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Project WM-10

Mr. Jerry Rowe
Golder Associates, Inc.
2950 Northup Way
Bellevue, WA 98004

Dear Mr. Rowe:

Attached are copies of editorial requirements for SCA products, including Chapter 4 (Groundwater), Appendices, and Site Issue Analyses. This memorandum is for your use in preparing all contracted products under the current contract, NRC-02-82-045.

The action taken by this letter is considered to be within the scope of the current contract (NRC-02-82-045). No change to costs or delivery of contracted products is authorized. Please notify me immediately if you believe this letter would result in changes to costs or delivery of contract products.

Sincerely,

ORIGINAL SIGNED BY

Tilak R. Verma, Ph.D.
High-Level Waste Technical
Development Branch
Division of Waste Management

Attachment:
As stated

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

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WMHT: 3104.1

MEMORANDUM FOR: BWIP Group Coordinators, Review Team

FROM: Ronald B. Uleck
High-Level Waste Technical
Development Branch
Division of Waste Management

SUBJECT: EDITORIAL RECOMMENDATIONS FOR SCA PRODUCTS

In the BWIP group coordinators' meeting on November 10th, NRC editors Marthe Singh and Ann Thomas identified several principles that need to be followed in preparing SCA products:

- Use present tense as a general rule - e.g., "the staff finds that DOE ..." Use past tence only when necessary, such as when describing past events.

- Use "the staff" as the voice throughout, rather than "we" or the "NRC", unless specifically required otherwise. "The staff" refers to NRC staff; use appropriate descriptors for personnel of other organization.

- Personal communications (e.g., telephone calls) may be used in the text, but they must be clearly indicated as such in the text. Personal communications need not be cited formally in the reference lists.

- See NUREG 0770 and 0544 for a uniform use of terms, acronyms, and abbreviation.s

- SCA products should be totally defensible intellectually.

- Need help? Don't hesitate to call NRC editors;

  Marthe Singh, X29435

  Ann Thomas, X28903

To minimize the amount of rewriting and editing required to develop the SCA, it is important that all authors make every effort to adhere to these principles.

Ronald B. Uleck

Ronald B. Uleck
High-Level Waste Technical
Development Branch
Division of Waste Management

FORMAT

NUREG REPORT (DRAFT)

left margin ______	12
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paper ______	8½ x 11 [plain]
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PAGE NUMBER according to Section; if section 1, page numbers will be 1-1, 1-2, 1-3, etc.; if section is 22, page numbers will be 22-1, 22-2, 22-3, etc.; if section is Appendix B, page numbers will be B-1, B-2, etc.

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<u>Specific Instructions</u>

Each MAIN SECTION [1, 2, 3, etc.] starts a new page/job; title at sink line;
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Document Name:
NU0822 OYSTER CREEK SEC 1

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INTEGRATED PLANT SAFETY ASSESSMENT
SYSTEMATIC EVALUATION PROGRAM
OYSTER CREEK NUCLEAR GENERATING STATION

1 INTRODUCTION

1.1 Background

In the late 1960s and early 1970s, the U.S. Atomic Energy Commission's (now Nuclear Regulatory Commission) scope of review of proposed power reactor designs was evolving and somewhat less defined than it is today. The requirements for acceptability evolved as new facilities were reviewed. In 1967, the Commission published for comment and interim use proposed General Design Criteria for Nuclear Power Plants (GDC) that established minimum requirements for the principal design standards. The GDC were formally adopted, though somewhat modified, in 1971, and have been used as guidance in reviewing new plant applications since then. Safety guides issued in 1970 became part of the Regulatory Guide Series in 1972. These guides describe methods acceptable to the staff for implementing specific portions of the regulations, including certain GDC, and formalize staff techniques for performing a facility review. In 1972, the Commission distributed for information and comment a proposed "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," now Regulatory Guide 1.70. It provided a standard format for these reports and identified the principal information needed by the staff for its review. The Standard Review Plan (SRP, NUREG-75/087) was published in December 1975 and updated in July 1981 (NUREG-0800) to provide further guidance for improving the quality and uniformity of staff reviews, to enhance communication and understanding of the review process by interested members of the public and nuclear power industry, and to stabilize the licensing process. For the most part, the detailed acceptance criteria prescribed in the SRP are not new; rather they are methods of review that, in many cases, were not previously published in any regulatory document.

Because of the evolutionary nature of the licensing requirements discussed above and the developments in technology over the years, operating nuclear power plants embody a broad spectrum of design features and requirements depending on when the plant was constructed, who was the manufacturer, and when the plant was licensed for operation. The amount of documentation that defines these safety-design characteristics also has changed with the age of the plant--the older the plant, the less documentation and potentially the greater the difference from current licensing criteria.

Although the earlier safety evaluations of operating facilities did not address many of the topics discussed in current safety evaluations, all operating facilities have been reviewed more recently against a substantial number of major safety issues that have evolved since the operating license was issued. Conclusions of overall adequacy with respect to these major issues (for example, emergency core cooling system, fuel design, and pressure vessel design) are a matter of record. On the other hand, a number of other issues (for example, seismic considerations, tornado and turbine missiles, flood protection, pipe break effects inside containment, and piping whip) have not been reviewed against today's acceptance criteria for many operating plants, and documentation for them is incomplete.

1.2 Systematic Evaluation Program Objectives

The Systematic Evaluation Program (SEP) was initiated by the U.S. Nuclear Regulatory Commission (NRC) in 1977 to review the designs of older operating nuclear reactor plants in order to reconfirm and document their safety. The review provides (1) an assessment of the significance of differences between current technical positions on safety issues and those that existed when a particular plant was licensed, (2) a basis for deciding on how these differences should be resolved in an integrated plant review, and (3) a documented evaluation of plant safety.

The original SEP objectives were:

(1) The program should establish documentation that shows how the criteria for each operating plant reviewed compare with current criteria on significant

safety issues, and should provide a rationale for acceptable departures from these criteria.

(2) The program should provide the capability to make integrated and balanced decisions with respect to any required backfitting.

(3) The program should be structured for early identification and resolution of any significant deficiencies.

(4) The program should assess the safety adequacy of the design and operation of currently licensed nuclear power plants.

(5) The program should use available resources efficiently and minimize requirements for additional resources by NRC or industry.

The program objectives were later interpreted to ensure that the SEP also provides safety assessments adequate for conversion of provisional operating licenses (POLs) to full-term operating licenses (FTOLs). The final version of this report and a supplement that will address the status of all applicable generic activities (TMI and USI), including those that formed the basis for deletion of specific SEP topics, will form a part of the basis for the Commission's consideration of the license conversion.

Many of the plants selected for review were licensed before a comprehensive set of licensing criteria had been developed. They include five of the oldest nuclear reactor plants and seven plants under NRC review for the conversion of POLs to FTOLs. The plants to be considered under the original Phase II program were

(1) Yankee Rowe (FTOL PWR)
(2) Haddam Neck (FTOL PWR)
(3) Millstone 1 (POL BWR)
(4) Oyster Creek (POL BWR)
(5) Ginna (POL PWR)
(6) LaCrosse (POL BWR)

(7) Big Rock Point (FTOL BWR)
(8) Palisades (POL PWR)
(9) Dresden 1 (FTOL BWR)
(10) Dresden 2 (POL BWR)
(11) San Onofre (POL PWR)

The SEP review of Dresden 1 has been deferred because the plant is undergoing an extensive modification and is not scheduled for restart before June 1986. Therefore, the total number of plants being reviewed for Phase II is 10.

1.3 Description of Plant

The Oyster Creek Nuclear Generating Station, located in Ocean County, New Jersey, is a boiling-water reactor designed by General Electric. The licensees are GPU Nuclear Corporation and Jersey Central Power & Light Company (JCP&L). JCP&L, hereinafter referred to as the licensee, filed the application for a construction permit and operating license on March 24, 1964. The construction permit was issued on December 15, 1964. The initial submittal of the Final Safety Analysis Report was filed on January 25, 1967, and the initial provisional operating license was issued on April 9, 1969. In March 1972 the licensee applied for a full-term operating license. The licensed thermal power rating currently is 1,930 megawatt-thermal (MWt). The Oyster Creek primary coolant system consists of the reactor vessel, recirculation system, main steam system, and isolation condenser. The recirculation and isolation condenser systems are shown in Figure 1.1.

The reactor is a single-cycle, forced-circulation boiling-water reactor producing steam for direct use in the steam turbine. The reactor vessel contains internal components, which include the necessary equipment for separating steam and water flowpaths.

The recirculation system provides for forced flow through the reactor core to facilitate heat removal capability. Water that is separated from the steam in the reactor vessel and mixes with water provided by the feedwater system is drawn from outside the core, passes through the recirculation pumps, and reenters the reactor vessel below the core. The water then flows upward through the core where boiling produces a steam-water mixture.

The main steam system directs the steam generated in the reactor vessel to the turbine generator for conversion to electrical power. The steam-water mixture travels from the reactor core, through the steam-separating equipment into the main steam lines. The steam then passes through the main steam lines to the turbine. Included in the main steam system are the relief and safety valves, which provide overpressure protection for the reactor vessel and associated piping systems. The relief valves are also designed to rapidly depressurize the reactor vessel so that the emergency cooling systems will function. The reactor relief valves are located upstream of the first isolation valve and discharge directly to the pressure-suppression pool; the safety valves are located on the steam lines inside the primary containment and discharge to the drywell atmosphere.

The isolation condenser system, which consists of two condensers, will provide reactor core cooling if the reactor should become isolated from the main condenser because of closure of the main steam isolation valves. The isolation condenser operates by natural circulation. During operation steam flows from the reactor, condenses in the tubes of the isolation condenser, and flows back to the reactor by gravity.

The containment systems provide a multibarrier pressure-suppression containment composed of a primary containment, a Mark I pressure-suppression system, and a secondary containment, the reactor building.

The primary containment system is designed (1) to provide a barrier that will control the release of fission products to the secondary containment and (2) to rapidly reduce the pressure in the containment resulting from a loss-of-coolant accident. The system consists of a drywell, which houses the reactor vessel and recirculation loops; the pressure-suppression pool, which contains the large volume of water used to condense the accident steam release; and the connecting vent systems. The drywell, which is in the shape of a light bulb and is constructed of steel plate, varies in diameter from 70 ft to 33 ft and is approximately 64 ft high. The shell thickness varies from approximately 3/4 to 2-3/4 in. The pressure-suppression chamber is a steel pressure vessel in the shape of a torus with an inside diameter of 30 ft, a water volume of approximately 83,400 cubic feet, and an air volume of approximately 127,000 cubic feet.

The reactor building is designed to provide containment during reactor refueling and maintenance operations when the primary containment system is open. The building will also provide secondary containment when the primary containment is required to be in service. The reactor building consists of the monolithic reinforced concrete floors and walls enclosing the nuclear reactor, primary containment, and reactor auxiliaries, and the building superstructure with sealed panel walls and precast concrete roof.

1.4 Summary of Operating History and Experience

The Oyster Creek plant received a provisional operating license on April 9, 1969, achieved initial criticality on May 3, 1969, and began commercial operation on December 23, 1969. The reactor has a licensed thermal power of 1,930 MWt and a design electric rating of 650 MWe.

1.4.1 Summary of Oak Ridge National Laboratory Report

To ensure that the plant's operating history, including plant transients, was appropriately evaluated and factored into the NRC staff evaluation, the staff requested the Oak Ridge National Laboratory (ORNL) to perform a detailed review. A copy of the ORNL report is attached as Appendix F.

Table 1.1 presents the Oyster Creek reactor availability and plant capacity factors. From 1970 through 1981, the reactor availability factor at Oyster Creek averaged 74.4% and the unit capacity factor averaged 61.4%, both of which were above average for commercial nuclear power plants. Startup tests reduced the values in 1969, but the availability and capacity factors remained high from 1970 through 1979. The figures for 1980 and 1981 were low because of extended refueling and maintenance outages. During these shutdowns, Oyster Creek performed the 10-year ASME Code hydrostatic test on the reactor vessel and coolant piping and made TMI modifications.

Of the 203 forced shutdowns and power reductions between 1969 and 1981 at Oyster Creek, 55 were design-basis events of one of the following 10 types:

(1) turbine trip (15)
(2) loss of normal feedwater (9)
(3) recirculation pump trip (9)
(4) loss of condenser vacuum (7)
(5) inadvertent closure of main steam isolation valve (MSIV)
(6) pressure regulator failure resulting in decreased steam flow (3)
(7) decreased feedwater temperature (2)
(8) pressure regulator failure resulting in increased steam flow (2)
(9) inadvertent opening of turbine valve (2)
(10) loss of external load (1)

The trend for the number of reportable event reports submitted by Oyster Creek is generally upward with peak years of 1974, 1980, and 1981, with 65, 75, and 72 events, respectively. The causes of reportable events have been primarily inherent equipment failures, which accounted for 64% of all reported events. Human error (including administrative, design, fabrication, installation, maintenance, and operator error) accounted for 34% of the reported events. Other causes, such as adverse environmental conditions, were responsible for the remaining 2%. There is no apparent trend in the causes of reported events.

The major contributor to the significant events was human error, which caused 15 of 17 of these events. The remaining events were caused by equipment failures (valve failures) and occurred early in Oyster Creek's operating history. Since 1976, the frequency at which significant events have occurred has steadily increased. This increased rate of occurrence is directly related to the increased frequency of containment integrity violations. When events pertaining to the loss of containment integrity are disregarded, no trend is apparent in the rate of occurrence of significant events.

Between 1969 and 1974, Oyster Creek experienced a variety of recurring mechanical problems with the MSIVs, including bent valve stems, packing leaks, and sticking pilot valves. Each problem was corrected by proper equipment modification. Various problems also occurred with the torus-to-reactor-building and torus-to-drywell vacuum breaker valves.

Reactor vessel cracks were noted three times throughout the history of Oyster Creek. In 1974, an inservice inspection revealed cracking in reactor head cladding. However, no cracks propagated into the reactor vessel base material. Later in 1974, a small leak was noted in a field weld between the incore housing and the vessel lower head. Since its repair, no further cracking has been noted.

Condenser tube leakage problems began in 1970. Through 1975, recurring power reductions were necessary to repair or plug leaking tubes. During a shutdown in the first part of 1976, condensers were retubed using welded titanium tubing. With the exception of a limited number of vibration-induced tube failures, these titanium tubes have functioned satisfactorily.

Outdated or inadequate procedures caused or at least complicated 24 of the reported events at Oyster Creek. The number of events averaged one per year between 1972 and 1977. The average increased to four per year between 1978 and 1981. Four of the events between 1978 and 1981 were significant events.

1.4.2 Operating Experience Since January 1, 1982

The Oyster Creek Nuclear Generating Station has operated from January 1 to July 31, 1982 with a unit availability factor of 47.2%, a capacity factor of 30.7%, and a forced outage factor of 52.8%, as compared with the cumulative factors of 71.1% availability, 60.0% capacity, and 11.5% forced outage. The plant entered a forced shutdown on December 9, 1981, when valve steam damage was discovered on isolation condenser isolation valves with Limitorque motor operators. The damage, which impaired valve operability, was caused by frequent backseating of the valves to stop packing leaks. Valve repairs were completed in January 1982, but the startup was delayed by control rod drive hydraulic pump failures, emergency diesel generator air cooler leaks, and required refueling cycle surveillances. The reactor was started up on April 16, 1982, on completion of required maintenance and testing. Power was limited to about 67% because of the unavailability of one of three condensate pumps. The plant was again forced to shut down from May 23 to May 27, 1982 because of a leak on a steam reheater manway cover. The third condensate pump was returned to service on June 5, 1982, and power was increased to about 85% limited by available core reactivity. The plant operated through July 1982 in coastdown with a refueling and modification outage scheduled to begin January 15, 1983.

During this period, the licensee requested deferment of 17 generic modifications for which commitment had been made for completion during the 1983 Cycle X refueling. The licensee requested that these items be deferred until the Cycle XI refueling outage proposed for early 1985.

1.4.2.1 Operating and Regulatory Performance Since January 1, 1982

A management meeting was held with the licensee on April 16, 1982 to discuss the findings of NRC Region I's Systematic Assessment of Licensee Performance (SALP) program. The SALP review included Oyster Creek's operating performance and NRC inspection activities from November 1, 1980 to October 31, 1981. Through the assessment period, the staff considered the licensee's operational and regulatory performance generally acceptable and directed toward safe facility operation; however, the staff found that the areas of maintenance and surveillance were in need of increased management attention.

An enforcement conference was held with the licensee management on May 4, 1982 to discuss NRC's concerns pertaining to violations related to the inoperability of the reactor-building-to-suppression-chamber vacuum breakers and isolation condenser isolation valves. These violations were the result of inadequate management controls over maintenance testing and surveillance activities.

During this period, an emergency preparedness appraisal identified the need to (1) upgrade the emergency support facilities, (2) improve postaccident coolant and containment atmosphere sampling capabilities, and (3) upgrade emergency response training and retraining.

The licensee has committed to increased staffing and management reorganization to improve the overall quality and control of maintenance, surveillance, and modification/construction activities.

Forty-two licensee event reports (LERs) were submitted from January 1 to July 31, 1982. The number and nature of the LERs are shown in Table 1.2.

Figure 1.1 Recirculation, steam and isolation condenser schematic
Source: Oyster Creek FSAR

Table 1.1 Oyster Creek Availability and capacity factors

Table 1.2 Number and nature of licensee event reports (LERs) from January 1 to July 31, 1982

Type of event	Number of LERs
Personnel error	6
Design, manufacture, contractor/ installation error	1
External cause	0
Defective procedures	4
Component failure	11
Other	20
Total	42

Document Name:
OYSTER CREEK NU0822 / A

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JPK

Author's Name:
Wang/Mejac

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TOPIC: II-1.A Exclusion Area Authority and Control

(1) Definition:

The establishment of the exclusion area and the licensee's control over it are reviewed at the construction permit/operating license stage. Thereafter, the licensees are required to report any changes with safety implications. The concern exists, however, that (1) the original review may not have been as thorough as currently done, or (2) changes may have occurred but have not been reported and reviewed. In particular, new activities within the exclusion area (for example, new recreational facilities or offshore oil drilling) and topographical changes (for example, changes in water levels) may need to be reviewed.

(2) Safety Objective:

To assure that appropriate exclusion area authority and control is maintained by the licensee.

(3) Status:

Selective reviews have been performed (San Onofre Nuclear Generating Station Unit 1) or are under way (Fort Calhoun) where changes in exclusion area boundary have become necessary.

(4) References:

1. Title 10, "Energy," Code of Federal Regulations, Part 100*
2. NUREG-75/087, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants - LWR Edition, "December 1975,"** Section 2.1.2

*Hereafter referred to as 10 CFR.
**Hereafter referred to as Standard Review Plan.

TOPIC: II-1.B Population Distribution

(1) Definition:

Population distribution in the vicinity of operating plants may have changed since the initial review was performed at the construction permit stage. Special attention should be given to new housing and commercial, military, or institutional installations established since the initial population-distribution review.

(2) Safety Objective:

New population distributions may require revision of low-population zone (LPZ) and population center to assure appropriate protection for the public by complying with the guidelines of 10 CFR Part 100. Adjustments may have to be made in emergency plans. New accident analyses may have to be performed to determine consequent conformance with 10 CFR Part 100 at new LPZ distances. Potential need for additional engineered safety features (for example, chemical sprays or better filters) exists.

(3) Status:

Has been done on a selective basis only, that is, Pilgrim Unit 1 new population center.

(4) References:

1. 10 CFR Part 100
2. Standard Review Plan, Section 2.1.3

TOPIC: II-1.C Potential Hazards or Changes in Potential Hazards Due to Transportation, Institutional, Industrial, and Military Facilities

(1) Definition:

For operating plants there are three concerns:

(a) New hazards created since the facility was licensed,

(b) Hazards considered for licensing but that have expanded beyond projections or which were not reviewed against current criteria, and

(c) Hazards that were not analyzed at the licensing stage because of lack of regulatory criteria at the time.

Nearby transportation, institutional, industrial, and military facilities may be threats to safe plant operation due to:

(a) Control room infiltration of toxic gases,

(b) Onsite fires triggered by transport of combustible chemicals from offsite releases,

(c) Shock waves due to detonation of stored or transported explosives and military ordnance firing, and

(d) Onsite aircraft impact.

(2) Safety Objective:

To assure that the control room is habitable at all times and that the postulated hazards will not result in releases in excess of the 10 CFR Part 100 guidelines by disabling systems required for safe plant shutdown.

(3) Status:

Action has been taken on a selective basis only, for example, curbing of military air activity in the vicinity of the Big Rock Point Plant. Liquid natural gas (LNG) hazards at Calvert Cliffs are under review. The review of older plants did not consider offsite hazards in detail (for example, aircraft traffic in the vicinity).

(4) Reference:

Standard Review Plan, Sections 2.2.1 and 2.2.2

TOPIC: II-2.A Severe Weather Phenomena

(1) Definition:

Safety-related structures, systems, and components should be designed to function under all severe weather conditions to which they may be exposed. Meteorological phenomena to be considered include tornadoes, snow and ice loads, extreme maximum and minimum temperatures, lightning, combinations of meteorology and air-quality conditions contributing to high corrosion rates, and effects of sand and dust storms.

(2) Safety Objective:

To assure that the designs of safety-related structures, systems, and components reflect consideration of appropriate extreme meteorological conditions and severe weather phenomena. This effort would identify deficiencies in designs and/or operation that may contribute to accidental releases of radioactivity to the atmosphere resulting in doses to the public in excess of 10 CFR Part 100 or Part 20 guidelines (as appropriate to the design of the component or system).

(3) Status:

Generic studies have been initiated to develop guidelines for extreme temperatures and lightning, and to the review the current Branch Positions on snow loads. Estimated completion dates are 6/1/78 or later.

(4) References:

1. 10 CFR Part 100 or Part 20
2. Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants"
3. Standard Review Plan, Section 2.3.1
4. Branch Technical Position, "Winter Precipitation Loads," March 24, 1975
5. Inquiry by Chairman Rowden Concerning Lightning Protection, July 9, 1976
6. 10 CFR Part 50

TOPIC: II-2.B Onsite Meteorological Measurements Program

(1) Definition:

To review the onsite meteorological measurements program to determine the extent that the licensee complies with 10 CFR Part 50, Appendix E and Appendix I.

(2) Safety Objective:

To assure that adequate meteorological instrumentation to quantify the offsite exposures from routine releases is available and maintained.

(3) Status:

Onsite meteorological measurements programs are being reviewed as a part of the Appendix I evaluations.

(4) References:

1. 10 CFR Part 50, Appendix E and Appendix I
2. Regulatory Guide 1.97, Rev. 1, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident"

3. Regulatory Guide 1.23, "Onsite Meteorological Programs"
4. Standard Review Plan, Section 2.3.3

(5) <u>Basis for Deletion (Related TMI Task, Unresolved Safety Issue (USI), or Other SEP Topic)</u>:

(a) <u>TMI Action Plan Task II.F.3, "Instrumentation for Monitoring Accident Conditions" (NUREG-0660)</u>

Task II.F.3 requires that appropriate instrumentation be provided for accident monitoring with expanded ranges and a source term that considers a damaged core capable of surviving the accident environment in which it is located for the length of time its function is required. Regulatory Guide 1.97, Revision 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident," issued December 1980, contains the required meteorological instrumentation to quantify the offsite exposure.

(b) <u>TMI Action Plan Task III.A.1, "Improve Licensee Emergency Preparedness - Short Term" (NUREG-0660)</u>

Task III.A.1 requires the evaluation of 10 CFR Part 50, Appendix E, backfit requirements in accordance with NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants." Backfit requirements include review of the Onsite Meteorological Measurement Program.

The evaluations required by Tasks II.F.3 and III.A.1 are identical to SEP Topic II-2.B; therefore, this SEP topic has been deleted.

TOPIC: II-2.C Atmospheric Transport and Diffusion Characteristics for Accident Analysis

(1) Definition:

To review the atmospheric transport and diffusion characteristics assumed to demonstrate compliance with the 10 CFR 100 guidelines with respect to plant design, control room habitability, and doses to the public during and following a postulated design-basis accident. This effort would examine the assumptions for:

(a) Effects of explosive concentrations from onsite or offsite releases of hazardous material for consideration in structural design,

(b) Calculation of relative concentration (x/Q) values for releases of radioactivity and toxic chemicals for consideration in control room habitability, and

(c) Calculations of doses to the public resulting from releases of radioactivity to the atmosphere during and following a postulated design-basis accident.

This effort is considered necessary because most original reviews were performed using the assumptions provided in Regulatory Guides 1.3 and 1.4 which have been found to be generally nonconservative based on evaluation of over 50 sites with actual meteorological observations.

(2) Safety Objective:

To assure that the atmospheric transport and diffusion characteristics originally assumed to demonstrate compliance with the 10 CFR 100 guidelines are appropriate, considering additional onsite meteorological data and results of recent atmospheric diffusion experiments.