEGMENT - An area is identified by a Sector and Zone designator. Thus, area NL is that area which lies between 348 3/4 and 11 1/4 degrees true north from the facility out to a radius of 1 mile. Area SE4 would be that area between 123 3/4 to 146 1/4 degrees and the 3- and 4-mile arcs from the facility.

*The letters I and O have been omitted from these sector designators so as to eliminate possible confusion between letters and numbers.

J. Protective Response (continued)

Evaluation Criteria	Applicability and Cross Reference to Plans		
	Licensee	State	Local
e. Provisions for the use of radioprotective drugs, particularly for emergency workers and institutionalized persons within the plume exposure EPZ whose immediate evacuation may be infeasible or very difficult, including quantities, storage, and means of distribution.		X	X
f. State and local organizations' plans should include the method by which decisions by the State Health Department for administering radioprotective drugs to the general population are made during an emergency and the predetermined conditions under which such drugs may be used by offsite emergency workers;[1]		X	X
g. Means of relocation;		X	X
h. Relocation centers in host areas which are at least 5 miles, and preferably 10 miles, beyond the boundaries of the plume exposure emergency planning zone; (See J.12).		X	X
i. Projected traffic capacities of evacuation routes under emergency conditions;		X	X
j. Control of access to evacuated areas and organization responsibilities for such control;		X	X
k. Identification of and means for dealing with potential impediments (e.g., seasonal impassability of roads) to use of evacuation routes, and contingency measures;		X	X
l. Time estimates for evacuation of various sectors and distances based on a dynamic analysis (time-motion study under various conditions) for the plume exposure pathway emergency planning zone (See Appendix 4); and		X	X

1/ See DHEW (new DHHS) Federal Register notice of December 15, 1978 (43 FR 58798) entitled "Potassium Iodide as a Thyroid-Blocking Agent in a Radiation Emergency." Other guidance concerning the storage, stockpiling, and conditions for use of this drug by the general public, is now under development by the Bureau of Drugs, DHHS.

J. Protective Response (continued)

Evaluation Criteria	Applicability and Cross Reference to Plans: Licensee	State	Local
m. The bases for the choice of recommended protective actions from the plume exposure pathway during emergency conditions. This shall include expected local protection afforded[2] in residential units or other shelter for direct and inhalation exposure, as well as evacuation time estimates.	X	X	
11. Each State shall specify the protective measures to be used for the ingestion pathway, including the methods for protecting the public from consumption of contaminated foodstuffs. This shall include criteria for deciding whether dairy animals should be put on stored feed. The plan shall identify procedures for detecting contamination, for estimating the dose commitment consequences of uncontrolled ingestion, and for imposing protection procedures such as impoundment, decontamination, processing, decay, product diversion, and preservation. Maps for recording survey and monitoring data, key land use data (e.g., farming), dairies, food processing plants, water sheds, water supply intake and treatment plants and reservoirs shall be maintained. Provisions for maps showing detailed crop information may be by including reference to their availability and location and a plan for their use. The maps shall start at the facility and include all of the 50-mile ingestion pathway EPZ. Up-to-date lists of the name and location of all facilities which regularly process milk products and other large amounts of food or agricultural products originating in the ingestion pathway Emergency Planning Zone, but located elsewhere, shall be maintained.		X	

2/ The following reports may be considered in determining protection afforded.

(1) "Public Protection Strategies for Potential Nuclear Reactor Accidents" Sheltering Concepts with Existing Public and Private Structures" (SAND 77-1725), Sandia Laboratory.

(2) "Examination of Offsite Radiological Emergency Measures for Nuclear Reactor Accidents Involving Core Melt" (SAND 78-0454), Sandia Laboratory.

(3) "Protective Action Evaluation Part II, Evacuation and Sheltering as Protective Actions Against Nuclear Accidents Involving Gaseous Releases" (EPA 520/1-78-001B). U. S. Environmental Protection Agency.

J. Protective Response (continued)

Evaluation Criteria	Applicability and Cross Reference to Plans		
	Licensee	State	Local
12. Each organization shall describe the means for registering and monitoring of evacuees at relocation centers in host areas. The personnel and equipment available should be capable of monitoring within about a 12 hour period all residents and transients in the plume exposure EPZ arriving at relocation centers.		X	X

K. Radiological Exposure Control

Planning Standard

Means for controlling radiological exposures, in an emergency, are established for emergency workers. The means for controlling radiological exposures shall include exposure guidelines consistent with EPA Emergency Worker and Lifesaving Activity Protective Action Guides.

Evaluation Criteria	Applicability and Cross Reference to Plans: Licensee	State	Local
1. Each licensee shall establish onsite exposure guidelines consistent with EPA Emergency Worker and Lifesaving Activity Protective Actions Guides (EPA 520/1-75/001) for:			
a. removal of injured persons;	X		
b. undertaking corrective actions;	X		
c. performing assessment actions;	X		
d. providing first aid;	X		
e. performing personnel decontamination;	X		
f. providing ambulance service; and	X		
g. providing medical treatment services.	X		
2. Each licensee shall provide an onsite radiation protection program to be implemented during emergencies, including methods to implement exposure guidelines. The plan shall identify individual(s), by position or title, who can authorize emergency workers to receive doses in excess of 10 CFR Part 20 limits. Procedures shall be worked out in advance for permitting onsite volunteers to receive radiation exposures in the course of carrying out lifesaving and other emergency activities. These procedures shall include expeditious decision making and a reasonable consideration of relative risks.	X		

K. Radiological Exposure Control (continued)

Evaluation Criteria	Applicability and Cross Reference to Plans		
	Licensee	State	Local
3.a. Each organization shall make provision for 24-hour-per-day capability to determine the doses received by emergency personnel involved in any nuclear accident, including volunteers. Each organization shall make provisions for distribution of dosimeters, both self-reading and permanent record devices.	X	X	X
b. Each organization shall ensure that dosimeters are read at appropriate frequencies and provide for maintaining dose records for emergency workers involved in any nuclear accident.	X	X	X
4. Each State and local organization shall establish the decision chain for authorizing emergency workers to incur exposures in excess of the EPA General Public Protective Action Guides (i.e., EPA PAGs for emergency workers and lifesaving activities).		X	X
5.a. Each organization as appropriate, shall specify action levels for determining the need for decontamination.	X	X	X
b. Each organization, as appropriate, shall establish the means for radiological decontamination of emergency personnel wounds, supplies, instruments and equipment, and for waste disposal.	X	X	X
6. Each licensee shall provide onsite contamination control measures including:			
a. area access control;	X		
b. drinking water and food supplies;	X		
c. criteria for permitting return of areas and items to normal use, see Draft ANSI 13.12.	X		

K. Radiological Exposure Control (continued)

Evaluation Criteria	Applicability and Cross Reference to Plans		
	Licensee	State	Local
7. Each licensee shall provide the capability for decontaminating relocated onsite personnel, including provisions for extra clothing and decontaminants suitable for the type of contamination expected, with particular attention given to radioiodine contamination of the skin.	X		

L. Medical and Public Health Support

Planning Standard

Arrangements are made for medical services for contaminated injured individuals.[1]

Evaluation Criteria	Applicability and Cross Reference to Plans		
	Licensee	State	Local
1. Each organization shall arrange for local and backup hospital and medical services having the capability for evaluation of radiation exposure and uptake, including assurance that persons providing these services are adequately prepared to handle contaminated individuals.	X	X	X
2. Each licensee shall provide for onsite first aid capability.	X		
3. Each State shall develop lists indicating the location of public, private and military hospitals and other emergency medical services facilities within the State or contiguous States considered capable of providing medical support for any contaminated injured individual. The listing shall include the name, location, type of facility and capacity and any special radiological capabilities. These emergency medical services should be able to radiologically monitor contamination personnel, and have facilities and trained personnel able to care for contaminated injured persons.		X	
4. Each organization shall arrange for transporting victims of radiological accidents to medical support facilities.	X	X	X

1/ The availability of an integrated emergency medical services system and a public health emergency plan serving the area in which the facility is located and, as a minimum, equivalent to the Public Health Service Guide for Developing Health Disaster Plans, 1974, and to the requirements of an emergency medical services system as outlined in the Emergency Medical Services System Act of 1973 (P.L. 93-154 and amendments in 1979 P.L. 96-142), should be a part of and consistent with overall State or local disaster control plans and should be compatible with the specific overall emergency response plan for the facility.

M. Recovery and Reentry Planning and Postaccident Operations

Planning Standard

General plans for recovery and reentry are developed.

Evaluation Criteria	Applicability and Cross Reference to Plans		
	Licensee	State	Local
1. Each organization, as appropriate, shall develop general plans and procedures for reentry and recovery and describe the means by which decisions to relax protective measures (e.g., allow reentry into an evacuated area) are reached. This process should consider both existing and potential conditions.	X	X	X
2. Each licensee plan shall contain the position/title, authority and responsibilities of individuals who will fill key positions in the facility recovery organization. This organization shall include technical personnel with responsibilities to develop, evaluate and direct recovery and reentry operations. The recovery organization recommended by the Atomic Industrial Forum's "Nuclear Power Plant Emergency Response Plan" dated October 11, 1979, is an acceptable framework.	X		
3. Each licensee and State plan shall specify means for informing members of the response organizations that a recovery operation is to be initiated, and of any changes in the organizational structure that may occur.	X	X	
4. Each plan shall establish a method for periodically estimating total population exposure.	X	X	

N. Exercises and Drills

Planning Standard

Periodic exercises are (will be) conducted to evaluate major portions of emergency response capabilities, periodic drills are (will be) conducted to develop and maintain key skills, and deficiencies identified as a result of exercises or drills are (will be) corrected.

Evaluation Criteria	Applicability and Cross Reference to Plans		
	Licensee	State	Local
1.a. An exercise is an event that tests the integrated capability and a major portion of the basic elements existing within emergency preparedness plans and organizations. The emergency preparedness exercise shall simulate an emergency that results in offsite radiological releases which would require response by offsite authorities. Exercises shall be conducted as set forth in NRC and FEMA rules.	X	X	X
b. An exercise shall include mobilization of State and local personnel and resources adequate to verify the capability to respond to an accident scenario requiring response. The organization shall provide for a critique of the annual exercise by Federal and State observers/evaluators. The scenario should be varied from year to year such that all major elements of the plans and preparedness organizations are tested within a five-year period. Each organization should make provisions to start an exercise between 6:00 p.m. and midnight, and another between midnight and 6:00 a.m. once every six years. Exercises should be conducted under various weather conditions. Some exercises should be unannounced.	X	X	X

N. Exercises and Drills (continued)

Evaluation Criteria	Applicability and Cross Reference to Plans: Licensee	State	Local
2. A drill is a supervised instruction period aimed at testing, developing and maintaining skills in a particular operation. A drill is often a component of an exercise. A drill shall be supervised and evaluated by a qualified drill instructor. Each organization shall conduct drills, in addition to the annual exercise at the frequencies indicated below:			
a. Communication Drills Communications with State and local governments within the plume exposure pathway Emergency Planning Zone shall be tested monthly. Communications with Federal emergency response organizations and States within the ingestion pathway shall be tested quarterly. Communications between the nuclear facility, State and local emergency operations centers, and field assessment teams shall be tested annually. Communication drills shall also include the aspect of understanding the content of messages.	X	X	X
b. Fire Drills Fire drills shall be conducted in accordance with the plant (nuclear facility) technical specifications.	X		
c. Medical Emergency Drills A medical emergency drill involving a simulated contaminated individual which contains provisions for participation by the local support services agencies (i.e., ambulance and offsite medical treatment facility) shall be conducted annually. The offsite portions of the medical drill may be performed as part of the required annual exercise.	X		X

N. Exercises and Drills (continued)

Evaluation Criteria	Applicability and Cross Reference to Plans		
	Licensee	State	Local
d. Radiological Monitoring Drills			
Plant environs and radiological monitoring drills (onsite and offsite) shall be conducted annually. These drills shall include collection and analysis of all sample media (e.g., water, vegetation, soil and air), and provisions for communications and record keeping. The State drills need not be at each site. Where appropriate, local organizations shall participate.	X	X	X
e. Health Physics Drills			
(1) Health Physics drills shall be conducted semi-annually which involve response to, and analysis of, simulated elevated airborne and liquid samples and direct radiation measurements in the environment. The State drills need not be at each site.	X	X	
(2) Analysis of inplant liquid samples with actual elevated radiation levels including use of the post-accident sampling system shall be included in Health Physics drills by licensees annually.	X		
3. Each organization shall describe how exercises and drills are to be carried out to allow free play for decisionmaking and to meet the following objectives. Pending the development of exercise scenarios and exercise evaluation guidance by NRC and FEMA the scenarios for use in exercises and drills shall include but not be limited to, the following:			
a. The basic objective(s) of each drill and exercise and appropriate evaluation criteria;	X	X	X

N. Exercises and Drills (continued)

Evaluation Criteria	Applicability and Cross Reference to Plans: Licensee	State	Local
b. The date(s), time period, place(s) and participating organizations;	X	X	X
c. The simulated events;	X	X	X
d. A time schedule of real and simulated initiating events;	X	X	X
e. A narrative summary describing the conduct of the exercises or drills to include such things as simulated casualties, offsite fire department assistance, rescue of personnel, use of protective clothing, deployment of radiological monitoring teams, and public information activities; and	X	X	X
f. A description of the arrangements for and advance materials to be provided to official observers.	X	X	X
4. Official observers from Federal, State or local governments will observe, evaluate, and critique the required exercises. A critique shall be scheduled at the conclusion of the exercise to evaluate the ability of organizations to respond as called for in the plan. The critique shall be conducted as soon as practicable after the exercise, and a formal evaluation should result from the critique.	X	X	X
5. Each organization shall establish means for evaluating observer and participant comments on areas needing improvement, including emergency plan procedural changes, and for assigning responsibility for implementing corrective actions. Each organization shall establish management control used to ensure that corrective actions are implemented.	X	X	X

O. Radiological Emergency Response Training

Planning Standard

Radiological emergency response training is provided to those who may be called on to assist in an emergency.

Evaluation Criteria	Applicability and Cross Reference to Plans		
	Licensee	State	Local
1. Each organization shall assure the training of appropriate individuals.	X	X	X
a. Each facility to which the plant applies shall provide site specific emergency response training for those offsite emergency organizations who may be called upon to provide assistance in the event of an emergency.1/	X		
b. Each offsite response organization shall participate in and receive training. Where mutual aid agreements exist between local agencies such as fire, police and ambulance/rescue, the training shall also be offered to the other departments who are members of the mutual aid district.		X	X
2. The training program for members of the onsite emergency organization shall, besides classroom training, include practical drills in which each individual demonstrates ability to perform his assigned emergency function. During the practical drills, on-the-spot correction of erroneous performance shall be made and a demonstration of the proper performance offered by the instructor.	X		

1/ Training for hospital personnel, ambulance/rescue, police and fire departments shall include the procedures for notification, basic radiation protection, and their expected roles. For those local services support organizations who will enter the site, training shall also include site access procedures and the identity (by position and title) of the individual in the onsite emergency organization who will control the organizations' support activities. Offsite emergency response support personnel should be provided with appropriate identification cards where required.

O. Radiological Emergency Response Training (continued)

Evaluation Criteria	Applicability and Cross Reference to Plans		
	Licensee	State	Local
3. Training for individuals assigned to licensee first aid teams shall include courses equivalent to Red Cross Multi-Media.	X		
4. Each organization shall establish a training program for instructing and qualifying personnel who will implement radiological emergency response plans.2/ The specialized initial training and periodic retraining programs (including the scope, nature and frequency) shall be provided in the following categories:			
a. Directors or coordinators of the response organizations;	X	X	X
b. Personnel responsible for accident assessment;	X	X	*
c. Radiological monitoring teams and radiological analysis personnel;	X	X	*
d. Police, security and fire fighting personnel;	X	*	X
e. Repair and damage control/correctional action teams (onsite);	X		
f. First aid and rescue personnel;	X	*	X
g. Local support services personnel including Civil Defense/Emergency Service personnel;	X		X
h. Medical support personnel;	X	X	X
i. Licensee's headquarters support personnel;	X		
j. Personnel responsible for transmission of emergency information and instructions.	X	X	X

2/ If State and local governments lack the capability and resources to accomplish this training, they may look to the licensee and the Federal government (FEMA) for assistance in this training.

* NRC and FEMA encourage State and local governments which have these capabilities to continue to include them in their training programs.

O. Radiological Emergency Response Training (continued)

Evaluation Criteria	Applicability and Cross Reference to Plans		
	Licensee	State	Local
5. Each organization shall provide for the initial and annual retraining of personnel with emergency response responsibilities.	X	X	X

P. Responsibility for the Planning Effort: Development, Periodic Review and Distribution of Emergency Plans

Planning Standard

Responsibilities for plan development and review and for distribution of emergency plans are established, and planners are properly trained.

Evaluation Criteria	Applicability and Cross Reference to Plans		
	Licensee	State	Local
1. Each organization shall provide for the training of individuals responsible for the planning effort.	X	X	X
2. Each organization shall identify by title the individual with the overall authority and responsibility for radiological emergency response planning.	X	X	X
3. Each organization shall designate an Emergency Planning Coordinator with responsibility for the development and updating of emergency plans and coordination of these plans with other response organizations.	X	X	X
4. Each organization shall update its plan and agreements as needed, review and certify it to be current on an annual basis. The update shall take into account changes identified by drills and exercises.	X	X	X
5. The emergency response plans and approved changes to the plans shall be forwarded to all organizations and appropriate individuals with responsibility for implementation of the plans. Revised pages shall be dated and marked to show where changes have been made.	X	X	X
6. Each plan shall contain a detailed listing of supporting plans and their source.	X	X	X

P. Responsibility for the Planning Effort: Development, Periodic Review and Distribution of Emergency Plans (continued)

Evaluation Criteria	Applicability and Cross Reference to Plans		
	Licensee	State	Local
7. Each plan shall contain as an appendix listing, by title, procedures required to implement the plan. The listing shall include the section(s) of the plan to be implemented by each procedure.	X	X	X
8. Each plan shall contain a specific table of contents. Plans submitted for review should be cross-referenced to these criteria.	X	X	X
9. Each licensee shall arrange for and conduct independent reviews of the emergency preparedness program at least every 12 months. (An independent review is one conducted by any competent organization either internal or external to the licensee's organization, but who are not immediately responsible for the emergency preparedness program). The review shall include the emergency plan, its implementing procedures and practices, training, readiness testing, equipment, and interfaces with State and local governments. Management controls shall be implemented for evaluation and correction of review findings. The result of the review, along with recommendations for improvements, shall be documented, reported to appropriate licensee corporate and plant management, and involved Federal, State and local organizations, and retained for a period of five years.	X		
10. Each organization shall provide for updating telephone numbers in emergency procedures at least quarterly.	X	X	X

APPENDIX 1

U. S. NUCLEAR REGULATORY COMMISSION

EMERGENCY ACTION LEVEL GUIDELINES

FOR NUCLEAR POWER PLANTS

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BASIS FOR EMERGENCY ACTION LEVELS FOR NUCLEAR POWER FACILITIES

Four classes of Emergency Action Levels are established which replace the classes in Regulatory Guide 1.101, each with associated examples of initiating conditions. The classes are:

Notification of Unusual Event

Alert

Site Area Emergency

General Emergency

The rationale for the notification and alert classes is to provide early and prompt notification of minor events which could lead to more serious consequences given operator error or equipment failure or which might be indicative of more serious conditions which are not yet fully realized. A gradation is provided to assure fuller response preparations for more serious indicators. The site area emergency class reflects conditions where some significant releases are likely or are occurring but where a core melt situation is not indicated based on current information. In this situation full mobilization of emergency personnel in the near site environs is indicated as well as dispatch of monitoring teams and associated communications. The general emergency class involves actual or imminent substantial core degradation or melting with the potential for loss of containment. The immediate action for this class is sheltering (staying inside) rather than evacuation until an assessment can be made that (1) an evacuation is indicated and (2) an evacuation, if indicated, can be completed prior to significant release and transport of radioactive material to the affected areas.

The example initiating conditions listed after the immediate actions for each class are to form the basis for establishment by each licensee of the specific plant instrumentation readings (as applicable) which, if exceeded, will initiate the emergency class.

Potential NRC actions during various emergency classes are given in NUREG-0728, Report to Congress: NRC Incident Response Plan. The NRC response to any notification from a licensee will be related to, but not limited by, the licensee estimate of severity; NRC will consider such other factors as the degree of uncertainty and the lead times required to position NRC response personnel should something more serious develop.

Prompt notification of offsite authorities is intended to indicate within about 15 minutes for the unusual event class and sooner (consistent with the need for other emergency actions) for other classes. The time is measured from the time at which operators recognize that events have occurred which make declaration of an emergency class appropriate.

Class	Licensee Actions	State and/or Local Offsite Authority Actions
NOTIFICATION OF UNUSUAL EVENT Class Description Unusual events are in process or have occurred which indicate a potential degradation of the level of safety of the plant. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs. Purpose Purpose of offsite notification is to (1) assure that the first step in any response later found to be necessary has been carried out, (2) bring the operating staff to a state of readiness, and (3) provide systematic handling of unusual events information and decisionmaking.	1. Promptly inform State and/or local offsite authorities of nature of unusual condition as soon as discovered 2. Augment on-shift resources as needed 3. Assess and respond 4. Escalate to a more severe class, if appropriate or 5. Close out with verbal summary to offsite authorities; followed by written summary within 24 hours	1. Provide fire or security assistance if requested 2. Escalate to a more severe class, if appropriate 3. Stand by until verbal closeout

EXAMPLE INITIATING CONDITIONS: NOTIFICATION OF UNUSUAL EVENT

1. Emergency Core Cooling System (ECCS) initiated and discharge to vessel
2. Radiological effluent technical specification limits exceeded
3. Fuel damage indication. Examples:
   a. High offgas at BWR air ejector monitor (greater than 500,000 uci/sec; corresponding to 16 isotopes decayed to 30 minutes; <u>or</u> an increase of 100,000 uci/sec within a 30 minute time period)

   b. High coolant activity sample (e.g., exceeding coolant technical specifications for iodine spike)

   c. Failed fuel monitor (PWR) indicates increase greater than 0.1% equivalent fuel failures within 30 minutes
4. Abnormal coolant temperature and/or pressure or abnormal fuel temperatures outside of technical specification limits
5. Exceeding either primary/secondary leak rate technical specification or primary system leak rate technical specification
6. Failure of a safety or relief valve in a safety related system to close following reduction of applicable pressure
7. Loss of offsite power or loss of onsite AC power capability
8. Loss of containment integrity requiring shutdown by technical specifications
9. Loss of engineered safety feature or fire protection system function requiring shutdown by technical specifications (e.g., because of malfunction, personnel error or procedural inadequacy)
10. Fire within the plant lasting more than 10 minutes
11. Indications or alarms on process or effluent parameters not functional in control room to an extent requiring plant shutdown or other significant loss of assessment or communication capability (e.g., plant computer, Safety Parameter Display System, all meteorological instrumentation)
12. Security threat or attempted entry or attempted sabotage
13. Natural phenomenon being experienced or projected beyond usual levels
    a. Any earthquake felt in-plant or detected on station seismic instrumentation

    b. 50 year floor or low water, tsunami, hurricane surge, seiche

    c. Any tornado on site

    d. Any hurricane

14. Other hazards being experienced or projected

    a. Aircraft crash on-site or unusual aircraft activity over facility

    b. Train derailment on-site

    c. Near or onsite explosion

    d. Near or onsite toxic or flammable gas release

    e. Turbine rotating component failure causing rapid plant shutdown

15. Other plant conditions exist that warrant increased awareness on the part of a plant operating staff or State and/or local offsite authorities or require plant shutdown under technical specification requirements or involve other than normal controlled shutdown (e.g., cooldown rate exceeding technical specification limits, pipe cracking found during operation)

16. Transportation of contaminated injured individual from site to offsite hospital

17. Rapid depressurization of PWR secondary side.

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Class	Licensee Actions	State and/or Local Offsite Authority Actions
ALERT **Class Description** Events are in process or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant. Any releases expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels. **Purpose** Purpose of offsite alert is to (1) assure that emergency personnel are readily available to respond if situation becomes more serious or to perform confirmatory radiation monitoring if required, and (2) provide offsite authorities current status information.	1. Promptly inform State and/or local authorities of alert status and reason for alert as soon as discovered 2. Augment resources and activate on-site Technical Support Center and on-site operational support center. Bring Emergency Operations Facility (EOF) and other key emergency personnel to standby status 3. Assess and respond 4. Dispatch on-site monitoring teams and associated communications 5. Provide periodic plant status updates to offsite authorities (at least every 15 minutes) 6. Provide periodic meteorological assessments to offsite authorities and, if any releases are occurring, dose estimates for actual releases 7. Escalate to a more severe class, if appropriate 8. Close out or recommend reduction in emergency class by verbal summary to offsite authorities followed by written summary within 8 hours of closeout or class reduction	1. Provide fire or security assistance if requested 2. Augment resources and bring primary response centers and EBS to standby status 3. Alert to standby status key emergency personnel including monitoring teams and associated communications 4. Provide confirmatory offsite radiation monitoring and ingestion pathway dose projections if actual releases substantially exceed technical specification limits 5. Escalate to a more severe class, if appropriate 6. Maintain alert status until verbal closeout or reduction of emergency class

EXAMPLE INITIATING CONDITIONS: ALERT

1. Severe loss of fuel cladding

   a. High offgas at BWR air ejector monitor (greater than 5 ci/sec; corresponding to 16 isotopes decayed 30 minutes)

   b. Very high coolant activity sample (e.g., 300 uci/cc equivalent of I-131)

   c. Failed fuel monitor (PWR) indicates increase greater than 1% fuel failures within 30 minutes or 5% total fuel failures.

2. Rapid gross failure of one steam generator tube with loss of offsite power

3. Rapid failure of steam generator tubes (e.g., several hundred gpm primary to secondary leak rate)

4. Steam line break with significant (e.g., greater than 10 gpm) primary to secondary leak rate (PWR) or MSIV malfunction causing leakage (BWR)

5. Primary coolant leak rate greater than 50 gpm

6. Radiation levels or airborne contamination which indicate a severe degradation in the control of radioactive materials (e.g., increase of factor of 1000 in direct radiation readings within facility)

7. Loss of offsite power <u>and</u> loss of all onsite AC power (see Site Area Emergency for extended loss)

8. Loss of all onsite DC power (See Site Area Emergency for extended loss)

9. Coolant pump seizure leading to fuel failure

10. Complete loss of any function needed for plant cold shutdown

11. Failure of the reactor protection system to initiate and complete a scram which brings the reactor subcritical

12. Fuel damage accident with release of radioactivity to containment or fuel handling building

13. Fire potentially affecting safety systems

14. Most or all alarms (annunciators) lost

15. Radiological effluents greater than 10 times technical specification instantaneous limits (an instantaneous rate which, if continued over 2 hours, would result in about 1 mr at the site boundary under average meteorological conditions)

16. Ongoing security compromise

17. Severe natural phenomena being experienced or projected

    a. Earthquake greater than OBE levels

    b. Flood, low water, tsunami, hurricane surge, seiche near design levels

    c. Any tornado striking facility

    d. Hurricane winds near design basis level

18. Other hazards being experienced or projected

    a. Aircraft crash on facility

    b. Missile impacts from whatever source on facility

    c. Known explosion damage to facility affecting plant operation

    d. Entry into facility environs of uncontrolled toxic or flammable gases

    e. Turbine failure causing casing penetration

19. Other plant conditions exist that warrant precautionary activation of technical support center and placing near-site Emergency Operations Facility and other key emergency personnel on standby

20. Evacuation of control room anticipated or required with control of shutdown systems established from local stations

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Class

SITE AREA EMERGENCY

Class Description

Events are in process or have occurred which involve actual or likely major failures of plant functions needed for protection of the public. Any releases not expected to exceed EPA Protective Action Guideline exposure levels except near site boundary.

Purpose

Purpose of the site area emergency declaration is to (1) assure that response centers are manned, (2) assure that monitoring teams are dispatched, (3) assure that personnel required for evacuation of near-site areas are at duty stations if situation becomes more serious, (4) provide consultation with offsite authorities, and (5) provide updates for the public through offsite authorities.

Licensee Actions

1. Promptly inform State and/or local offsite authorities of site area emergency status and reason for emergency as soon as discovered
2. Augment resources by activating on-site Technical Support Center, on-site operational support center and near-site Emergency Operations Facility (EOF)
3. Assess and respond
4. Dispatch on-site and offsite monitoring teams and associated communications
5. Dedicate an individual for plant status updates to offsite authorities and periodic pressure briefings (perhaps joint with offsite authorities)
6. Make senior technical and management staff onsite available for consultation with NRC and State on a periodic basis
7. Provide meteorological and dose estimates to offsite authorities for actual releases via a dedicated individual or automated data transmission
8. Provide release and dose projections based on available plant condition information and foreseeable contingencies
9. Escalate to general emergency class, if appropriate

   or

10. Close out or recommend reduction in emergency class by briefing of offsite authorities at EOF and by phone followed by written summary within 8 hours of closeout or class reduction

State and/or Local Offsite Authority Actions

1. Provide any assistance requested
2. If sheltering near the site is desirable, activate public notification system within at least two miles of the plant
3. Provide public within at least about 10 miles periodic updates on emergency status
4. Augment resources by activating primary response centers
5. Dispatch key emergency personnel including monitoring teams and associated communications
6. Alert to standby status other emergency personnel (e.g., those needed for evacuation) and dispatch personnel to near-site duty stations
7. Provide offsite monitoring results to licensee, DOE and others and jointly assess them
8. Continuously assess information from licensee and offsite monitoring with regard to changes to protective actions already initiated for public and mobilizing evacuation resources
9. Recommend placing milk animals within 2 miles on stored feed and assess need to extend distance
10. Provide press briefings, perhaps with licensee
11. Escalate to general emergency class, if appropriate
12. Maintain site area emergency status until closeout or reduction of emergency class

000845

EXAMPLE INITIATING CONDITIONS: SITE AREA EMERGENCY

1. Known loss of coolant accident greater than makeup pump capacity

2. Degraded core with possible loss of coolable geometry (indicators should include instrumentation to detect inadequate core cooling, coolant activity and/or containment radioactivity levels)

3. Rapid failure of steam generator tubes (several hundred gpm leakage) with loss of offsite power

4. BWR steam line break outside containment without isolation

5. PWR steam line break with greater than 50 gpm primary to secondary leakage and indication of fuel damage

6. Loss of offsite power and loss of onsite AC power for more than 15 minutes

7. Loss of all vital onsite DC power for more than 15 minutes

8. Complete loss of any function needed for plant hot shutdown

9. Transient requiring operation of shutdown systems with failure to scram (continued power generation but no core damage immediately evident)

10. Major damage to spent fuel in containment or fuel handling building (e.g., large object damages fuel or water loss below fuel level)

11. Fire compromising the functions of safety systems

12. Most or all alarms (annunciators) lost and plant transient initiated or in progress

13. a. Effluent monitors detect levels corresponding to greater than 50 mr/hr for 1/2 hour or greater than 500 mr/hr W.B. for two minutes (or five times these levels to the thyroid) at the site boundary for adverse meteorology

    b. These dose rates are projected based on other plant parameters (e.g., radiation level in containment with leak rate appropriate for existing containment pressure) or are measured in the environs

    c. EPA Protective Action Guidelines are projected to be exceeded outside the site boundary

14. Imminent loss of physical control of the plant

15. Severe natural phenomena being experienced or projected with plant not in cold shutdown

    a. Earthquake greater than SSE levels

b. Flood, low water, tsunami, hurricane surge, seiche greater than design levels or failure of protection of vital equipment at lower levels

c. Sustained winds or tornadoes in excess of design levels

16. Other hazards being experienced or projected with plant not in cold shutdown

a. Aircraft crash affecting vital structures by impact or fire

b. Severe damage to safe shutdown equipment from missiles or explosion

c. Entry of uncontrolled flammable gases into vital areas. Entry of uncontrolled toxic gases into vital areas where lack of access to the area constitutes a safety problem

17. Other plant conditions exist that warrant activation of emergency centers and monitoring teams or a precautionary notification to the public near the site

18. Evacuation of control room and control of shutdown systems not established from local stations in 15 minutes

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Class

GENERAL EMERGENCY

Class Description

Events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

Purpose

Purpose of the general emergency declaration is to (1) initiate predetermined protective actions for the public, (2) provide continuous assessment of information from licensee and offsite organization measurements, (3) initiate additional measures as indicated by actual or potential releases, (4) provide consultation with offsite authorities and (5) provide updates for the public through offsite authorities.

Licensee Actions

1. Promptly inform State and local offsite aut orities of general emergency status and reason for emergency as soon as discovered (Parallel notification of State/local)
2. Augment resources by activating on-site Technical Support Center, on-site operational support center and near-site Emergency Operations Facility (EOF)
3. Assess and respond
4. Dispatch on-site and offsite monitoring teams and associated communications
5. Dedicate an individual for plant status updates to offsite authorities and periodic press briefings (perhaps joint with offsite authorities)
6. Make senior technical and management staff onsite available for consultation with NRC and State on a periodic basis
7. Provide meteorological and dose estimates to offsite authorities for actual releases via a dedicated individual or automated data transmission
8. Provide release and dose projections based on available plant condition information and foreseeable contingencies
9. Close out or recommend reduction of emergency class by briefing of offsite authorities at EOF and by phone followed by written summary within 8 hours of closeout or class reduction

State and/or Local Offsite Authority Actions

1. Provide any assistance requested
2. Activate immediate public notification of emergency status and provide public periodic updates
3. Recommend sheltering for 2 mile radius and 5 miles downwind and assess need to extend distances. Consider advisability of evacuation (projected time available vs. estimated evacuation times)
4. Augment resources by activatin primary response centers
5. Dispatch key emergency personn including monitoring teams and associated communications
6. Dispatch other emergency personnel to duty stations within 5 mile radius and alert all others to standby status
7. Provide offsite monitoring results to licensee, DOE and others and jointly assess them
8. Continuously assess information from licensee and offsite monitoring with regard to changes to protective actions already initiated for public and mobilizing evacuation resources
9. Recommend placing milk animals within 10 miles on stored feed and assess need to extend distance
10. Provide press briefings, perhaps with licensee
11. Maintain general emergency status until closeout or reduction of emergency class

0000849

EXAMPLE INITIATING CONDITIONS: GENERAL EMERGENCY

1. a. Effluent monitors detect levels corresponding to 1 rem/hr W.B. or 5 rem/hr thyroid at the site boundary under actual meteorological conditions

   b. These dose rates are projected based on other plant parameters (e.g., radiation levels in containment with leak rate appropriate for existing containment pressure with some confirmation from effluent monitors) or are measured in the environs

   Note: Consider evacuation only within about 2 miles of the site boundary unless these site boundary levels are exceeded by a factor of 10 or projected to continue for 10 hours or EPA Protective Action Guideline exposure levels are predicted to be exceeded at longer distances

2. Loss of 2 of 3 fission product barriers with a potential loss of 3rd barrier, (e.g., loss of primary coolant boundary, clad failure, and high potential for loss of containment)

3. Loss of physical control of the facility

   Note: Consider 2 mile precautionary evacuation

4. Other plant conditions exist, from whatever source, that make release of large amounts of radioactivity in a short time period possible, e.g., any core melt situation. See the specific PWR and BWR sequences below.

   Notes: a. For core melt sequences where significant releases from containment are not yet taking place and large amounts of fission products are not yet in the containment atmosphere, consider 2 mile precautionary evacuation. Consider 5 mile downwind evacuation (45° to 90° sector) if large amounts of fission products (greater than gap activity) are in the containment atmosphere. Recommend sheltering in other parts of the plume exposure Emergency Planning Zone under this circumstance.

   b. For core melt sequences where significant releases from containment are not yet taking place and containment failure leading to a direct atmospheric release is likely in the sequence but not imminent and large amounts of fission products in addition to noble gases are in the containment atmosphere, consider precautionary evacuation to 5 miles and 10 mile downwind evacuation (45° to 90° sector).

   c. For core melt sequences where large amounts of fission products other than noble gases are in the containment atmosphere and containment failure is judged imminent, recommend shelter for those areas where evacuation cannot be completed before transport of activity to that location.

d. As release information becomes available adjust these actions in accordance with dose projections, time available to evacuate and estimated evacuation times given current conditions.

5. Example PWR Sequences

a. Small and large LOCA's with failure of ECCS to perform leading to severe core degradation or melt in from minutes to hours. Ultimate failure of containment likely for melt sequences. (Several hours likely to be available to complete protective actions unless containment is not isolated)

b. Transient initiated by loss of feedwater and condensate systems (principal heat removal system) followed by failure of emergency feedwater system for extended period. Core melting possible in several hours. Ultimate failure of containment likely if core melts.

c. Transient requiring operation of shutdown systems with failure to scram which results in core damage or additional failure of core cooling and makeup systems (which could lead to core melt)

d. Failure of offsite and onsite power along with total loss of emergency feedwater makeup capability for several hours. Would lead to eventual core melt and likely failure of containment.

e. Small LOCA and initially successful ECCS. Subsequent failure of containment heat removal systems over several hours could lead to core melt and likely failure of containment.

NOTE: Most likely containment failure mode is melt-through with release of gases only for dry containment; quicker and larger releases likely for ice condenser containment for melt sequences. Quicker releases expected for failure of containment isolation system for any PWR.

6. Example BWR Sequences

a. Transient (e.g., loss of offsite power) plus failure of requisite core shut down systems (e.g., scram). Could lead to core melt in several hours with containment failure likely. More severe consequences if pumps trip does not function.

b. Small or large LOCA's with failure of ECCS to perform leading to core melt degradation or melt in minutes to hours. Loss of containment integrity may be imminent.

c. Small or large LOCA occurs and containment performance is unsuccessful affecting longer term success of the ECCS. Could lead to core degradation or melt in several hours without containment boundary.

d. Shutdown occurs but requisite decay heat removal systems (e.g., RHR) or non-safety systems heat removal means are rendered unavailable. Core degradation or melt could occur in about ten hours with subsequent containment failure.

7. Any major internal or external events (e.g., fires, earthquakes, substantially beyond design basis) which could cause massive common damage to plant systems resulting in any of the above.

APPENDIX 2

METEOROLOGICAL CRITERIA FOR EMERGENCY PREPAREDNESS AT OPERATING NUCLEAR POWER PLANTS

Introduction

10 CFR Part 50.47 requires that the Emergency Plan shall provide "(A)dequate methods, systems, and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition ..."

The basic functions needed to comply with the meteorological aspects of these requirements are:

1. A capability for making meteorological measurements.

2. A capability for making near real-time predictions of the atmospheric effluent transport and diffusion.

3. A capability for remote interrogation of the atmospheric measurements and predictions by appropriate organizations.

A staged schedule is provided in Annex 1 to this appendix for implementation of the meteorological elements addressing emergency preparedness requirements.

Meteorological Measurements

The emergency facilities and equipment as stated in Appendix E to 10 CFR Part 50 shall include "(E)quipment for determining the magnitude of and for continuously assessing the impact of the release of radioactive materials to the environment." To address this requirement, in part, the nuclear power plant operator shall have meteorological measurements from primary and backup systems.

Each site with an operating nuclear power plant shall have a primary meteorological measurements system. The primary system shall produce current and record historical local meteorological data. These data will provide a means to estimate the dispersion of radioactive material due to accidental radioactive releases to the atmosphere by the plant. The acceptance criteria for meteorological measurements are described in the proposed Revision 1 to U. S. NRC Regulatory Guide 1.23.

Each site with an operating nuclear power plant shall have a viable backup meteorological measurements system. The backup system shall provide meteorological information when the primary system is out of service and, thus, assurance that basic meteorological information is available during and immediately following an accidental airborne radioactivity release. The acceptance criteria for the backup meteorological measurements system are described in the proposed Revision 1 to U. S. NRC Regulatory Guide 1.23.

Atmospheric Transport and Diffusion Assessment

Appendix E to 10 CFR Part 50 states that "(T)he means to be used for determining the magnitude of and for continually assessing the impact of the release of radioactive materials shall be described ..." To address this requirement, in part, all licensees with operating nuclear power plants shall provide the description of their system for making current, site-specific estimates and predictions of atmospheric effluent transport and diffusion during and immediately following an accidental airborne radioactivity release from the nuclear power plant. The purpose of these predictions is to provide an input to the assessment of the consequences of accidental radioactive releases to the atmosphere and to aid in the implementation of emergency response decisions.

Near real-time, site-specific atmospheric transport and diffusion models shall be used when accidental airborne radioactive releases occur. Two classes of models are appropriate. The first, Class A, is a model and calculational capability which can produce initial transport and diffusion estimates for the plume exposure EPZ within 15 minutes following the classification of an incident. The second, Class B, is a numerical model which represents the actual spatial and temporal variations of plume distribution and can provide estimates of deposition and relative concentration of radioactivity within the plume exposure and ingestion EPZs for the duration of the release.

The Class A model shall use actual 15 minute average meteorological data from the meteorological measurements systems maintained by the licensee. The selected data shall be indicative of the conditions within the plume exposure EPZ. The Class A model shall provide calculations or relative concentrations (X/Q) and transit times within the plume exposure EPZ. Atmospheric diffusion rates shall be based on atmospheric stability as a function of site-specific terrain conditions. Site-specific local climatological effects on the trajectories, such as seasonal, diurnal, and terrain-induced flows shall be included. Source characteristics (release mode, and building complex influence) shall be factored into the model. The output from the Class A model shall include the plume dimensions and position, and the location, magnitude, and arrival time of (1) the peak relative concentration and (2) the relative concentrations at appropriate locations. The bases and justification for these model(s) and input data shall be documented. The performance and limitations of the model(s) shall also be included in the documentation.

The essential elements of the input, of model components, and of output to be incorporated in the Class A model are given to provide guidance for meteorological system implementation. Additional guidance will be prepared to outline the staff position on dose assessment capabilities to be used for emergency response.

Remote Interrogation

Appendix E to 10 CFR Part 50 states that there shall be "(P)rovisions for communications among the nuclear power reactor control room, the onsite technical support center and the near-site emergency operations facility ...." There shall also be "(P)rovisions for communications by the licensee with the NRC Headquarters and the appropriate NRC Regional Office Operations Center from the nuclear power reactor control room, the onsite technical support center, and the near-site emergency operations facility" and "... among the nuclear facility, the principal State and local emergency operations centers ...."

To address this requirement with respect to the meteorological information, all systems producing meteorological data and effluent transport and diffusion estimates at sites with operating nuclear power plants shall have the capability of being remotely interrogated. This will provide current meteorological data and transport and diffusion estimates to the licensee, emergency response organizations, and the NRC staff, on-demand, during emergency situations.

Proposed Revision 1 to Regulatory Guide 1.23 identifies the meteorological data that shall be available. The information that shall be available from the transport and diffusion assessment include the model outputs, input variables, model identification and data source information, plant identification, and data from other sources, as available.

The capability to make transport and diffusion calculations with specific inputs shall be provided. The primary and backup communications systems shall have a data transmission rate of 1200 BAUD and the rate(s) and other specifications indicated in proposed Revision 1 to Regulatory Guide 1.23.

Documentation for procedures to access and use the system shall be provided to the emergency response organizations and the NRC, and shall be available in the control room, the Technical Support Center (TSC) and the Emergency Operations Facility (EOF).

ANNEX 1 TO APPENDIX 2

<u>SCHEDULES TO IMPLEMENT THE METEOROLOGICAL ELEMENTS ADDRESSING EMERGENCY PLANNING RULES</u>

Schedule for Operating Reactors -- For operating reactors the following implementation milestones shall be met to address the functional requirements.

Milestones are numbered and tagged with the following code; <u>a-date</u>, <u>b-activity</u>, <u>c-minimum acceptance criteria</u>. They are as follows:

(1) a. January 2, 1981

b. Submittal of radiological emergency response plans

c. A description of the emergency plan which addresses the meteorological functions shall be provided

(2) a. March 1, 1981

b. Submittal of implementing procedures

c. Methods, systems, and equipment to assess and monitor actual or potential offsite consequences of a radiological emergency condition shall be provided

(3) a. April 1, 1981

b. Implementation of radiological emergency response plans

c. Three functions of Appendix 2 with the exception of the Class B model of the assessment capability

Alternative to milestone (3) requiring compensating actions:

A meteorological measurements system which is consistent with the existing technical specifications as the baseline or a primary system and/or a backup system of Appendix 2, or two independent backup systems shall provide the basic meteorological parameters (wind direction and speed and an indicator of atmospheric stability) on display in the control room. An operable dose calculational methodology (DCM) shall be in use in the control room and at appropriate emergency response facilities. The following compensating actions shall be taken by the licensee for this alternative:

(i) if only a primary <u>or</u> a backup system is in use:

- The licensee (a person who will be responsible for making offsite dose projections) shall check communications with the cognizant National Weather Service (NWS) first order station and NWS forecasting station on a monthly basis to ensure that routine meteorological observations and forecasts can be accessed.

- The licensee shall calibrate the meteorological measurements at a frequency no less than quarterly and identify a readily available source of meteorological data (characteristic of site conditions) to which they can gain access during calibration periods.

- During conditions of measurements system unavailability, an alternate source of meteorological data which is characteristic of site conditions shall be identified to which the licensee can gain access.

o The licensee shall maintain a site inspection schedule for evaluation of the meteorological measurements system at a frequency no less than weekly.

o It shall be a reportable occurrence if the meteorological data unavailability exceeds the goals outlined in Proposed Revision 1 to Regulatory Guide 1.23 on a quarterly basis.

(ii) The portion of the DCM relating to the transport and diffusion of gaseous effluents shall be consistent with the characteristics of the Class A model outlined in the assessment capability of Appendix 2.

(iii) Direct telephone access to the individual responsible for making offsite dose projections (Appendix E to 10 CFR Part 50(IV)(A)(4)) shall be available to the NRC in the event of a radiological emergency. Procedures for establishing contact and identification of contact individuals shall be provided as part of the implementing procedures.

This alternative shall not be exercised after July 1, 1982. Further, by July 1, 1981, a functional description of the upgraded capabilities and schedule for installation and operation shall be provided (see milestones 4 and 5).

(4) a. April 1, 1982

b. Installation of Emergency Response Facility meteorological hardware and software

c. Three functions of Appendix 2, with exception of the Class B model of the assessment capability

(5) a. July 1, 1982

b. Full operation of milestone 4

c. The Class A model (designed to be used out to the plume exposure EPZ) may be used in lieu of a Class B model out to the ingestion EPZ. Compensating actions to be taken for extending the application of the Class A model out to the ingestion EPZ include access to supplemental information (meso and synoptic scale) to apply judgment regarding intermediate and long-range transport estimates. The distribution of meteorological information by the licensee should be as follows by July 1, 1982:

Meteorological Information	CR	TSC	EOF	NRC and Emergency Response Organizations
Basic Met. Data (e.g., 1.97 Parameters)	X	X	X	X (NRC)
Full Met. Data (1.23 Parameters)		X	X	X
DCM (for Dose Projections)	X	X	X	X
Class A Model (to Plume Exposure EPZ)	X	X	X	X
Class B Model or Class A Model (to Ingestion EPZ)		X	X	X

(6) a. July 1, 1982 or at the time of the completion of milestone 5, whichever is sooner

b. Mandatory review of the DCM by the licensee

c. Any DCM in use should be reviewed to ensure consistency with the operational Class A model. Thus, actions recommended during the initial phases of a radiological emergency would be consistent with those after the TSC and EOF are activated

(7) a. September 1, 1982

b. Description of the Class B model provided to the NRC

c. Documentation of the technical bases and justification for selection of the type Class B model by the licensee with a discussion of the site-specific attributes

(8) a. June 1, 1983

b. Full operation of the Class B model

c. Class B model of the assessment capability of Appendix 2

o <u>Schedule for Near-Term OLs</u>

For applicants for an operating license at least milestones 1, 2, and 3 shall be met prior to the issuance of an operating license. Subsequent milestones shall be met by the same dates indicated for operating reactors. For the alternative to milestone 3, the meteorological measurements system shall be consistent with the NUREG-75/087, "Standard Review Plan For the Review of Safety Analysis Reports for Nuclear Power Plants," Section 2.3.3 program as the baseline or primary system and/or backup system.

APPENDIX 3

MEANS FOR PROVIDING PROMPT ALERTING AND NOTIFICATION OF RESPONSE ORGANIZATIONS AND THE POPULATION

NRC and FEMA recognize that the responsibility for activating the prompt notification system called for in this section is properly the responsibility of State and local governments. NRC and FEMA also recognize that the responsibility for demonstrating that such a system is in place rests with the facility licensee.

The initial notification when appropriate, of the affected population within the plume exposure pathway Emergency Planning Zone (EPZ) must be completed in a manner consistent with assuring the public health and safety. The design objective for the system shall be to meet the acceptance criteria of section B of this Appendix. This design objective does not, however, constitute a guarantee that early notification can be provided for everyone with 100% assurance or that the system when tested under actual field conditions will meet the design objective in all cases.

The plan shall include:

- o The specific organizations or individuals, by title, who will be responsible for notifying response organizations and the affected population and the specific decision chains for rapid implementation of alerting and notification decisions;

- o A capability for 24-hour per day alerting and notification;

- o Provision for the use of public communications media or other methods for issuing emergency instructions to members of the public, and

- o A description of the information that would be communicated to the public under given circumstances, for continuing instructions on emergency actions to follow, and updating of information.

A. Concept of Operations

Commercial broadcast messages are the primary means for advising the general public of the conditions of any nuclear accident. The primary means for alerting the public to an impending notification by public authorities may be any combination of fixed, mobile or electronic tone generators which will convey the alerting signal with sufficient timeliness and intensity to permit completion of notification by broadcast media in a timely manner. Since the timeliness of notification is a function of the accident severity, to be effective, appropriate systems, such as EBS and NOAA weather radio, should be placed on alert prior to the physical need for a public broadcast. The second or "Alert" category of events in Appendix 1 would ordinarily trigger the placing of broadcast media on alert, pending further instructions from State and local officials.

It is desirable for the public notification system to have a phasing capability. The arrangements for phasing are a function of the case-by-case population distribution or topography around each nuclear power station, and the details of each site-specific preparedness plan of State and local government.

B. Criteria for Acceptance

1. Within the plume exposure EPZ the system shall provide an alerting signal and notification by commercial broadcast (e.g., EBS) plus special systems such as NOAA radio. A system which expects the recipient to turn on a radio receiver without being alerted by an acoustic alerting signal or some other manner is not acceptable.

2. The minimum acceptable design objectives for coverage by the system are:

   a) Capability for providing both an alert signal and an informational or instructional message to the population on an area wide basis throughout the 10 mile EPZ, within 15 minutes.

   b) The initial notification system will assure direct coverage of essentially 100% of the population within 5 miles of the site.

   c) Special arrangements will be made to assure 100% coverage within 45 minutes of the population who may not have received the initial notification within the entire plume exposure EPZ.

   The basis for any special requirements exceptions (e.g., for extended water areas with transient boats or remote hiking trails) must be documented. Assurance of continued notification capability may be verified on a statistical basis. Every year, or in conjunction with an exercise of the facility, FEMA, in cooperation with the utility operator, and/or the State and local governments will take a statistical sample of the residents of all areas within about ten miles to assess the public's ability to hear the alerting signal and their awareness of the meaning of the prompt notification message as well as the availability of information on what to do in an emergency. The system plan must include a provision for corrective measures to provide reasonable assurance that coverage approaching the design objectives is maintained. The systems shall be operable no later than July 1, 1981. The lack of a specific design objective for a specified percent of the population between 5 and 10 miles which must receive the prompt signal within 15 minutes is to allow flexibility in system design. Designers should do scoping studies at different percent coverages to allow determination of whether an effective increase in capability per unit of cost can be achieved while still meeting the objective of item 2.a. above.

3. Public Notifications

A prompt notification scheme shall include the capability of local and State agencies to provide information promptly over radio and TV at the time of activation of the alerting signal. The Emergency Plans shall include evidence of such capability via agreements, arrangements or citation of applicable laws which provide for designated agencies to air messages on TV and radio in emergencies. Initial notifications of the public might include instructions to stay inside, close windows and doors, and listen to radio and TV for further instructions.

C. Physical Implementation

1. Communications Supporting Alerting and Notification Systems

Policy Objective

Federal, State and local government and utility authorities must develop and maintain plans, systems, procedures and relationships that are effective in mobilizing responsible authorities and operating elements in alerting and notifying the general public and in assuring appropriate and effective responses by the public.

Incident Alert Notification

The triggering of processes to mobilize forces and warn the public is dependent upon the communication between the nuclear power facility and government authorities (Federal, State and Local). The communications net must feature the following capacity:

a. Coverage: 24 hour coverage at the facility and at the primary points to receive and act upon notification.

b. Points to be Linked: Appendix 1 describes the conditions for assured dissemination of alert and warning information by the nuclear power plant to appropriate local and State warning points at all times and under all conditions. The system should include identical communications capabilities at primary and alternate operating locations.

c. Net Control: To assure effective utilization, net discipline and availability, one location should be assigned responsibility for net control and an alternate designated. The primary and alternate location should be a State or local civil government activity. It should issue and update procedures on testing, net access, and discipline and maintenance and repair.

d. System Availability and Reliability: All stations/points on the network and the communications linkage must provide a capability for immediate dissemination, receipt and acknowledgment of alert and warning messages on a 24-hour basis. The system should be able to function notwithstanding adverse environmental conditions, such as floods and power outages. It should not be subject to pre-emption for lower priority purposes nor to failure due to traffic (subscriber) overloading. To the extent a single system does not meet these performance standards, alternate means must be in place which have dissimilar vulnerability characteristics.

e. <u>Information Sensitivity</u>: The system design should take into consideration that alert and warning information is highly sensitive and if monitored or intercepted by unauthorized personnel, is subject to misinterpretation and can lead to undesirable and counterproductive reactions. Therefore it is desirable not to cite specific radio frequencies in public planning documents.

f. <u>System Features</u>: Dissemination should be rapid and reliable and provide acknowledgment and verification of message content. It is desirable for voice traffic to be supported by hard copy verification.

g. <u>Multipurpose Use</u>: Whatever system is designed and installed to meet all of the above capabilities for accident alerting may be used for communication in support of other response functions. However, systems designed for other purposes should not be adapted to incident alert notification unless (a) all of the criteria are met and (b) such adaptation does not compromise their primary purpose. Exception may be justified when a system designed for other purposes is adapted to incident alert notification to serve as a back-up to the primary system.

2. Notification of Response Organizations

a. Assigned Responsibility: Plans should clearly designate the responsibility and means of notifying response organizations by either the nuclear power plant or by the State or local warning points designated to receive initial alert notification.

b. Dissemination Time: Warning points cannot be encumbered by sequential call down processes nor can response organizations accept the time lost by such processes. This second level notification by warning points should be a one call process to all assigned organizations to be notified. Acknowledgement and message verification is essential. Message content must be clear, and brief. A preferred procedure is to communicate a posture code which calls for various predetermined responses for each organization based on its mission.

c. Capability of Organizations to be Notified: Organizations with immediate response functions must also have a 24-hour capability of receiving and acting upon a notification.

d. Internal Alerting: Each organization with response functions must develop reliable procedures for internal alerting and mobilization of forces. The system should account for the non-emergency nature of some organizations and the routine posture of key staff elements.

3. Sirens

Wherever proposed as part of a system, subject to later testing by statistical sampling, the design concept and expected performance must be documented as part of plans submitted by licensees, States and local governments. The designs of such systems must take into account the demography and topography of the areas being considered.

Some institutional alerting mechanisms are already in place (e.g., in schools, factories, hospitals, shopping centers, jails, and centralized offices). Siren systems should complement rather than substitute for these already in place.

The basic criterion needed for the design of a siren system is the acceptable dissonant sound level as described in "Outdoor Warning Systems Guide," Report No. 4100, by Bolt, Beranek and Newman, Inc., June 1979 (FEMA publication number CPG-1-17).

As an acceptable criteria at most locations 10db above average daytime ambient background should be a target level for the design of an adequate siren system. In cases involving industrial operations, a special survey to determine design sound level targets or an inside system may be needed to provide an audible 10db dissonant differential. Sirens on vehicles may be used to supplement fixed alert systems outside the inner five mile radius of the plume exposure EPZ.

Siren systems should be designed considering the demography and topography of an area, and taking into account other alert or notification systems in place or planned. The maxium sound levels received by any member of the public should be lower than 123db, the level which may cause discomfort to individuals.

a. The 10db dissonant differential is a conservative use of the 9db differential which is discussed in FEMA document CPG-1-17. Research has shown that a person is capable of being alerted by such a differential above or below the background ambient in the case of a predominately narrow band 300 to 800 Hz emitted by large sirens. The achievement of a positive differential of 10db has been a basic objective (although not always attained) of a wide range civil defense system.

b. In considering siren applications for nuclear power stations, the actual population density must be considered. The average population density around such stations is well below 2000 persons/ per square mile. Therefore, any use of population based criteria such as Figure 1 of CPG-1-17 is improper because the actual population density is predominately low.

c. The 10db differential above daytime ambient is meant to provide a distinguishable signal inside of average residential construction under average conditions. Where special individual cases require a higher alerting signal, it should be provided by other means than a generally distributed acoustic signal.

d. In keeping with the policy that sirens may only be a portion of a complete public notification system, NRC and FEMA believe that organizations proposing their use retain the responsibility for cost/benefit decisions which might involve the use alternative methods in thinly populated areas where such methods are cost effective while meeting the notification criteria for the Plume Exposure EPZ. Where sirens are proposed, the design may be based either on handbook values for background, or alternatively on field surveys.

e. <u>For Organizations Proposing Systems Without Field Surveys</u>

It may be very difficult, expensive, and time consuming to determine the average-day-time ambient for an EPZ. Sound level change with season, location, weather, traffic, ground cover, etc. If in combination with the uncertainties in siren performance, it is doubtful whether the predictability of detection would be increased above what could be obtained using existing date to develop standards, 50db(a) is a conservative estimate of the average day time ambient in areas with population below 2000 person/per square mile. For organizations proposing systems without field surveys, the following requirements apply:

That Figure 1 of CPG-1-17, "Outdoor Warning Systems Guide" published by FEMA, be used as the design criterion for siren systems in areas with population densities above 2000 persons/mi^2.

For areas with population densities below 2000 persons/mi^2 the siren system must be designed to produce a minimum of 60db(c). An attenuation factor of 10db loss per distance doubled should be used to determine siren range in the absence of special geographical features. Those organizations applying the criteria should document the basis for their selection of appropriate values to include:

* population densities, location of major transportaiton routes and heavy industry

* attenuation factors with distance

* siren output db(c) at 100 feet vs. assumed range and acoustic frequency spect a.

* maps showing siren location, size of coverage and any features that could affect siren performance (e.g., hills)

* mounting heights of sirens

* special weather conditions such as expected heavy snow which might modify the design assumptions

f. For Organizations Proposing Systems With Field Surveys

Instead of a 50db(a) estimate of average daytime background for areas with relatively low population (less than 2000 persons/per square mile), the average daytime (7 am to 10 pm) background may be measured.

The 10 db above average daytime ambient background may then be applied against these measurements.

Background db should be determined in a band about the siren signal frequency. Inclusion of background noise energy from outside this band could be misleading.

Figure 1 of CPG-1-17, "Outdoor Warning System Guide," should be used as the design criterion for siren systems in areas with population densities above 2000 persons/mi^2.

Organizations choosing to measure background ambients should document the basis for their selection to include:

* The basic requirements described in paragraph e concerning population densities, attenuation factors, siren output and spectra, and maps with terrain features

* Values of measured average daytime ambient background used as a basis for siren selection, to include survey location, how locations were selected, frequency range measured and measurement time span

* How seasonal changes were taken into account

g. <u>General Considerations</u>

NRC's licensees are urged to cooperate with State and local governments in the use of cost effective combinations of systems, including those already in place, as a means of satisfying this objective.

The siren signal shall be a 3 to 5 minute steady signal as described in Paragraph IV E of CPG-1-17 and capable of repetition.

h. <u>Siren Testing Guidance</u>

(1) Types of tests and suggested frequency are:

* Silent Test	every two weeks - log entry
* Growl Test (or equipment)	quarterly and when preventive maintenance is performed
* Complete Cycle Test	at least annually, and as required for formal exercises

(2) Oversight

* FEMA will receive an annual statement from the cognizant State or local authority that silent and growl tests have been performed. This may in turn be based on utility certification if the utility has directed responsibility for maintenance.

* FEMA will observe or receive a statement of the annual statistical sample of population in the EPZ hearing a test based on a field test or in conjunction with an exercise. FEMA will approve corrective measures necessary to provide assurance that siren systems are meeting the objectives for alerting the population (where they are the specific means for such alerting) approved jointly by NRC and FEMA.

4. Other Systems

a. The Emergency Broadcast System (EBS)

The Emergency Broadcast System (EBS) exists to furnish an expedited means of furnishing real time communications to the public in the event of war, threat of war, or grave national, or regional or local crisis.

To activate the EBS at the State level, a request may be directed to an Originating Primary Relay Station (usually an FM station located near the State capital) by the Governor, his desingated representative, the National Weather Service, the State Civil Preparedness or Emergency Services Office, or other designated State authority.

At the local level, a request for activation may be directed to the Common Program Control Station (CPCS-1), by designated officials of local government or the National Weather Service.

In either case, communications facilities developed for use in contacting and providing emergency program material may include any of the following: telephone, remote pickup units, NOAA Weather Wire Service or NOAA Weather Radio, police and fire communications, amateur and citizens band radio. Station management at the Originating Primary Relay Station and/or the Common Program Control Station authenticates the validity of all requests to activate the system. Other broadcast stations may activate the EBS on an individual basis as needed. This is important since station management is responsible for all program material broadcast to the public.

The Originating Primary Relay Station at the State level, or the Common Program Control Station at the local level, will take the following steps to activate the EBS:

1. Take action to broadcast emergency programming which may include recording the emergency message for use later.

2. Broadcast an initial statement.

3. Transmit the two-tone Attention Signal.

4. Broadcast the emergency announcement.

All other participating stations, alerted via their off-the-air monitoring of the two-tone signal, repeat the above procedures.

The State and local EBS is available for public officials who have specifically been designated "activating officials." These designees are responsible to the community for determining the appropriateness of activating the EBS for disseminating emergency public information. In this regard, the activating official could determine that an early alert to the broadcasters was advisable, because of certain actual or contemplated adverse conditions at a nuclear power plant. Such a decision could be implemented by the activating official notifying the broadcasters by available communications. The bottom line of the early alert would be to notify stations that are off the air, that there may be a need for activation, which in turn would cause the stations to notify appropriate personnel to stand by.

Alerting and notification systems around nuclear facilities must be integrated with the State and local EBS Operational Area Plan. Operational Area EBS plans involve agreements with the Common Program Control Stations (CPCS-1) and local emergency preparedness organizations while the State EBS plan is coordinated with the State emergency communications chairman. It may be necessary for utility organizations to sign agreements with CPCS-1 stations in order to cover a fast breaking general emergency described in Appendix 1. However, actual public notices would only take place upon authorization of governmental authorities.

b. National Oceanic and Atmospheric Administration (NOAA) Weather of Emergency Alert

Receivers compatible with Weather or Emergency Alert transmitters can be obtained commercially. Where transmitters or repeaters are not available, such could be provided independently, or perhaps by negotiation with the National Oceanic and Atmospheric (NOAA) or the Federal Communications Commission (FCC). Receivers and servicing thereof could be offered as a service.

c. Telephone Automatic Dialers

Systems are available whereby pre-selected telephone numbers could be dialed automatically, and a recorded announcement played when a telephone is answered. After a fixed number of rings, the next number is dialed automatically; the unanswered numbers are redialed at the end of the quene. This system could be most cost-effective and secure for warning to principal response officials, school systems, selected industrial complexes, downstream water works or isolated farms.

d. Aircraft with Loudspeakers

Hiking trails and hunting areas are illustrative of areas where it may not be feasible to provide a prompt notification by any other means except by aircraft equipped with powerful sound systems or by dropping prepared leaflets. Such would not work in bad weather, of course, but such areas are less likely to be used in bad weather. These areas should be reached on a best effort basis.

APPENDIX 4

EVACUATION TIME ESTIMATES WITHIN THE PLUME EXPOSURE PATHWAY EMERGENCY PLANNING ZONE

The following is an example of what shall be included in an evacuation times assessment study and how it might be presented. The example includes a complete outline of material to be covered, but only a few typical tables and explanations are provided. The requirements are intended to be illustrative of necessary considerations and provide for consistency in reporting. Because the evacuation time estimates will be used by those emergency response personnel charged with recommending and deciding on protective actions during an emergency the evacuation time estimates should be updated as local conditions change (e.g., change in type or effectiveness of public notification system).

I. INTRODUCTION

This section of the report should make the reader aware of the general location of the nuclear power plant and plume exposure pathway emergency planning zone, and generally discuss how the analysis was done.

A. Site Location and Emergency Planning Zone

A vicinity map showing the plant location shall be provided along with a detailed map of the plume exposure pathway emergency planning zone (EPZ). The map shall be legible and identify transportation networks, topographical features and political boundaries. (See planning element J.10.a.)

B. General Assumptions

All assumptions used in the analysis shall be provided. The assumptions shall include such things as automobile occupancy factors, method of determining roadway capacities, and method of estimating populations.

C. Methodology

A description of the method of analyzing the evacuation times shall be provided. If computer models are used, a general description of the algorithm shall be provided along with a source for obtaining further information or documentation.

II. DEMAND ESTIMATION

The objective of this section is to provide an estimate of the number of people to be evacuated. Three potential population segments shall be considered: permanent residents, transients, and persons in special facilities. Permanent residents includes all people having a residence in the area, but not in institutions. Transients shall include tourists, employees not residing in the area, or other groups that may visit the area. Special facility residents include those confined to institutions such as hospitals and nursing homes. The school population shall be evaluated in the special facility segment. Care should be taken to avoid double counting.

A. Permanent Residents

The number of permanent residents shall be estimated using the U. S. Census data or other reliable data, adjusted as necessary, for growth. (See planning element J.10.b.). This population data shall then be translated into two subgroups: 1) those using autos and those without autos. The number of vehicles used by permanent residents is estimated using an appropriate auto occupancy factor. A range of two to three persons per vehicle would probably be reasonable in most cases.

An alternative approach is to calculate the number of vehicles based on the number of households that own vehicles assuming one vehicle per household is used in evacuation. Regardless of the approach used, special attention must be given to those households not having automobiles. The public transport-dependent population must, therefore, be considered as a special case.

B. <u>Transient Populations</u>

Estimates of transient populations shall be developed using local data such as peak tourist volumes and employment data for large factories. Automobile occupancy factors would vary for different transient groups. Tourists might have automobile occupancy factors in the range of three to four while a factory would probably have a factor of less than 1.5 persons per vehicle. This population segment along with the permanent population subgroup using automobiles constitute the general population group for which an evacuation time estimate shall be made.

C. <u>Special Facility Population</u>

An estimate for this special population group shall usually be done on an institution-by-institution basis. The means of transportation are also highly individualized and shall be described. Schools shall be included in this segment.

D. Emergency Planning Zone and Sub-areas

The sub-areas for which evacuation time estimates are required must encompass the entire area within the plume exposure EPZ. Additionally, evacuation time estimates are also required for simultaneous evacuation of the entire plume exposure pathway. The areas to be considered are as follows:

Radius	Area
about 2 miles	four 90° sectors
about 5 miles	four 90° sectors
about 10 miles (EPZ)	four 90° sectors
about 10 miles (EPZ)	entire EPZ

When making estimates for the outer sectors, assume that the inner adjacent sectors are being evacuated simultaneously. The boundaries of the sub-areas shall be based upon the same factors as the EPZ, namely demography, topography, land characteristics, access routes, and local jurisdictions. To the extent practical, the sector boundaries shall not divide densely populated areas. Where meteorological conditions such as dominant wind directions, warrant special consideration, an additional sub-area may need to be defined and a separate estimate made for this case. The EPZ and its sub-areas shall be identified by mapping on United States Geological Survey (USGS) 7-1/2-minute series quadrant maps when available. Special facilities shall also be noted on these maps, to the extent that their locations can be geographically specified.

Populations shall be provided by evacuation areas as specified in planning element J.10.b. For the purpose of determining evacuation times it may also be useful to summarize population data by sector and distance from the plant. Figure 1 is an example of such a summary. Separate totals shall be provided for the three population segments. Figure 2 shows the population totals translated into the number of vehicles estimated to be used in evacuation.

III. TRAFFIC CAPACITY

This section of the report shall show the facilities to be used in evacuation. It shall include their location, types, and capacities. A complete review shall be made of the road network. Analyses shall be made of travel times and potential locations for serious congestion in potential corridors. (The analyses may be simplified in extreme rural areas.) The entire road network shall be used but local routes shall be carefully selected and analyzed to minimize their impact on the major routes should queuing or cross traffic conflicts occur. Care shall be taken to avoid depending only on high-capacity interstate and similar type routes because of limitations of on-ramp capacities. Alternatively, special traffic management plans may be developed to effectively utilize available capacity. Evacuation shall be based on general radial dispersion.

A. Evacuation Roadway Network

A map showing only those roads used as primary evacuation routes shall be provided. Figure 3 is an example. The map need not show local access streets necessary to get to the evacuation routes. Each segment of the network shall be numbered in some manner for reference.

The sector and quadrant boundaries shall also be indicated. (See planning elements J.10.a and b.).

B. Roadway Segment Characteristics

A table such as example Table 1 shall be provided indicating all the evacuation route segments and their characteristics, including capacity. The characteristics of a segment shall be given for the narrowest section or bottleneck if the roadway is not uniform in the number of lanes throughout the segment.

IV. ANALYSIS OF EVACUATION TIMES

As indicated previously, evacuation time is composed of several components. Each of these components shall be estimated in order to determine the total evacuation time.

A. Reporting Format

Table 2 shows the desired format for presenting the data and results for each type of evacuation. Each of the evacuation time components is presented along with the total evacuation time. Two conditions -- normal and adverse -- are considered in the analyses. Adverse conditions would depend on the characteristics of a specific site and could include flooding, snow, ice, fog or rain. The adverse weather frequency used in this analysis shall be identified and shall be severe enough to define the sensitivity of the analysis to the selected events. These conditions will affect both travel times and capacity. More than one adverse condition may need to be considered. That is, a northern site with a high summer tourist population should consider rain, flooding, or fog as the adverse condition as well as snow with winter population estimates.

The text accompanying the table shall clearly indicate the critical assumptions which underlie the time estimates; e.g., day versus night, workday versus weekend, peak transient versus off-peak transient, and evacuation on adjacent sectors versus nonevacuation. The relative significance of alternative assumptions shall be addressed, especially with regard to time dependent traffic loading of the segments of the evacuation roadway network.

Some modification of the reporting format may be appropriate, depending on local circumstances.

B. Methodology

The method for computing total evacuation time shall be specified. Two approaches are acceptable. The simplest approach is to assume that events are sequential. That is to say, for example, that no one begins to move until all persons are warned and prepared to leave before anyone starts moving. The time is estimated by simply adding the maximum time for each component. This approach tends to overestimate the evacuation time.

The second approach, which is more complex and will be discussed further, is to combine the distribution functions for the various evacuation time components. This second approach may result in reduced time estimates due to more realistic assumptions. The added complexity of analysis, therefore, may be warranted at sites with long evacuation times. When distribution functions are used, estimates are made of the likelihood that each stage in an evacuation sequence will be accomplished within a given period of time. These conditional probabilities depend upon completion of the preceding stage. For example, formulation of family units or other evacuation groups does not commence until notification is received. Some of these distribution functions must be based on the judgment of the estimators. Computation of the joint distribution functions of evacuation times are made. Typically, the joint distribution assumes the form of an S-shape curve as shown in Figure 4. The evacuation time function is fairly smooth for large homogeneous population segments such as the general public. Special facilities, such as hospitals and industrial centers, produce less smooth functions, or discontinuous ones. The assessment of evacuation time may be easily updated should further analyses be conducted, assumptions changed, or new plans developed.

When distributions are used, distribution functions for notification of the various categories of the evacuee population shall be developed. The distribution functions for the action stages after notification predict what fraction of the population will complete a particular action within a given span of time. There are separate distributions for auto-owning households, school population, and transit dependent populations. These distribution functions can be constructed in a variety of ways, depending greatly on the kinds of data available for the actual site being studied. The previously developed conditional distributions are combined to develop the time distributions for the various population segments departing their home or other facility from which they are being evacuated. For example, for the auto-owning population segment, these vehicles are then loaded onto the roadway network in order to compute travel times and delays.

Regardless of the means by which the time and amount of traffic to be loaded on the network is determined (i.e., sequentially or using distribution functions), it is necessary to calculate the on-road travel and delay times. In this step, traffic from each sector is assigned to available evacuation routes, and, if assigned volumes exceed capacity, delay times must be calculated using a queuing analyses. Traffic queue (backup) locations and estimated delay times should be indicated on the area map.

An estimate of the time required to evacuate that segment of the non-car-owning population dependent upon public transport shall be made, in a similar manner to that used for the auto-owning population. This estimate shall include consideration of any special services which might be initiated to serve this population subgroup. Such services might include fixed-route departures from designated assembly points.

Estimates for special facilities shall be made with consideration for the means of mobilization of equipment and manpower to aid in evacuation, and the needs for designated employees or staff to delay their evacuation in order to shut down industrial facilities. Each special facility shall be treated on an individual basis. Weather conditions and time of day conditions shall be considered. Consideration shall be given to the impact of peak populations including behavioral aspects.

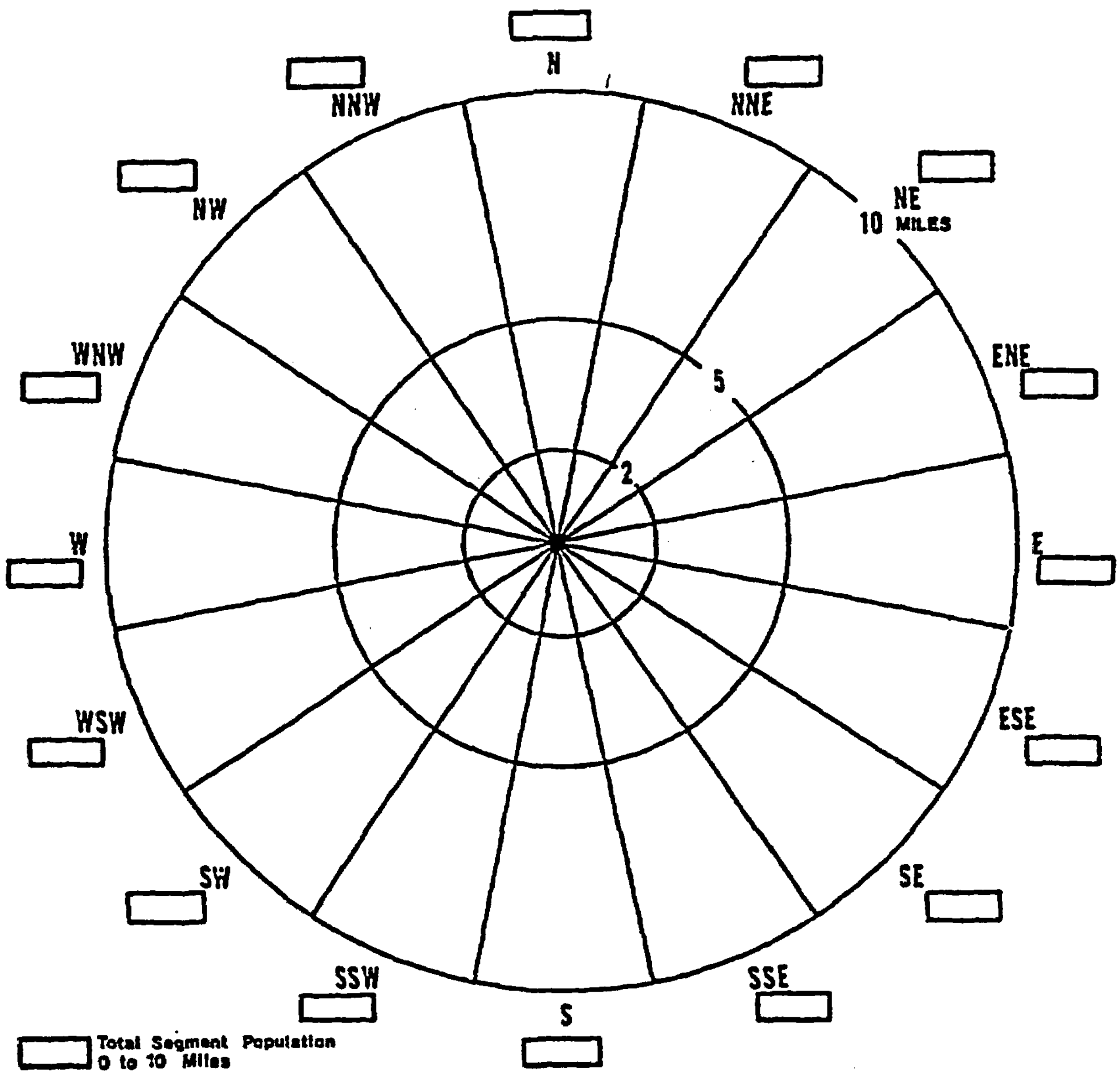
All of the results shall be reported in the format previously indicated. This format summarizes the maximum time for each component and for each sector. The components may or may not be directly additive based on the methodology used and stated in the report. Where distribution functions are used the percentage of the population as a function of time should be reported (See Figure 4 for an example format).

V. OTHER REQUIREMENTS

The time required for confirmation of evacuation shall be estimated. Candidate methods include visual confirmation by aircraft or ground vehicles and telephone confirmation.

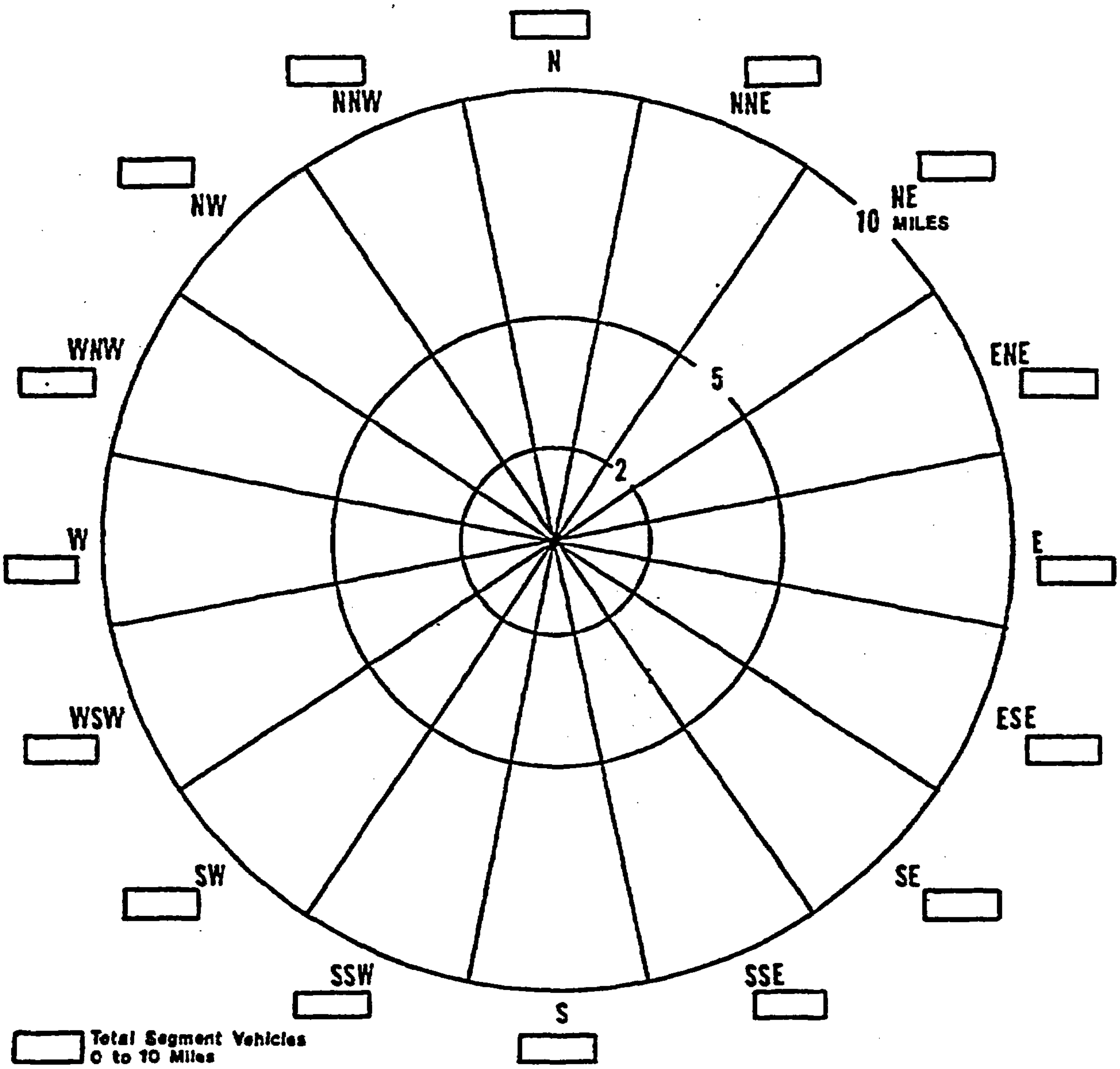
Specific recommendations for actions that could be taken to significantly improve evacuation time shall be given. Where significant costs may be involved, preliminary estimates of the cost of implementing these recommendations shall be given.

A review of the draft submittal by the principal organizations (State and local) involved in emergency response for the site shall be solicited and comments resulting from such review included with the submittal.



POPULATION TOTALS			
RING, MILES	RING POPULATION	TOTAL MILES	CUMULATIVE POPULATION
0-2		0-2	
2-5		0-5	
5-10		0-10	

Figure 1: Example of Format for Presentating Population Data By Sector



VEHICLES TOTALS			
RING MILES	RING VEHICLES	TOTAL MILES	CUMULATIVE VEHICLES
0-2		0-2	
2-5		0-5	
5-10		0-10	

Figure 2: Example of Format for Presenting Vehicle Data By Sector

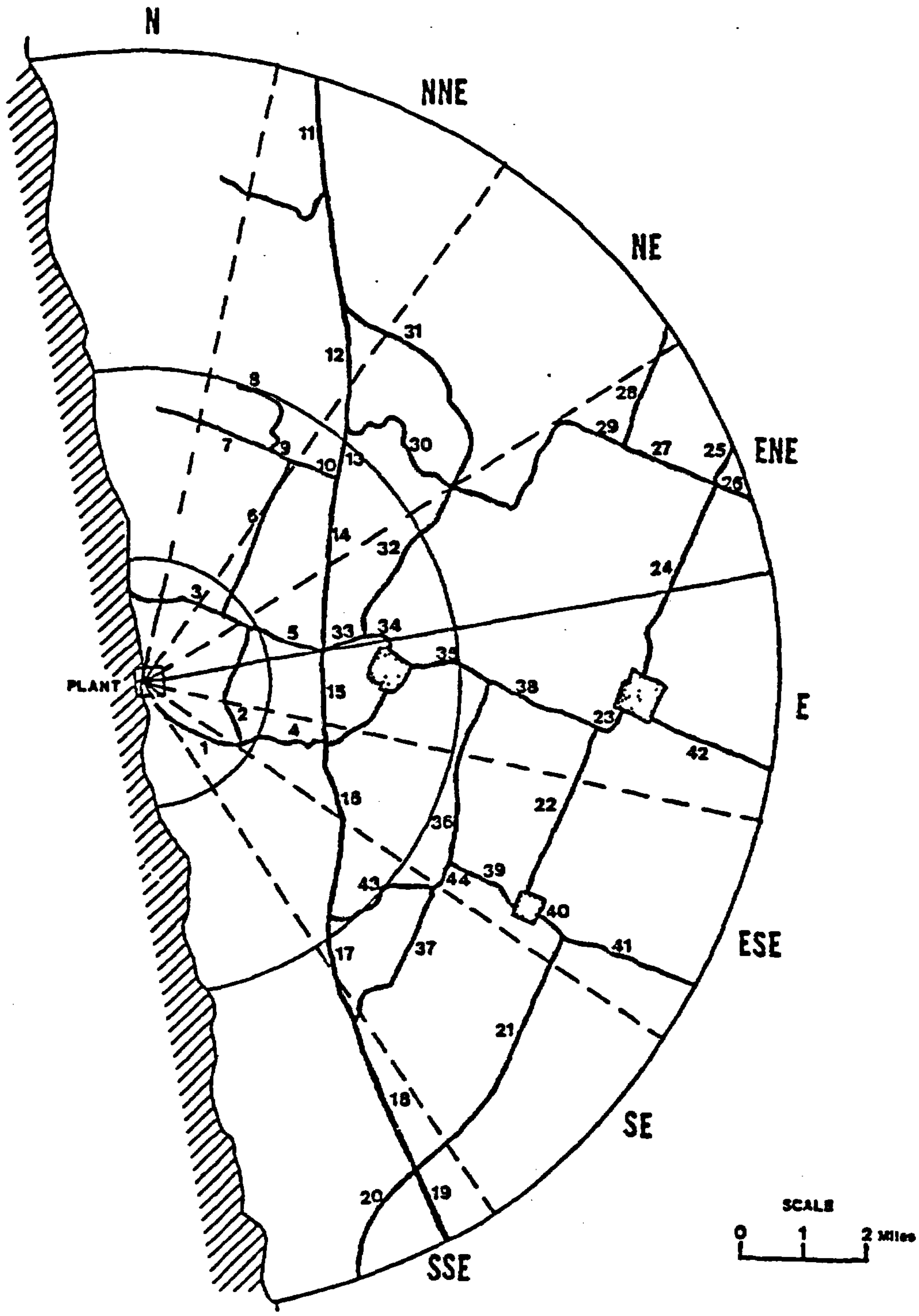


Figure 3: Example of Evacuation Roadway Network

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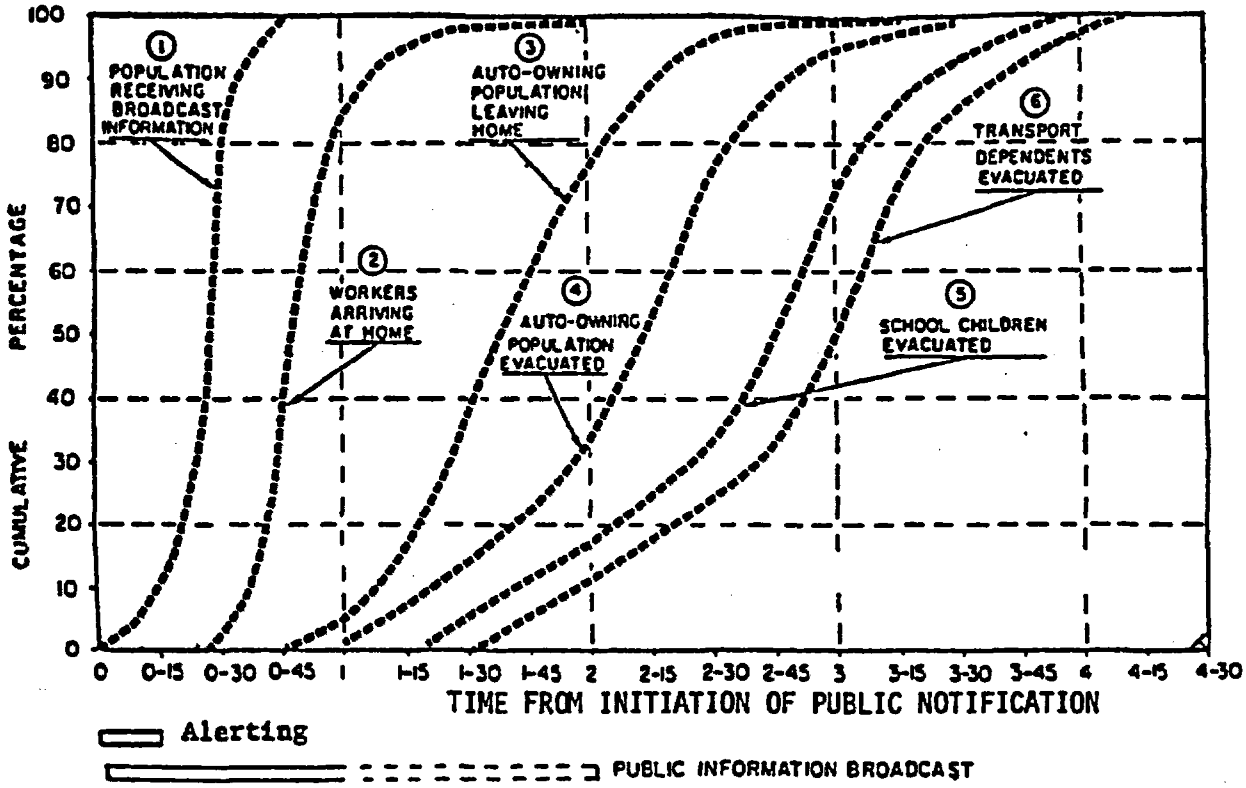

Figure 4: Example of Additional Reporting Format for Time Estimates of Population Evacuation When Probability Distributions Are Used

Note: These curves are suggestive of a hypothetical 10-mile radius EPZ. Similar curves can be developed for sub-areas of the entire EPZ. The horizontal displacement of these curves along the time axis as well as the slope of the curves will vary depending upon the characteristics of the EPZ or sub-areas of the EPZ.

Table 1: Example of Roadway Characteristics

Segment	Number[1] of Lanes	Type[2]	Capacity[3]	Comments[4]

NOTES: [1]Total number of through lanes in both directions. If roadway cross section is not uniform, use section with least number of lanes

[2]F = Freeways and Expressways
U = Urban Streets
R = Rural Highways

[3]If known

[4]Indicate any special conditions that may affect roadway capacity.

Table 2: Example of Summary of Results of Evacuation Times Analysis

AREAS	Permanent Population	Permanent Pop. Vehicles	Transient Population	Transient Pop. Vehicles	Evacuation Capacity Per Hour	Notification Time	Preparation Time	Permanent Pop. Response Normal Condit: ns	Permanent Pop. Response Adverse Contitions	Transient Pop. Response Normal Conditions	Transient Pop. Response Adverse Conditions	General Pop. Evac. Time Normal Conditions	General Pop. Evac. Time Adverse Conditions	Confirmation Time	Special Pop. Evac. Time Normal Conditions	Special Pop. Evac. Time Adverse Conditions
WITHIN TWO MILES																
WITHIN FIVE MILES																
WITHIN TEN MILES																

APPENDIX 5

GLOSSARY

Three major "organizations" are identified by the three columns headed "Licensee", "State", and "Local" in Part II of this document. "Organizations" are also indicated in the document generally with a modifying word preceeding the term "organization", e.g.: -

Principal (organizations):
Federal, State, Local agencies or departments or executive offices and nuclear utilities (licensees) having major or lead roles in emergency planning and preparedness.

Sub - (organizations):
Any organization such as agencies, departments, offices or local jurisdictions having a supportive role to the principal or lead organization(s) in emergency planning and preparedness.

Federal (organizations):
Agencies, departments or their components, of the U. S. Federal government, having a role in emergency planning and preparedness.

State (organization):
The State government agency or office having the principal or lead role in emergency planning and preparedness. There may be more than one State involved, resulting in application of the evaluation criteria separately to more than one State. To the extent possible, however, one State should be designated lead.

<u>Local</u> (organization):

The <u>local</u> government agency or office having the <u>principal</u> or <u>lead</u> role in emergency planning and preparedness. Generally this will be the County government. Other local government entities (e.g., towns, cities, municipalities, etc.), are considered to be sub-organizations with supportive roles to the <u>principal</u> or lead local government organization responsible for emergency planning and preparedness. In some cases there will be more than one lead organization at the local level, but designation of one lead local organization is preferable.

<u>Private Sector</u> (organizations):

Industry, volunteer, quasi-governmental etc. having a role in emergency planning and preparedness.

It is not possible to totally specify each class or type of organization that may be involved in the total emergency planning and preparedness scheme. Nor is it possible to define the particular roles, function and responsibilities of "principal organizations" and "sub-organizations". This is a matter that is best defined by the various parties involved in developing plans and preparedness for each nuclear site. Where the guidance in this document indicates a function that must be performed, emergency planners at all levels, must decide and agree among themselves, which organization is to perform such function. As a minimum, one lead agency at the State level and one lead local government agency having 24 hour manning is required.

Onsite Technical Support Center (TSC) and Licensees Near-Site Emergency Operations Facility (EOF)

For description and functional criteria for the TSC and EOF, see "Functional Criteria for Emergency Response Facilities" (NUREG-0696), U. S. Nuclear Regulatory Commission.

Consequences	The results or effects (especially projected doses or dose rates) of a release of radioactive material to the environment.
Core Melt Accident	A postulated reactor accident in which the fuel melts because of overheating.
Emergency Planning Zone (EPZ)	A generic area defined about a nuclear facility to facilitate offsite emergency planning and develop a significant response base. It is defined for the plume and ingestion exposure pathways. During an emergency response best efforts are made making use of plan action criteria without regard to whether particular areas are inside or outside EPZs.
Ingestion Exposure Pathway	The principal exposure from this pathway would be from ingestion of contaminated water or foods such as milk or fresh vegetables. The duration of principal exposures could range in length from hours to months.

Planning Basis

Guidance in terms of (1) Size of Planning Area (Distance); (2) Time Dependence of Release; and (3) Radiological Characteristics of Releases.

Planning Standard

The standard that must be met for onsite and offsite emergency plans and preparedness. (Ref: 10 CFR 50 section 50.47 Emergency Plans, 45 FR No. 162 pp 55409; and proposed 44 CFR 350 section 350.5 Criteria for Review and Approval of State and Local Radiological Emergency Plans and Preparedness, 45 FR No. 123 pp 42344).

Plume Exposure Pathway

The principal exposure sources from this pathway are: (a) whole body external exposure to gamma radiation from the plume and from deposited materials and (b) inhalation exposure from the passing radioactive plume. The duration of principal potential exposures could range in length from hours to days.

Projected Dose

An estimate of the radiation dose which affected individuals could potentially receive if protective actions are not taken.

Protective Action	An action taken to avoid or reduce a projected dose. (Sometimes referred to as protective measure).
Protective Action Guide	Projected absorbed dose to individuals in the general population which warrants protective action.

NRC FORM 335
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U.S. NUCLEAR REGULATORY COMMISSION

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16. ABSTRACT (200 words or less)

The purpose of this guidance and upgraded acceptance criteria document is to provide a basis for NRC licensees, State and local governments to develop radiological emergency plans and improve emergency preparedness. The guidance is the product of the joint FEMA/NRC Steering Committee established to coordinate the agencies' work in emergency preparedness associated with nuclear power plants. This document is consistent with NRC and FEMA regulations and supersedes other previous guidance and criteria published by FEMA and NRC on this subject. It will be used by reviewers in determining the adequacy of State, local and nuclear power plant licensee emergency plans and preparedness.

17. KEY WORDS AND DOCUMENT ANALYSIS
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radiological emergency planning
radiological emergency
emergency response plans

17a DESCRIPTORS
radiological emergency response plans

17b IDENTIFIERS/OPEN-ENDED TERMS

18 AVAILABILITY STATEMENT
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NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

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NUREG-0654, REV. 1
FEMA-REP-1

CRITERIA FOR PREPARATION AND EVALUATION OF RADIOLOGICAL EMERGENCY
RESPONSE PLANS AND PREPAREDNESS IN SUPPORT OF NUCLEAR POWER PLANTS

NOVEMBER 1980

000901

December 8, 2000

Mr. Robert J. Barrett
Site Executive Officer
New York Power Authority
Indian Point 3 Nuclear Power Plant
Post Office Box 215
Buchanan, NY 10511

SUBJECT: INDIAN POINT 3 -EVALUATED EMERGENCY PREPAREDNESS EXERCISE - NRC INSPECTION REPORT NO. 05000286/2000-010

Dear Mr. Barrett:

The enclosed report documents an inspection at the Indian Point 3 Nuclear Power Plant. The inspectors evaluated the performance of your emergency response organization during the November 15, 2000, full-participation exercise, and the post-exercise critique as specified in the reactor oversight program. The inspectors discussed the findings of this inspection with Mr. F. Dacimo and other members of your staff on November 16, 2000.

This inspection was an examination of activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. Within these areas, the inspection consisted of a selected examination of procedures and representative records, observations of activities, and interviews with personnel.

Based on the results of this inspection, no significant issues were identified.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

000902

Should you have any questions regarding this report, please contact Mr. Richard J. Conte at (610) 337-5183.

Sincerely,

/RA/ Daniel H. Dorman for:

Wayne D. Lanning, Director
Division of Reactor Safety

Docket No. 05000286
License No: DPR-64

Enclosures:
1. Inspection Report No. 05000286/2000-010
2. NRC's Revised Reactor Oversight Process

cc w/encls:
C. D. Rappleyea, Chairman and Chief Executive Officer
E. Zeltmann, President and Chief Operating Officer
J. Knubel, Chief Nuclear Officer and Senior Vice President
F. Dacimo, Plant Manager
H. P. Salmon, Jr., Vice President of Engineering
W. Josiger, Vice President - Special Activities
J. Kelly, Director - Regulatory Affairs and Special Projects
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T. Morra, Executive Chair, Four County Nuclear Safety Committee
Chairman, Committee on Corporations, Authorities, and Commissions
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000905

U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No:	05000286
License No:	DPR- 64
Report No:	05000286/2000-010
Licensee:	New York Power Authority P.O. Box 215 Buchanan, NY 10511
Facility:	Indian Point 3 Nuclear Power Plant
Dates:	November 14 - 16, 2000
Inspectors:	D. Silk, Senior Emergency Preparedness Inspector, DRS (Lead) N. McNamara, Emergency Preparedness Inspector, DRS L. James, Resident Inspector, Indian Point 3, DRP P. Bissett, Senior Operations Engineer R. Nimitz, Health Physicist, DRS R. Bores, State Liaison Officer (FEMA RAC Member)
Approved by:	Richard J. Conte, Chief Operational Safety Branch Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000286/2000-010, on 11/14-16/2000; New York Power Authority; Indian Point 3 Nuclear Power Plant. Emergency Preparedness exercise.

This inspection was conducted by region based inspectors and the resident inspector. The significance of issues is indicated by their color (green, white, yellow, red) and was determined by the Significance Determination Process (Enclosure 2).

Cornerstone: Emergency Preparedness

- No significant findings were identified.

Report Details

1. REACTOR SAFETY

Cornerstone: Emergency Preparedness (EP)

1EP1 Drill, Exercise, and Actual Events

a. Inspection Scope

The inspectors reviewed:

- The exercise scenario to determine if the exercise would test major elements of the licensee's emergency plan.
- The licensee's biennial full-participation exercise performance by focusing on risk-significant activities in the control room simulator, the technical support center, and the emergency operations facility (EOF). The risk significant areas are emergency classifications, offsite notification, radiological assessment, and protective action recommendations (PARs).
- The licensee's exercise performance in the above mentioned facilities, as well as, the operations support center and the emergency news center.
- The emergency response organization's (ERO) recognition of abnormal plant conditions, classification of emergency conditions, notification of offsite agencies, development of PARs, command and control, communications, utilization of repair and field monitoring teams, and the overall implementation of the emergency plan.
- The post-exercise critique to evaluate the licensee's self-assessment of the exercise.

b. Issues and Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed licensee findings pertaining to the recent drills, and the last licensee biennial exercise critique to determine if significant performance trends exist and to determine the effectiveness of licensee corrective actions based upon ERO performance during the exercise.

b. Issues and Findings

No findings of significance were identified.

4OA6 Exit Meeting

The inspectors presented the inspection results to Mr. F. Dacimo and other members of your staff at the conclusion of the inspection on November 16, 2000. The licensee had no objections to the NRC findings.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

R. Barrett, Site Executive Officer
F. Dacimo, Plant Manager
A. Grosjean, Senior Emergency Planner
R. Martin, Emergency Plan Engineer
M. Wilson, Emergency Planning Coordinator

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

None

Closed

None

Discussed

None

LIST OF ACRONYMS USED

DEP	Drill and Exercise Performance
EAL	Emergency Action Level
EOF	Emergency Operations Facility
ERO	Emergency Response Organization
GE	General Emergency
PAR	Protective Action Recommendation

ENCLOSURE 2

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revised its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety

- Initiating Events
- Mitigating Systems
- Barrier Integrity
- Emergency Preparedness

Radiation Safety

- Occupational
- Public

Safeguards

- Physical Protection

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.

TECHNICAL MEMORANDUM

NUREG/CR-2859
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(Distribution Codes:
RE and XA)

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EVALUATION OF AIRCRAFT CRASH HAZARDS ANALYSES FOR NUCLEAR POWER PLANTS

by

C. A. Kot, H. C. Lin, J. B. van Erp,*
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Components Technology Division

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U. S. Nuclear Regulatory Commission
Washington, D. C. 20555
under Interagency Agreement DOE 40-550-75

NRC FIN No. A2076

* Reactor Analysis and Safety Division, ANL
** ATResearch Associates, Inc., Glen Ellyn, Illinois

ABSTRACT

The state of knowledge concerning aircraft crash hazards to nuclear power plants is critically evaluated. This effort is part of a study to analyze the potential effects of offsite hazards upon the safety of nuclear power plants and to develop a technical basis for the assessment of siting approaches for such facilities. The evaluation includes the deterministic modeling of aircraft crash scenarios and threat environments, the estimation of the effects on and the response of the vital plant systems, and the probabilistic aspects of the crash problem, i.e., data bases and statistical methodologies. Also critically reviewed are past licensing experience and regulatory practice with respect to aircraft crash hazards.

In general it is found that the data bases, methodologies and modeling approaches are adequate to estimate the threat and plant response. However, this knowledge is not always fully used in specific applications. Siting of nuclear power plants relative to aircraft hazards is a risk based procedure that considers both probabilities of crash occurrence and their consequences. In this context it appears feasible to improve the site screening procedures and to develop exclusion zones from controlled air spaces (airports, airways, etc.) based solely on local aviation statistics and independent of plant design. Methodologies for treating complex aviation environments such as multiple airports and overlapping airways are needed, as are guidelines for crash target calculations. Further investigations of crash scenarios, particularly those that could lead to multiple or propagating failures, should be pursued.

NRC FIN No.	Title
A2076	Analysis of Offsite Hazards and Their Effects on Nuclear Facilities

PREFACE

This report presents the results of an investigation conducted for the U.S. Nuclear Regulatory Commission (NRC), Office of Nuclear Regulatory Research, Division of Health, Siting and Waste Management. The work was performed under a Standard Order for DOE Work (FIN No. A2076) and was a joint effort of the Argonne National Laboratory (ANL) and ATResearch Associates, Inc. of Glen Ellen, Illinois. The project monitor was Mr. R. P. Grill, NRC/RES; his helpful suggestions and reviews are gratefully acknowledged. The authors also express their thanks to Mr. C. Campe, NRC/NRR for his review and comments on the report manuscript.

C. A. Kot, Manager
Structural Systems Analysis Section
Components Technology Division-335
Argonne National Laboratory
June 1982

steel is determined, for example, by the von Mises criterion: $I_2 - \frac{2}{3}\bar{\sigma}^2(k) = 0$, where $\bar{\sigma}(k)$, the uniaxial tensile yield stress, is a function of a hardening parameter k. Figure 10(d) shows a typical curve for kinematic hardening. Failure in steel bars occurs when the ultimate tensile strain is reached.

6.2.2 Material Nonlinearity Effects on Structural Response

Zimmermann et al. [42] investigated the effects of material nonlinearities on response spectra resulting from the impact of a Boeing 707-320 on the secondary containment of a BWR reactor such as shown in Fig. 11. They used a finite-element model which considered concrete cracking and crushing as well as steel yielding for the analysis. The resulting displacement time histories are shown in Fig. 12. Comparison of the nonlinear and linear displacement time histories shows a significant increase in the vertical displacement (28%) in the vicinity of impact zone, which fades out rapidly away from the impact point as expected, since the response far away from the impact area is primarily elastic behavior. Therefore, if the impact loading is sufficient to produce any permanent deformation, a more complicated constitutive equation must be used in order to obtain the real structural response. Since there is no consensus theory which can predict all material behavior of concrete, such as tension, compression, crushing, microcracking, creeping, etc., the choice should depend on the most important.

6.3 Local Structural Response

6.3.1 Local Failure Mechanisms

The impact of an aircraft upon a concrete containment of a nuclear power plant generally may result in the damage to concrete walls. The damage may be local or may produce an overall dynamic response of the target wall. Kennedy [43] presented a detail review of procedures for the analysis and design of concrete structures to withstand missile impact effects. Missile velocities generated by aircraft crashes may be between 100 and 1500 ft/sec. The local damage due to aircraft impact consists of spalling of concrete from the front (impacted) surface and scabbing of concrete from the rear surface of the target together with missile penetration into the target as shown in Fig. 13. If the damage is sufficient, the missile may perforate and pass through the target.

As the velocity of the impacting missile increases, pieces of concrete are spalled off from the impacted surface of the target. This spalling creates a spall crater that can extend over an area substantially greater than the

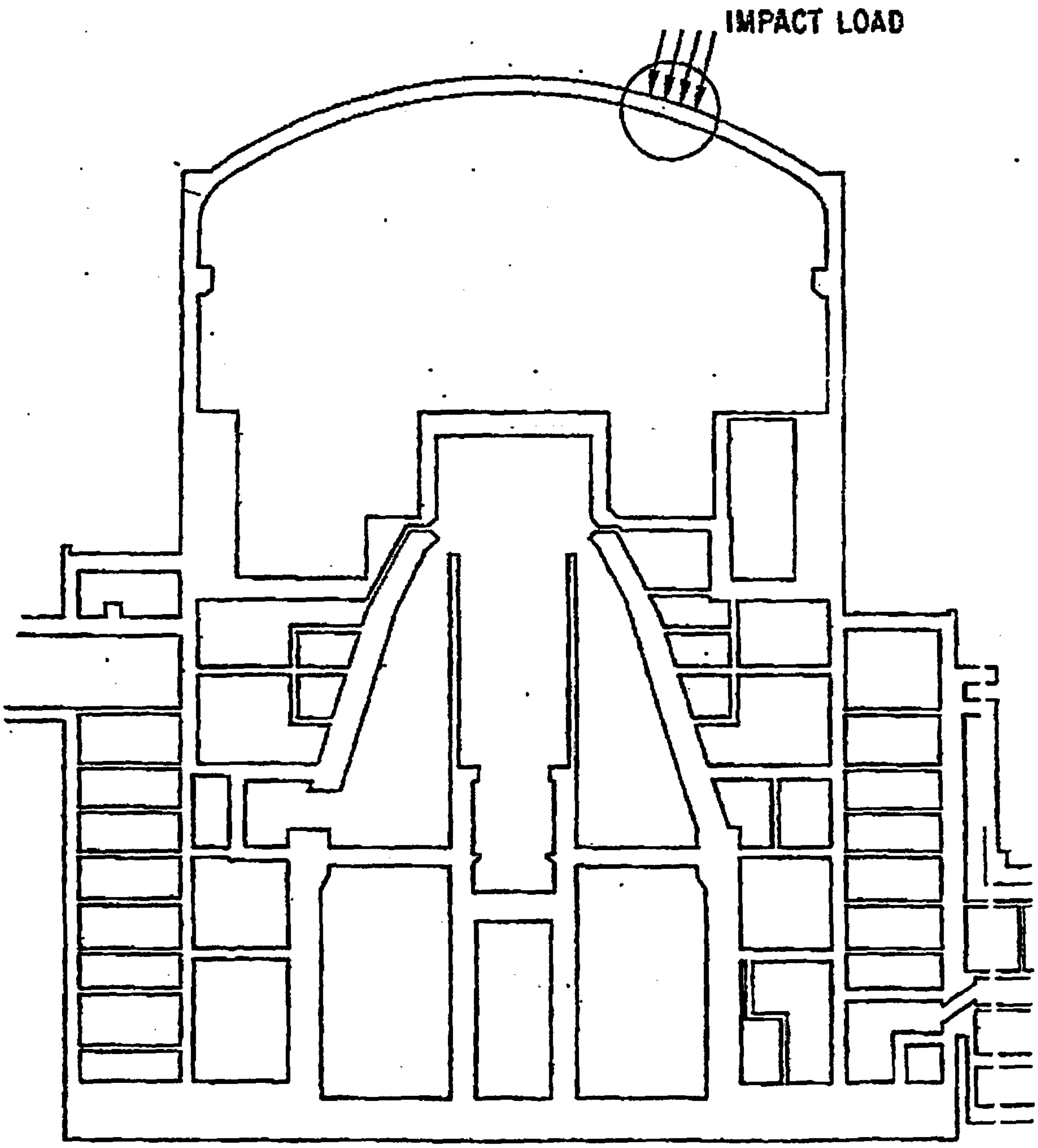

Fig. 11 Impact on Reactor Building [42]

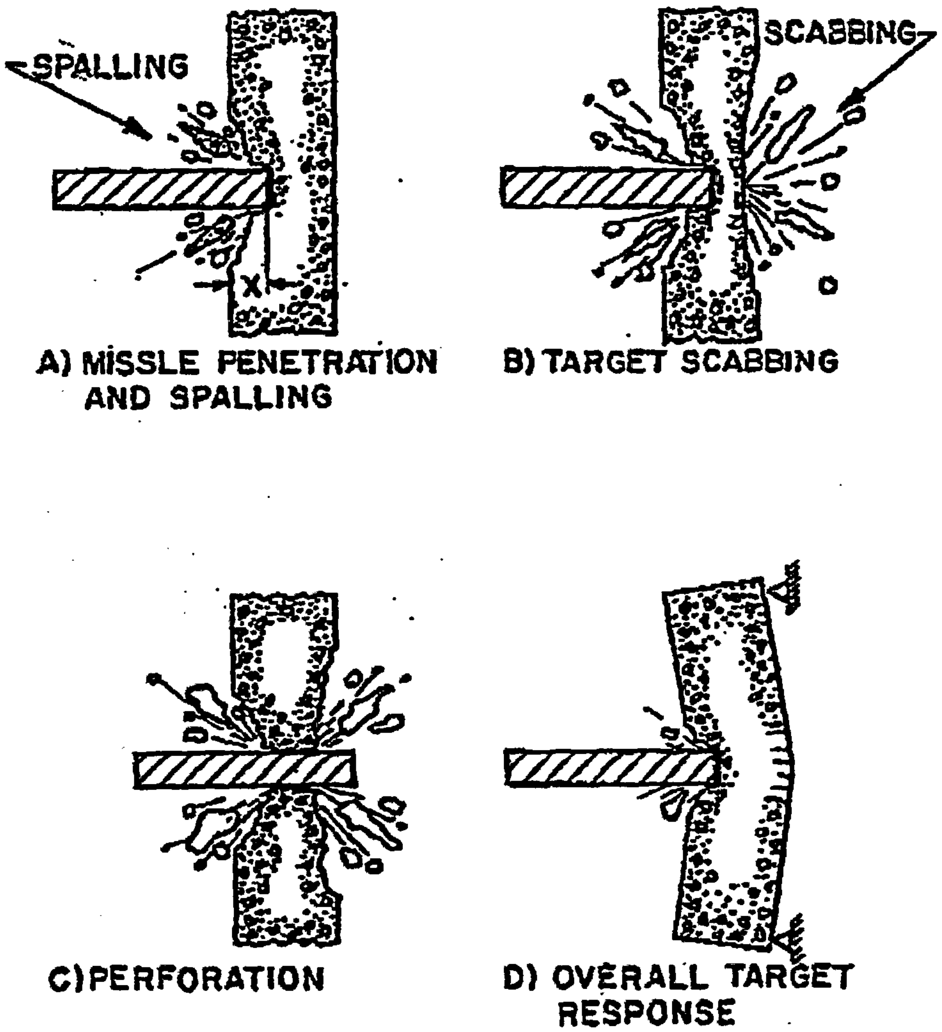

Fig. 13 Missila Impact Phenomena [43]

cross-sectional area of the striking missile. As the velocity increases, the missile will penetrate the target to depths beyond the depth of the spall crater, forming a cylindrical hole with diameter slightly greater than the missile diameter. As the penetration continues, the missile will stick to the concrete target; this is called plastic impact. Further increases in velocity produce cracking of the concrete on the rear surface followed by scabbing of concrete from this rear surface. The zone of scabbing will generally be much wider, but not as deep as the front surface spall crater.

Once scabbing begins, the depth of penetration will increase rapidly. For barrier thickness to missile diameter ratios less than five, the pieces of scabbed concrete can be large and have substantial velocities. As the missile velocity increases further, perforation of the target will occur as the penetration hole extends through to the scabbing crater. Still higher velocities will cause the missile to exit from the rear surface of the target. Upon plastic impact, portions of the kinetic energy of the impacting missile are converted to strain energy associated with deformation of the missile and energy losses associated with target penetration. The remaining energy is absorbed by the impact target. This absorbed energy results in an overall target response that includes flexural deformation of the target barrier and the subsequent deformation of its supporting structures. A review of commonly used empirical procedures for determining local missile impact effects such as penetration depth, perforation thickness, and scabbing thickness for concrete targets subjected to hard-missile impact can be found in [43]. Note that these empirical formulas were developed by the Army Corps of Engineers, the National Defense Research Committee, and others many years ago based on experimental observation. Today, with the advent of the finite-element method and after intensive research in fracture mechanics, it is possible to predict these phenomena analytically. The above discussion deals with concrete structures only. If the aircraft impact on a steel structure, then only penetration, perforation, and overall response will occur. The numerical approach to various target geometries of this type can be found in [44].

6.3.2 Failure-Mode Analysis Using Plastic Shells of Revolution Theory

Degen, Furrer, and Jemielewski [45] have investigated the effects of a large commercial airplane crashing perpendicularly on the surface of a spherical reactor building dome. They obtained the carrying capacity of the structure under an equivalent static load using the yield-line theory of circular plates, and calculated the sectional forces using linear-elastic shell theory. They then calculate the failure load and distribution of sectional forces using the plastic shell theory. The analysis was performed using the computer code STARS-2P developed by Svalbonas and Levine [46]. This code performs plastic analysis of shells of revolution. Plastic effects are approximated using the initial strain appproach, and different modes of hardening may be taken into account. From the results, they obtained the failure zone mechanism at the apex of a spherical shell subjected to aircraft impact over a finite loading area. The results are shown in Fig. 14.

Degen et al. [45] also presented failure mode analysis by the finite-element program TRIDI [47] which utilizes three-dimensional elements for concrete and one-dimensional elements for reinforcing steel. This program considered nonlinear stress-strain relationships for concrete under multiaxial stress, cracking and crushing under a triaxial stress state, and elastic-plastic behavior for reinforcing steel. The calculation of collapse load using yield-line theory for plates, STARS-2P for shell of revolution, and three-dimensional TRIDI are in the pressure range of p = 11 to 25, 30 to 35, and 25 to 30 kg/cm^2, respectively as reported by Degen et al.

Since the calculated collapsed load was assumed to be distributed over a certain contact area, the impacting total load corresponding to a range of 30-35 kg/cm^2 results in 28,000-33,000 tons, using the peak load-velocity relationship; the crushing velocity of a large commercial airplane which the structure under consideration could still sustain may be between 480 and 530 km/hr. If the impact velocity further increases, part of the energy (not absorbed by the structure) will be retained in the falling object. Figure 15 shows the maximum remaining loads as a function of crash velocity. Within the velocity range of 480 to 750 km/hr, only part of the peak load may act on the structure, but over 750 km/hr the total peak load must be used. Carlton and Bedi [48] and Gupta and Seaman [49] also studied the local response of reinforced concrete to missile impacts using a different computer code. The analysis appears to be adequate for the description of failure mode mechanisms.

6.4 Structural System and Equipment Response

There are many studies [50-58] concerning the comparison of the dynamic response of a typical nuclear power plant subjected to a modest earthquake and to the impact of aircraft crashes. Ahmed et al. [50-51] used a finite-element beam model and modal superposition techniques to obtain the time history response and the corresponding floor response spectra of the structure/component. The effect of soil-structure interaction is considered in that study. Figure 16 shows the structural idealization of the nuclear power plant in the finite-element model. Figure 17 shows the comparison of

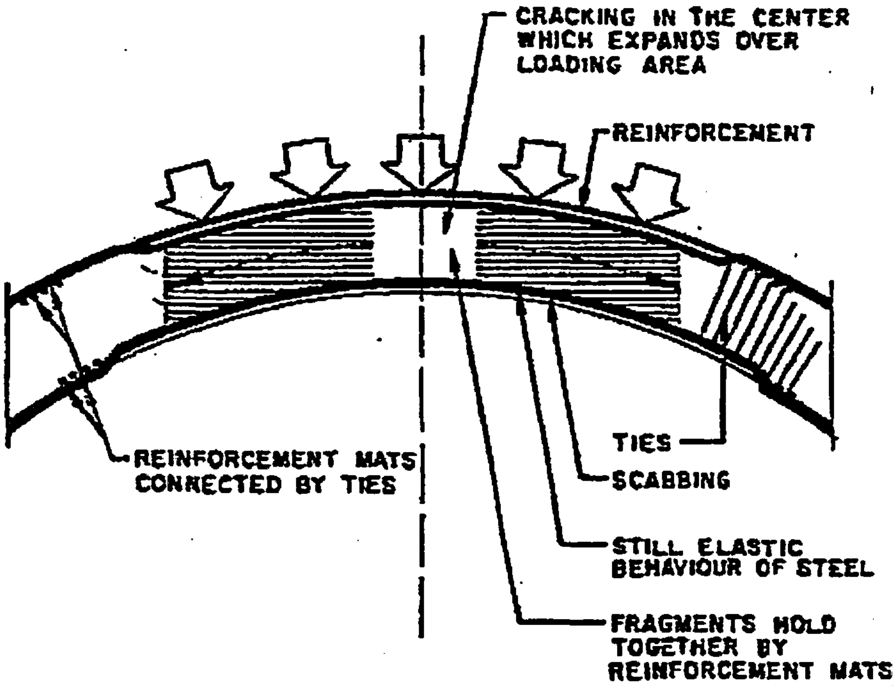

Fig. 14 Failure Zone at the Apex [45]

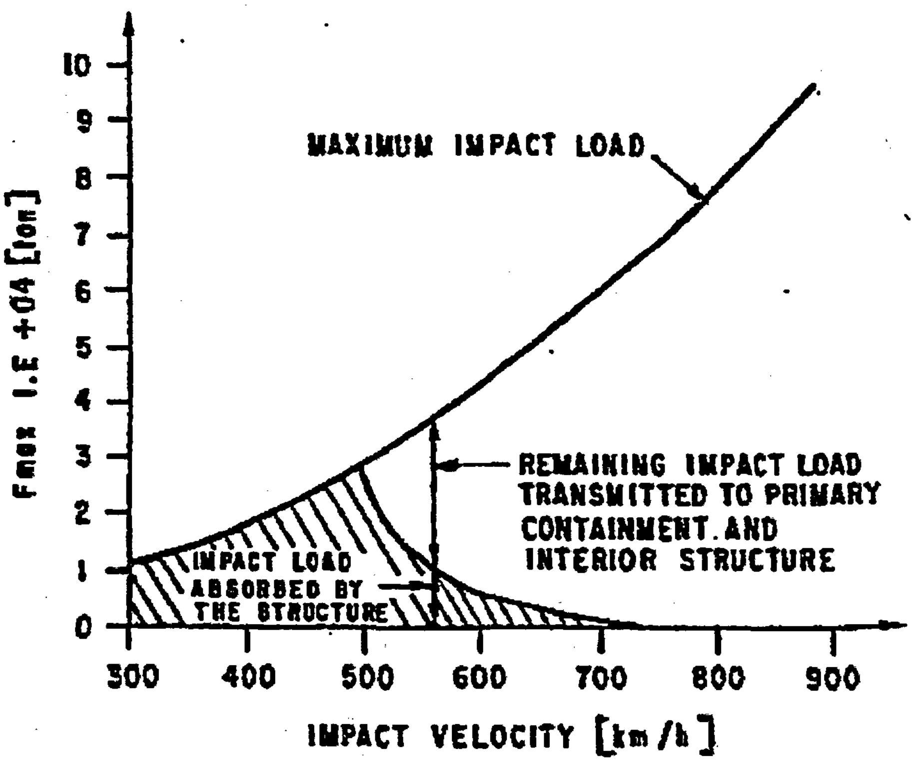

Fig. 15 Maximum Remaining Impact Load as a Function
of Impact Velocity [14]

7. FIRE AND EXPLOSION HAZARD ASSOCIATED WITH AN AIRCRAFT CRASH

The crash of an aircraft, at least those events in which the aircraft structure is significantly damaged, will release large quantities of fuel in the general vicinity of the crash site. A significant fraction of the maximum aircraft takeoff weight is fuel; thus, quantities of the order of 50,000 lb of fuel can be expected to be released by large military aircraft such as an FB-111 fighter. Even larger quantities of fuel are used in large commercial aircraft. The fuels are, typically, JP-1, JP-4, or kerosene. These fuels are not highly volatile, but they burn readily and when properly mixed with air can explode.

Crash events which consist of relatively long ground traverses frequently sever or puncture fuel tanks (i.e., wing structures), and the leaking fuel is sprayed and spilled out over rather long distances forming vapor clouds and liquid pools. Crash events which consist of the abrupt arresting of the entire aircraft, and, therefore, providing essentially total structural collapse of the aircraft in a few tenths of a second, release their fuel very rapidly, spilling the fuel on the impact point (structure) and the immediate area. Again a portion of the fuel will tend to mix with the surrounding air forming a potentially explosive cloud. A major portion of the fuel will form pools or wet down the adjacent surfaces.

The crash event, being rather catostrophic, will be associated with the release of significant amounts of energy, heat, and sparks such that ignition sources will generally be present; it is therefore most likely that a fuel fire will occur. These fires will be local events and last for periods of time of the order of many minutes, perhaps a few tens of minutes. They will generate a significant amount of heat (thermal radiation and hot gases) and combustion products (smoke and toxic fumes). The hot combustion products, largely gases, will be transported upward due to buoyancy forces and will move downwind. Thus, these gases have the potential of reaching nearby intake vents of the surrounding facilities.

In addition to the above potential combination and toxic hazards, which appear to be tolerable in many instances, at least for adequately designed facilities, it is important to examine the crash event and the local impact area for unique situations which may cause an unacceptable hazard. For example, in the case of an impact on a double enveloped containment structure it may be possible to deposit a significant adequate quantity of fuel between the two envelopes. The subsequent vaporization and ignition of the resulting vapor-air mixture could lead to a rather violent explosion environment and impose upon the primary containment relatively severe loads. These loads are different in character than those imposed by the impact process, but may be just as severe. Furthermore, these loads will occur shortly after the impact load, and, therefore, the response of the structure to the combined load event should be examined.

A relatively large body of data and analysis methodologies exists relating to fires resulting from the crashes of aircraft. This data base resides primarily in the FAA domain and is supported by a yet larger data base dealing with fire and fire effects in general. The quantification of fires and their effects, expecially pool fires, has been developed to a stage where the general characteristics (i.e., flame height, duration, radiative flux, etc.) are known. While it is still difficult to predict with precision the outcome of various aircraft fuel-spill fires, the influence of many major parameters such as fuel properties and wind effects is understood. The major difficulties generally lie in the complex nature of the fuel distribution, the influence of random effects, and the somewhat extreme geometries which may be encountered in any realistic aircraft crash at a plant site (i.e., cluster of buildings).

The explosion hazard resulting from the crash of an aircraft is difficult to define for several reasons. One is that the basic phenomenon is very complex, and many or varied degrees of energy release or combustion can occur. The other is that the dissemination of the fuel and its partial mixing with the surrounding air to form an explosive cloud are virtually impossible to predict with any acceptable degree of accuracy. The approach used by Eichler and Napendensky [59] and others in dealing with a broad class of accidental vapor cloud explosions was to define, from accident and experimental data, reasonably conservative TNT equivalence factors for these events. Because of the very dynamic fuel dispersion and the low vapor pressure of aviation fuels, the applicability of the TNT equivalency approaches to the explosion hazards from catastrophic aircraft crashes must be carefully evaluated. This is particularly true for the effects close-in to the explosion. Napadensky and Takata [60], while examining train accidents involving the release of combustible materials for a 10-year period in which a fire and/or an explosion occurred, observed that approximately 36 percent of the events involved both fire and explosion, while approximately 56 percent of the events involved only fire. The remaining 8 percent of the events involved only an explosion.

It is clear that a broad spectrum or mix of fire and explosion events can occur, and while the amount of fuel involved in any explosion event may be quite small, the occurence of such events must be considered. If only one percent of the fuel, say 500 lb for the FB-111 fighter plane, is involved in such an event, the blast environment will be equivalent to the detonation of approximately 1000 lb of TNT. The local blast characteristics of a vapor cloud are substantially different from those of a TNT explosion; however, at longer ranges the equivalency concept is appropriate. For the above explosion the "safe" overpressure of 1 psi will exist at a range of approximately 120 m.

It is difficult to obtain a complete and perhaps correct picture of the design review and acceptance process as it applies to any given offsite hazard feature, since the details are frequently divided between many diverse documents in the dockets and in the iterative question and answer format which is employed. Using the fire hazard analysis of the Seabrook Station [37], the following level of treatment appears to be typical. The production of a combustible vapor is dismissed as being insignificant (in quantity) on the basis that the atomization process takes place over the 0.3-sec. impact (load) duration. This duration is not representative of the vapor production period. Clearly a number of vapor production mechanisms will exist. For example, some fuel will be sprayed into the atmosphere and then fall as "rain" settling at a rate much less than 0.1 m/s, depending upon droplet size. Furthermore, fuel can be expected to be thrown over large elevated surfaces with subsequent flow downward over these surfaces due to the action of gravity.

Depending upon the surface temperature of these exposed surfaces (exposed perhaps to the sun) and the possible presence of fire, the vaporization rate can be amplified significantly and the vaporization period may last for many minutes. Fires are usually treated in a more comprehensive manner than explosions since a variety of pool conditions can be postulated, and using a vaporization rate (for a burning pool) of approximately 0.004 cm/s, the durations of the fires can be estimated. Flame temperatures, radiative flux levels, and fire durations can then be used (but usually not explicitly used) to claim that fires do not constitute a threat to the facility. The probability of fuel entering the relatively few openings (vent stack, air intake vents, steam line tunnels, etc.) to these collective structures will generally be quite low simply on an area basis, although specific values are frequently not cited. Account has been taken [61] of the internal concrete wall which acts as a missile barrier when present to prevent flames and fuel from directly entering the air intake. It would appear, however, that this is too optimistic since vaporized fuel, hot gaseous reaction products, and to a certain extent portions of liquid (fuel) streams will flow around such obstructions.

Based on the review of past licensing experience, it appears that fire and explosion hazards have been treated with much less care than the direct aircraft impact and the resulting structural response. Therefore, the claim that these fire/explosion effects do not represent a threat to nuclear power plant facilities has not been clearly demonstrated.

targets subjected to missile impact. Simplified procedures are defined for determining the dynamic response of the target wall and for eventing overall failure of the wall.

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Offsite Hazards: Aircraft Crash
Type of Model: Deterministic
Author: Krutzik, N. J.
Title: Analysis of Aircraft Impact Problems
Reference: Advanced Structural Dynamics, Ed. by Donea, J. Applied Science Publishers, Ltd., London, 1978, pp 337-386

Brief Description:

This paper presented the characterization of the load case induced by various aircraft impacting on the nuclear power plants. Also the influence of elastoplastic deformation in the area of impact on load function is discussed. The dynamic structural investigations for reactor building are presented using beam and shell models. The modal damping, damping parameters, soil parameters are discussed. Investigation of two neighboring buildings of unequal sizes show that the presence of the smaller building has a damping effect on the dynamic response of the larger building, and the impact on the larger building excites oscillations in the smaller buildings. As far as the comparisons with an earthquake and an explosive shock wave, in the low frequency range (up to 5 Hz) the load case of an earthquake is governing whereas in the high frequency range (above 10 Hz) the load case of an aircraft crash dominated.

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Offsite Hazards: Aircraft Crash
Type of Model: Probabilistic
Authors: Niyogi, P. K., Boritz, R. C., and Bhattacharyya, A. K.
Title: Safety Design of Nuclear Power Plants Against Aircraft Impacts
Reference: United Engineers & Constructors, Inc., Philadelphia, PA.

Brief Description:

A nuclear power plant is considered adequately designed against aircraft hazards if the probability of aircraft accidents resulting in radiological consequences greater than 10 CFR part 100 guidelines is less than about 10^{-7} per year Otherwise an aircraft accident is considered a design basis event and the plant must be hardened up to the point at which the above criterion is met. In many cases it has been sufficient to demonstrate that the probability of an impact on a safety-related building is less than 10^{-7} per year. In other cases, it is necessary to take into account the intrinsic hardness of buildings and structures designed to withstand tornado, seismic, and manmade hazards in order to demonstrate that an aircraft impact presents an acceptable risk In some cases, however, it is necessary to consider aircraft impacts as design basis events and to specify the level of hardening required to satisfy the design criterion.

DD-02-XX

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION

Samuel J. Collins, Director

In the Matter of ENTERGY NUCLEAR OPERATIONS, INC. (Indian Point Nuclear Generating Unit Nos. 1, 2, and 3)	)))))))	Docket Nos. 50-003, 50-247, and 50-286 License Nos. DPR-5, DPR-26, and DPR-64 (10 CFR 2.206)

PROPOSED DIRECTOR'S DECISION UNDER 10 CFR 2.206

I. Introduction

By letter dated November 8, 2001, as supplemented on December 20, 2001, Riverkeeper, Inc., et al. filed a Petition pursuant to Title 10 of the *Code of Federal Regulations*, Section 2.206 (10 CFR 2.206). The Petitioners requested that the U.S. Nuclear Regulatory Commission (NRC) take the following actions: (1) order the licensee to suspend operations, revoke the operating license, or adopt other measures resulting in a temporary shutdown of the Indian Point Nuclear Generating Unit Nos. 2 and 3 (IP2 and 3); (2) order the licensee to conduct a full review of the facility's vulnerabilities, security measures, and evacuation plans; (3) require the licensee to provide information documenting the existing and readily attainable security measures which protect the IP facility against land, water, and airborne terrorist attacks; (4) immediately modify the IP2 and 3 operating licenses to mandate certain specified security measures sufficient to protect the facility; and (5) order the revision of the licensee's emergency response plan and Westchester County's radiological emergency response plan (RERP) to account for possible terrorist attacks and prepare a comprehensive response to multiple, simultaneous attacks in the region, which could impair the efficient evacuation of the a··a. In addition, the Petitioners requested that the NRC take prompt action to permanently retire the facility if, after conducting a full review of the facility's vulnerabilities, security measures, and evacuation plans, the NRC finds that the IP facility cannot be adequately protected against terrorist threats. Further, separately from the above issues, the Petitioners requested that the NRC order the licensee to undertake the immediate conversion of the current water-cooled spent fuel storage system to a dry cask system. The bases for the requests are that (1) the IP facility is a plausible target of future terrorist actions, (2) actual threats against nuclear power plants have been documented, (3) IP is currently vulnerable to a catastrophic terrorist attack, (4) a terrorist attack on IP2 and 3 would have significant public health, environmental, and economic impacts, and (5) the Westchester County's RERP is inadequate because it is based on erroneous assumptions.

In a letter dated December 20, 2001, the NRC informed the Petitioners that their request for a full review of the facility's vulnerabilities, security measures, and evacuation plans at IP2 and 3 were approved, in part, because the NRC had already taken action to require licensees to enhance security and the Commission had directed the staff to undertake a comprehensive review of plant security. In light of the facility's defense-in-depth, the heightened security measures implemented in response to the events of September 11, and the NRC's ongoing reevaluation of its safeguards regulations and programs, the NRC did not consider the immediate closure of IP2 and 3 to be necessary to provide adequate protection of the public health and safety. Further, the NRC informed the Petitioners that the issues in the Petition were being referred to the Office of Nuclear Reactor Regulation (NRR) for appropriate action.

In its December 20 letter, the NRC told the Petitioners that a public meeting or telephone conference with the NRR Petition Review Board (PRB) was not necessary or appropriate at the time since the Petitioner's request was already being treated as a 2.206

Petition and because of the sensitive nature of the information. Under normal circumstances, the NRC would closely follow Management Directive (MD) 8.11, "Review Process for 10 CFR 2.206 Petitions," when reviewing requests for enforcement action; however, since the Petition involved sensitive security information, the NRC deferred application of certain public aspects of the MD 8.11 process pending further developments of our security review.

On December 20, 2001, the Petitioners provided a declaration from Dr. Gordon Thompson dated December 7, 2001, and requested that the declaration be included as a supplement to their Petition. The NRC treated the declaration as a supplement to the Petition. Although the NRC had initially withheld the Petition from public distribution pending Commission guidance about public dissemination of potential security information, the NRC has now determined that the Petition can be made publicly available. Therefore, the documents are available in the NRC's Agencywide Documents Access and Managment System (ADAMS) for inspection at the Commission's Public Document Room (PDR), located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records are also accessible from the ADAMS Public Electronic Reading Room on the NRC Web site http://www.nrc.gov/reading-rm/adams.html. Persons who do not have access to ADAMS or have problems in accessing the documents located in ADAMS should contact the NRC PDR reference staff by telephone at 1-800-397-4209 or 301-415-4737 or by e-mail to pdr@nrc.gov.

Entergy Nuclear Operations, Inc., responded to the Petition on February 11, 2002, and the staff considered the information in reviewing the Petition.

II. Discussion

Full Review of Vulnerabilities and Security Measures

In the Petition, as supplemented, the Petitioners requested that the NRC order the licensee to conduct a full review of the facility's vulnerabilities, security measures, and evacuation plans. The Petitioners stated that the reactor, spent fuel, control rooms, and electrical switching were vulnerable to terrorist attack. The Petitioners' request was based on the following assertions: (1) IP2 and 3 are a plausible target because of the population density of the surrounding area and the proximity to New York City, (2) news releases have documented threats against nuclear facilities, (3) an operational plant is more vulnerable, (4) an attack could damage cooling to the spent fuel pools and/or drain the pools, leading to fuel cladding oxidation, fire, and release of radioactive materials, and (5) the design-basis threat did not consider a terrorist attack. The Petitioners also stated that the facility is not currently equipped to defend itself from terrorist attacks, the licensee has a poor record in security and emergency preparedness, and nuclear industry security forces have repeatedly failed to repell mock attacks. The Petitioners also believe that an attack on an operating reactor would force plant operators to face competing interests from safe operations and physical security.

Staff Response

The NRC and its licensees have dealt with the issue of protection of licensed facilities against sabotage or attack for a number of years. Security against sabotage has been an important part of the NRC's regulatory activities, with defense-in-depth as the guiding principle. NRC regulations ensure that nuclear power plants are among the most hardened and secure industrial facilities in our nation. The many layers of protection offered by robust plant design features, sophisticated surveillance equipment, professional security forces, and NRC regulatory oversight provide an effective deterrence against potential terrorist activities that could target equipment vital to nuclear safety.

The NRC requirements for the defense of nuclear power plants are defined by the "design basis threat" (DBT). The DBT is specified in general terms in 10 CFR 73.1 and in greater detail in sensitive documents. The DBT was prepared by safeguards experts on the basis of information from the Department of Energy and the intelligence community about terrorist-related information both abroad and in the United States. Before September 11, the DBT defined the licensee's defense obligation and the NRC's assessment of the reasonably likely sabotage threat.

In 10 CFR Part 73, "Physical Protection of Plants and Materials," the NRC provides detailed requirements designed to protect nuclear power plants against acts of radiological sabotage, prevent the theft of special nuclear material, and protect safeguards information against unauthorized release. The requirements of Part 73 are as follows:

1. The licensee permits only authorized activities and conditions within established protected areas, material access areas, and vital areas by using controls and procedures, defined boundaries, detection, communication and surveillance subsystems, and by establishing schedules of authorized operations.
2. The licensee prevents unauthorized access of persons, vehicles and objects into protected and vital areas by using detection and barrier systems.
3. The licensee provides for authorized access and assures detection of and response to unauthorized penetrations of the protected area.
4. The licensee permits only authorized control and movement of special nuclear material.
5. The licensee provides response capabilities to assure that NRC requirements are achieved.
6. The licensee maintains a well-equipped and highly trained security organization.
7. The licensee installed physical barriers to protect vital equipment and material.
8. The licensee installed detection, surveillance, and alarm systems capable of sensing unauthorized penetrations of isolation zones and ensuring a prompt response action.

9. The licensee provides access authorization (e.g., background checks, routine worker screening, badging, etc.) programs and procedures.

10. The licensee ensures that all guards and armed response individuals have the ability to communicate with a continuously manned alarm station.

11. The licensee established an effective testing and maintenance program to verify that all physical barriers, and detection and alarm systems are capable of meeting NRC requirements.

The current NRC regulations require all licensees to establish a physical protection system and a security organization. These requirements are necessary to prevent the unauthorized access of persons, vehicles, and materials into protected and vital areas and ensure that security personnel respond to unauthorized penetrations of the protected area. Licensees are also required to develop physical security plans (PSPs) and submit these plans to the NRC for approval before implementing them. (NRC regional security teams conducted routine inspections for compliance with commitments made in approved PSPs and to assess the capabilities of the licensees' security programs. Although these commitments were intended to ensure that the security organizations were able to protect against the DBT, the inspections carried out to evaluate compliance with these commitments did not provide for performance testing of tactical response capabilities or evaluation of the effectiveness of these commitments to protect against the DBT.) Performance testing has been done by the staff through the Operational Safeguards Response Evaluation, which have been done at all sites. In addition, the licensees are required to establish a liaison with local law enforcement organizations for added assistance in the event of an attack.

Shortly after September 11, the NRC recognized the need to reexamine the basic assumptions underlying the current nuclear facility security and safeguards programs.

Chairman Richard A. Meserve, with the full support of the Commission, directed the staff to undertake a top-to-bottom review of the NRC's security regulations and programs. The security review includes the NRC's participation with the Office of Homeland Security, the Federal Bureau of Investigations (FBI), Departments of Transportation and Energy, and others to keep the NRC advised of the current threat environment. The NRC's participation with these agencies allows the NRC to communicate its actions to other Federal agencies, ensuring an appropriate and balanced response throughout the nation's entire critical energy infrastructure.

Attacks like September 11 were of a type that have not been part of the NRC's planning (or that of any other agency with similar responsibilities). Moreover, there are other aspects of the September 11th attack and the subsequent assessments that require the NRC and its licensees to reevaluate the type of assault that might be mounted against a nuclear power plant. As a result, on February 25, 2002, the NRC issued Orders to all operating power reactor facilities to require that certain interim compensatory security measures be taken beyond those called for by current regulations. These interim measures are the result of the NRC's initial review of current safeguards and security plan requirements and a review of information provided by the intelligence community. Although licensee responses to the prior Threat and Safeguards Advisories Safeguards (which provided information about potential threats and possible prompt actions for the licensee to consider implementing) were adequate to provide reasonable assurance of adequate protection of public health and safety, the NRC also determined that certain compensatory measures were prudent to address the generalized high-level threat environment in a consistent manner throughout the nuclear reactor industry. The Orders formalized a series of steps that nuclear power plant licensees had been advised to take by the NRC in the aftermath of the terrorist attacks on September 11 and added certain security enhancements. For security reasons, the details of the security requirements cannot be made public. In general, the requirements include additional personnel access controls; enhanced requirements for guard forces; new requirements for searches of vehicles approaching nuclear facilities; enhanced capability to respond to and mitigate any large fires or explosions on site; and heightened coordination with appropriate local, State, and Federal authorities. The Order also directed licensees to evaluate and address potential vulnerabilities to maintain or restore cooling to the core, containment, and spent fuel pool and to develop specific guidance and strategies to respond to an event resulting in damage to large areas of the plant due to explosions or fire. These security strategies are intended to help identify and utilize any remaining equipment and capabilities to maintain or restore core, containment, and spent fuel pool cooling, including both onsite and offsite resources. These requirements will remain in effect until The NRC notifies licensees that the threat environment has significantly changed or until the NRC determines, as a result of the ongoing comprehensive reevaluation of current safeguards and security programs, that other changes are needed. In addition, pursuant to 10 CFR 2.202, the NRC concluded that in the circumstances described above, the public health, safety, and interest require that these Orders be made effective immediately.

As part of the comprehensive review of safeguards vulnerabilities, the NRC will reexamine the DBT and modify it as appropriate. As in the past, the NRC will coordinate its evaluation with various other Government agencies and discuss resource commitments with the military, the States, and local law enforcement. If a credible vulnerability is identified that is not addressed by the actions of another Federal agency, the NRC staff will consider additional physical protection, material control, and other appropriate requirements. The NRC will continue to be assisted by the Office of Homeland Security and other Federal agencies in evaluating threats beyond the defensive capabilities of NRC licensees. Because of the budgetary obligations that might be associated with any new responsibilities, the Office of

Homeland Security will be a central player in discussions to define the appropriate boundary between the private and public sector in the defense of nuclear facilities.

Although the NRC cannot rule out the possibility of future terrorist activity directed at a licensee's site before implementing any further enhancements to its safeguards programs, the NRC believes that these facilities can continue to operate safely. Nuclear power plant design is based on defense-in-depth principles, and includes many features to protect public health and safety. For example, reinforced containment buildings and redundant safety systems would help trained operators prevent or limit the release of radioactive material in the event of a terrorist attack. In addition, NRC requirements for coping with fires and station blackout (loss of offsite and onsite power) provide added capability to bring the plant to safe shutdown conditions assuming such aspects as loss of the control room to fire or failure of the emergency diesel generators. (The control rooms for IP2 and 3 are also located in separate buildings.)

The NRC requires careful background checks (to minimize the risk of insider assistance) and facility access controls, delay barriers, and intrusion detection systems (to detect potential attackers). The NRC also requires licensees to be able to respond with force to a group of armed attackers, using protective strategies involving layers of defense. Therefore, the NRC believes that the facilities are adequate to withstand many of the challenges from safety or safeguards events, such as armed assaults.

Regarding the issue of whether a terrorist could gain employment at a nuclear power plant, the regulations require that every employee who will have access to safety equipment have passed various background checks (past employment, references, credit history, and an FBI criminal record check) and have undergone psychological testing. During the course of employment, each employee is also subject to fitness-for-duty requirements, including random drug and alcohol testing. Behavioral monitoring of employees is also required to ensure that aberrant actions receive appropriate attention. As in the past, access to the plants is controlled, and there are portal detectors for metals and explosives. As part of the ongoing review, the NRC is considering whether to supplement these requirements.

Full Review of Radiological Emergency Preparedness and Evacuation Planning

In its December 20 supplement, the Petitioners cited a prior NRC study prepared by Sandia National Laboratory and discussed source terms and potential radiological consequences of an attack on IP. The Petitioners were concerned about the economic and environmental consequences of an attack causing a massive release of radioactive materials.

Regarding emergency preparedness planning, the Petitioners believe that the IP onsite and offsite emergency plans did not envision an act of terrorism of the magnitude seen on September 11, 2001. Additionally, the Petitioners state that the Westchester County RERP is inadequate and does not consider the possibility of multiple simultaneous attacks on vital infrastructure relied on in the current plan.

Staff Response

The overall objective of emergency response planning is to minimize the dose to the public for a spectrum of accidents that could produce offsite doses in excess of protective action guidelines. No single accident sequence should be isolated as the one for which to plan because each accident could have different consequences, both in nature and degree. Emergency plans are intended to be broad and flexible enough to respond to a wide spectrum of events. The plans are then designed to manage any radiobiological accident, regardless of the source of the release, types of nuclides released, or magnitude, timing, or duration of release.

The NRC and the Federal Emergency Management Agency (FEMA) are the two Federal agencies responsible for evaluating emergency preparedness at and around nuclear power plants. The NRC is responsible for evaluating the adequacy of onsite emergency plans developed by the utility, while FEMA is responsible for assessing the adequacy of offsite (State and local) radiological emergency planning and preparedness activities. The NRC requires licensees to have detailed procedures for responding to events, making timely notifications to appropriate authorities, and providing accurate radiological information. For the offsite plans, the NRC relies on FEMA's findings in determining whether there is reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. The licensee, local and county emergency response officials, and State emergency management officials discuss and agree on the facility's emergency response plan.

NRC regulations require the establishment of a plume exposure pathway emergency planning zone (EPZ) about 10 miles in radius and an ingestion exposure pathway EPZ about 50 miles in radius around each nuclear power plant site. The size of the EPZs chosen represents a judgment on the extent of detailed planning which must be performed to ensure an adequate response in the event of a radiological emergency. In one emergency, protective actions may be restricted to a small part of the planning zones. On the other hand, the response measures established within the 10-mile and 50-mile EPZs provide a planning basis for expanding the protective actions if conditions of a particular accident warrant.

In the event of a severe reactor accident with offsite consequences, NRC guidance calls for the prompt evacuation of the population within a 2-mile radius of the plant and about 5 miles in the downwind direction. The guidance states that these protective actions would be expanded, as necessary, based on further assessment of plant conditions, dose assessment, and field monitoring information. At longer distances, shelter is usually the appropriate protective action, followed by relocation of segments of the population if warranted by the results and analysis of radiological measurements taken in the field. The main protective action planned for the 50-mile EPZ is protection of the public from the ingestion of contaminated food and water. It is considered extremely unlikely that evacuation would be required at a distance of 50 miles even after the most severe accident. The planning established for the 10-mile and 50-mile EPZs, the decreasing consequences and increasing time available for taking protective actions as the distance from the plant increases, and the availability of monitoring data on which to base protective action decisions provide assurance that appropriate protective actions would be taken to protect the population within 50 miles of a site.

NRC regulations also require that the applicant for a nuclear power reactor operating license provide an analysis of the time required to evacuate and take other protective actions within the plume exposure pathway EPZ. This analysis is referred to as the "evacuation time estimate" (ETE). There are no preset minimum evacuation times that a nuclear power plant site must meet. However, the NRC expects that the ETEs for a site are a reasonably accurate reflection of the time it would take to evacuate the site environs under normal and adverse conditions. ETEs are mostly used to identify potential traffic bottlenecks so that appropriate traffic control plans can be developed. Nuclear power reactor licensees are expected to review and revise their ETEs for their sites. The revisions must take into account changes in population, road capacities, potential traffic impediments, and other factors affecting the ETEs. The ETEs are assessment tools used by decision makers for determining whether evacuation is the preferred protective action option for the general public under specific accident and offsite conditions. There are no minimum required evacuations times.

On August 1, 2001, the NRC issued Regulatory Issue Summary (RIS) 2001-16, "Update of Evacuation Time Estimates," to all holders of operating licenses for nuclear power plants. In this RIS, the NRC alerted licensees of the possible need to update ETEs as the results of the

2000 Census are published. The licensee is currently preparing a new ETE report for IP2 and 3.

The Petitioners refer to the 1982 Sandia National Laboratory (SNL) report "Calculation of Reactor Accident Consequences" (CRAC-2 Report). The reactor siting studies in the CRAC-2 report were performed as part of research on the sensitivity of various plant siting parameters. The studies used generic postulated releases of radioactivity from a spectrum of severe (core melt) accidents, independent of probabilities or mitigation mechanisms. The studies were never intended to be realistic assessments of accident consequences. The estimated deaths and injuries resulted from assuming the most adverse condition for each parameter in the analytical code. In the cited studies, the number of resulting deaths and injuries also reflected the assumption that no protective actions were taken for the first 24 hours. The studies did not, and were never intended to, reflect reality or serve as a basis for emergency planning. The CRAC-2 report analyses used more simplistic models than current technologies. The two basic conclusions from the SNL siting studies were that the mean estimated number of health effects from the assumed releases for all reactor sites varied by up to more than 4 orders of magnitude and that the financial costs of the releases were dominated by clean-up costs and replacement power costs. The SNL studies provided a useful measure to compare sites, not to analyze plant-specific accident consequences.

FEMA has established the Radiological Emergency Preparedness Program to (1) ensure that the health and safety of citizens living around commercial nuclear power plants can be adequately protected in the event of a nuclear power plant accident, (2) inform and educate the public about radiological emergency preparedness, and (3) make findings and determinations as to the adequacy of State and local plans and the capability of State and local governments to effectively implement these plans and preparedness measures. Such findings and determinations, where appropriate, are submitted to the NRC for use in licensing proceedings. In accordance with a Presidential Directive and Federal mandates, FEMA issues policy and guidance to assist State and local governments in developing and implementing their radiological emergency response plans and procedures. Federal agencies also have plans in place to coordinate their response activities and share their resources in support of State and local officials during an emergency. Coordination of activities includes joint planning and training sessions and exercise participation.

The emergency planning and preparedness framework, which is set forth in the emergency plans, integrates a number of key elements, including division of responsibilities and authorities, management controls, provisions for timely and informed decision making, coordination of response organizations, adequate primary and backup communication systems, adequate assessment capabilities, adequate notification capabilities, written procedures to guide emergency response personnel, adequate public radiological emergency information and the dissemination of information to the public, and training for emergency response personnel. These key elements apply to any type of emergency, including terrorist initiated events. Emergency planning and preparedness also makes emergency workers more aware of the complex nature of emergency response and fosters a better understanding not only of individual response tasks but also of how the separate tasks combine to form diverse response capabilities. Trained responders are extremely flexible in handling the disruptions caused by natural phenomena, such as severe weather, and flexibility implicitly extends to handling disruptions from potential terrorist activities. Further, emergency planning and preparedness is a dynamic process. Emergency plans are continually improved based on experience gained through plan implementation and as a result of exercises, drills, and actual events.

In late January 2002, the State of New York issued its annual letter of certification to FEMA. By this letter, the State informed FEMA that specific preparedness activities have been completed including training and the updating of State and local plans. However, the updating of State and local plans is an ongoing activity. The NRC staff understands that the State and counties are presently addressing the adequacy of evacuation plans through their required review process in preparation for the scheduled exercise in September 2002 and, in doing so, will review evacuation-related procedures in light of changes in demographics and conditions.

Regarding the Petitioners' assertion that the emergency plans do not contemplate multiple attacks on the infrastructure, the NRC finds that the existing emergency response plans allow considerable flexibility to respond to a wide variety of adverse conditions, including the results of a terrorist attack. The NRC advisories and the Order issued since September 11 directed licensees to take specific actions to improve existing emergency response plans, including heightened coordination with local, State, and Federal authorities.

Information about Security Measures to Protect Against Terrorist Attacks

The Petitioners requested that the NRC require the licensee to provide information documenting the existing and readily attainable security measures which provide IP with protection against land, water, and airborne terrorist attacks. This information should provide sufficient basis for the NRC to determine that physical barriers, intrusion alarms, and other measures are in place or constructed and are sufficient to meet realistically expected threats.

Staff Response

The NRC and its licensees have taken a number of steps since September 11 to increase security at NRC-licensed facilities, including safeguards advisories. At IP, the Entergy security force was augmented by the New York State Police and the National Guard (including Hudson River patrols) and local law enforcement personnel.

The NRC issued Orders on February 25, 2002, to all operating commercial nuclear power plants to implement interim compensatory security measures for the current threat environment. Some of the requirements made mandatory by the Orders formalized the security measures that NRC licensees had taken in response to advisories issued by the NRC in the aftermath of the September 11th terrorist attacks. The Orders also imposed additional security enhancements, which have emerged from our ongoing security review. The requirements will remain in effect until the NRC determines that the level of threat has diminished, or that other security changes are needed. The NRC views these compensatory measures as prudent interim measures to address the current threat environment in a consistent manner throughout the nuclear reactor industry. The specific actions are sensitive, but generally include requirements for increased patrols, augmented security forces and capabilities, additional security posts, installation of additional physical barriers, enhanced coordination with law enforcement and military authorities, more restrictive site access controls for all personnel, and enhanced capability to respond to and mitigate any large fires or explosions on site. The Orders also require additional security measures pertaining to the owner-controlled land outside of the plants' protected areas. Currently, the New York State Naval Militia provides security measures to detect and deter watercraft access from entering the exclusion area around the IP plants.

In its report on security, the State of New York Office of Public Security (OPS) provided recommendations to enhance security at IP. Many of the measures suggested have been implemented by the licensee and others are currently under advisement. The measures are recommendations by OPS to further enhance security and are not requirements in current NRC regulations. As stated in the NRC's letter of March 13, 2002, the NRC cannot release this information because of the sensitivity of the material.

PROPOSED

The NRC understands that there must be a balance between security and openness. The NRC has sought to achieve public confidence through a variety of methods. The most effective method is NRC's policy of transparency. The NRC's open decision-making processes enable the public to be fully informed of the issues before the NRC. However, the events of September 11 have made clear the need to rethink just how open the NRC can and should be with respect to physical security issues. In this process, there are two vital, but competing, interests. The public's right to know is grounded in law and is one of the most cherished principles of our democracy. On the other hand, the NRC needs to keep sensitive information away from those whose purpose is to destroy that democracy. The NRC is striving to strike an appropriate balance between openness and security.

As stated in its letter to the Petitioners of March 13, 2002, the NRC is currently reviewing documents related to security to judge whether any of the information could provide a level of assistance to a potential adversary. In general, if a terrorist could use the information for threat analysis, target identification, or vulnerability analysis, the information will be redacted from the public record or withheld. The NRC believes that it is to no one's benefit to discuss perceived vulnerabilities and current or planned security measures in the public domain.

<u>Mandate Security Measures Sufficient to Protect the Facility</u>

The Petitioners requested the NRC to mandate, at a minimum, the following security measures sufficient to protect the facility:

1. Obtainment of a permanent no-fly zone from the Federal Aviation Administration (FAA) in the air space within 10-nautical miles of the IP facility..
2. A defense and security system sufficient to protect and defend the no-fly zone.
3. A defense and security system sufficient to protect the entire facility, including the containment and spent fuel storage buildings, control room and electrical equipment.

The Petitioners also believe that a terrorist attack on an operating unit would force the plant operators to face competing interests from both safe operations and physical security.

Staff Response

Since September 11, the NRC's safeguards analysts have been working continuously with the intelligence and law enforcement agencies to assess the general threat environment. The NRC, with assistance from Federal, State, and local law enforcement, has examined unusual incidents, such as flyovers and unsubstantiated threats.

Both the NRC and the FAA have provided direction regarding flyovers of nuclear power plants to NRC licensees and general aviation pilots. On September 26, 2001, the FAA issued a Notice to Airmen (NOTAM) that advised pilots to avoid the airspace above or in the proximity to various structures, including nuclear power plants. It also indicated that pilots "should not circle as to loiter in the vicinity of such facilities." This NOTAM was reissued on December 19, 2001, to include military facilities. On October 6, 2001, the NRC advised licensees to report any flyovers that are considered too close to their sites or that are of a suspicious nature to the local FAA, local FBI, local law enforcement, and the NRC. This direction remains in effect today.

The NRC is also reviewing measures to bolster defenses and to establish new antiterrorism strategies in a thorough and systematic manner. The NRC is taking a realistic and prudent approach toward assessing the magnitude of the potential threat and the strength of licensee defenses.

Since September 11, there have been no specific credible threats of a terrorist attack on a nuclear power plant. In light of the high general threat environment, the NRC and facility licensees have maintained a high security posture. The NRC has started a comprehensive review of its security program to ensure that the right protections are in place for the long term.

NRC licensees must defend nuclear power plants against the DBT. September 11 showed that the NRC and its licensees must reevaluate the scope of potential assaults of all types. However, there are limits to what can be expected from a private guard force, even assisted by local law enforcement. Even if it is determined that nuclear power plants should be defended against aircraft attack, the NRC cannot expect licensees to acquire and operate antiaircraft weaponry. Protection against this type of threat may be provided by other means within the Federal government. Similarly, there might be other types of attacks which should properly involve governmental response because of the size of the assumed attacking force or the equipment that must be employed in defense. As a result, in developing policy, the NRC must differentiate between the licensee's defensive obligation and that which must be undertaken by the government. Any gap between licensee capability and the assumed threat must be assumed by the government, and the government must prepare for this. As noted by the licensee in its February 11, 2002, response to the Petition, prior NRC proceedings have concluded that a licensee is entitled to rely on settled and traditional governmental assistance in handling an attack or sabotage by enemies of the state. In light of the difficulty in protecting the numerous specific potential targets of an aircraft attack, the NRC believes that the Nation's resources devoted to protection against terrorist attacks by air should be primarily directed toward enhancing security at airports and within airplanes in flight.

As part of the ongoing comprehensive security review, the NRC is examining the new threat environment in coordination with the new Office of Homeland Security, the FBI, FEMA, the FAA, the military, the intelligence community, and the Department of Energy, among others. The NRC will need to discuss government support with the military, the States, and local law enforcement organizations about the provision of governmental assets at appropriate times. These organizations will define the appropriate boundary between the public and private sector in the defense of nuclear facilities. The NRC has communicated with the Governors of New York and about 40 States to ensure that any state defensive assets (National Guard or State police) are used as needed to augment licensee defensive strategies.

Dry Cask Spent Fuel Storage System

The Petitioners request that the NRC order the licensee to immediately convert the current spent fuel storage from water-cooled spent fuel pools to a dry cask storage system in a bunkered structure. As the basis for the request, the Petitioners state that this action would reduce the long-term risk of potential exothermic oxidation in the existing fuel storage facility. The Petitioners state that the NRC has never established that the spent fuel storage facility at IP is secure against foreseeable attacks nor can the NRC be certain that the spent fuel storage facility is sufficiently sound to preclude the possibility of a spent fuel fire in the event of an airborne, land, or water-based assault. The Petitioners' concerns were based, in part, on information in an NRC report, "Final Technical Study of Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," dated October 2000, and on the Petitioners' evaluation of the consequences of a terrorist attack on the spent fuel pool buildings. In their December 20, 2001, supplement, the Petitioners state that the NRC has not performed an environmental impact statement or probabilistic risk analysis assuming all modes of water loss from the spent fuel pools, including terrorist attack, and the Petitioners further discuss the probability and consequences of exothermic oxidation of the spent fuel cladding.

Staff Response

The NRC staff believes that spent fuel can be safely stored at the IP reactor site in the current system of spent fuel pools. Although the spent fuel storage buildings at IP are not as hardened as the reactor containment structures, the spent fuel pools themselves are robust, and relatively small structures, that are partially below ground level. The spent fuel is stored in racks resting on the floor of the pools and is covered by more than 20 feet of water. The pools are designed to prevent a rapid loss of water with the structure intact, and the pool water level and cooling system are monitored and alarmed in the control rooms. Thus, the response time for events involving the SFP is significantly longer. It is also easier to add water to the SFP from various sources because it is an open pool. The robust design and small size of the pools minimize the likelihood that a terrorist attack would cause damage of a magnitude sufficient to result in an offsite release of radioactive material. Further, offsite resources can be brought onsite to assist the response to an event.

When the NRC staff completes its reevaluation of the physical security requirements, the NRC will be able to judge whether modifications to the spent fuel pool structures and enclosures are warranted and whether additional safeguards measures should be established. If so, the NRC will act accordingly. In the meantime, the NRC has issued Orders to all operating nuclear power plants requiring certain interim compensatory measures to augment security and strengthen mitigation strategies. The spent fuel pools are within the protected area of the facility and therefore protected from certain external threats under the security provisions identified in the physical security plans (PSPs).

During the NRC review of the transfer of the licenses for IP1 and 2, the licensee indicated that it was evaluating the possible construction of an independent spent fuel storage facility. In a public meeting on March 14, 2002, the licensee stated that it was expediting its engineering review for this facility.

The regulations in 10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste," establish requirements, procedures, and criteria for the issuance of licenses to receive, transfer, and possess power reactor spent fuel, power reactor-related greater-than-class C waste, and other radioactive materials associated with spent fuel storage in an independent spent fuel storage installation (ISFSI).

The NRC authorizes storage of spent nuclear fuel at an ISFSI under two licensing options: site-specific licensing and general licensing. Under a site-specific license, an applicant submits a license application to The NRC and the NRC performs a technical review of all the safety aspects of the proposed ISFSI. If the application is approved, the NRC issues a license that is valid for 20 years. A spent fuel storage license contains technical requirements and operating conditions (fuel specifications, cask leak testing, surveillance, and other requirements) for the ISFSI and specifies what the licensee is authorized to store at the site.

A general license authorizes a nuclear power plant licensee to store spent fuel in NRC-approved casks at a site that is licensed to operate a power reactor under 10 CFR Part 50. The licensee is required to perform evaluations of its site to demonstrate that the site is adequate for storing spent fuel in dry casks. These evaluations, including analysis of earthquake intensity and tornado missiles, must show that the cask certificate-of-compliance conditions and technical specifications can be met. The licensee must also review its security program, emergency plan, quality assurance program, training program and radiation protection program, and make any necessary changes to incorporate the ISFSI at its reactor site.

Dry cask storage allows spent fuel that has already been cooled in the spent fuel pool to be surrounded by inert gas inside a container called a cask. The casks are typically steel cylinders that are either welded or bolted closed. The steel cylinder provides a leak-tight containment of the spent fuel. Each cylinder is surrounded by additional steel, concrete, or other material to provide radiation shielding to workers and members of the public.

III. Conclusion.

As stated in its letter to the Petitioners on December 20, 2001, the NRC has, in effect, partially granted the Petitioners' request for an immediate security upgrade at IP2 and 3. On September 11, 2001, the NRC took action to enhance security at all nuclear facilities, including IP2 and 3. Immediately after the attacks, the NRC advised all nuclear power plants to go to the highest level of security, which they promptly did. These facilities have remained at a heightened security level since. The NRC continues to work with other Federal agencies and is monitoring relevant information it receives on security matters at nuclear facilities. The NRC is prepared to make immediate adjustments as necessary to ensure adequate protection of the public.

On February 25, 2002, the NRC issued Orders to IP and all other operating commercial nuclear power plants to implement interim compensatory security measures for the high-level threat environment. Some of the requirements formalized a series of security measures that NRC licensees had taken in response to advisories issued by the NRC, and others are security enhancements which have emerged from the Commission's ongoing comprehensive security review. The Commission issued the Orders because the generalized high-level threat environment had persisted longer than expected and, as a result, it is appropriate to maintain the security measures within the established regulatory framework. The details of those security requirements are sensitive and will not be provided to the public. In general, the requirements include additional personnel access controls, enhanced requirements for guard forces, enhanced capability to respond to and mitigate any large fires or explosions on site, and heightened coordination with appropriate local, State, and Federal authorities. Therefore, the Petitioners' request that the licensee conduct a full review of the facility's vulnerabilities, security measures, and evacuation plans has been, in effect, granted.

PROPOSED

The NRC finds that the existing emergency response plans are flexible enough to respond to a wide variety of adverse conditions, including a terrorist attack. The NRC advisories and the Order issued since September 11 directed licensees to take specific actions deemed appropriate to ensure continued improvements to existing emergency response plans. The Petitioners' concern that the emergency plans do not contemplate multiple attacks on the infrastructure is alleviated by the fact that the emergency plans are intended to be broad and flexible enough to respond to a wide spectrum of events. Further, the Petitioners' request that the applicable emergency plans be revised to account for possible terrorist attacks has been, in effect, granted.

As stated above, the NRC in its February 25, 2002, Order required IP and other plants to implement interim compensatory security measures for the high-level threat environment. The Order also directed licensees to evaluate and address potential vulnerabilities to maintain or restore cooling to the core, containment, and spent fuel pool and to develop specific guidance and strategies to respond to an event that damages large areas of the plant due to explosions or fire. These strategies are intended to help licensees to identify and utilize any remaining onsite or offsite equipment and capabilities to maintain or restore core, containment, and spent fuel pool cooling. If NRC's ongoing security review recommends any other security measures, the NRC will take appropriate action.

The NRC denies the Petitioners' request regarding the defense of a no-fly zone by the licensee. This is the responsibility of the Federal government. Further, the current security requirements, along with the enhancements in the February 25 Order, provide reasonable assurance of the protection of the facility.

The NRC finds that the current spent fuel storage system and the security provisions at IP adequately protect the spent fuel. However, the licensee has stated its intention to install an ISFSI. The Petitioners' request to order the installation is denied.

As provided in 10 CFR 2.206(c), a copy of this Director's Decision will be filed with the Secretary of the Commission for the Commission to review. As provided for by this regulation, the decision will constitute the final action of the Commission 25 days after the date of the decision unless the Commission, on its own motion, institutes a review of the decision within that time.

Dated at Rockville, Maryland, this day of 2002.

FOR THE NUCLEAR REGULATORY COMMISSION

Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

PROPOSED

Before the
UNITED STATES NUCLEAR REGULATORY COMMISSION
Washington, D.C. 20555

In the Matter of:	:	TO:	EXECUTIVE DIRECTOR FOR OPERATIONS
ENTERGY CORPORATION	:		
(Indian Point Nuclear Power Station, 50-Units No. 2 and 3; Facility Operating Licenses DPR-26 and DPR-64)	:		Docket Nos. 50-003, 50-247, 286
			August 9, 2002

RIVERKEEPER, INC., et al,
Petitioners

COMMENTS ON MAY 16' 2002 PROPOSED DIRECTOR'S DECISION ON RIVERKEEPER'S NOVEMBER 8TH PETITION 2.206 REQUEST FOR EMERGENCY SHUTDOWN OF INDIAN POINT UNITS 2 AND 3

Restatement of Request for Action

Riverkeeper, Inc. and the individual and organizational petitioners identified on the attached page (collectively, "Petitioners") hereby submit the following comments in response to the NRC's proposed Director's Decision to petitioner's November 8th petition. Since September 11, the fundamental assumptions about the safety of nuclear power plants and the nature and likelihood of an assault on such plants have changed. The Commission is faced with a choice between protecting the investment of nuclear plant operators, such as Entergy, who knowingly took on the economic risks of operating a nuclear power plant, and assuring the health and safety of civilian populations surrounding such plants, who have never been given the choice of whether to assume these new risks to nuclear plant operations. Unfortunately, the proposed decision would protect the operators' economic interests at the expense of the safety and security of the surrounding population. The proposed Director's Decision fails to provide assurance of the public health and safety with the continued operation of the Indian Point nuclear power facility in the face of plausible terrorist attack scenarios following the September 11 attacks on the World Trade Center and the state of war with the Al Qaeda terrorist organization. Accordingly, Petitioners request that the Commission modify its proposed decision in order to afford the relief originally requested by Petitioners, including an immediate shutdown of the Indian Point nuclear power plant, with its restarting contingent upon implementation of adequate security measures to protect the plant against airborne and seaborne terrorist attacks.

According to a recent National Research Council report, "the potential for a September 11-type surprise attack in the near term using U.S. assets such as airplanes appears to be high." National Research Council, Making the Nation Safer - The Role of Science and Technology in Countering Terrorism, at p. 50 (available at http://books.nap.edu/html/stct/index.html). The Commission, in its proposed decision, acknowledges the "gap" between the licensee's capability to protect against air attacks and the protection afforded by the government. Proposed Decision at 21. Yet, despite this gap, in its Proposed Decision, the Commission essentially proposes to do nothing to protect the public from the very threat of aircraft attack that the National Research Council has ranked as "high." This failure of the Commission to act, if incorporated in a final decision, can only be characterized as a complete abdication of the Commission's statutory duty to protect the public health and safety. See Atomic Energy Act § 103, 42 U.S.C. § 2133(d).

Petitioners repeat their request that the NRC implement the following immediate actions:

1. Order the Indian Point licensee to suspend operations, revoke the operating license, or adopt other measures resulting in a temporary shutdown of Indian Point Unit 2 and Unit 3, as per 10 CFR § 2.202, and order the licensee to conduct a full review of the facility's vulnerabilities, security measures and evacuation plans.

2. Require the licensee to provide information, as contemplated by 10 CFR § 2.204(a), documenting the existing and readily attainable security measures which provide the Indian Point facility with protection against land, water, and airborne terrorist attacks. Such information should provide, at a minimum, sufficient basis for the Commission to determine that physical barriers, intrusion alarms, and other measures are in place or may be easily constructed, and are sufficient to meet realistically expected threats.

3. Immediately modify the licensee's operating license for Units 2 and 3 to mandate, at minimum, the following security measures sufficient to protect the facility as required by 10 CFR § 73.55:

   a. obtainment of a permanent no-fly zone from the Federal Aviation Administration in the air space within 10 nautical miles of the Indian Point facility;

   b. a defense and security system sufficient to protect and defend the no-fly zone;

c. a defense and security system sufficient to protect the entire facility, including the containment and spent fuel storage buildings, control room and electricity equipment, from a land or water based terrorist attack. The security review described above should contemplate retaining these measures on a permanent basis, and/or discuss reasonable alternatives of equal efficacy.

4. Order the revision of licensee's Emergency Response Plan and Westchester County's Radiological Emergency Response Plan in order to account and prepare for possible terrorist attacks. These reviews must contemplate not only realistic and catastrophic effects of a terrorist attack on the Indian Point facility, but a comprehensive response to multiple attacks in the region which may impair the efficient evacuation of the area. Examples of such attacks include destruction of the Tappan Zee Bridge, loss of power to passenger railroads, and other events which deny use of necessary infrastructure.

5. If, after conducting a full review of the facility's vulnerabilities, security measures and evacuation plans, the NRC cannot sufficiently ensure the security of the Indian Point facility against terrorist threats, the Commission should take prompt action to permanently retire the facility.

6. Separate and apart from the above, the Commission must order the Indian Point licensee to undertake the immediate conversion of the current spent fuel storage technology from a water cooled system to a dry cask system in a bunkered structure in order to reduce the long-term risk associated with potential exothermic oxidation within the existing spent fuel storage facility.

Petitioners will demonstrate in the following comments that security provided by Entergy and Wackenhut Services and security and intelligence provided by various federal agencies cannot defend against an attack on Indian Point of the scale, sophistication, and coordination demonstrated on September 11, 2001. Based on this threat, the Petitioners renew their request that the United States Nuclear Regulatory Commission suspend the operating licenses for all the Indian Point units.

I. There is a Gap Between the Present Terrorist Threat and the Indian Point Nuclear Power Facility's Security Measures.

In light of the September 11 attacks, much of the Nation has been alerted to the necessity of protecting sensitive infrastructure from possible terrorist attack. Efforts to upgrade security and safety around the United States continue, often with mixed results.

Yet, at Indian Point, efforts to upgrade security seem to be lagging more than the norm. Despite measures taken between the September 11 attacks and the present day, the Indian Point nuclear facility remains vulnerable to terrorist attack. The Nuclear Regulatory Commission even acknowledged on page 9 of its proposed decision that "although the NRC cannot rule out the possibility of future terrorist activity directed at a [nuclear power plant] licensee's site before implementing any further enhancements to its safeguard programs, the NRC believes that these facilities can continue to operate safely." The Commission further acknowledges that "Any gap between the licensee capability and the assumed threat must be assumed by the government, and the government must prepare for this." Proposed Decision at 21.

In other words, while the NRC acknowledges that a gap in the security of the Indian Point facility exists, NRC is still willing, at least for the immediate future, to live with this gap that leaves the 20 million people residing in the Hudson River Valley and New York City vulnerable to nuclear catastrophe. The petitioners are unwilling to accept this conclusion.

A. NRC Cannot Rely on the Lack of a Specific Credible Threat of a Terrorist Attack on Indian Point, as the National Research Council has Ranked the "Near Term" Risk of a Terrorist Attack on a Nuclear Power Plant as "High."

Despite acknowledging the real risk of catastrophic results of an aerial terrorist attack on Indian Point and the gap between air defense provided by the plant operator and that needed for effective defense, the Commission, in its proposed decision, is nonetheless willing to accept the risk of a terrorist attack occurring before this gap can be filled. Apparently, the Commission would take comfort in imposing this risk on the population around Indian Point because "since September 11, there have been no specific credible threats of a terrorist attack on a nuclear power plant." Proposed Decision at 20. The Commission's proposed decision thus ignores the nature of terrorist attacks (which are not usually preceded by a "specific credible threat"). This premise of the proposed decision is also directly contradicted by a recent report of the National Research Council, which ranks the "near term" threat of a terrorist surprise air attack on a nuclear power plant as "high." National Research Council, Making the Nation Safer - The Role of Science and Technology in Countering Terrorism, p. 50. (See attached Exhibit A)

By their very nature, terrorist attacks are not preceded by "specific credible threats" identified by United States intelligence agencies. Certainly, the World Trade Center attack was not the subject of such a "specific credible threat"; nor was the bombing of the U.S.S. Cole in Yemen. The mere lack of advance intelligence warning does not make an attack on a U.S. nuclear plant unlikely, or excuse the Commission from taking immediate measures to protect public safety from the effects of an attack that now appears likely, if unpredictable.[1]

In fact, the National Research Council has performed a recent, detailed assessment of the likelihood of various radiological attacks by terrorists, and has concluded that "the potential for a September 11-type surprise attack in the near term using U.S. assets such as airplanes appears to be high." Id. at p. 50. The report notes that such plants "may present a tempting, high visibility target for terrorist attack." Id. at p. 50. There is no more highly visible and tempting nuclear power plant target in the country than the Indian Point nuclear power generating station. And, as the National Research Council Report notes, "such attacks could potentially have severe consequences."

Petitioners have thus identified a potential incident – airborne terrorist attack – for which there is a "high risk" as assessed by the National Research Council, and for which the Indian Point plants have no protection. This is not a case where the Commission is being asked to take extraordinary measures to respond to a miniscule risk; rather, the Commission is being asked to take immediate measures to respond to a risk that is "high" in the "near term." Ignoring this risk is an abdication of the Commission's duty to protect the public.

B. Airspace around Indian Point is Not Secure

Examination of the current security measures in and around Indian Point show that the present measures are not sufficient to deal with the threat of terrorist attack that now exists. Firstly, the airspace around Indian Point is far from secure. Numerous incidents involving

[1] Moreover, contrary to the assertion in the Proposed Decision, there have been specific credible threats of an attack on U.S. nuclear power plants. On May 24, 2002: The Nuclear Regulatory Commission sent a special advisory to the nation's 103 commercial nuclear power plants. The advisory, triggered by information gained by the intelligence community, warned nuclear power plant operators to be on the lookout and to report anything suspicious to the operations center. A January 23, 2002 NRC memo alerted nuclear power plants that terrorists may be planning an attack on a nuclear power reactor using a hijacked commercial airliner. – "FBI headquarters has provided the following information to all field offices. During debriefings of an al Qaeda senior operative, he stated there would be a second airline attack in the U.S. The attack was already planned and three individuals were on the ground in the states recruiting non-Arabs to take part in the attack. The plan is to fly a commercial aircraft into a nuclear power plant to be chosen by the team on the ground."

violation of protected airspace, either of Indian Point or of other restricted sites, have occurred with alarming frequency since the September 11 attacks.

On April 18, 2002, Senator Clinton sent letters to the Nuclear Regulatory Commission and the Federal Aviation Administration demanding an explanation in response to a recent revelation that a reporter from Fox News, without displaying identification up front, was able to hire a pilot of a small plane to take him directly over the Indian Point nuclear facility for an extended period of time without interference. In letters to FAA Administrator Jane F. Garvey and to NRC Chairman Richard A. Meserve, Senator Clinton expressed her "grave concern" over the incident and stated that "like it or not, our nuclear facilities are potential targets for future terrorist activity. So we must be as vigilant as possible to ensure that these plants are not only operated safely, but that the plants and the communities in which they are situated are afforded the highest level of security, emergency planning, and preparedness against potential terrorist and criminal attacks."

If a civilian reporter can hire a small plane pilot to take him over the Indian Point facility for an extended period without interference, then it can hardly be asserted that the airspace over Indian Point is secure. Even a small plane flying into the Indian Point facility can do significant damage, especially one loaded with explosives.

Another piece of information showing that the airspace in the United States is not secure comes from an Associated Press news article. This article, titled "Planes Often Enter Prohibited Air" (See attached Exhibit B) and published on April 5, 2002, reported that, despite military patrols and tighter security, pilots had intruded into America's protected airspace at least 567 times in the seven months since Sept. 11, highlighting the continued challenges of thwarting a terrorist air attack. In each case, a pilot wrongly flew into one of the country's six prohibited flight zones, where no planes are allowed, or into one of many restricted zones where air traffic is limited because of sensitive military or nuclear operations or special events. As of this filing, more violations of protected airspace have no doubt occurred in the intervening three months.

Of all our American institutions, one would expect that the greatest effort to successfully secure airspace would be made for the White House. Unfortunately, even though the airspace over the White House has been heavily restricted for years and a new 15-mile no-fly zone was established after September 11, several unauthorized flyovers have occurred at the White House since the attacks on the World Trade Center and the Pentagon, including at least four commercial jetliners and a medical helicopter. "Planes Often Enter Prohibited Air" (See attached Exhibit B).

The latest of these flyovers occurred on June 20, 2002 and clearly illustrated the difficulty of relying even on the US Air Force to secure restricted airspace. On June 20, a small, single-engine plane wandered into the restricted airspace over Washington, D.C. and, despite repeated attempts, air traffic controllers in the area were unable to contact the pilot. Air Force F-16 fighter jets were immediately scrambled to intercept the plane, but by the time the fighters were airborne, the small plane had already left the restricted zone. Had the pilot actually intended to ram the White House, even the United States Air Force would have been unable to prevent it.

Clearly, these events show that securing restricted airspace in the wake of September 11 is a notoriously difficult exercise. The NRC's reliance on present methods of securing airspace seems woefully insufficient to protect the Indian Point facility from an attack from the air. Since the nation is unable to prevent aircraft from entering *restricted* airspace, the unrestricted airspace above the Indian Point nuclear power plant is clearly vulnerable.

C. Indian Point is Not Secure from Breaches in Airport Security

The NRC has stated on page 21 in its Proposed Decision that it will rely on airport security to safeguard the Indian Point facility from a terrorist attack from the air. In fact, the NRC went so far as to say "in light of the difficulty in protecting the numerous specific potential targets of an air attack, the NRC believes that the nation's resources devoted to protection against terrorist attacks by air should be primarily directed towards enhancing security at airports and within airplanes in flight."

However, this reliance on airport security will not enhance the safety of Indian Point in any way from civilian-owned planes that take off and land at smaller airports. While the Office of Homeland Security concentrates on commercial airliners and major airports, very little oversight has been put on civilian craft. There is nothing to prevent a terrorist from purchasing or renting a small single-engine plane, taking off from a small airport and taking a direct course to the Indian Point facility. The NRC said that a small plane would be unlikely to breach the Indian Point containment dome. This is still in dispute, but even if it was not, there are other ways in which a small plane could cause catastrophic damage to Indian Point, such as a small plane loaded with explosives. Such examples include merely driving the small plane into the spent fuel storage facility or even the control room. NRC has made no mention of how it intends to protect the Indian Point facility from such an attack.

Reliance on airport security at the Nation's major airports to protect against a suicide attack in a commercial airliner is similarly questionable. In March 2002, the Transportation Department inspector general released a report that found airport security screeners on several dozen occasions failed to catch guns and simulated explosives, even after the September terrorist attacks. Inspector General Kenneth Mead's report found screeners missed knives 70 percent of the time, guns 30 percent of the time and simulated explosives 60 percent of the time. Also, according to the Federal Aviation Administration, security breaches caused the government to evacuate 59 airport concourses or terminals between October 30, 2001 and March 7, 2002, forcing 2,456 flights to be delayed or canceled. Passengers on another 734 flights had to leave their seats and go through security a second time. "Airport Security Gets an 'F,'" CBSNews, March 25, 2002 (See attached Exhibit B)

This is a huge failure rate that exposes the fallacy of relying on airport security to protect nuclear power plants. With this sort of ease of smuggling weapons through the airports, terrorists could effortlessly hijack an airliner and use it as a weapon against any nuclear facility, including Indian Point.

D. Indian Point's Spent Fuel Storage Facility and Cooling Water Intakes are also Vulnerable to Attack

Terrorists need not fly an airliner directly into the containment dome of a nuclear power plant to cause a breach of containment and catastrophic release of radiation. An attack on the spent fuel water-cooling pools of a nuclear plant would be enough to cause loss of coolant to the point where the highly radioactive used fuel melts and releases huge amounts of radiation. Indian Point's spent fuel facility is particularly vulnerable to attack as the roofs of these storage buildings are constructed out of insubstantial sheet metal. The spent fuel storage buildings at Indian Point were also constructed with rather thin walls. This sort of building is not sturdy enough to stand up to a determined terrorist attack, whether by a hijacked airplane or by an armed group of attackers on the ground who detonate explosives.

Recent information also shows that plant cooling intakes could be vulnerable to a scuba-based terrorist attack. On May 24, 2002, the FBI issued another warning saying scuba divers might be used in terror attacks, but the FBI was vague about the likely scenarios. "Recent information has determined that various terrorist elements have sought to develop an offensive scuba diver capability," the FBI said in a bulletin issued by its National Infrastructure Protection Center and sent to state and local law enforcement agencies. "While there is no evidence of operational planning to utilize scuba divers to carry out attacks within the United States, there is a body of information showing the desire to obtain such capability." "Subways, Scuba Divers and Landmarks," ABCNews, May 27, 2002. (See attached Exhibit C)

Can present security measures at Indian Point defend against an attack involving scuba divers detonating explosives to cripple the cooling intake technology? It is doubtful this question has ever before been conceived of before September 11, let alone actively evaluated by the NRC and the various licensees of the Nation's nuclear power plants. This vulnerability at Indian Point also has to be addressed.

E. Indian Point's Design Basis Threat (DBT) Does Not Adequately Address Present Terrorist Threats to the Indian Point Facility

Indian Point's design basis threat did not consider safeguarding the facility from methods of deliberate attack by terrorists, whether by land, water or suicide attack. On March 25, 2002, U.S. Congressional Rep. Edward Markey released a report entitled "Security Gap: A Hard Look At the Soft Spots in Our Civilian Nuclear Reactor Security" that analyzed more than 100 pages of Nuclear Regulatory Commission (NRC) correspondence sent to the Congressman in response to several letters. The report indicates that in no case has any U.S. licensee considered the possibility of a deliberate aircraft impact such as the one that occurred on September 11, 2001. Twenty-one U.S. nuclear reactors are located within 5 miles of an airport, but 96% of all U.S. reactors, including Indian Point, were designed without regard for the potential for impact from even a small aircraft. According to the NRC Response, only 4 U.S. reactors include any design features calculated to withstand the impact of an airplane. The Limerick (Philadelphia, PA) and Seabrook (Portsmouth, NH) reactor designs were evaluated to consider impacts from aircraft weighing up to 12,500 pounds – less than 3-5 percent of the weight of the Boeing 757s/767s aimed at the World Trade Center and the Pentagon. Only the Three Mile Island units 1 and 2 near Harrisburg, PA, were designed with the impact of a large airliner in mind. According to the NRC Response, Unit 1 was designed with "reinforcement of outer walls, thickening of concrete sections, and unique internal features. In addition, special fire protection and ventilation features were provided to cope with aircraft crashes. Similar features were incorporated in Three Mile Island Unit 2." The design features were made so that the reactors could withstand the impact of planes weighing up to 200,000 pounds. The NRC Response to Congressman Markey's inquiries states that the U.S. chose not to require additional protection against the impact of an aircraft because "The likelihood of an airplane accidentally crashing onto a reactor site in the U.S. is typically much lower than in Europe." See Footnote 11 of "Security Gap: A Hard Look At the Soft Spots in Our Civilian Nuclear Reactor Security"

Aircraft impact to the containment structure of a nuclear reactor is not the only way an aircraft could cause a full-scale core meltdown. The NRC Response acknowledges that there are buildings other than the core of the reactor (which is a hardened structure) that could lead to a core meltdown if destroyed by the impact of a commercial aircraft: "The NRC recognizes that aircraft crashes may result in multiple-failure initiating events, and that non-safety system malfunctions could contribute to such events." If all electrical power to a reactor was cut off (by a deliberate crash of an aircraft into the power generating systems, for example), the time it would take for damage to the reactor core to begin is estimated by the NRC to be about two hours. Support systems for the reactor, such as the cooling system, are not located within buildings that are hardened (such as the reactor core) and "are not designed to withstand the direct impact of a large commercial aircraft." The destruction of some of these buildings could lead to core damage. These acknowledgments by the NRC are highly significant, because they indicate that claims by the nuclear industry that existing plants would be able to withstand a terrorist aircraft or other attack due to the strength of containment structures are irrelevant to the very real risk that terrorists might target critical support infrastructure whose destruction could result in a catastrophic nuclear accident.

Since September 11, the Federal government has issued many warnings that Al Qaeda terrorists may try to hijack another plane and use it as a weapon in the same exact manner as they did on the World Trade Center and the Pentagon. This makes it plainly obvious that the DBT as it is now established is simply not adequate to deal with present possible threats to nuclear power plants in general and the Indian Point facility in particular.

II. NRC's Proposed Actions are Insufficient to Close the Security Gap Now Present At Indian Point.

A. The FAA Notice to Airmen is Insufficient to Protect the Airspace around Indian Point

As the NRC noted in its proposed decision, the FAA issued a Notice To Airmen (NOTAM) on September 26, 2001 that advised all pilots to avoid the airspace above and around sensitive buildings, including those of nuclear power plants. However, as was noted above, this NOTAM has not prevented hundreds of unauthorized penetrations of restricted airspace from occurring. Since September 11, there have been nearly a dozen unauthorized flyovers in the vicinity of the White House alone.

For the NRC to say that the NOTAM is a security measure that is sufficient to protect our nuclear power plants from aerial attack is simply not credible in light of these hundreds of airspace violations. The idea that terrorist attackers bent on destroying the Indian Point plant would refrain from entering the airspace because a "Notice to Airmen" warns them not to is facially ludicrous.

The NRC also claimed in its proposed decision that it issued a warning to all nuclear power plant licensees to "report any flyovers that are considered too close to their sites or that are of a suspicious nature to the local FAA, local FBI, local law enforcement, and the NRC." Proposed Decision at 20. While the NRC may believe this to be an effective measure, the reality of the situation is quite different. The difficulty and time consumption of the notification of "suspicious aircraft" to these several different agencies should be plainly obvious. It is also unclear how the NRC expects the efforts of these varied agencies will be coordinated in response to a suspicious aircraft. Nor it is clear what actions these agencies will take in response to being notified of a suspicious aircraft. Also, it is simply not credible that notifying these agencies of a suspicious aircraft while it is hurtling towards a nuclear power plant at speeds in excess of 500 miles an hour will be able to prevent a nuclear catastrophe once the aircraft impacts the power plant.

B. Reliance on the US Intelligence Agencies Will Not Suffice to Ensure Security at Indian Point

The NRC mentions in its proposed decision that it will coordinate its efforts to help secure the Nation's nuclear power plants with several federal agencies, including the US intelligence services such as the Central Intelligence Agency and the Federal Bureau of Investigation. This is all well and good, but an over-reliance on the intelligence agencies to prevent terrorist attacks on our nuclear power plants is questionable.

Intelligence failures on the part of the FBI and the CIA concerning their inability to pool their resources and coordinate their anti-terrorism efforts have been reported at length in the media since September 11. Several FBI agents in Minnesota and Arizona have disclosed that they took notice of suspicious actions by Middle Eastern men who showed a special interest in learning how to fly commercial airplanes, but the agents' superiors did not react quickly to the warnings. According to U.S. officials, the CIA learned in early 2000 that two of the men who would eventually become the September 11 hijackers held a meeting in Malaysia. Unfortunately, the CIA did not inform domestic authorities (including the FBI) to watch for these two men until three weeks before the September 11 attacks. A former senior official of the FBI said this about the Bureau: "The FBI is the greatest in the world at investigating a crime after it happened, but it is not equipped to prevent crimes. It wasn't in the 90's, it wasn't on 9/11. We didn't know what we knew." June 2, 2002 NY Times, "Wary of Risk, Slow to Adapt, FBI Stumbles in Terror War."

While it is good that the NRC wishes to coordinate its efforts of securing the Nation's nuclear power plants with the US intelligence services, it is clear that much more remains to be done in reorganizing the coordination efforts of the intelligence services so that they can prevent further terrorist attacks. However, the intelligence agencies have other important responsibilities to maintain besides that of preventing terrorist attacks so any reorganization of the intelligence services will have to keep their old responsibilities intact as well as prepare them for terrorist attack prevention.

It must also be realized that the FBI is designed to investigate crimes (including terrorist crimes) *after* they happen; it is not geared towards preventing such attacks. The same can be said of the other post-9/11 intelligence agencies. This is at odds with the NRC's mission, which is to protect the public health and safety by *preventing* such attacks on our nuclear power plants before they can occur. Considering the conflicting goals of the NRC and the nation's intelligence services and the fact that the reorganization of the intelligence services to their new tasks will take a significant amount of time, it would be unwise for the NRC to simply pass the responsibility of prevention to the intelligence services in the interim.

C. Reliance on Airport Security is Not Enough to Prevent an Aerial Terrorist Attack on Indian Point

As noted above, the NRC stated in its proposed decision that it intends to rely on enhancement of airport security by the Federal government as a valid security measure that will help protect the Indian Point nuclear facility from an aerial terrorist attack. Unfortunately, there is a great deal of evidence that airport security today, even in the wake of improvements made since September 11, is still found wanting.

On April 15, 2002, the Washington Post reported that there is significantly less security at the cargo handling and private plane sections of the Nation's airports than there is for the commercial airline passengers. Furthermore, airport workers with access to these restricted areas could move from there into the commercial areas of the airports unscreened by airport security. No metal detectors are even present in the restricted areas of the airports. Experts interviewed by the Washington said they were worried that terrorists might try to exploit these weaknesses to gain access to commercial aircraft. Since Sept. 11, beefed-up security at airports has concentrated on passengers, right down to their shoes, but not on the "back doors" of airports. "It doesn't take a rocket scientist to come up with the conclusion that if I devote all my resources and attention to one segment of security [of security]...and delay attention [elsewhere], I'm asking for trouble," Capt. Bob Miller, a pilot for United Parcel Service and president of the Coalition of Airline Pilots Associations, told the Washington Post. April 15, 2002 Washington Post "Security Gaps Remain at Dulles Airport"

On Tuesday, April 23, 2002, Federal authorities rounded up 94 workers at Washington-area airports on a variety of charges from illegal immigration to lying about a criminal background, Attorney General John Ashcroft announced. The arrests at Dulles and Reagan National airports were part of a continuing post-Sept. 11 crackdown by U.S. law enforcement and transportation authorities on airport security lapses. Ashcroft said the workers allegedly gained access to secure areas of the airports "by lying on security applications," using false Social Security numbers or committing "various immigration frauds." The April 15th operation was a joint effort that included the FBI, the Immigration and Naturalization Service, federal prosecutors and the Transportation Department's inspector general. Similar arrests have occurred in the weeks leading up to the April 15th operation in Phoenix, Las Vegas, Salt Lake City and

San Francisco. At the time of the bust, about 400 workers have been arrested since Sept. 11. The investigation, called Operation Tarmac, had spread to 10 airports before the April 15th arrests. Most of the workers arrested had security badges allowing them to get onto planes, ramps, runways and cargo areas, law enforcement officials said. They were employed by private companies, such as those that clean the airplanes or operate airport restaurants. While law enforcement officials said none of those arrested have been linked to terrorism, some aviation experts said the workers were in a position to help smuggle bombs or weapons aboard aircraft. "Dozens of Airport Workers Arrested" CBSNEWS.com, April 23, 2002. (See attached Exhibit D)

Time Magazine conducted an investigation of airport security in March 2002 and discovered that recent security measures had not improved the Nation's security. An excerpt of this article states: "Random screenings and camouflaged soldiers in airports have not made flying more secure. Sensible proposals long sought by aviation experts – such as requiring carriers to match all bags to passengers on connecting flights – have not been adopted. The congressional mandate to install 2,200 explosive-detection devices in all 429 airports by the end of the year has been scaled down. While the new Transportation Security Administration plans to buy almost 5,000 trace-detection devices, little is being done in the meantime. The TSA is having trouble recruiting more than 40,000 new screeners. So far, government-trained screeners have taken up positions in exactly one airport. Some experts say the United States' haphazard security procedures may only invite terrorists to try their luck. Because airports, carriers and the government haven't yet implemented a methodical system for identifying potential terrorists, everyone from pilots to grandmothers is subject to random screening. In the long run, that can work in the enemy's favor. 'The U.S. has the bad guys celebrating this inefficient use of resources,' says Lior Zoucker, who heads an aviation-security firm. 'Terrorists like a system that treats everyone the same.'" May 27, 2002 Time Special Report "While America Slept"

These and many other instances show that airport security is still insufficient to protect commercial airlines from being taken over by terrorists and used against American targets.

D. Without Public Oversight, Recent Secret NRC Orders Issued to all Operating Nuclear Power Facilities May Not Ensure that Security at Indian Point is Actually Enhanced.

In the 2.206 petition Riverkeeper submitted, the petitioners asked that the NRC require Entergy to provide information documenting the present and readily attainable security measures which could be put in place at the Indian Point facility to protect against terrorist attack. In its proposed decision, the NRC said that it received a number of security recommendations from the New York Office of Public Security (OPS) but that these recommendations could not be revealed publicly because they are "not required under the current NRC regulations" (Proposed Decision at 18) and because "of the sensitivity of the material." (Proposed Decision at 18.)

While the petitioners can certainly appreciate the need for keeping sensitive information from falling into the hands of terrorists, NRC's refusal to allow this information to be publicly released creates a danger that Entergy will be given a free pass to pay only minor lip service to making the Indian Point more secure against terrorist attack. If nothing about present or future security measures at Indian Point is allowed to come into the public domain, then the public cannot be reasonably informed about these measures and public notice and commentary and public oversight of Indian Point security becomes impossible.

The greatest danger that worries the petitioners in this case is that Entergy and the NRC will claim that further measures to secure Indian Point are underway, while in reality, Entergy and the NRC will use the excuse of "national security" to obscure a failure to implement sufficient security upgrades at Indian Point. This would be far more dangerous to the public health and safety than allowing piecemeal knowledge of security upgrades at Indian Point to fall into the hands of terrorists. If some knowledge of security upgrades at Indian Point that was made public did fall into the hands of terrorists, the security upgrades still have a chance of defeating the terrorists should they attack. But if *nothing* more is done, then the Indian Point facility remains that much more vulnerable to a terrorist attack.

In is in the interests of all the concerned parties (Entergy, the NRC, the petitioners and the general public), to allow public notice and comment of any new security measures to be made at Indian Point.

III. Petitioners' Requested Actions will Suffice to Close Indian Point's Security Gap & Ensure the Public's Health and Safety

A. Temporarily Shutting Down the Indian Point Reactors As the Facility Undergoes a Full Review of Indian Point's Security Measures and Vulnerabilities Will Provide Greater Security to the Public Health and Safety In and Around Indian Point.

As has been demonstrated above, the Indian Point facility is vulnerable to terrorist attack by several different means, either by an aerial attack, a ground attack or an attack via the Hudson River. An aerial attack by a hijacked plane could release radioactive material from the IP facility by triggering a loss of coolant scenario whereby the fuel in the reactor core or the spent fuel pool building(s) suffers a fuel meltdown. A ground attack by an armed force of terrorists could allow these attackers to take over the IP control room or detonate explosives adjacent or within the spent fuel pool building(s), which could lead to catastrophic damage. An aquatic attack by terrorist scuba divers could damage the facilities cooling intakes.

A shutdown of the operating nuclear reactors at Indian Point would vastly reduce the threat of catastrophic nuclear release in a terrorist attack. According to a preliminary analysis conducted by the Nuclear Control Institute (NCI), after a shutdown of twenty days – which would greatly reduce the radioactive inventory in the core through decay – the number of acute fatalities (within a 10-mile radius) from a core meltdown and breach of containment could be reduced by 80% and the number of long-term cancer deaths (within a 50-mile radius) by 50%. A reactor core's inventory of short-lived radioisotopes is substantially reduced within a few days of shutdown, thus reducing the potential incidence of early health effects and thyroid cancers in surrounding populations if a release occurs.[2]

In addition, removing the fuel from the reactors – something than can be done approximately a week after shutdown – will allow security forces to focus their protection on the irradiated fuel pool where this highly radioactive used fuel is stored. A plant that is closed is no longer producing the irradiated fuel rods, which are most dangerous in the first six months upon removal from the reactor.

It is easier to protect and monitor a reactor that is shut down. The site is most vulnerable while the reactor is operating. There are a number of ways to cause a meltdown of the reactor: cutting off-site power, destroying the coolant intakes, sabotage/destruction of safety systems, destruction of the control room, as well as crashing a jet into the reactor. The propensity of a reactor core to melt, if the flow of cooling water to the core is interrupted, is substantially reduced within a few hours of shutdown.

[2] Nuclear Control Institute, The Impact of Nuclear Plant Shutdown on Sever Accident Consequences, February 12, 2002.

With so many exploitable vulnerabilities, it would make sense for the NRC to order the temporary shutdown of Indian Point Units 2 and 3. If both units were shut down, a ground attack on the control rooms would be likely to cause less damage. If the reactors in Units 2 and 3 were placed into cold shutdown, less radioactive material would be released in the event a large airliner pierced the containment dome, damaged the reactor core cooling system and triggered a release of radiation. Furthermore, if Units 2 and 3 were shut off, this would allow Entergy to concentrate more on security especially in relation to safeguarding spent fuel pools. Since shutting down Units 2 and 3 would make the Indian Point facility more secure, it would also be prudent to conduct a full review of Indian Point's security measures while the two units are shut down.

B. Requiring Entergy to Reveal Information Regarding Present and Easily Attainable Security Measures at Indian Point will Help Determine How to Enhance Security at Indian Point.

Title 10, Section 2.204(a) authorizes the NRC to demand from any licensee "information for the purpose of determining whether an order under § 2.202 should be issued, or whether other action should be taken." It is clear that security at Indian Point needs to be enhanced, especially in the wake of September 11. Indian Point's DBT did not address the possibility of a terrorist attack, either by land, water or air. The DBT likewise did not address a suicide attack of any means by terrorists. Entergy has struggled with security at the Indian Point facility and some of its other plants. All these instances point up the necessity of upgrading security at Indian Point.

However, as noted above, the NRC claimed in its proposed decision that it has already taken measures to upgrade security at the Nation's power plants, including Indian Point, but cannot reveal the information concerning these new measures "due to the sensitivity of the material." While the petitioners can certainly appreciate this, a hallmark of NRC's ability to fulfil its responsibility of protecting the public health and security has been public oversight.

NRC indicates its desire to take its time with upgrading security at Indian Point in its proposed decision. While acknowledging the security gap that exists between present security measures at Indian Point and the current atmosphere of possible terrorist attack on the Indian Point facility, NRC said quite clearly that it is willing to live with the gap until new security measures have been implemented. By keeping those same measures secret with the invocation of "national security," NRC has a too-convenient opportunity to pay only lip service to improvement of the security of the Nation's nuclear power plants while real action could languish. This is a matter of great public concern and needs to be discussed openly.

Unfortunately, the Operational Safeguard Response Evaluation tests designed to measure security at nuclear power plants have been suspended. Without the OSRE tests, the only way for Indian Point's security to be evaluated to the public's satisfaction would be for NRC to compel Entergy to release any and all necessary information relevant to determining whether Indian Point can be secured from terrorist attack. Anything less would be a less than complete effort to maintaining the public health and security.

C. Modifying Entergy's License to Mandate Measures to Defend Indian Point's Airspace will Help Secure the IP Facility from Aerial Attack.

It has been shown above that the airspace around Indian Point is not secure from an aerial attack. This situation has to changed as quickly as possible because a successful attack on either Indian Point's reactors or its spent fuel storage building could result in a catastrophic release of radioactive contamination that would directly affect the immediate surroundings and quite possibly extend to New York City.

While the NRC has stated, correctly, that Entergy is unqualified to place and operate anti-aircraft weaponry in and around Indian Point to protect its airspace, there are other available measures that can be readily implemented to speedily secure the airspace around the Indian Point nuclear facility and thereby prevent disaster. Similarly, while the petitioners concede that Entergy is not qualified to operate and man anti-aircraft defenses, this does not preclude the possibility of deployment of those defenses to protect Indian Point, and conditioning Indian Point's continued operation on obtaining these appropriate defenses from the federal government.

In October 2001, in the aftermath of the September 11 attacks, the nation of France deployed anti-aircraft weaponry around its most vulnerable nuclear facilities. Other measures, such as a no-fly zone and military fighter protection, were also implemented. While the anti-aircraft weapons around two of France's nuclear facilities were removed, the other measures remain in place.

Entergy's claim that anti-aircraft weaponry cannot be deployed around Indian Point because its employees are not qualified to operate such equipment is refuted by France's

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deployment of similar equipment around its own nuclear plants. The employees working at France's nuclear power plants were likewise unqualified to operate military equipment. But what Entergy's claim does not foresee is that the petitioners are not asking Entergy or its employees to operate the anti-aircraft weaponry. Such a task can only fall to qualified military personnel serving in the US Army. In France the same thing was done: qualified military personnel, not the nuclear plant employees, operated the anti-aircraft weaponry while it was deployed around France's nuclear plants.

Also, the removal of the aircraft weaponry in France does not negate the petitioners' argument in favor of anti-aircraft weaponry deployed to protect Indian Point for several reasons. France evidently decided that the threat of aerial attack on its nuclear power plants had passed; the same is not true for the United States. Since September 11, the Office of Homeland Security and FBI has issued numerous warnings of possible and even imminent terrorist attack. Al Qaeda has made similar statements that it intends to strike at the United States in the near future. It is clear that a terrorist threat via an aerial attack to our Nation's nuclear facilities still exists, so deploying a defense system to protect our nuclear power plants, especially Indian Point, would be a prudent measure, both for now and in the foreseeable future.

Alternatively, deploying passive defenses such as a massive array of barrage balloons (interim) and tall poles linked by steel cables (long term) can play an important role in protecting Indian Point from an air attack. A barrage balloon is anchored singly or in a series over a potential target to block passage of attacking aircraft. Adding large earthen berms around the entire plants, which also can make an air attack much more difficult, especially if used in addition to barrage balloons. Earth berms also protect against attacks by rocket-propelled grenades and many other possible scenarios. Also, undersea netting needs to be installed to protect against submarines and scuba diver assaults.

Another measure the NRC would be wise to enact is the establishment of a permanent no-fly zone within 10 nautical miles of the Indian Point facility. As shown above, the NRC's warning to all nuclear power plant licensees to "report suspicious aircraft" is not sufficient to protect Indian Point, especially when such a plane could be under terrorist control and hurtling towards a nuclear power plant at speeds in excess of 500 mph. However, the establishment of a permanent no-fly zone around Indian Point would make all pilots aware of the necessity of avoiding the airspace around Indian Point. Combined with the deployment of anti-aircraft weaponry around Indian Point, the 10-mile no-fly zone would serve to give defenders of the nuclear facility more lead time to make a proper decision on how best to defend Indian Point should a plane violate the facility's airspace.

It is the NRC's interest and in the interests of all the people living in the vicinity of Indian Point to have these measures enacted so that they can be protected from nuclear catastrophe by means of a terrorist attack. Furthermore, these actions are necessary and required under 10 CFR 73.55 which mandates physical protection of nuclear power plants from radiological sabotage.

D. Transferring Indian Point's Spent Fuel Facility to a Dry Cask System Will Greatly Improve Public Health and Safety.

As was noted above, Indian Point's spent fuel storage facility is vulnerable to attack, which could lead to devastating consequences. If the cooling water inside the facility is reduced, whether by an aerial attack or an attack by a group of armed attackers on the ground, the result will be the same: the remaining water will heat up and evaporate, causing a chain reaction of events that could lead to a spent fuel rod assembly fire.

The NRC admits in its proposed decision that "the spent fuel storage buildings are not as hardened as the reactor containment structures." Proposed Decision at 22. Yet it also goes on to say that the structures are, in fact, "robust." Id. at 22. It seems disingenuous for the NRC to claim that the spent fuel storage buildings, which consist of thin concrete walls and sheet metal roofs, are "robust" enough to withstand a terrorist attack, whether by a hijacked plane or by armed attackers on the ground.

The NRC also claims that the cooling pools in the spent fuel storage facility "are designed to prevent a rapid loss of water *with the structure intact.*" (Id. at 22, emphasis added). But what if the structure is *not* intact? Damage to the structure itself is eminently possible in the event of a terrorist attack. Indeed, should the terrorists use a plane (whether a small commuter plane or a hijacked airliner) to ram the storage facility, damage to the structure sufficient to reduce the water in the cooling pools is virtually assured.

It is for these reasons that the petitioners also respectfully request that NRC order Entergy to transfer all spent fuel over five years old from wet storage to hardened on-site dry storage systems. The dry cask system greatly reduces the risk of radioactive contamination in the event of a terrorist attack on the storage facility. Mandating this conversion also enhances the NRC's ability to fulfil its duty to protect the public health and safety.

E. If NRC Will Not Order Entergy to Undertake the Requested Protective Actions, then NRC Should Mandate the Immediate and Permanent Shutdown of IP Units 2 and 3 to Protect the Public Health and Security.

The vulnerability of Indian Point as its security measures are now is clear. The Indian Point nuclear facility is in danger of an attack that could have devastating consequences for hundreds of thousands, if not millions, of people living in the vicinity of Indian Point. NRC has been charged with protecting the public health and safety. In order to fulfil this duty, it must undertake the actions which have been outlined above.

However, if there is any reason that the NRC cannot undertake the outlined actions, there is only one alternative available which would enable NRC to fulfil its duty of protecting the public health and safety. If the NRC cannot comply with the petitioners' requests, then the NRC must mandate the immediate and permanent shutdown of Indian Point Unit Two and Unit Three.

Possible radioactive contamination from a successful terrorist attack on the Indian Point facility is substantial, especially if security measures at Indian Point are not upgraded. However, if the reactors are shut down, the high risk of contamination from a rupturing of the containment dome goes down precipitously. Likewise, shutting off the reactors also reduces the risk of releasing radioisotopes, which was responsible for many radiation illnesses in the wake of the Chernobyl accident. Finally, turning off Units Two and Three will make Indian Point less attractive as a terrorist target. This is so because terrorists are interested in "economy of force," that is, causing the maximum amount of damage for the least effort expended. By shutting down the Indian Point facility, the potential for causing catastrophic damage that could take the lives of many Americans is markedly reduced. As such, the damage that could be caused by attacking Indian Point after Units Two and Three have been shut down will not be worth the effort necessary to conduct the attack. Thus, shutting down Units Two and Three makes Indian Point a target that terrorists will deem not worth attacking. This thereby leaves the people living in and around Indian Point safe from nuclear disaster, thus enhancing the public health and safety.

IV. NRC has Broad Discretionary Powers to Order and Implement Petitioner's Request

A. 10 CFR § 2.202(a) Allows the NRC to Modify, Suspend or Revoke a License or to Take Any Proper Actions to Fulfil Its Duty to Maintain the Public Health and Safety.

It has been shown that the NRC needs to undertake the actions requested by the petitioners. Pursuant to 10 CFR § 2.202(a), the NRC has the authority to "institute a proceeding to modify, suspend or revoke a license or to take such actions as may be proper." This clearly gives the NRC the authority to order Entergy to comply with the requested actions.

The petitioners have requested that Entergy's license be modified to mandate upgrading the security measures at Indian Point. This modification of Entergy's license is permitted under the language of 10 CFR § 2.202(a). The petitioners also requested that if Entergy's license could not be modified to mandate the security measures, then the NRC must suspend Entergy's license, thereby forcing the shutdown of Units Two and Three. The wording of 10 CFR § 2.202(a) also allows suspension of a nuclear power plant's license. Finally, the petitioners also requested that the NRC force Entergy to transfer its spent fuel cooling pool storage system to a much safer dry cask storage system. The petitioners have shown that such an action would be proper in order for the NRC to maintain its commitment to the public health and safety. Thus, 10 CFR § 2.202(a) also grants the NRC the authority to mandate transfer of these spent fuel rods to dry storage as a necessary action.

For all these reasons, it is clear that 10 CFR § 2.202(a) grants the NRC the authority to undertake the actions the petitioners have requested.

B. § 161(b) of the Atomic Energy Act empowers the NRC to "establish rules, regulations and orders" to "protect health or to minimize danger to life or property.

The NRC has rightly said that it does not have the authority to establish a no-fly zone over the Nation's nuclear power plants. That authority only exists with the FAA. However, considering that the FAA and the NRC are both federal agencies that are coordinating with the Office of Homeland Security, it seems reasonable to assume that the NRC can coordinate with the FAA to establish no-fly zones over the Nation's nuclear power plants in order to secure them from terrorist attack.

No one can doubt that securing the airspace over nuclear power plants goes a long way towards protecting the public health and security and Entergy's own assets. The seeming impasse here results from the FAA having control over the Nation's airspace, while the NRC must protect the public health and security in all matters involving nuclear energy.

However, Section 161 of the Atomic Energy Act allows the NRC to establish rules and orders to protect the public health. Certainly, the NRC can write up a proposed rule to the FAA that nuclear power plants should be protected by a no-fly zone. But it would be the FAA's order that would actually create the no-fly zone. Thus, the NRC would be "establishing" the no-fly zone rule, but it would be the FAA's authority that *implements* the no-fly zone. Under this scenario, there would be no conflict of authority, as the NRC would be fulfilling its mandate under Section 161 to protect the public health and safety while the FAA would maintain full control of American airspace.

V. Conclusion

Since September 11, the NRC has been confronted with a new challenge: how to protect our nuclear power plants from terrorist attack and how to prevent such attacks from resulting in the release of radioactive contamination that could threaten countless lives. Despite the fact that this daunting challenge is unlike any the Nuclear Regulatory Commission has faced before, the stakes are so high that the NRC is compelled to act.

The actions the petitioners have outlined are needed and necessary to protect the public health and safety from the danger of radioactive contamination resulting from an attack on the Indian Point nuclear facility. The NRC must mandate that Entergy immediately act to improve the present state of security at Indian Point to protect it from terrorist attack whether via air, land or by water. Furthermore, if, for any reason, security at Indian Point cannot be made to protect the nuclear facility from terrorist attack, then the NRC must suspend Entergy's license and shut down the reactors at Indian Point known as Unit Two and Unit Three to make the facility less of an inviting target to terrorists. Lastly, the NRC must order Entergy to transfer spent fuel rods to a dry storage system from a cooling pool system, which is vulnerable to terrorist attack, to a dry cask system, which is decidedly more secure.

These actions are reasonable, within NRC's ability to achieve and mandated by NRC's responsibility to protect the public health, property and environment of New York. To uphold that responsibility, the NRC must undertake these actions.

Dated: August 9, 2002

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<u>The Following Organizations and Individuals Join in the Forgoing Comments on NRC Director's Proposed Decision</u>

Nuclear Control Institute
STAR Foundation
Waterkeeper Alliance
Hudson River Sloop Clearwater

Eliot Engel, U.S. Congress (D)
Maurice Hinchey, U.S. Congress (D)
Jerrold Nadler, U.S. Congress (D)

Eric Schneiderman, NY State Senate (D)
Thomas Marahan, NY State Senate (R)
Suzi Oppenheimer, NY State Senate (D)
Richard Brodsky, NY State Assembly (D)
Samuel Colman, NY State Assembly (D)
Alexander Gromack, NY State Assembly (D)
Naomi Matusow, NY State Assembly (D)
Amy Paulin, NY State Assembly (D)
Ronald C. Tocci, NY State Assembly (D)

Stanley Michels, NY City Council (D)
Jim Gennaro, NY City Council (D)

Scott Vanderhoef, Rockland County Executive (R)
Tom Abinanti, Westchester County Board of Legislators (D)
George Latimer, Westchester County Board of Legislators (D)
Vincent Tamagna, Putnam County Board of Legislators (R)
Sam Oliverio, Putnam County Board of Legislators (D)
Harriet Cornell, Rockland County Board of Legislators (D)

Paul Feiner, Town Supervisor, Greenburgh (D)
Greenburgh Town Board
Charles Holbrook, Town Supervisor, Clarkstown (D)
John Dinin, Town Supervisor, Bedford (R)
Christopher P. St. Lawrence Town Supervisor, Ramapo (D)

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