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May 7, 1981



Mr. W. Paulson
U. S. Nuclear Regulatory Commission
Washington, D. C.

Dear Mr. Paulson:

Subject: Oyster Creek Nuclear Generating Station,
Systematic Evaluation Program, Docket No. 50-219-1

Jersey Central Power & Light Company (JCP&L) letter, dated February 4, 1981, to the U. S. Nuclear Regulatory Commission (NRC) advised that JCP&L will undertake to prepare approximately 20 SEP draft Topic Assessments by June, 1981. In accordance with that commitment, we are transmitting herewith for your review the following draft Topic Assessments:

A. Design Basis Event Evaluations

Topic XV-1, "Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow and Inadvertent Opening in a Steam Generator Relief or Safety Valve"

Topic XV-3, "Loss of External Load, Turbine Trip, Loss of Condenser Vacuum, Closure of Main Steam Isolation Valve (BWR), and Steam Pressure Regulatory Failure (Closed)."

Topic XV-4, "Loss of Non-Emergency A-C Power to the Station Auxiliaries."

Topic XV-5, "Loss of Normal Feedwater Flow."

Topic XV-9, "Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate."

Topic XV-20, "Radiological Consequences of a Fuel-handling Accident."

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Topic XV-11, "Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position (BWR)."

Topic XV-13, "Spectrum of Rod Drop Accidents (BWR)."

Topic XV-14, "Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory."

Topic XV-15, "Inadvertent Opening of a PWR Pressurizer Safety/Relief Valve or a BWR Safety/Relief Valve."

Topic XV-18, "Radiological Consequences of Main Steam Line Failure Outside Containment (BWR)."

Topic XV-19, "Loss of Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary."

Topic XV-24, "Loss of All AC Power."

B. Seismology and Geology Evaluations

Topic II-4. "Tectonic Province."

Topic II-4.B, "Proximity of Capable Tectonic Structures in Plant Vicinity."

Topic II-4.C, "Historical Seismicity Within 200 Miles of Plant."

C. Site Hazard Evaluations

Topics II-3.A, II-3.B and II-3.B.1, "Hydrologic Description, Flooding Potential and Ability to Cope with the Design Basis Flood."

Topic II-3.C, "Safety-Related Water Supply (Ultimate Heat Sink (UHS))."

Topic III-2, "Wind and Tornado Loadings."

Topic III-3.A, "Effects of High Water Level on Structures."

Topic III-4.D, "Site Proximity Missiles (Including Aircraft)."


Topic II-1.A, "Exclusion Area Authority and Control."

Topic II.1.B, "Population Distribution."

Topic II.1.C, "Potential Hazards or Changes in Potential Hazards Due to Transportation, Institutional, Industrial, and Military Facilities."

We will be pleased to meet with you to discuss these draft assessments if you desire.

Sincerely,


Ivan R. Finfrock, Jr.
Vice President

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TOPIC ASSESSMENT
DESIGN BASIS EVENT EVALUATION
OYSTER CREEK NUCLEAR GENERATING STATION

Docket No. 50219-1

April 1981

DESIGN BASIS EVENT EVALUATION

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DESIGN BASIS EVENT EVALUATION

A. Introduction

The safety philosophy used in the design of reactor plants has traditionally been based on the concept of "defense-in-depth." The approach begins with a conservative design, using components of high quality. Redundant and diverse systems are used to ensure that a single failure will not prevent system functions. The reactor systems are designed to prevent unforeseen occurrences, and to mitigate the consequences of such events should they happen.

One important means of protecting the public from exposure to the radioactive products produced by nuclear fission in the fuel is by providing multiple barriers between the fuel and the public. The three main layers of defense are the physical barriers of the reactor fuel clad, the reactor coolant system pressure boundary, and the reactor containment building.

System disturbances and malfunctions or equipment failures can occur during plant operation and challenge the integrity of the three barriers. These are analyzed to determine the capability of the plant design and installed plant systems to prevent breaching these barriers.

The American Nuclear Society has classified plant conditions into four categories in accordance with anticipated frequency of occurrence and potential radiological consequences to the public. In general, this classification is also followed in the NRC Standard Review Plan Chapter 15 review procedure for plant accidents and transients. The four categories are:

Condition I: Normal operation and operational transients

Condition II: Faults of moderate frequency

Condition III: Infrequent faults

Condition IV: Limiting faults

The basic principle applied in relating design requirements to each of the conditions is that the most probable occurrences should yield the least radiological risk to the public, and those extreme situations having the potential for the greatest risk to the public shall be those least likely to occur. The impact of various single failures on the course of an accident or transient is also considered.

For a new plant under review for an operating license, the approach outlined in Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plant," Chapter 15 is used to:

1. Ensure that a sufficiently broad spectrum of initiating events has been considered,

2. Categorize the initiating events by type and expected frequency of occurrences so that only the limiting cases in each group need to be quantitatively analyzed, and
3. Permit the consistent application of specific acceptance criteria for each postulated initiating event.

To accomplish these goals, a number of disturbances of process variables and malfunctions or failures of equipment should be postulated. Each postulated initiating event falls into one of the following categories:

1. Increase in heat removal by the secondary system (turbine plant)
2. Decrease in heat removal by the secondary system (turbine plant)
3. Decrease in reactor coolant system flow rate
4. Reactivity and power distribution anomalies
5. Increase in reactor coolant inventory
6. Decrease in reactor coolant inventory
7. Radioactive release from a subsystem or component

One of the items of information that is considered for each initiating event relates to its expected frequency of occurrence. Each initiating event within the seven major

categories (see previous list) is assigned to one of the following frequency groups:

1. Incidents of moderate frequency
2. Infrequent incidents
3. Limiting faults

The initiating events for each combination of category and frequency group are evaluated to identify the events that are limiting. The intent is to reduce the number of initiating events that need to be quantitatively analyzed. That is, not every postulated initiating event needs to be completely analyzed. In some cases, a qualitative comparison of similar initiating events is made to identify the specific initiating event that leads to the most limiting consequences. Only that initiating event is analyzed in detail.

This approach is used in the reevaluation of accidents and transients for the Systematic Evaluation Program (SEP) facilities. The accident and transient analyses for the Oyster Creek plant are discussed and evaluated in the following subsections. In accordance with the SEP review method, the evaluation includes an assessment of the expected system response and the ability of the plant to adequately mitigate the event. The current regulatory criteria used in the accident and transient evaluations are those found in Chapter 15 of the Standard Review Plan.

The sections which follow present a history of DBE evaluations from the initial plant FDSAR to present, a description of the various computer codes and analytical models used, and individual assessments of the various DBEs. A complete list of applicable references is also included. The assessments included cover SEP Topics XV-1, -3, -4, -5, -7, -9, -11, -13, -14, -15, -18, -19 and -24. Although specifically applicable only to PWRs, available information covering topics XV-2 and XV-6 is also provided. (Assessments of SEP Topics XV-16 and XV-20 will be covered in a separate report.)

B. DBE Documentation History

The Oyster Creek FSAR (Reference 1) was issued in January 1967. The submittal was prepared by the licensee, Jersey Central Power and Light Company, and the vendor, General Electric. Most transients and accidents were analyzed in this report at a power level of 1600 MWt.

During 1970, additional analyses were provided at power levels of 1690 MWt and 1930 MWt. These results were transmitted in References 3 and 4 to support power upratings.

Starting in 1974, for cycle 5 operations, the core was reloaded with Exxon fuel. Exxon submitted topical reports on their methods (Reference 6), comparisons to GE results (Reference 8), and their calculations for 7 X 7 (Reference 9) and 8 x 8 fuel (Reference 11). Analysis of the rod withdrawal event is in Reference 10. For these submittals, analyses at 1930 MWt were performed. These reports were submitted with the cycle 5 reload report (Reference 7).

The result of these analyses was the MCHFR for each event, calculated with the XN-1 correlation. In 1975, the XN-2 correlation was developed, which includes the Tong factor to handle nonuniform axial power distribution effects.

During operation, the MCPR is of more interest, since this gives the power margin, and the MCPR is more readily determined using the on line computer from the existing operating conditions. In Reference 12, MCHFR results with XN-2 were presented for limiting events. Technical

Specifications were altered so that an MCPR of 1.40 (as determined from the MCHFR) was assured for any transient. This necessitated a power limit of 1820 MWt. The staff evaluation in Reference 16 approved the use of an MCPR of 1.4 and these supporting analyses.

Later in 1975, additional analyses were done (Reference 14) to directly calculate MCPR so that the full power rating of 1930 MWt could be supported. Limiting events (rod withdrawal, turbine trip, w/o bypass, 5 pump trip, and slow loss of feedwater heating) were assessed. Additional test data obtained also allowed operation with a MCPR of 1.32 (7 x 7) or 1.34 (8 x 8). These values provide a 95%/95% confidence that fuel rods will not experience boiling transition.

The staff evaluation of this analysis is presented in Reference 21. This SER accepted use of the XN-2 correlation to calculate MCPR, and use of the MCPR's given above as the design limits.

These results form the reference analyses since reloads after cycle 6 (1976) were conducted under the provisions of 10 CFR 50.59.

LOCA Analysis

The original LOCA analysis was presented in Chapter XIII of the FSAR. This analysis considered large double-ended recirculation line breaks.

Later review determined that, since Oyster Creek does not have a high pressure coolant injection system, small breaks may be more limiting. The effects of the emergency condensers, auto-relief system and core spray systems were included, as well as the control rod drive pumps and the feedwater pumps (if offsite power is available).

Starting in 1974, Exxon supplied analyses to support their reload fuel. Modifications to the models were required, however, to conform to Appendix K. No credit is taken for some systems such as the CRD pumps which are not safety-grade.

In 1975, documentation of the Exxon WREM-Based NJP-BWR ECCS Evaluation Model was submitted (Reference 17). This model was used to determine LOCA performance for Oyster Creek. Staff evaluation of the evaluation model is shown in Reference 18. The model was found to be in conformance with the requirements of Appendix K.

Calculations for Exxon fuel types were done, late in 1975 and early in 1976, to show conformance with 10 CFR 50.46. These analyses in References 19 and 20 used the approved models. Two input parameter changes, use of less than design limit LHGR's and use of 100% of the spray coefficients for 8 X 8 fuel were approved by the staff in Reference 21.

The spray coefficients assumed in the Oyster Creek analyses are based on the minimum spray flow provided to any assembly by spray distributed from a single spray system. However, the minimum flow per assembly may be reduced due to a narrowing of the cone angle at the spray nozzles. This effect could result in reduced spray coefficients for BWRs if the minimum spray flow per assembly falls below the minimum value previously assumed. However, Oyster Creek has modified their spray systems such that no single failure could preclude operation of either spray system.

The Exxon analysis discussed above used as input results from NSSS vendor blowdown analysis and then Exxon heatup results. Exxon later submitted a model which included both blowdown and heatup calculations in reference 22. The NRC accepted this topical report for reference to plant applications in reference 23.

Analysis for Oyster Creek using this model was submitted in reference 24. The results from this report are the reference analysis for LOCA for the Oyster Creek plant.

C. Codes and Models

Analyses in the FSAR were done with GE plant transient codes. For the limiting events, which are reanalyzed for reloads, the Exxon plant transient code PTS-BWR/MOD 002 (Reference 6) was used.

The PTSBWR2 digital computer code is used to assess transient performance of non-jet pump BWR's. The codes utilize the basic transient fluid conservation equations for mass, energy and one-dimensional momentum. A point kinetics reactivity model is used with feedback from Doppler, voiding and control rods. Axial weighting factors on power or feedback can be used. The program calculates fluid conditions, such as quality, pressure and flow as well as heat flux, power and reactivity. Control functions such as reactor scram, relief and safety valve flow, isolation valve closure, water level controller and steam pressure regulator can be modeled. The code was latter modified to include an isolation condenser model for the analysis of the loss of feedwater transient (reference 26).

As discussed in Section B, this code calculates the MCPR during the event. The transient analyses were performed with the following initial conditions and assumptions unless otherwise specified in the individual DBE section:

- 1930 MWt = 100% power

- End of cycle void coefficient for heatups

- Beginning of cycle void coefficient for cold water addition events

- 25% uncertainty factor on void coefficient

- 1035 psig reactor pressure

- bounding scram reactivity curve

The analyses assume an initial power level of 100% (1930MWt). The Standard Review Plan requires use of a 2% power measurement uncertainty. This deviation is being assessed by the NRC on a generic basis for all BWR/2's and BWR/3's, but an implementation position has not been established.

As required by Appendix K, the LOCA analysis assumes an initial power of 102%.

The LOCA codes are discussed in Section 8 and Section 7.1.

D. DBE Performance

1.0 Group I Events

These moderate frequency events involve either an increase in heat removal by the secondary system or an increase in core flow.

1.1 Decrease in Feedwater Temperature (Topic XV-1)

A decrease in feedwater temperature can result from a failure of a feedwater heater. The analysis assumes an instantaneous loss of all feedwater heating with the temperature dropping to 135°F.

The enthalpy reduction results in a power increase due to the negative void coefficient. An overpower trip occurs. Pressure does not increase due to continued turbine demand until the reactor trip.

A second case was also considered, a slow loss of heating resulting from a turbine trip. This event was found to be essentially the same as a turbine trip, since the power and pressure transient (due to the turbine trip) is over before the colder water reaches the core (delay time for water to reach the bottom of the downcomer). These results as well as those for the instantaneous loss are given References 9, 11 and 15.

A slow loss of heating not accompanied by a turbine trip causes a more gradual decrease in feedwater enthalpy, with the temperature dropping from 315°F to 100°F in 60 seconds. This slow loss is more severe in terms of MCPR reduction because the peak heat flux is higher. The complete loss of feedwater heating results in a quick neutron flux increase and thus an overpower trip is reached sooner. If a trip did not occur, the heater loss would be a more severe event.

The sequence of events for these events is summarized below.

Slow loss, with turbine trip

<u>Time (sec)</u>	<u>Event</u>
0	turbine trip, tertiary heaters lost
1	overpower reactor trip
1+	bypass valve opens

Slow loss, without turbine trip

<u>Time (sec)</u>	<u>Event</u>
0	IW heaters lost
30	APRM overpower trip

Instantaneous Loss

<u>Time (sec)</u>	<u>Event</u>
0	loss of IW heating
5	colder water reaches the core
5.5	overpower trip

Following the scram, the plant is in a condition from which a safe shutdown can be achieved.

Operator procedure in the event one or more heaters are lost is to reduce reactor power by control rod insertion. Failure to do so would make an overpower reactor trip occur sooner than it might otherwise.

1.2 INCREASE IN FEEDWATER FLOW (TOPIC XV-1)

An increase in feedwater flow can result from a malfunction of the feedwater controller. Power increases in response to void reduction reactivity.

The analyzed events are initiated from approximately 53% power, 42% flow, since these conditions permit the largest increase in flow, and thus feed/steam mismatch.

The transient is terminated by the high water level turbine trip in 8 seconds. Scram occurs during closure of the turbine stop valves. The bypass valves automatically opens due to the increasing reactor perssure.

This event has been assessed by Exxon in References 9 and 11.

The generic reload topical report for GE BWR's considered the feed-water controller failure to maximum demand as a potentially limiting transient. The event is assumed to be initiated from full power, and proceeds until the high level turbine trip produces a reactor scram. The event can be thermally limiting since the turbine trip occurs from an elevated power level. The bypass is assumed to function normally. The increase in feed flow from full power has not been analyzed for Oyster creek. However, based on comparisons from other BWR's, results of this analysis are not expected to require changes to a setpoint or the initial MCPR.

1.3 Increase in Steam Flow (Topic XV-1)

An increase in steam flow can occur due to a pressure regulator malfunction. The increase heat removal causes reactor cooldown, pressure decrease and void formation. Neutron flux decreases via the void reactivity feedback.

Transients were considered from three power levels, 1860 MWt, 960 MWt and hot standby. In each case, the turbine control valves are opened to 110% of full rated. A mechanical stop prevents further valve opening.

The hot standby event has the fastest depressurization rate, as would be expected.

The decrease in pressure would result in MSIV closure (825 psig). A reactor scram is initiated by low reactor water level. The MSIV closure would also trip the reactor upon their reaching 10% closed.

Pressure regulator malfunctions are considered in Section VII-8 of the FSAR (Reference 1) and Amendment 14 (Reference 2). Since there are only minor consequences compared to other possible transients, no reanalysis was performed for later cycles.

1.4 STARTUP OF AN IDLE LOOP (SOPIC XV-9)

The startup of an idle loop could result in a cold water addition to the core, which could cause a power increase through void collapse.

Oyster Creek is a non-jet pump plant with five recirculation loops, each with isolation valves. Procedures and interlocks on pump starting and valve opening effectively prevent such a transient. Multiple failures are required for a severe idle loop startup event to occur.

Procedures require that the loop discharge valve be closed and the suction valve open before the pump is started (a small bypass line prevents dead-heading the pump). After the pump is running (at 30%), the operator can open the discharge valve. When it is fully open the pump can be placed in automatic speed control.

Normally a loop will not be isolated when its pump is shut down so that reverse flow via the bypass valve will keep the loop hot. A 50°F differential requires reactor shutdown to restart the pump.

However, an analysis of this event has been performed by Exxon at 1930 Mwt in reference 25. The following assumptions are made:

- 1) Water in the isolated loop is 100°F.
- 2) The suction and bypass valves are opened at the same time.
- 3) Coincident with valve opening, the pump is started and quickly brought up to speed.

- 4) The discharge valve is opened as soon as the pump is isolated.
- 5) 100% power and 100% flow from 4 running pumps.
- 6) Scram setting is at 116% power.

Cases were also assessed from reduced power and flow conditions. The full power case was the most severe. The results show that a high flux scram occurs 7 seconds after the cold water reaches core. The MCPR limit is not violated.

A malfunction of the speed controller can cause the scoop tube positioner to move at its maximum speed in the direction of increasing pump flow, a 10%/sec increase. A malfunction of the master flow controller is less severe since it has its own rate limit which is less than that of individual scoop positioner.

The transient is initiated from 52% flow, 67.5% power, to allow the maximum possible increase in flow. Neutron flux increases due to the void coefficient. Heat flux does not increase above the steady-state value due to thermal lag. Power and flow re-establish themselves at a new equilibrium value.

The consequences of this event are relatively mild, and resemble normal load followed by flow control. No thermal limits are approached.

This event was assessed in the FSAR, as well as in References 9 and 11 by Exxon.

1.6 Main Steam Line Isolation Valve Closure (Topic XV-3)

Inadvertent closure of the main steam line isolation valves can result in vessel overpressurization and loss of the steam removal path through the steam line to the turbine. Alternate paths are needed for heat removal.

The relief valves located upstream of the MSIV's open automatically on high pressure after 4 seconds to relieve the pressure transient. Reactor scram occurs on 10% closure of the isolation valves at 0.3 seconds into the event. For long-term cooling the relief valves and isolation condensers are adequate for removing decay heat.

For a conservative analysis of the pressure, the fastest closure time (3 seconds) is assumed, and no credit is taken for heat removal by the isolation condensers. The safety valve lift setpoint is not reached.

This event was analyzed by Exxon in references 9 and 11. The results showed that the pressure consequences were less severe than those predicted by the turbine trip without bypass event (The relief valve sizing transient).

BWR's covered by the GE Generic Reload Methodology assess MSIV closure with reactor trip due to the high flux scram as the limiting overpressure event. This sequence of events generally results in a higher pressure than a turbine trip or load rejection with bypass failure (with scram on closure of turbine stop valves or control valves). Oyster Creek

analyzes an even more severe event to assess the adequacy of safety valve sizing. The turbine stop valve closes, no scram is assumed to occur, and the relief valves, bypass valves and isolation valves on the isolation condensers are all assumed to fail. The safety valves are able to maintain pressure below 1375 psig (110% of reactor vessel design pressure).

2.0 GROUP II RESULTS

The moderate frequency events in group II are characterized by a decrease in heat removal by the secondary system.

2.1 Loss of Load (Topic XV-3)

Upon loss of electrical load, the turbine generator overspeeds, causing a rapid closure of the turbine control valves. Reactor trip occurs on sensing the load rejection (acceleration relay on the turbine control valve system). The bypass automatically accepts the steam load when the turbine is isolated.

The loss of load transient behaves similarly to a turbine trip event. However, since the steam flow to the turbine is initially reduced by action of the throttle valve rather than sudden closure of the stop valve, the pressure and power transients are milder. The isolation condenser is not actuated since the pressure increase does not last as long as the 15 seconds actuation delay.

Thus, this event is bounded by the turbine trip, and is not analyzed separately.

2.2 Turbine Trip (Topic XV-3)

A turbine trip can result from a variety of malfunctions of the turbine generator such as overspeed, or from inadvertent closure of the turbine stop valves. The sudden loss of the reactor heat sink leads to pressurization and heatup of the reactor.

Protection is afforded by a reactor scram upon 10% closure of the turbine stop valves. The normal path of heat removal would be through the bypass valve to the main condenser. Use of this method requires offsite power to cool the condenser. For a conservative calculation, bypass failure is assumed, and alternative heat removal means are used, such as the relief valves and isolation condenser. These systems do not require AC power.

Exxon analyzes three turbine trip events (references 9 and 11).

The turbine trip from full power with scram on closure of the turbine stop valves and operation of the bypass valves. Pressure remains well below the lowest safety valve lift set point.

The turbine trip from full power with failure of the bypass valves (relief valve sizing transient). Scram occurs on closure of the turbine stop valves. The relief valves are adequate to keep pressure below the lowest safety valve lift set point.

The turbine trip from full power with combined failure of bypass, isolation condensers and relief valves is used to size the safety valves. No reactor scram is assumed to occur. The safety valves are adequate to keep pressure below 110% of the design limit.

The sequence of events is summarized below for a turbine trip, bypass failure, reactor scram or 10% turbine stop valve closure:

<u>Time (sec)</u>	<u>Event</u>
0	turbine trip
0+	reactor trip
1.5	relief valve opens
9.0	relief valve closes

For the safety valve sizing event, the sequence is as follows:

<u>Time (sec)</u>	<u>Event</u>
0	turbine trip
0.5	neutron flux peak
2+	heat flux peak
2.0	safety valves have opened and reduced pressure

The Exxon analysis showed that the turbine trip with bypass failure was the worst overpressurization event challenging the relief valves. The relief valve capacity is sufficient to prevent the safety valves from lifting even if the bypass is neglected. As discussed previously the pressure rise due to a turbine trip without scram or relief valves can be accommodated by the safety valves.

Single failure of pressure relieving devices are considered in the analysis.

Operator action in response to a turbine trip would be to verify automatic plant responses, such as reactor trip, bus transfer and relief valve operation and then proceed with normal plant cooldown using

the isolation condensers and relief valves if the main condenser is unavailable or if the bypass fails.

2.3 Loss of Condenser Vacuum (Topic XV-3)

A loss of condenser vacuum results in a loss of the main heat sink for the reactor. A turbine trip occurs at 20" Hg, and a reactor scram at 22" Hg. This event behaves similarly to a turbine trip with bypass failure, since the bypass to the condenser is automatically blocked upon receiving the loss of vacuum. Relief valves and the isolation condenser are used to remove decay heat.

2.4 Turbine Pressure Regulator Failure (Topic XV-3)

A steam pressure regulator failure in the direction of decreasing flow is mitigated by the actions of the backup regulator. As pressure increases, the backup regulator will take over to control reactor pressure. This event induces a very mild transient on the plant.

2.5 LOSS OF NORMAL FEEDWATER FLOW (Topic XV-5)

A loss of feedwater flow could occur from pump failures, feedwater controller failures or operator error. Loss of feedwater results in a reduction of vessel inventory which causes water level to drop. The level drop is terminated by isolation of the steam system. Reactor protection is provided by trips on low and low low water levels.

The Exxon analysis for the loss of feedwater transient was submitted in reference 26. The analysis conservatively assumed full power and an initial water level one foot below normal operating level. The transient is more severe from high power conditions because the rate of reactor vessel decrease is greatest and the amount of stored heat to be dissipated is highest. Having the water level one foot below operating conditions minimizes the initial system coolant inventory. All automatic actions are assumed to occur.

The sequence of events is:

<u>TIME (Sec)</u>	<u>EVENT</u>
0	Loss of Feedwater occurs
3.5	Feedwater flow decreases to zero
4.5	Reactor Water Level reaches low level setpoint and reactor scram occurs

<u>TIME (SEC)</u>	<u>EVENT</u>
15.0	Reactor water level reaches low-low level setpoint and the following events occur <ol style="list-style-type: none"> 1. Main Steam Isolation valves (MSIVs) begin closing (10 second closing time). 2. Main recirculation pumps trip. 3. Isolation condenser return valves signaled to open 4. Core spray pumps are signaled to start
35.0	Minimum downcomer water level of 5.36 feet above the top of the active fuel is reached.

After MSIV closure, the isolation condenser system initiates system depressurization. The maximum dome pressure during the transient is 1047 psia, below the setpoint of the relief valves (1065 psia). The minimum critical power ratio does not decrease below its initial steady state value.

Beyond the first 125 seconds of the transient analyzed above, the sequence is straightforward. The limited amount of inventory makeup available from the control rod drive flow is not expected to raise downcomer level at a sufficient rate to clear the low-low level indication. Since the various safety systems have been actuated at this level, no credit is taken for operator intervention.

The system will continue to depressurize until core spray flow is introduced to the vessel at approximately 285 psig. The water level in the core at this point has been calculated to exceed the low-low-low level setpoint (4'8" above the active fuel). This level estimation is based upon fully collapsed level of the fluid mass at saturation conditions. Following initiation of core spray the level will recover and the event terminated.

The corrective functions for this event are automatic. The operator performs a monitoring function to verify the automatic actions and attempts to restore feedwater flow. The operator will manually cycle the isolation condenser operation to maintain a vessel cooldown rate of less than 100°F/hour when reactor level and pressure are under control.

The LOF had been previously analyzed without taking credit for the cooling and depressurization effects of the isolation condenser, references 8, 9, and 11. In these analyses the relief valves lifted. Calculations show that the core remains well covered after reaching equilibrium at a pressure below the relief valve set point. Hence failure of the isolation condensers, which would be the normal method of cooling, has been analyzed.

Only a complete loss of feedwater incident is analyzed, since other transients such as a trip and restart of a feedwater pump are less severe. The consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which fuel, pressure or containment are designed.

On 5/2/79, at 98 percent power, a pressure spike on reactor high pressure scram switches caused a reactor scram. A complete loss of feedwater led to decreasing reactor water level. An operator manually initiated MXIV closure and isolation condenser operation was initiated. However, because of closed recirculation line discharge valves on all five loops, reactor triple low water level was reached at about three minutes. Reactor level was recovered when a feedwater pump and a recirculation pump were restarted at about 40 minutes. This event is described in reference 27. As a result of this event administrative controls were established to insure that at least two discharge valves on recirculation lines are open to insure adequate communication between downcomer and core regions.

2.6 Feedwater System Pipe Break (Topic XV-6)

These line breaks are considered as a subset of the recirculation line breaks in section 7.1.

3.0 GROUP III EVENTS

Group III events are infrequent or limiting events with a low probability of occurrence. These events are due to steam system piping failures.

3.1 Steam Line Break Inside the Drywell (Topic XV-2)

These line breaks are considered as a subset of the recirculation line breaks in section 7.1.

3.2 Steam Line Break Outside the Drywell (Topic XV-2)

A steam line break results in a sharp increase in steam flow and system depressurization. The flow limiters in the main steam line limit the blowdown rate to 200% of rated steam flow.

The turbine admission valves are closed by the controller in response to the decreasing steam pressure. The main steam isolation valves begin to close within 0.5 sec in response to high steam flow or high pipe tunnel temperature. The maximum 10 second valve closure time is considered.

A reactor scram occurs upon 10% closure of the MSIV's, with low water level trip as a backup means of protection.

Closure of the MSIV's halts the blowdown. The analysis shows that the core remains covered, even assuming no makeup flow from feedwater or control rod drive systems. Further, the amount of fuel damage prior to isolation is small so that the radiological consequences are minimal.

This event is analyzed in the FSAR and the radiological effects were later scaled up to 1930 Mwt. Since the plant response to a break outside the drywell is not sensitive reload fuel characteristics, the analysis is considered acceptable for reloads.

3.3 Radiological Consequences of a Steam Line Break Outside Containment (Topic XV-

The radiological consequences for this event were calculated using the criteria of regulatory guide 1.5 in reference 29. The calculation used the limit for primary coolant activity concentrations and the maximum closing time for main steam isolation valves as specified in the technical specifications. The results of that calculation are listed below:

Site Boundary (414m)
(2 hours)

Whole Body	0.17 Rem
------------	----------

Thyroid	13.6 Rem
---------	----------

Low Population Distance (3,218m)
(30 days)

Whole Body	0.027 Rem
------------	-----------

Thyroid	2.18 Rem
---------	----------

Although these results are higher than those given in reference 5, the consequences are still limited to a small fraction (less than or equal to 10%) of the 10CFR part 100 exposure guidelines.

The analysis of the main steam line break accident depends on the operating thermal-hydraulic parameters of the overall reactor such as the pressure, and the overall factors affecting the consequences, such as primary coolant activity. The primary coolant activity is assumed

to be at the limiting value stated in the Technical Specifications.
Insertion of 8X8 reload fuel will not change any of these parameters
so the results of the analysis discussed above will not change.

4.0 GROUP IV EVENTS

Group IV events involve a loss of ac power. Loss of power to auxiliaries occurs with moderate frequency; a complete loss of ac power occurs infrequently.

4.1 LOSS OF AUXILIARY POWER (TOPIC XV-4)

A loss of auxiliary power could occur due to electrical power distribution malfunctions. A reactor trip would occur upon loss of AC power to the reactor protection system.

Loss of auxiliary power causes loss of condenser cooling water, trip of feedwater pumps and trip of the recirculation pumps. Turbine trip and reactor trip ensue.

The bypass is assumed to be available for 1.5 seconds, reducing the power/pressure transient so that the transient is less severe than the turbine trip with bypass failure. The bypass valves trip shut when the main condenser vacuum reaches 10 inches Hg. Reactor operating experience has shown that vacuum does not drop below the 10 inch set point until after 1.5 seconds.

The relief valves and isolation condenser would be available for decay heat removal. The diesel generators would be available to supply emergency power with a loss of offsite power. The diesels automatically start upon opening of the breakers. A control rod drive hydraulic pump, powered by the diesel, can supply 110 gpm makeup flow to the reactor. However,

the analysis shows that even without this makeup flow the core remains well covered.

This event was analyzed in reference 5. As discussed in reference 13, this event causes a less severe isolation than a turbine trip without bypass, since the bypass is assumed to function immediately after the trip.

4.2 LOSS OF ALL AC POWER (STATION BLACKOUT) (TOPIC XV-24)

This event is being considered as a generic item. No licensing position has been established; therefore this topic is not being addressed in the SEP.

5.0 GROUP V EVENTS

These events involve a decrease in reactor coolant system flow rate. A reactor coolant pump rotor seizure occurs infrequently, and a loss of flow because of a loss of power occurs with moderate frequency.

5.1 PUMP TRIP (Topic XV-7)

A loss of reactor coolant flow can result from loss of power to the pump, failure of drive motor connections, M-G set breakers or pump failure. The decreasing core flow causes a core heatup due to the flow-power mismatch.

The increased void formation inserts negative reactivity to drop power back to a level compatible with the lower flow. No reactor trips occur due to the decreased flow. If the loss of flow was due to a loss of power, a scram may occur due to loss of power to the reactor protection system. The MCPR decreases, but does not reach the limit.

A loss of auxiliary power can cause all five M-G set drive motors to stop, leading to a five pump flow coastdown.

One, two or three recirculation pumps could be lost by a failure of drive motor connections of the M-G sets to the buses. The coastdown from these events is less severe than for the 5 pump trip.

5.2 PUMP STALL (TOPIC XV-7)

The seizure of one recirculation pump is also considered. The flow

reduction results in increased core enthalpy, and a reduction in MCPR.

Reverse flow begins through the pump with the shaft seizure. The flow-power mismatch is slightly more severe than that due to a five pump trip, but the results are still acceptable since the MCPR does not drop below the limits. No reactor trip occurs. Reactor power decreases in response to the reduced recirculation flow.

The loss of flow events have been analyzed by Exxon in references 9, 11 and 14. The results show that these events leave more margin to thermal limits than other transients.

Oyster Creek is permitted by technical specifications to operate at full power with four recirculation pumps. The effect of a one pump stall from 4 pump operation was evaluated in reference 28. While the one-of-four pump stall is slightly more severe than the one-of-five pump stall, there is still considerable margin to thermal limits.

6.0 Group VI Events

Group VI events involve reactivity and power distribution anomalies associated with control rod malfunctions.

6.1 Rod Withdrawal at Power (Topic XV-13)

The inadvertent withdrawal of a control rod because of operator error or rod controller malfunction causes an increase in core power level and heat flux. Severe local peaking can also occur. Protection is afforded by a rod block from the average power range monitoring (APRM) system of the reactor protection system. The APRM system uses signals from the local power range priorities (LPRM's) to measure core power. When increasing power is detected, rod block signals are generated.

The Exxon analysis (reference 14) of a single rod withdrawal was performed with the following assumptions:

1. No xenon or samarium present
2. Peak core reactivity
3. Control rod pattern which maximizes the reactivity insertion.
4. transient rod is fully inserted, adjacent rods withdrawn

The APRM rod block terminates the rod withdrawal at a set point of 106% power. No MCPR limits are reached for this event. This event determines the steady-state MCPR operating limits since the CPR is greatest for this event. Bypassing of APRM channels and failures of LPRM detectors

are considered in this event. The combination of bypassed and/or failed detectors is limited to those permitted by the technical specifications.

6.2 Inadvertent Rod Withdrawal from Lower Power (XV-13)

An uncontrolled rod withdrawal during startup from a low power condition could result in a rapid increase in neutron flux. This increase could result in high heat generation in the fuel, and possible fuel damage.

Inadvertent continued withdrawal of a single control rod from low power conditions could result from operator error or a malfunction of the rod control system.

A rod block would be initiated either by the neutron monitoring system (NMS) or by the rod worth minimizer. The NMS, through the reactor protection system, inhibits rod withdrawal when too high a flux is detected by the intermediate range monitors. The rod worth minimizer initiates a rod block if an out-of-sequence rod is selected, or if a rod is withdrawn one notch beyond the pattern position.

The RWM system is basically provided to minimize the consequences of a rod drop accident. However, it also serves to protect against rod withdrawal errors during startup. The RWM is required for up to 10% power.

This equipment effectively prevents a continuous withdrawal which could be limiting. Analysis of this event, therefore, is considered only in the FSAR.

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The RWM system is basically provided to minimize the consequences of a rod drop accident. However, it also serves to protect against rod withdrawal errors during startup. The RWM is required for up to 10% power *or sequence verification by a second licensed operator.*

This equipment effectively prevents a continuous withdrawal which could be limiting. Analysis of this event, therefore, is considered only in the FSAR.

6.3 Rod Drop (XV-13)

A rod drop accident occurs when a rod is removed from the core at a more rapid rate than can be achieved using the drive mechanisms. A fully inserted rod is assumed to drop out after becoming disconnected from its drive. The rapid reactivity insertion cause a flux spike and rapid increase in energy deposition. Fuel damage could allow the release of fission products and have potential radiological consequences.

The adverse effects of a rod drop are limited by the rod velocity limiter and the rod worth minimizer. These devices ensure that the reactivity insertion due to the rod drop is as small as possible. The RWM is used only below 10% power.

No automatic reactor trips occur in the rapid time frame of the event. Doppler feedback is assumed to terminate the event.

The worse case occurs during hot standby conditions, with the vacuum pump operating, since this provides a direct pathway for release of radioactivity to the environment. Also, the rod worth is high for these conditions.

The rod drop event was analyzed in the FSAR using General Electric excursion analysis models. Fuel rods with enthalpies exceeding 170 cal/gm were assumed to experience eventual cladding damage. This is in agreement with the Standard Review Plant for BWR Rod Drop (15.4.9). The analysis was performed at 1600 Mwt. The radiological consequences were later scaled to a power level of 1930 Mwt.

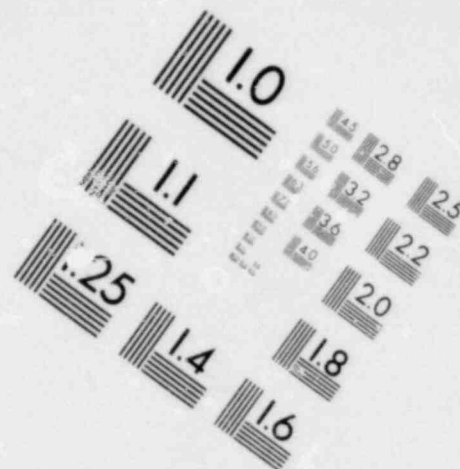
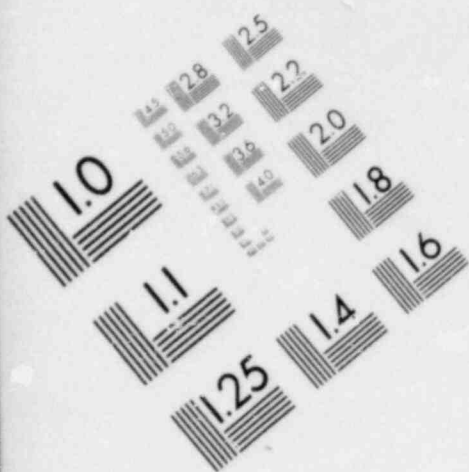
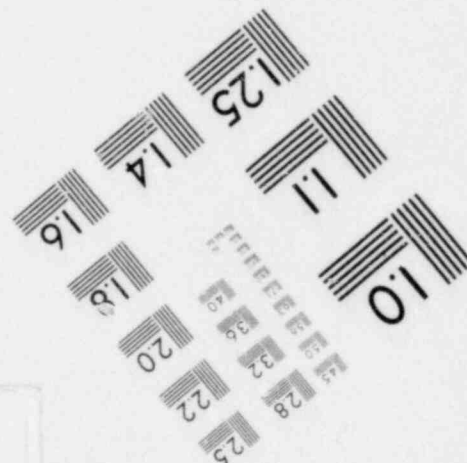
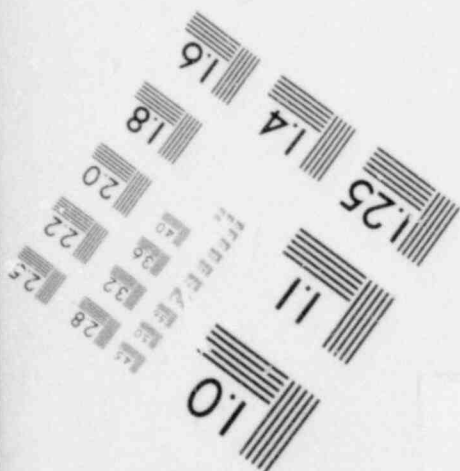
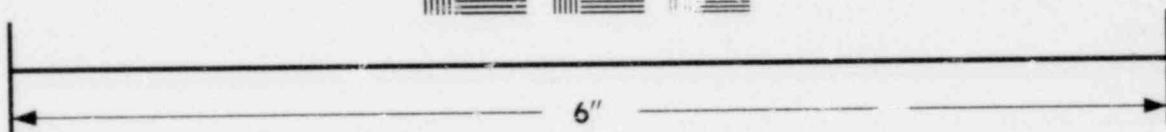
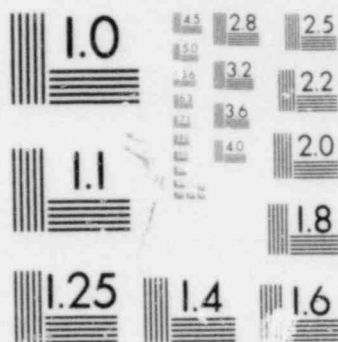


IMAGE EVALUATION
TEST TARGET (MT-3)



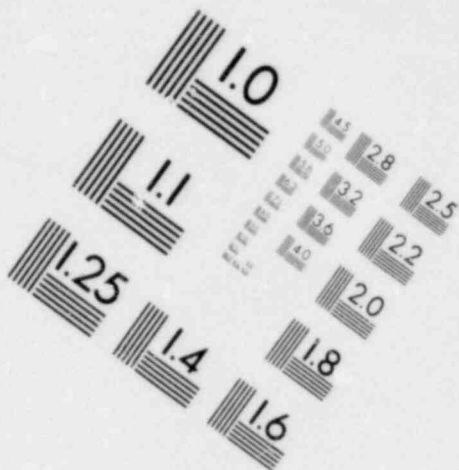
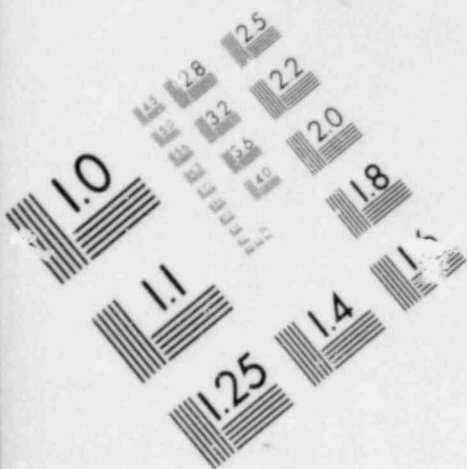
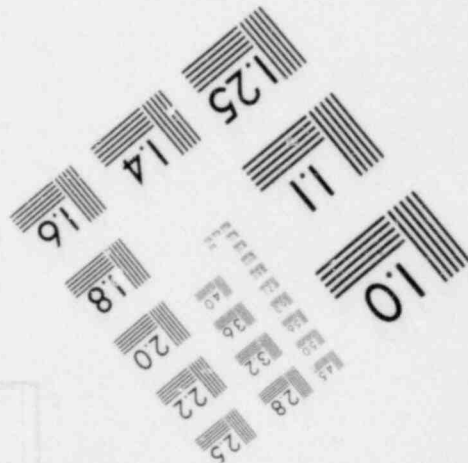
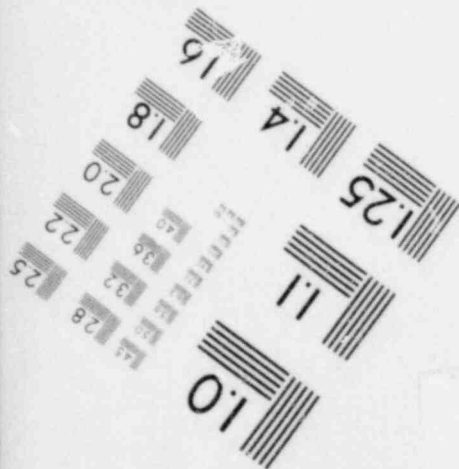
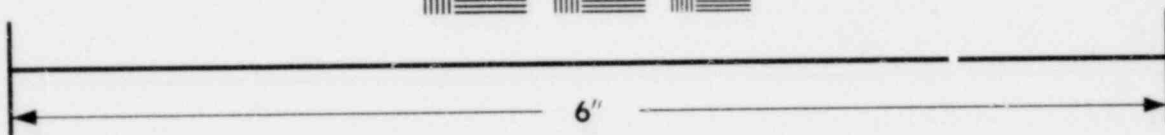
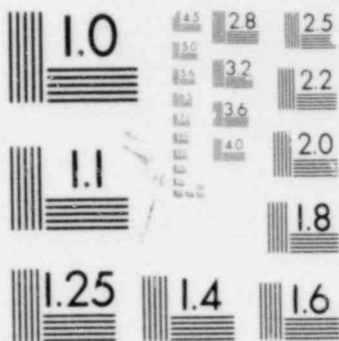


IMAGE EVALUATION
TEST TARGET (MT-3)



The acceptance criteria for this event as given in the SRP are that:

- (a) the reactivity excursion shall not result in radially averaged fuel rod enthalpy greater than 280 cal/gm at any axial location in any rod.
- (b) the maximum pressure during the transient shall be less than the design limit.

No fuel rods are predicted to have enthalpies greater than 220 cal/gm.

For the hot standby case, the increase in steam flow due to the energy release can be handled by the turbine bypass, so pressure does not approach the limit.

Thus, the criteria are satisfied for this event.

Due to the fast response to this event, no immediate operator actions can mitigate the consequences. Following the event, operator response is directed to recovering the rod; and maintaining the reactor in a safe condition.

6.4 Radiological Consequences

The consequences of a rod drop accident were calculated in reference 1 and updated to 1930 MW in reference 5.

The consequences have been evaluated and the design of the plant has been found to assure that the recovery from the accident is sufficiently rapid and effective to limit the activity releases. The evaluation of

radiological consequences has been performed using an analytical model based upon a conservative description of the plant response to the accidents. The calculated doses are presented in table 6.4-1, and are well within (taken to be less than or equal to 25%) the 10 CFR Part 100 exposure guidelines.

TABLE 6.4-1

Calculated Doses for Rod Drop Accidents

	<u>Peak Off Site Doses (rem)</u>	
	<u>2 hr.</u>	<u>Total</u>
Whole Body Dose	5.0×10^{-1}	1.0
Thyroid Dose	2.2×10^{-4}	8.3×10^{-4}

These postulated accidents result from a loss of coolant, in excess of makeup capacity, due to piping breaks in the reactor coolant pressure boundary.

7.1

Loss-of-Coolant Accident (Topic XV-19)

A loss-of-coolant accident is caused by a leak or rupture of lines containing primary coolant. A full spectrum of break sizes are considered, up to the complete double-ended rupture of a recirculation line. The loss of energy and mass from the system causes vessel depressurization. Continued loss of coolant would lead to core uncover, and would prevent heat removal from the reactor.

Protection is afforded by emergency core cooling systems, which are designed to reflood the core following a break. The reactor trips on low reactor water level or high drywell pressure.

The course of the accident depends on the break size and location, whether offsite power is available, and the assumed single failure. The analysis considers ECCS actuation with the diesel generators supplying the power.

For a small break, the feedwater system is used to maintain vessel level if offsite power is available. The isolation condenser are used to remove decay heat from the system. The alternative is the automatic depressurization system (ADS), which reduces reactor pressure so that the low pressure systems can operate.

The small break model assumes loss of offsite and auxiliary power coincident with the break. This results in coastdown of the recirculation pumps, trip of the main feedwater pumps and closure of the main steam isolation valves. The condenser bypass valves are assumed to allow steam relief for 1.71 seconds after loss of power. The closure of the main steam line isolation valves results in an immediate reactor trip. The small break spectrum analyzed is from 0.1 ft^2 to 1.0 ft^2 . The breaks were assumed to have occurred downstream of the venturi with failure of the isolation condenser valve in the broken loop.

Small break spectrum LOCA analyses using ENC's NIP-BWR-EM model have shown appreciable PCT reductions relative to previous analyses which demonstrated the maximum PCT conditions in the small break spectrum. This condition arose due to the use of the Ellion pool film boiling correlation during the blowdown transient in the earlier model. In the present model, the blowdown flow calculation results in a nucleate boiling interval during which much of the core stored energy is removed which results in considerably reduced clad temperatures during the subsequent core spray interval. Thus the highest PCT is reached for the large break.

For very small breaks ($< 0.1 \text{ ft}^2$), the CRD pumps may be used to supply cooling flow. These pumps are automatically sequenced on the emergency diesels. The analysis, however, does not take credit for this system since it is not safety grade.

In the 1.0 ft² blowdown, the intact loop emergency condenser remains on once initiated since the valve flow magnitude is less than the automatic shutoff criteria (flow \geq 275 lb/sec) 35 seconds low water level signal. The automatic depressurization systems (ADS) is not activated. Core rated spray is reached 88.6 seconds after break initiation.

For the smaller breaks (0.35 ft² and 0.1 ft²), the emergency condenser shuts off early in the transient. Because of the slower depressurization the ADS system is activated after its 120 second delay from the low-low level signal in each of these breaks. For the 0.35 ft² and 0.10 ft² small breaks wherein failure of the emergency condenser valve on the broken loop was assumed, the time of core rated spray was 186.7 and 392.3 seconds, respectively.

The assumption of the ADS valve failure rather than an emergency condenser valve failure has the effect of slowing the rate of depressurization late in the transient. For the 0.35 ft² break the time of core spray initiation was delayed 5.2 seconds and the time of core rated spray was delayed 9.3 seconds from 186.3 seconds to 197.0 seconds after break initiation.

Heatup calculations have been performed for each of the four small breaks discussed above. ENC 8 X 8 reload fuel at 7.0 GWD/MTM burn-up with octant symmetry was used in these calculations. The results indicate that the time of hot plane uncover to a large extent governs the

final PCT. The assumption of ADS failure rather than emergency condenser failure for the 0.35 ft^2 break was found to result in a PCT reduction of 16.0 degrees.

For larger breaks, depressurization through the break drops pressure below the low pressure system shutoff pressure so that core spray cooling can commence. ADS is not required for depressurization. Heat is removed from the drywell by the containment spray cooling system.

Double-ended guillotine (DEG) breaks with discharge coefficients (C_d) of 0.4, 0.6 and 1.0 were considered as well as split breaks with break areas of 1.0, 2.5, 4.0 and 6.292 ft^2 . The worst single failure was the loss of one emergency condenser. The operable emergency condenser was connected to the intact recirculation loop recirculation loop based on sensitivity calculations showing this to be the worst location.

The worst large break (limiting break) was determined to be the complete double-ended guillotine break ($C_d = 0.4$) of a recirculation pump discharge line downstream of the venturi. The worst single failure is loss of one condenser.

A steam line break analysis was performed to confirm the assumption that recirculation line breaks result in the most severe LOCA PCT's. The break area assumed was that for a full steam line pipe diameter (2.54 ft^2). A PCT of 682°F was calculated for the steam line break as opposed to a PCT of 2200°F for a recirculation line DEG with similar initial operating conditions.

Two additional assumed breaks were calculated using the approved NJP-BWR ECCS Evaluation Model, a core spray line break and a feedwater line break in reference 24 . In the core spray line break, the spray line, the spray header ring, and the spray nozzles were modeled in the RELAP4-EM calculation. The break was assumed to be an open-ended break with a Discharge Coefficient (CD) of 1.0. The small break model was used in the analysis (break area .196 ft²). The analysis showed that the reactor depressurized rapidly until the system pressure reached the pressure equal to the saturation conditions of the lower plenum. After which the system pressure remained nearly constant until the Automatic Depressurization System (ADS) valves opened, permitting the system to again depressurize. Core rated spray, 465 lbm/sec, was calculated to occur at 527 seconds. During the course of the blowdown transient the core remained covered with water. The peak clad temperature calculated for the core spray line break is 1335°F.

For the feedwater line break, the feedwater line, the feedwater sparger ring, and the sparger orifices, were modeled in the RELAP4-EM calculation. The break was assumed to be a guillotine break with a discharge coefficient of 1.0. The guillotine break permitted flow out two of the four lines feeding the feedwater sparger. The feedwater sparger is located in the downcomer of the reactor. The break flow area was (.998 ft²) hence, the small break model was used to analyze the transient. This transient behaved similarly to that for the core spray line break. The reactor depressurized rapidly to the lower plenum saturation pressure. The calculated pressure was then nearly constant until the ADS came on. The transient calculation was terminated at this time since the core was still covered with water and the transient fluid conditions are nearly identical to the core spray line break. The subsequent depressurization rate is controlled by the ADS flows.

therefore, the core was not expected to uncover as shown earlier by the core spray line analysis, and a peak clad temperature similar to the 1335°F calculated for the core spray line break will result from this transient.

The core spray line break and feedwater line break are clearly not limiting for Oyster Creek.

The consequences for this event were evaluated in the FSAR and updated for operation at 1930 MW in reference 5. The consequences were re-evaluated using the assumptions of safety guide 1.3 in reference 29. This calculation was also done for full power operation at 1930 MW.

The LOCA evaluation using the assumptions from reference 29 results in the off-site doses tabulated below:

Site Boundary (2 hours)	Thyroid	145.0 Rem
	Whole Body	9.5 Rem
Low Population Distance (30 days)	Thyroid	117.0 Rem
	Whole Body	4.5 Rem

These doses are well below the guidelines of 10 CFR 100 even when the conservative assumptions of the Safety Guide are used.

8.0 GROUP VIII EVENTS

These events are infrequent occurrences that lead to possible radioactive releases from fuel damaged by dropping a heavy load or through fuel handling.

8.1 Radiological Consequences of a Fuel-handling Accident (TOPIC XV-70)

The refueling accident analyzed in the operating license application state occurs when a fuel bundle is accidentally dropped onto the top of the core during fuel handling operations. This is not the accident required by Regulatory Guide 1.25, but the resultant fuel failure and radioactive fission product release is at least as extensive. A comparison of the safety analysis for this accident and Regulatory Guide 1.25 is provided reference 32.

There are a number of area of non-compliance with the Regulatory Guide, but the methods and numerical values used are well documented and justified in the General Electric Topical Report, APED-5756; "Analytical Methods for Evaluating the Radiological Aspects of the General Electric Boiling Water Reactor".

The potential doses to off-site persons corresponding to the fission product release for the fuel handling accident are well below the 10CFR 100 guideline limits. Additional evaluation, with much higher fission product release levels, has been performed (see the response to Question I-13 in reference 33) and the resulting dosages were still below the limits of 10CFR 100.

The analysis of the refueling accident involves the mechanical damage caused by a fuel bundle falling back onto the top of the core while it is being removed, and the subsequent release of radioactive fission products. The severity of the consequences depends on the fission product inventory in the fuel and various factors affecting the amount and kind of releases to the atmosphere. There will be no change in the total quantity of fission products due to the change in the reload fuel design since it will be operating at no higher power level, but there will be slight changes in the relative amount of different constituents because of the presence of gadolinium in the Type VB bundles instead of boron in the curtains. The effects of these differences will be small and undetectable when the various reduction factors are applied to determine off-site doses. The most significant difference introduced by the Type VB fuel design is a substantial reduction in the quantity of gaseous fission products in the rod gas spaces due to the lower fuel operating temperature. Given the sample bundle and exposure history, the fission gas inventory potentially available for release in the 8X8 bundle gas spaces will be approximately 40 percent of the inventory in the 7X7 bundle gas spaces. This being the case, the previous analysis of this accident conservatively applies to the 8X8 fuel.

9.0 GROUP IX EVENTS

These moderate frequency events are essentially small break LOCA's. However, due to the postulated frequency of occurrence, they should satisfy the fuel clad integrity criteria.

9.1 INADVERTENT OPENING OF A BWR SAFETY/RELIEF VALVE (TOPIC XV-15)

The inadvertent opening of a safety or relief valve results in a reactor coolant inventory decrease and a decrease in reactor coolant system pressure. Neutron flux decreases due to additional void formation.

The pressure regulator senses the pressure decrease and partially closes the turbine control valves. No trip occurs, and conditions stabilize at a power level near the initial power. The feedwater system is used to makeup the continuing loss of inventory.

Opening of the turbine bypass valve is less severe since the capacity is less and the pressure regulator can respond faster as the turbine pressure drops.

If the pressure regulator fails to respond, the increased steam flow would cause a decrease in steam pressure and close the MSIVs, such as discussed in section 1.3.

If a relief bypass valve sticks open, the continued depressurization could require a reactor scram. If possible, the operator should reset the pressure regulator downward to that as much steam as possible is diverted through the steam line to the main condenser (MSIV open and offsite power), rather than through the relief valve to the suppression pool provided the $\leq 100^\circ \text{ F/hr.}$ cooldown rate can be maintained.

The licensee performed analyses of the opening of a relief valve in Section B.IV-2.3 and opening of a bypass valve in Section B.XI-3.4 of Reference 5. The safety valve opening event would be very similar to a relief valve opening since their capacities are about the same.

10.0 GROUP X EVENTS

Group X events have a moderate frequency of occurring and lead to an increase in primary coolant inventory. These events could cause an increase in pressure and power.

10.1 Inadvertent Operation of ECCS Increasing Core Inventory (Topic XV-14)

The high pressure emergency cooling systems for Oyster Creek are isolation condensers, which rely on natural circulation, and the feedwater system. The low pressure core spray system cannot deliver flow to the vessel until pressure drops to approximately 350 psig. Thus, this event is not analyzed. The increase in feedwater flow transient, which increases coolant inventory, is considered in section 1.2.

11.0 GROUP XI EVENTS

The Group XI events involve misloading of fuel assemblies in the core. Undetected errors could lead to power distribution anomalies and exceeding fuel limits.

11.1 FUEL MISLOADING (TOPIC XV-11)

The inadvertent loading of a fuel assembly in an improper position could increase the fuel assembly power because of the difference in the water gap and the exposure/enrichment mismatch. There are two types of inadvertent loadings considered in this analysis; mislocated and misoriented. A misoriented fuel assembly is incorrectly rotated 90° or 180° within the fuel cell. The mislocated fuel assembly is loaded into an incorrect location within the core. Both fuel loading errors result from multiple operator errors.

To insure a fuel assembly is properly positioned in the reactor core, the following visual checks are performed:

- 1) The channel fastener is located at one corner of each fuel assembly adjacent to the center of the control rod.
- 2) The identification boss on the fuel assembly bail points towards the adjacent control rod.
- 3) The channel spacing buttons are adjacent to the control rod travel area.
- 4) The fuel assembly serial number, located on the bail, are all readable from the direction of the center of the fuel cell.
- 5) There is cell-to-cell replication.

The independent verification identifies each fuel assembly unique serial number and compares its location the loading pattern.

A loading error could also be detectable by an inventory of the discharged fuel assemblies.

The effects of a mislocated fuel assembly at the Oyster Creek Nuclear Power Station has never been analyzed. However, General Electric has performed numerous fuel assembly misloading analyses for generic and plant cycle specific BWRs. The results of these analyses have never resulted in the postulated violation of the safety limit critical power ratio. As a result of these analyses, General Electric informed the NRC (Ref. 30) that they will discontinue performing plant-cycle specific mislocated bundle analysis.

The misoriented fuel assembly accident has been analyzed and documented for the Oyster Creek Nuclear Power Station on numerous occasions. The initial analysis appeared in the FSAR, Ref. 1, for the original 7X7 fuel. This worst case analysis concluded that a fuel bundle misoriented 180° would result in an increased bundle power of 29%. This increase in bundle power does not result in exceeding MCHF limits.

In Ref. 31, an updated analysis concluded that a misoriented bundle would result in an increase bundle power of 21% which remained within MCHF limits.

In Ref. 10, it was shown that a misoriented Exxon 8X8 VB fuel assembly would experience a 17% power increase in the worst case analysis and is conservatively bounded by previous analysis.

REFERENCES

1. Facility Description and Safety Analysis Report, Final, Amendment 3 for the Oyster Creek Nuclear Power Plant Unit 1, January 1967.
2. Amendment 14 to the Preliminary Safety Analysis Report, December 19, 1967.
3. Amendment 55 to PSAR, May 5, 1970.
4. Amendment 63 to PSAR, September 17, 1970.
5. Amendment 65, December 1970.
6. XN-74-6, Revision 1, September 1974.
7. T. S. Change Nos. 33, 35, 36, January - March 1975.
8. XN-74-38, Revision 1, September 1974 (Note: 8, 9 and 10 transmitted as part of Amendment 76, January 31, 1975).
9. XN-74-41, Revision 2, January 1975.
10. Amendment 76 (Supp. 1), March 25, 1975
11. XN-74-43, Revision 2, January 1975.
12. Amendment 76 (Supp. 3), May 1975.
13. Answers to NRC Questions, May 8, 1975.
14. Amendment 76 (Supp. 4), October 20, 1975.
15. Additional Information Re New Proposed MCPR Limits, April 15, 1976.

16. License Amendment 9, Change No. 25 to Tech. Specs., May 24, 1975.
17. LOCA Analysis Model Documentation, March 25, 1975.
18. License Amendment 8, May 24, 1979
19. Amendment 15 to POL, February 24, 1976.
20. LOCA Analysis Re-evaluation, Change Request #40, December 23, 1975
21. Amendment 16, July 26, 1976.
22. XN-75-55 (A) and Supplements 1 and 2, dated August, 1976
23. Letter to ENC, March 28, 1977
24. XN-NF-77-55 Revision 1, March 1978
25. Letter to NRC, answers to questions 2/5/76
26. Letter to NRC, transient of May 2, 1979, 5/12/79
27. Letter to NRC, Transient of May 2, 1979, 5/12/79
28. Letter to Mr. S. Norwicki (NRC), from G. R. Bond (GPUSC),
May 17, 1979.
29. Amendment 68, March 6, 1972
30. Letter to NRC (T. A. Ippolito) from GE (R. Engel), Change in General
Electric Methods for Analysis of Mislocated Bundle Accident, dated
November 14, 1980.

31. Oyster Creek Facility Change Request No. 4, dated January 18, 1973.
32. Amendment 68 (Supplement 6), November 1, 1973
33. Amendment 11, June 21, 1967

TABLE 1REACTOR PROTECTION SYSTEM (RPS) SETPOINTS

<u>PARAMETER</u>	<u>SETPOINT</u>
High Neutron Flux	115.7%
High Reactor Pressure	1060 psig
High Containment Pressure	≤ 2 psig
Low Reactor Water Level	11'5" above top of active fuel
Low Condenser Vacuum	≥ 23 " Hg
Main Steam Line High Radiation	10 times background
Scram Discharge Volume High Level	37 gallons
Loss of AC Power to RPS	--
Closure of MSIV's	10% closure
Turbine Trip	10% stop valve closure
Load Rejection	Loss of oil pressure from turbine acceleration relay
Manual	--
Rod Block	106%
Scram delays:	0.2 sec Scram Mechanism 0.3-0.9 Isolation Valve Closure (depending on closure time)

TABLE 2ENGINEERED SAFETY FEATURES

<u>Parameter</u>	<u>Setpoint</u>	<u>Systems Started</u>
Low low reactor water level	7'2" above top of active fuel	Core Spray Containment Spray Automatic Depressurization System Isolation Condenser (if signal persists for 3 seconds)
High Containment Pressure	2 psig	Core Spray Containment Spray in Conjunction with Low Low Reactor Water Level Automatic Depressurization System in conjunction with low-low-low Reactor Water level
Reactor High Pressure	1060 psig	Isolation Condenser (if signal persists for 3 seconds)
Low Reactor Pressure Permissive	≥ 285 psig	Core Spray

TABLE 3

CONTAINMENT ISOLATION

<u>Parameter</u>	<u>Setpoint</u>	<u>Systems Isolated</u>
High Steam Flow	≤ 20 psig ΔP	
High Condensate Return Flow	$\leq 27''$ ΔP	Isolation Condensers
High Steam Line Flow	120%	Main Steam Line Drain Main Steam
High drywell pressure	≤ 2 psig	Reactor Building Ventilation Sumps, Vent Purges Traversing In-core Probes
Reactor low-low water level	7'2" above top of active fuel	Main Steam Main Steam Line Drain Cleanup System Cleanup Auxiliat Pump System Shutdown System Reactor Buidling Ventilation
Steam Line High Radiation	10 times background	Main Steam Main Steam Line Drain

TABLE 3 (continued)

CONTAINMENT ISOLATION

<u>Parameter</u>	<u>Setpoint</u>	<u>Systems Isolated</u>
High Temperature in Steam Tunnel	50°F above ambient	Main Steam Main Steam Line Drain
High Reactor Pressure	> 120 psig	Shutdown Systems
High Reactor Building Radiation	≤ 17 mr/hr	Reactor Building Ventilation
High Radiation Reactor Building Operating Floor	≤ 100 mr/hr	" " "
Low Steam Pressure	> 825 PSIG	Main Steam

TABLE 4

ANALYSIS ASSUMPTIONS

<u>Event</u>	<u>Assumptions</u>
Decrease in Feedwater Temperature	135°F feedwater temperature, 100% power
Increase in feedwater flow	53% power, 42% flow, 110% feedwater flow
Increase in steam flow	110% of rated steam flow 96.4% power, 49.7% flow and hot standby
Startup of an inactive loop	100% power, 100% flow, 100°F water in isolated pump
Flow controller malfunction-increasing flow	53% power, 42% flow, 10%/sec change
Main steam line isolation valve closure	100% power, 3 sec. closure time, no credit for isolation condenser
Loss of load	100% power
Turbine trip	100% power failure of bypass failure of relief valves failure of isolation condenser (scram for safety valve sizing transient initiated by valve position switched at 90% open)
Loss of condenser vacuum	100% power
Turbine pressure regulator failure	100% power
Loss of feedwater flow	complete instantaneous loss of feed failure of isolation condenser, 100% power; Initial water level just above reactor alarm set point and about one foot below normal operating level.
Steam line break inside drywell	see LOCA
Steam line break outside drywell	100% power 3 sec. isolation valve closure time
Rod Withdrawal	No xenon or samrium peak core reactivity maximum rod worth transient rod initially fully inserted, surrounding rods withdrawn

Table 4 (Continued)

<u>Event</u>	<u>Assumptions</u>
Rod drop	Hot standby maximum rod worth
Loss-of-coolant accident	102% power, worst single failure of ECCS, loss of offsite power
Inadvertent opening of a relief valve	100% power
Inadvertent operation of ECCS increasing core inventory	see increase in feedwater flow
Loss of auxiliary power	100% power
Loss of forced flow	100% power

March 24, 1981

SEISMOLOGY AND GEOLOGY
OYSTER CREEK NUCLEAR GENERATING STATION

I. Introduction

The purpose of this safety evaluation report is to evaluate the adequacy of the original seismic design basis for the Oyster Creek Nuclear Generating Station as compared to current U.S. NRC regulatory criteria. Specifically, this evaluation includes an assessment of the following seismic topics of the Systematic Evaluation Program (SEP) for Oyster Creek:

- II-4A Tectonic Province
- II-4B Proximity of Capable Tectonic Structures in Plant Vicinity
- II-4C Historical Seismicity Within 200 Miles of Plant

This topic assessment includes a summary of the original Oyster Creek seismic design basis, results of more recent investigations by NRC and JCP&L consultants, and a comparison of the original seismic design basis with seismic design bases developed by methods currently used in the licensing of nuclear plants and prescribed in Appendix A of 10 CFR 100.

The evaluation presented herein is based on the regulatory criteria and guidelines presented in the following:

- Appendix A, "Seismic and Geologic Siting Criteria For Nuclear Power Plants," Code of Federal Regulations, Title 10, Part 100;

- U.S. NRC Regulatory Guide 1.60, "Design Response Spectra For Seismic Design of Nuclear Power Plants";
- U.S. NRC Standard Review Plan, Section 2.5.1, "Basic Geologic and Seismic Information";
- U.S. NRC Standard Review Plan, Section 2.5.2, "Vibratory Ground Motion";
- U.S. NRC Standard Review Plan, Section 3.7.1, "Seismic Input".

II. Background

The Oyster Creek seismic design basis was developed by Dr. G. W. Housner and is summarized in the Oyster Creek FDSAR, Reference 1, and in Appendix A of Reference 2, which was transmitted to the NRC (Mr. D. L. Ziemann) by JCP&L letter, dated July 9, 1979. The seismic design basis recommended by Dr. Housner considered the historical seismicity of the eastern United States through the period 1863 to 1956. The spectral intensity was set at 0.94 for the operating basis earthquake. The spectral shape was determined from strong motion records of several California earthquakes and resulted in the acceleration response spectrum shown in Figure 1 for the operating basis earthquake. The safe shutdown earthquake (SSE) was taken as two times the OBE and has a zero-period acceleration (ZPA) of 0.22 g.

Subsequent evaluations were made in support of the design of the Forked River Nuclear Generating Station (3,000 ft. to the west of Oyster Creek) and for the design of a new Radwaste building for Oyster Creek. The Forked River seismological evaluation is presented in Section 2.7 of Reference 3 and concludes that, based on seismological and historical seismicity in the eastern United States, the maximum probable earthquake for the Forked River/Oyster Creek sites does not exceed Modified Mercalli (MM) Intensity VII and the Oyster Creek ZPA of 0.22 g. The more recent Oyster Creek study by Woodward-Moorhouse & Associates is given in Reference 4 and concluded independently that the maximum historic earthquakes in the general region of Oyster Creek rank at the lower end of the broad definition of MM Intensity VII and would be characterized by a magnitude of less than 5.0. No attempt was made in this study to determine the peak ground acceleration corresponding to an Intensity VII, magnitude 5.0 earthquake, although the Woodward-Moorhouse report indicates that, based on more recent (1975) correlations, the 0.22 g peak ground acceleration selected by Housner as the Oyster Creek seismic design basis would be considered conservative.

In summary, the original Oyster Creek seismic design basis for the SSE is reported in Reference 1 and consists of a Housner response spectrum anchored at 0.22 g ZPA. Studies performed in the early 1970s for Forked River and for the

Radwaste building expansion independently determined that the maximum historical seismic event applicable to the Oyster Creek site is an MM Intensity of VII and confirmed that the peak ground acceleration of 0.22 g selected by Housner is conservative.

III. Description of Oyster Creek Site

The Oyster Creek site is located on the eastern shore of New Jersey in the Atlantic coastal plain tectonic province. The sub-surface geology at the Oyster Creek site has been extensively studied beginning with pre-construction investigations by Burns & Roe and others, followed in the early 1970s by evaluations for the Forked River site, and in 1975 in support of the Oyster Creek Radwaste building addition. These investigations indicate that the site is underlain by a series of unconsolidated and dense sediments lying on bedrock. The total thickness of this overburden is approximately 3,000 ft. The age of the sediments range from Quaternary through Tertiary and Cretaceous. The age of the bedrock is believed to be either early Paleozoic or Pre-Cambrian. The specific formations encountered during drilling at the site, together with relevant soil data, are shown in Figure 2, which is reproduced from Reference 5. In general, the soil underlying the site consists of clay strata and dense, fine to medium, and coarse sands.

The water table at the Oyster Creek site is at a depth of approximately 10 ft. The foundation of the reactor vessel

building is at a depth of approximately 47 ft. The ground surface at the plant site is at an elevation of about 23 ft.

IV. Evaluation

Since the initiation of the Systematic Evaluation Program in early 1978, the seismicity and regional tectonics of the eastern United States in general, and the Oyster Creek site in particular, have been reexamined in considerable detail. The principal investigations undertaken during this period are described below:

A. Development of Tectonics

The tectonics and seismicity of those portions of the eastern United States pertinent to sites of SEP plants were evaluated by Weston Geophysical for the SEP Owners Group. The results of this study are reported in Reference 6. This study, performed in accordance with Appendix A to 10 CFR, Part 100, confirmed that the controlling maximum design earthquake for the Oyster Creek site is an MM Intensity VII, based on the maximum historical earthquakes which have occurred in the Atlantic coastal plain tectonic province. There are no known capable faults or tectonic structures near the Oyster Creek site. The maximum earthquake magnitude range determined for the Oyster Creek site was 4.5 to 5.2, which is consistent with the results of the earlier studies discussed above.

B. Development of Site Specific Spectra

Additional evaluations were performed in 1979 and 1980 by URS/John A. Blume & Associates of the tectonics and seismicity of the Oyster Creek site. The primary purpose of these evaluations, presented in Reference 5, was to develop site specific response spectra appropriate to the Oyster Creek region and site characteristics. The main results of this study are as follows:

1. The Oyster Creek SSE should be based on an MM Intensity VII event, located in the immediate vicinity of the Oyster Creek site. This postulated earthquake is not associated with any known active fault.
2. The peak ground acceleration for the SSE is estimated based on a number of empirical intensity-acceleration correlations, including those developed by Gutenberg and Richter (1942), Trifunac and Brady (1975), and Murphy and O'Brien (1977). Based on extensive reviews of these correlations, it was concluded that the more recent Murphy and O'Brien correlation is the appropriate correlation to use for selection of the peak ground acceleration for Oyster Creek, primarily because it rests on a much larger data base and properly reflects the statistical distribution of the available acceleration data. The peak horizontal ground acceleration for the Oyster Creek SSE is calculated, using the Murphy-O'Brien correlation, to be 0.17 g, based on the 75th percentile of Intensity VII accelerations. This 75th percentile for Intensity VII is close to the mean acceleration of 0.18 g, which is calculated for an Intensity VIII event. As a result, the margin of safety associated with the peak ground acceleration for the Oyster Creek SSE of 0.17 g corresponds to almost one unit of earthquake intensity.
3. Site specific response spectra were developed based on statistical analysis of 36 strong motion records selected from sites comparable to the Oyster Creek site. The accelerograms selected were recorded at

deep sedimentary sites, had earthquake magnitudes in the range of 5 to 6, and epicentral distances comparable to those expected in the Oyster Creek province. The resulting spectra are shown in Figure 3. Vertical accelerations are taken to be two-thirds of the horizontal spectral accelerations.

4. A probabilistic assessment indicates that the mean recurrence interval for the SSE event described above is 10^4 years.

C. U.S. NRC Site Specific Spectra Program

Over the past 2 1/2 years, the U.S. NRC, through its consultants, Tera Corporation and the Lawrence Livermore Laboratory, undertook to develop recommended site specific spectra for all eastern SEP sites. The results of the Tera-Lawrence Livermore study are based on a uniform hazard analysis and are presented in Reference 7. The resulting site specific spectra are considered by the NRC to more accurately reflect true variations in real seismic hazards than those developed utilizing the deterministic approach of Appendix A of 10 CFR 100. The site specific spectra, recommended by the NRC for the Oyster Creek SSE, are also shown in Figure 3, together with the original design basis Housner spectra. These spectra are replotted for comparison on a log-log-scale in Figure 4.

D. U.S. NRC Deterministic Spectra

In addition to the uniform hazard analyses discussed above, the U.S. NRC presented in Reference 7 the results of a deterministic evaluation of the Oyster Creek site

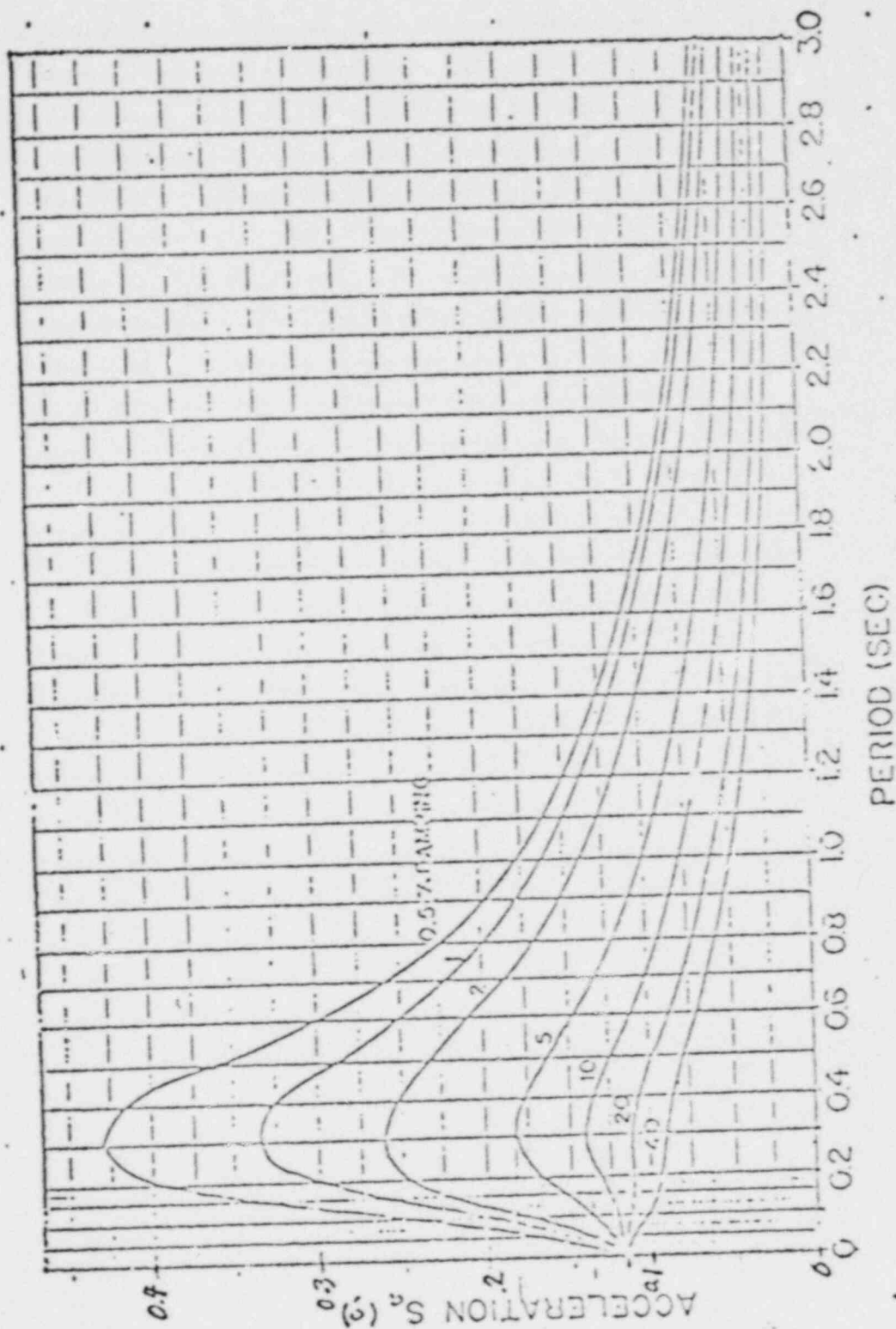
specific spectra in accordance with Appendix A to 10CFR, Part 100. This evaluation concludes that the appropriate SSE for Oyster Creek, based on the Appendix A approach, would be a Reg. Guide 1.60 spectrum anchored at 0.13g PGA. This deterministic spectra is shown in Figure 3 as the NRC Reg. Guide 1.60 spectrum.

V. Conclusions

Extensive reevaluations of the tectonics and seismicity of the Oyster Creek site conducted over the past 2-1/2 years indicate that the original seismic design basis for the Oyster Creek plant is conservative compared to site specific spectra developed by NRC and JCP&L consultants using currently available data and state-of-the-art technology. For most of the period range of interest, the original Housner spectra exceed those recommended by the NRC (Lawrence Livermore-Tera) and JCP&L (URS/Blume). In addition, the original Housner spectra for Oyster Creek are very close to the spectra obtained by the NRC using the deterministic approach of Appendix A to 10CFR, Part 100. Accordingly, it is concluded that the original Oyster Creek seismic design basis given in Reference 1 is comparable to that which would be selected today based on either uniform hazard or Appendix A methodology, and is therefore acceptable.

VI. References

1. Oyster Creek Nuclear Generating Station, "Facility Design and Safety Analysis Report," submitted to NRC in support of license application.
2. MPR Report, "Summary of Seismic Design Information, Oyster Creek Nuclear Generating Station," dated July 9, 1979.
3. Forked River Nuclear Generating Station, "Preliminary Safety Analysis Report," revised March 19, 1971.
4. "Geotechnical Study, Proposed Radwaste and Off-Gas Buildings, Oyster Creek Nuclear Generating Station," Woodward-Moorhouse & Associates Report 73C807B, dated February 4, 1975.
5. URS/J. A. Blume Report, "Site Specific Response Spectra for the Oyster Creek Nuclear Power Plant," dated December, 1980.
6. Weston Geophysical Corporation Report, "Eastern United States Tectonic Structures and Provinces Significant to the Selection of a Safe Shutdown Earthquake," dated August, 1979.
7. U.S. Nuclear Regulatory Commission Memorandum from R. E. Jackson (Chief, Geosciences Branch) to D. Crutchfield (Acting Chief, SEP Branch), dated June 23, 1980.



EARTHQUAKE ACCELERATION RESPONSE SPECTRUM

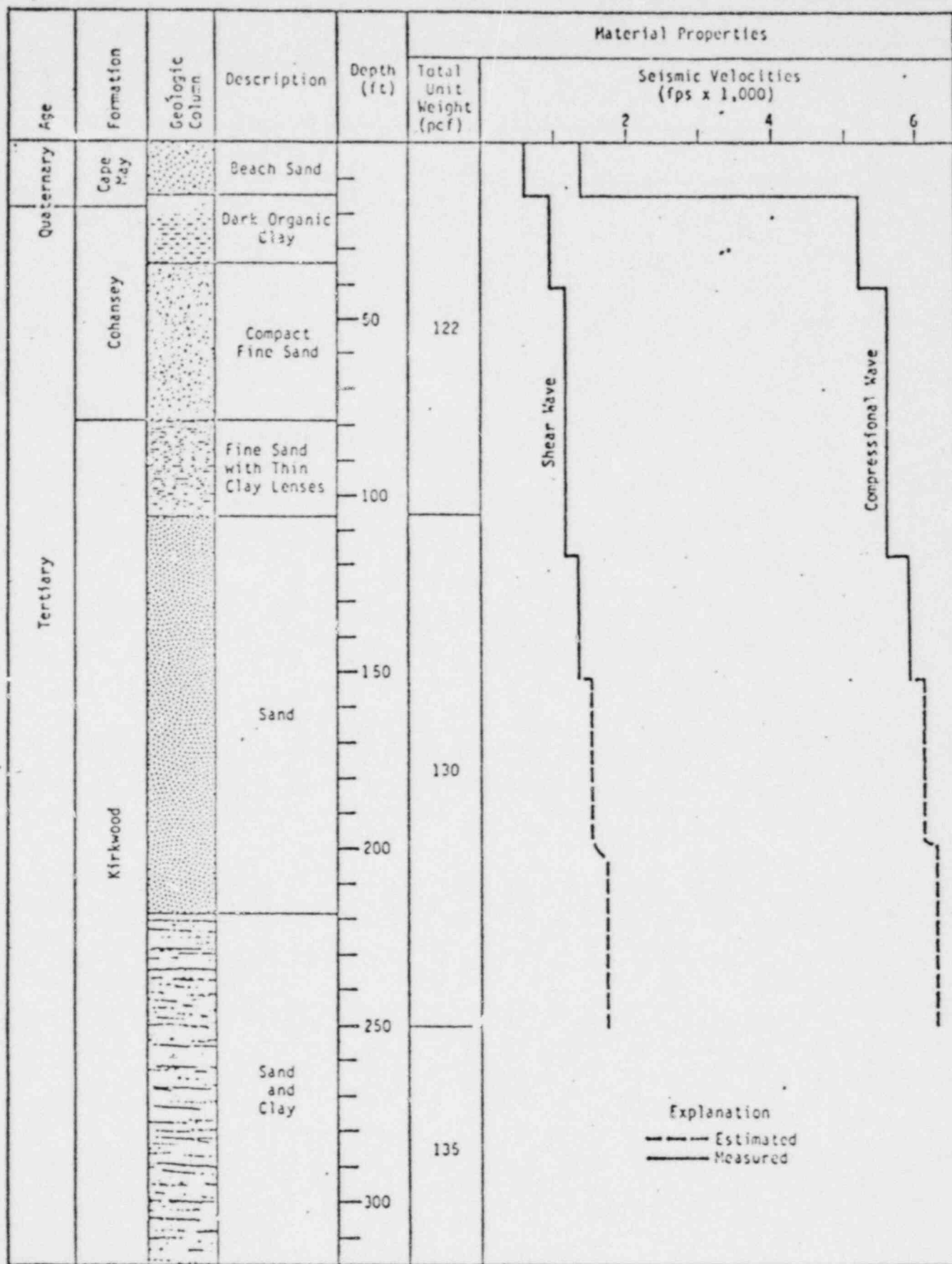
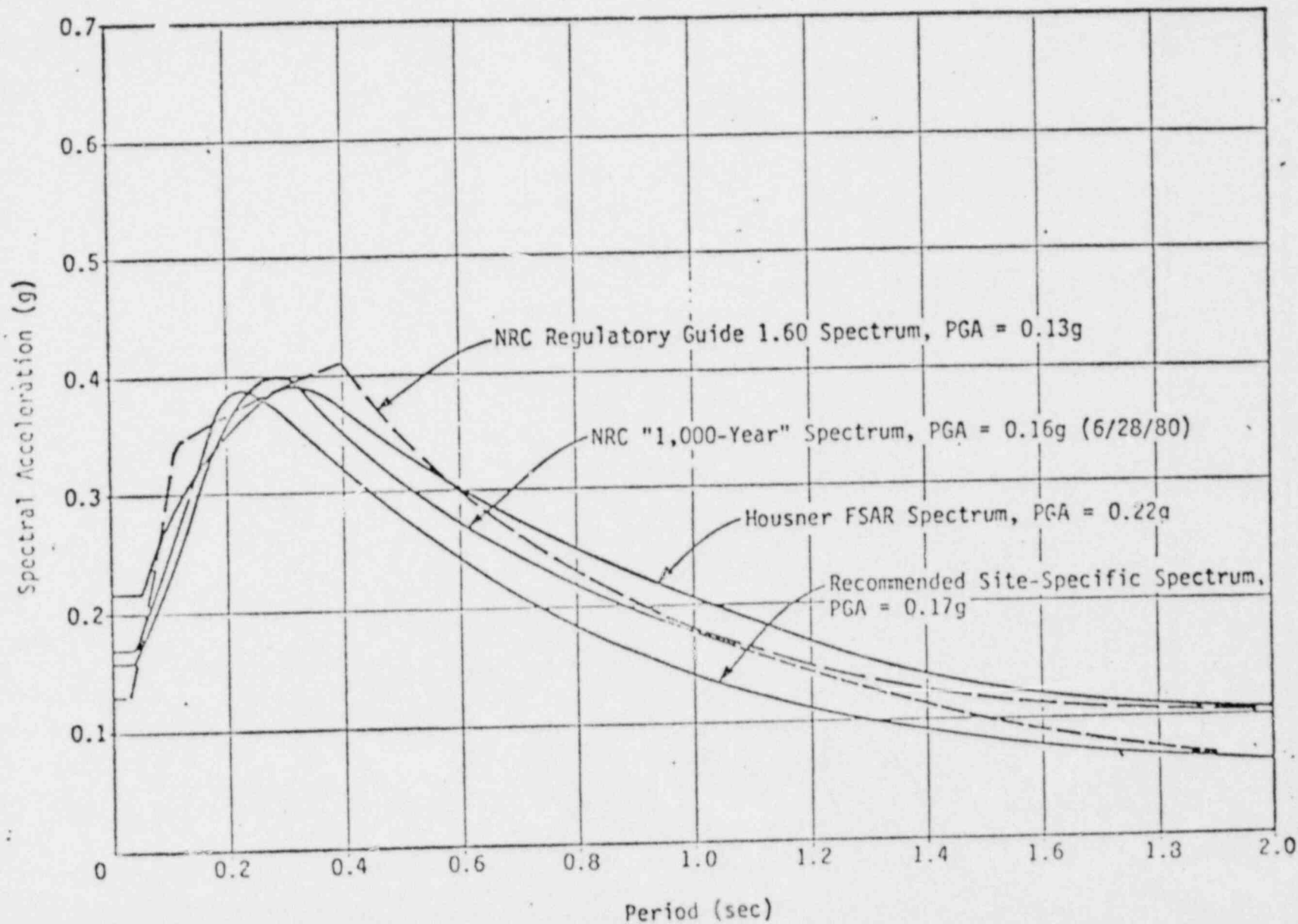


FIGURE 2 - OYSTER CREEK NUCLEAR POWER PLANT SITE: GENERALIZED GEOLOGIC COLUMN AND SOIL PROPERTIES TO A DEPTH OF 250 FEET



Note: PGA = peak ground acceleration.

FIGURE 3 - OYSTER CREEK NUCLEAR POWER PLANT SITE: COMPARISON OF 5%-DAMPED HORIZONTAL RESPONSE SPECTRA

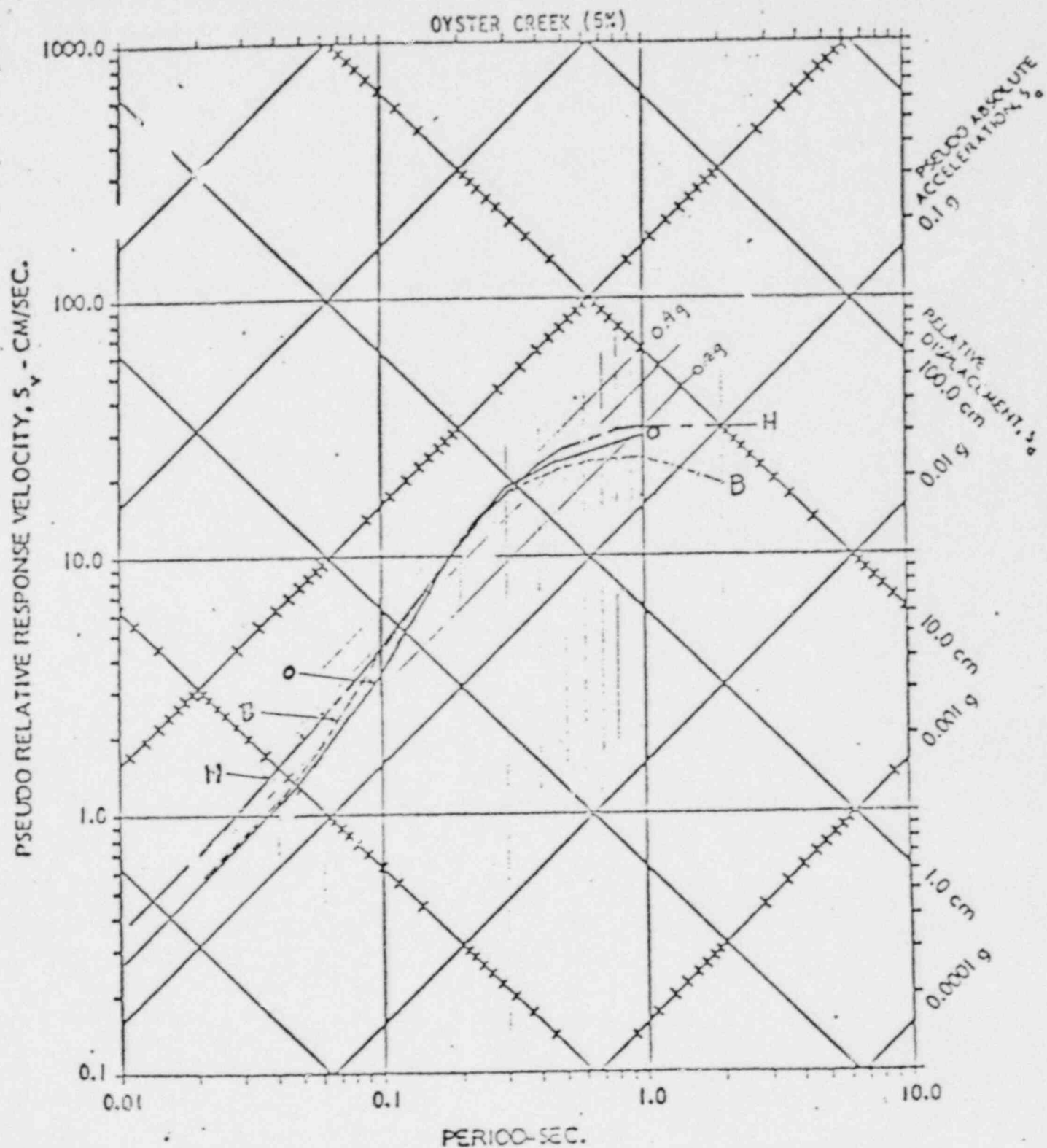


FIGURE 4

LEGEND

H = Original Housner 0.22 g Spectra

O = NRC Recommended Spectra

B = Blume Recommended Spectra

TOPIC II-3.A, II-3.B, AND II-3.B1

HYDROLOGIC DESCRIPTION, FLOODING POTENTIAL AND ABILITY TO COPE WITH THE DESIGN BASIS FLOOD

I. INTRODUCTION

This topic encompasses both the surface and groundwater and their interface with Oyster Creek Nuclear Power Plant safety related buildings and systems. It provides a brief description of the hydrologic features of the site and surrounding area. It also reviews the potential for flooding at the plant and the ability of the plant to reach a state of cold shutdown during the maximum probable flood according to NRC criteria. Required flood emergency procedures are also outlined. In addition to an external Design Basis Flood (off site source), local streams and rivers are investigated to determine their flooding potential for the site.

The review is based on docketed information for the Oyster Creek Plant and the Forked River Nuclear Station, as well as on design drawings, and reports on file. Any deviations from current licensing requirements will be identified and the significance of the deviations will be explained. Please note that all elevations referenced in this topic refer to a mean sea level datum.

II. BASIS FOR EVALUATION

The current criteria pertinent to this topic are the following:

Standard Review Plans: 2.4.1, 2.4.2, 2.4.3, 2.4.5, 2.4.10,
2.4.11, 2.4.13, 2.4.14, 3.4.1,
3.4.2

NRC Regulating Guides 1.59, 1.102, 1.127, 1.135

American National Standard Institute N170-1976

III. BACKGROUND

The Oyster Creek Nuclear Generating Station was constructed before 1970. The design basis high water level for the plant was established from a March 1962 storm, considered to have been the worst ever to strike New Jersey. Flood marks from this storm showed a high tide elevation of 4.5 ft. The intake structure was designed for this storm with an operating deck at elevation 6.0 ft.

Regulatory Guides 1.59 and 1.102 have been specifically identified by the NRC's Regulatory Requirements Review Committee as needing consideration for backfit on operating reactors. These guides as well as the other documents listed are used in determining whether the Oyster Creek Plant complies with current criteria; or makes use of some equivalent alternatives which could be considered.

The hydrologic characteristics of the Oyster Creek site have not changed since construction. Therefore, this review will include only a brief hydrological description of the site. Standard Review Plant 2.4.13 requires extensive groundwater surveys and analyses of the entire region. This information has not been compiled and is, therefore, unavailable.

IV. HYDROLOGIC DESCRIPTION

The Oyster Creek site is located on the Atlantic Coast of New Jersey as shown in Figure I. The site, about 800 acres, is partly in Lacey and partly in Ocean townships of Ocean County, New Jersey, about two miles inland from the shore of Barnegat Bay and about seven miles west-north-west of Barnegat Light. The site is approximately nine miles south of Toms River, New Jersey, thirty-five miles north of Atlantic City, New Jersey, forty-five miles east of Philadelphia, Pennsylvania, and sixty miles south of Newark, New Jersey.

The site is bounded on the east by the right-of-way of the Central Railroad of New Jersey and New Jersey Route 9; on the west by the Garden State Parkway; on the north by the South Branch of Forked River and on the south by Oyster Creek.

The Island Beach peninsula and Long Beach Island provide a barrier between Barnegat Bay and the Atlantic Ocean. This barrier, along with the shallowness of the Bay, minimize tidal fluctuations in the Bay. The 1966 U.S.G.S. Nautical Chart, 824-SC Sandy Hook to Little Egg Harbor, New Jersey, shows a mean tidal range at Oyster Creek of 0.6 feet. However, during storms tidal changes may be greater than 0.6 feet. In March, 1962, high tides accompanied a storm which is considered to have been the most disastrous ever to strike New Jersey. Flood marks recorded by the U.S. Geological Survey immediately north of the Oyster Creek site at Forked River showed a high tide elevation of 4.5 feet above MSL. Grade level at the site is 23 feet above MSL which is well above the 4.5 feet recorded, and there is no record of the site area being flooded or inundated even during storms with high tidal conditions.

Two minor fresh water streams are located at the plant site. Oyster Creek on the south and Forked River on the

north. Both Oyster Creek and Forked River flow west to east and have drainage areas consisting primarily of pine barrens, drainage areas are approximately 11.5 and 2 square miles respectively. Oyster Creek was dammed to form a small fire protection pond entirely within the plant property. The pond is also used as backup for certain emergency reactor systems. The small size and drainage areas of the streams along with the site topography preclude the possibility of their causing flooding of the plant site. Therefore, the only concern for flooding at the site is a high water level in Barnegat Bay resulting from a catastrophic hurricane.

Test borings at the site indicate ground water levels less than 10 ft. below grade elevation. The surface of the ground water was shown to slope from the west downward toward the bay and also from the high ground in the center of the site toward the streams on the north and south. Thus surface drainage at the site is toward the bay to the east, toward the south branch of Forked River to the north or toward Oyster Creek to the south.

IV. FLOODING

A report was prepared by Dames & Moore for Jersey Central Power & Light in 1972 to study the Probable Maximum Hurricane (PMH) at the Oyster Creek site and to establish design criteria for flood protection of Class I structures. Conclusions reached in the report for PMH conditions are:

- 1) The maximum stillwater elevation at the site is 22.0 ft.
- 2) With the surrounding top of the plant site fill at 23.0 ft., the plant structures are protected against wave runup.

The Probable Maximum Hurricane is defined as a "hypo-hurricane that might result from the most severe combination of hurricane parameters that is considered reasonably possible in the region involved, if the hurricane should approach the point under study along a critical path at an optimum rate of travel" (Ref. 2).

The return frequency of such a postulated storm would be well in excess of 1000 years. Mr. T.E. Haeussner, a Hydraulic Consultant, in a similar PMH study for the Forked River Nuclear Station estimated the return frequency for the PMH to be on the order of once in a million years. Flood records for the site indicate a record high tide elevation in the bay of 4.5 ft. A storm with an intensity return frequency of once in 250 years on a critical path

concurrent with astronomical high tide would result in a peak tide elevation of about 5.3 ft. on the inland shore of Barnégat Bay (Ref. 2). Based on the above data, it can be concluded that the likelihood of a PMH flood occurring at the site during the remaining plant life is extremely remote.

The Oyster Creek Nuclear Station has an emergency procedure designed to deal with the PMH so that in the unlikely event that the PMH does occur, the plant can be brought to a safe shutdown. Emergency Procedure No. 520 provides the plant operator with a series of actions to be taken during the development of a flooding situation. The procedure involves both the emergency shutdown of the reactor when the flood water level reaches elevation 6.0 ft., and the procedure for cooldown which is to be performed by operation of the isolation condensers. Since, the emergency diesel generators are protected against the PMH flood, they can supply power for the necessary pumps even in the event of loss of offsite power.

V. CONCLUSION

The only portion of the Oyster Creek Nuclear Generating Station subject to flooding due to the PMH is the intake structure. This is due to the fact that plant grade is at elevation 23.0 feet, while the flood level for the probable maximum hurricane is 22.0 feet. Even the inundation of the intake structure (operating deck elevation 6.0 feet) is unlikely since the PMH has an expected return frequency far in excess of 1000 years. A storm with a return frequency of 250 years would not be expected to affect any of the plant's intake pumps. If a flood affecting the ability of the plant's intake pumps to provide water to the plant did occur, Emergency Procedure 520 has been developed in order to establish the procedure for shutdown and cooldown of the reactor for floods up to the PMH flood elevation of 22.0 ft. Therefore, it can be concluded that flooding of any portion of the Oyster Creek Plant is very unlikely, and if the PMH flooding ever should occur, the plant can readily implement an established procedure to bring the plant to a cold shutdown state in spite of the flooded intake structure.

VI. REFERENCES

- 1) Report - "Probable Maximum Hurricane Flood analysis, Oyster Creek Nuclear Unit 1" prepared by Dames & Moore, 1972.
- 2) Report - "Determination of P.M.H. Flood Height for Forked River Unit 1 Nuclear Power Plant" prepared by T.F. Haussner, Hydraulic Engineer Consultant, 1970.

- 3) The Oyster Creek Nuclear Power Plant "Facility Description and Safety Analysis Report" (FSAR).

TOPIC II-3.C SAFETY-RELATED WATER SUPPLY
(Ultimate Heat Sink)

I. Introduction

This topic reviews a part of the Cooling Water System, the Safety-Related Water Supply or the Ultimate Heat Sink (UHS). Of particular concern is the ability of the UHS to provide an adequate amount of water for emergency shutdown and the maintenance of safe shutdown.

A review of the acceptability of the Oyster Creek Nuclear Generating Station UHS with respect to current pertinent regulatory criteria will be provided. Criteria considered will include NRC regulatory guides, standard review plans, general criteria, regulations and other documented staff positions used in the licensing of new nuclear plants. The review will be based on docketed information on Oyster Creek and as the nearby proposed Forked River Nuclear Station, as well as on design drawings, specifications and reports on file. Any deviations from current licensing requirements will be identified and the significance of the deviations will be explained.

II. Basis for Evaluation

The current criteria pertinent to this topic are the following:

Standard Review Plans 2.4.7,
2.4.8, 2.4.11, 9.2.1 and 9.2.5

NRC Regulatory Guides 1.27,
1.59, 1.102, 1.117, 1.127 and 1.135

NRC Branch Technical Position ASB 9-2

III. The Oyster Creek Ultimate Heat Sink

The UHS under review consists of a canal drawing water from Barnegat Bay. The canal is 140 ft. wide, 10 ft. deep and lined with riprap covered with a layer of 4 inches of crushed stone bonded with asphalt. The canal follows the South Branch of the Forked River. Discharge is through another canal, 100 ft. wide similarly lined, following Oyster Creek back to the bay. Based on our analyses, the Oyster Creek Plant UHS meets all applicable NRC criteria with the following notes:

- 1) SRP 9.2.5, "Ultimate Heat Sink" requires in Section III-5 that the reviewer verify that essential portions of the UHS are classified Seismic Category I, Quality Group C, and are tornado missile protected. The intake canals being an open channel of water to Barnegat Bay would not be threatened by tornado missiles. However, the slopes of the channel have not been designed for seismic conditions. This is not significant in that the lined canal banks should be very stable during a seismic event. Also, even if the banks collapsed, the dimensions of the channel 140 ft. wide by 10 ft. deep would preclude the possibility that an embankment collapse would significantly affect water flow and threaten the ability of the plant to reach a safe shutdown state.

An additional investigation was performed to study the affects of a seismic failure of the bridges which cross

- 2) One of the requirements of Regulatory Guide 1.27 is that technical specifications for the plant should include provisions for actions to be taken in the event that conditions threaten partial loss of the capability of the UHS. Even the partial loss of the UHS is not considered credible since the canal is open to Barnegat Bay. However, high flood levels could threaten the intake pumps. This problem has been addressed under the discussion of loss of intake pump function due to flooding. (The intake structure and pumps are not considered to be part of the UHS according to Regulatory Guide 1.27). The technical specification concerning loss of pumping capability is discussed under Topic III-3A, "Effects of High Water Level on Structures".

IV. Reference: Forked River Nuclear Station, Unit 1 PSAR, Volume 5, Question 2.12.

TOPIC III-2 Wind and Tornado Loading

I. Introduction

This topic reviews the wind and tornado loading (including pressure drop) capacity of the Oyster Creek Nuclear Generating Station and also its ability to withstand tornado generated missiles. The objective of the review is to assure that Category I structures, systems and components are adequately designed for tornado winds and pressure drop and that any damage to structures not designed for tornado generated forces will not endanger Category I structures.

Items involved in this topic have previously been addressed in amendments 11, 28, 31 and 32 of the Oyster Creek FSAR. Therefore, these amendments serve as a large part of the basis for this review.

II. Basis for Evaluation

The current criteria pertinent to this topic are the following:

Standard Review Plans 2.3.1, 3.3.1,
3.3.2, 3.5.1.4,
3.5.2, 3.8.1,
3.8.4, 9.2.1, 9.2.5

NRC Regulatory Guides 1.76, 1.117

III. Evaluation

a) Wind Design Load:

Reviews of the Oyster Creek plant buildings for wind loading indicate that they will be able to safely withstand a 100 year windstorm. This is based on the use of ANSI A58.1 with allowable stresses increased by one-third for load combinations which include wind. The 100 year windstorm considered has a velocity of 100 mph from 0 to 50 feet and 125 mph from 50 to 100 ft.

III.

b) Tornado Design Load

Regulatory Guide 1.76, "Design Basis Tornado", requires a maximum pressure drop capability of 3.0 psi and a maximum tornado wind speed of 360 mph. No design basis tornado had been established specifically for the site but the tornado frequency for the site had been estimated at 2190 years (see Ref. 2).

The tabulation below lists the various safety-related structures with their respective maximum permissible wind velocity and depressurization values. The allowable stresses do not exceed 90% of yield for reinforcing steel and 85% of the ultimate concrete strength and include the combined effect of dead loads plus normal operating loads.

<u>Structure</u>	<u>Wind/mph</u>	<u>Pressure/psi</u>
Reactor Bldg. Exterior Concrete Walls	300	2.0
Reactor Bldg. Insulated Metal Siding	160	0.53
Reactor Bldg. Roof Decking	280	0.68
Reactor Bldg. Steel for Craneway Enclosure	*190	0.68
Control Room - North Wall	160	0.53
Remainder	300	2.0
Intake Structure	300	2.0
Ventilation Stack	180	2.0
Diesel Generator and Oil Tank Vaults	300	2.0

* Based on siding drag - without siding steelwork can withstand 300 mph

Generally safety-related equipment is enclosed in the listed safety-related structures and is therefore protected within the limits shown. The outdoor Service Water Pumps and Startup Transformer are capable of withstanding 200 mph winds and a depressurization 2 psi.

The method of analysis to determine the protective capability of safety-related buildings and equipment against various sized missiles and missile penetration at tornado velocities was based on the Modified Petry Formula.

The missiles assumed were a wood utility pole, 35 feet long by 14 inches in diameter having a velocity of 200 mph and a 1-ton missile, such as a compact-type automobile traveling at 100 mph with a contact area of 25 square feet. Other missiles postulated by SRP 3.5.1.4 have not been investigated.

The results of the analysis indicate that no perforation of the 18-inch thick reactor building walls or the 12-inch thick control room walls will occur with the utility pole or the automobile, although spalling of the inside concrete face would be expected.

The control room, battery room, emergency diesel generator building, emergency switch gear and related electrical duct banks have been designed for tornado protection.

There is essentially no missile protection of the magnitude discussed above in the metal siding walls of the reactor building above the refueling floor and the equipment access opening. There is also no missile protection for the Class 1 pumps at the intake structure.

Another matter considered was how other structures would be affected by failure of the stack due to a tornado. An analysis was performed to determine the capability of the reactor building refueling floor to resist various sized stack sections. The analysis was performed for seismic considerations so the results concern the stack falling, not being wind driven, however the data is an indication of the safety levels involved. The results of the analysis are tabulated below indicating a size of stack section which the various structural elements can withstand without allowing penetration:

<u>Section Analyzed</u>	<u>Length and Weight of Stack Section</u>	
16" Floor Slab	20 feet,	34 KIPS
12" Floor Slab	15 feet,	20 KIPS
7' - 0" Thick Shield Plugs	40 feet,	86 KIPS

In each case the stack section was assumed to strike on end as a cylinder and was assumed to fall from the top of the stack.

The equipment necessary for emergency core and containment cooling and other equipment necessary to shut down the reactor, including heat exchangers and pumps, are located in the reactor building in compartments located below the refueling floor level. Thus, even such an improbable event as failure of the stack would not impair the ability to safely shut down.

Consideration was also given to the likelihood of damage to the spent fuel storage pool due to failure of the reactor building steel superstructure siding during a tornado. Conclusions reached indicate that the design of the reactor building and the arrangement of equipment is such that there is very little chance that either the spent fuel pool or the fuel stored in the pool could be seriously damaged as a result of the tornado effects on the building or its contents.

IV. Conclusion

The Oyster Creek Nuclear Generating Station conforms to the current wind loading criteria but does not meet the tornado wind and pressure drop loading to the absolute limits specified in Regulatory Guide 1.76. The safety margin is adequate when considering the historical records for tornado loadings at the site. The missile protection for the Category I pumps is not considered significant due to the fact that (a) the return frequency of the tornado is estimated at 2190 years and (b) the physical separation of the pumps is considered significant enough that the probability of a single or multiple missiles damaging all pumps of the same safety system at the same time is very remote.

V. References

- 1) Oyster Creek Facility Description and Safety Analysis Report
- 2) Amendment 11 to the FSAR, "Answers to 109 A.E.C. Questions Regarding Additional Plant Information".
- 3) Amendment 28 to the FSAR, "Response to A.E.C. Letters of Oct. 16, 1967 and Nov. 20, 1967"
- 4) Amendment 31 to the FSAR, "Corrections to Oyster Creek FSAR and Amendments"
- 5) Amendment 32 to the FSAR, "Response to A.E.C. Letter of Jan. 9, 1968".
- 6) Miller, D. R. and W. A. Williams, "Tornado Protection for the Spent Fuel Storage Pool" (Nov. 1968, a report by General Electric).

TOPIC III-3A EFFECTS OF HIGH WATER LEVEL ON STRUCTURES

I. Introduction

This topic reviews the effect of the present postulated high water level at the plant site as compared to the original design basis water level considered for construction of the plant buildings.

A design basis flood, the "Probable Maximum Hurricane," is developed using current criteria and compared to the design basis event that was used for construction. Deviations and their safety significances are discussed. The design of plant building structures and the design basis criteria are reviewed and compared to current criteria. The variations and safety significance of the variations are discussed.

The information used to perform the reviews is gathered from the Oyster Creek FSAR, the original plant design criteria and special reports. In case detailed information is not available, the analysis will be made using conservatively estimated parameters.

II. Base for Evaluation

The current criteria pertinent to this topic are the following:

Standard Review Plans: 2.4.2, 2.4.3, 2.4.5,
3.4.1, 3.4.2, 3.8.1,
3.8.4, and 3.8.5

NRC Regulatory Guides: 1.29, 1.59, 1.102

III. Background

The Oyster Creek Nuclear Generating Station was constructed before 1972. High water level considered for the design of the intake and discharge structures, due to the worst storm (March 1962), was EL. 4.5' above mean sea level. Hence, the circulating water pumps and the service water pumps were considered safe, since they are located on the operating deck at EL. 6.0 ft.

IV. High Water Level Due To Probable Maximum Hurricane (PMH)

As per Appendix C to Regulatory Guide 1.59, the new estimated PMH flood level at Oyster Creek site is EL. 23.8 feet above mean sea level. Probable Maximum Hurricane flood analysis (Ref. 1) was carried out by Mr. Philip Sherlock of Dames and Moore Engineers, Cranford, New Jersey, assuming very conservatively estimated parameters.

The report of PMH analysis, submitted to GPU Service Corp. in March, 1972, concluded that Oyster Creek will have flood level at EL. 22.0 feet above MSL.

V. Flood Protection Measures For Seismic Catagory I Structures

The plant site is located at grade EL. 23.0' above mean sea level, which is 1'-0" above PMH flood level EL. 22.0 feet. The major buildings of the plant are sealed against entry of flood water to EL. 23.0'.

Although the external walls of safety related structures that are below the plant grade elevation are protected by a water proofing membrane to EL. 5.0 ft., the PSAR indicates that ground water exists at an EL. 15.0'. The difference between the ground water elevation and the water proofing membrane is judged insignificant since the plant has not any ground water seepage problem. The piping penetrations and other openings in the walls below flood water elevation are protected with flood seals. These above factors and the fact that the wall thickness below grade for safety related structures ranges from 4'-6" to 3'-0", makes the flooding potential of the reactor building very small.

The walls of safety related buildings are designed to withstand the hydrostatic pressure due to ground water level at EL. 15.0' and horizontal seismic earth loading. Hence, the walls will resist the hydrostatic pressure due to PMH flood level and lateral soil pressures.

The intake structure which supports the circulating water pumps and the service water pumps, seismic Class I equipment, designated as a Class II structure.

- Analysis of the as-built intake structure shows that all elements of the intake structure satisfy the criteria for a Class I structure, except for PMH flood water level. (Ref. 2)

The intake structure was designed for high water level at EL. 4.5' above mean sea level (see paragraph 3.0). Hence, the operating deck level was established at EL. 6.0', giving a 1.5 ft. margin of safety. As a result, the intake structure and pumps represent the major problem for a safe shutdown of the reactor during PMH flood as per Regulator Guide 1.59. Due to this fact, and in response to question 4.J in Reference 2, Jersey Central Power and Light has committed that emergency procedure No. 520 will be followed when water level at the intake reaches to EL. 6.0', to safely shutdown the reactor and maintain it for the required period of time.

VI. Conclusion

The use of the hydrostatic load from design flood or high water level provides reasonable assurance that in the event of flood the structural integrity of the plant seismic Class I structures will not be impaired. As a result seismic Class I systems and the components located within these buildings will be adequately protected and will perform their necessary safety functions.

The intake structure, supporting Class I equipment e.g. service water pumps, etc., is protected only to water level EL. 6'0. Beyond that flood level, emergency procedure No. 520 shall be enforced for safe shutdown of the facility.

VII. References

1. Report - Probable Maximum Hurricane Flood analysis - Oyster Creek Nuclear Station Unit No. 1, by Mr. Philip Sherlock of Dames and Moore, Cranford, New Jersey (March 1972).
2. Oyster Creek Nuclear Power Plant Unit No. 1 Amendment 22. Additional information response to Category I questions of October 16, 1967 request. (Response to letter - Dr. Peter A. Morris, Director, Division of Research Licensing to Mr. John E. Logan, Vice President, Jersey Central Power & Lighting Co.)

TOPIC III-4.D

SITE PROXIMITY MISSILES (INCLUDING AIRCRAFT)

I. INTRODUCTION

This topic reviews the acceptability of the Oyster Creek Nuclear Power Plant against the missiles generated from accidental explosions in the vicinity of the site, and from aircraft accident. The objective of the review is to assure that Category I structures, systems and components are adequately designed for the site proximity missiles.

The results of study for the nearby Forked River Nuclear Plant are reviewed concerning the proximity of the site to the accidents under investigation.

Finally the evaluation is made concerning the consequence of the above accidents upon the plant and its operation.

Further a probability study of accidents, as required by Standard Review Plan, Section 3.5.1.6, is recommended to confirm that the plant is adequately protected against aircraft hazards.

II. BASIS FOR EVALUATION

The current criteria pertinent to this topic are as follows:

Standard Review Plans 2.2.1, 2.2.2, 2.2.3, 3.5.1.5, 3.5.1.6, 3.5.2, 3.5.3.

NRC Regulatory Guides 1.70, 1.91.

III. EVALUATION OF ACCIDENT

1. Missiles from Accidental Explosion

There is little industry within ten miles of the site. Small industry, consisting mainly of boat repair and commercial fishing, is scattered along coastal area. The remaining industry in the area is small and diversified. A list of the industrial locations within ten miles of the plant site is presented in Exhibit III-4.D-1. Therefore, the industrial facilities near the site do not pose any threat to the plant.

Other accidents resulting from explosions of nearby transportation - train, truck, ship or barge, and pipe-

line, are studied using the transportation system shown in Exhibit III-4.D-2. The major highways are the Garden State Parkway and U.S. Route 9. The Central Railroad of New Jersey track, alongside Route 9 in a north-south direction through the study area, is currently dismantled. Pipeline explosions are precluded because the existing 10" ϕ pipe line and a 12" ϕ pipe line planned in the future along Route 9 is considered less critical than the truck explosion. Therefore, the remaining sources of missiles to be studied are those from truck explosions and ship explosions. An explosion fed by as much as 10,000,000 pound ship load of TNT in Barnegat Bay is not considered hazardous due to the relative distance.

Initial evaluation of the plant relative to Reg. Guide 1.91 indicates the critical plant structures are within the critical distance of approximately 1600 ft., wherein truck explosion may be hazardous. However, further detailed study on peak incident pressure due to a truck explosion indicates that the pressure is only 2 psi and, therefore, is not critical.

2. Aircraft Hazards

There are several private airports serving the region. The Manahawkin Airport is located nine miles south of the site. The Robert J. Miller Airfield is located nine miles to the northwest near Dover Forge. In addition, there are two small dirt airstrips in the area. One is located two and one-half miles northeast of the site while the other, Beechwood, is eight miles NNE. The Federal Aviation Administration lists a restricted area at Warren Grove with its closest boundary 7.5 miles from the site. This area is an aerial target range used by the Massachusetts Air National Guard. However, bombs and rockets used are dummies without explosive charges.

The air corridor, known as New York 838, passes within 10 miles of the plant location and is used by the Air National Guard as a low level high speed military training route only to make approaches to the Warren Grove Range. After the practice runs over the target have been completed, the planes fly to higher altitudes to return to their base by routes other than New York 838.

Although an extensive study on the probability of aircraft accidents near the site has not been conducted to prove that aircraft hazards are eliminated as a design basis concern, it is anticipated that there will be no

significant safety implications. However, a more detailed survey of aircraft traffic near the site is required to substantiate this positive conclusion in accordance with the acceptance criteria in SRP 3.5.1.6.

IV. EVALUATION OF PLANT

The missile hazards to the plant due to accidental explosions in Item III(1) above are not likely as critical as the effects of tornado missile described in TOPIC III-2. Therefore, the plant modification as recommended in TOPIC III-2 is considered adequate.

Although consideration of aircraft impact on the plant was never required as a licensing basis for the plant, the fall of the plant stack on reactor building was evaluated. In this unlikely event, it was concluded that the reactor can be safely shut down. (See Amendment 31, CORRECTIONS TO OYSTER CREEK FSAR AND AMENDMENTS, JCP&L Co.)

V. CONCLUSIONS

The Oyster Creek Nuclear Power Station is adequately protected against site proximity missiles except for the Category I pumps on the intake structure (See TOPIC III-2). However, the probability of an aircraft hazard should be calculated in accordance with SRP 3.5.1.6 to confirm that aircraft hazards to the plant need not be considered.

VI. REFERENCES

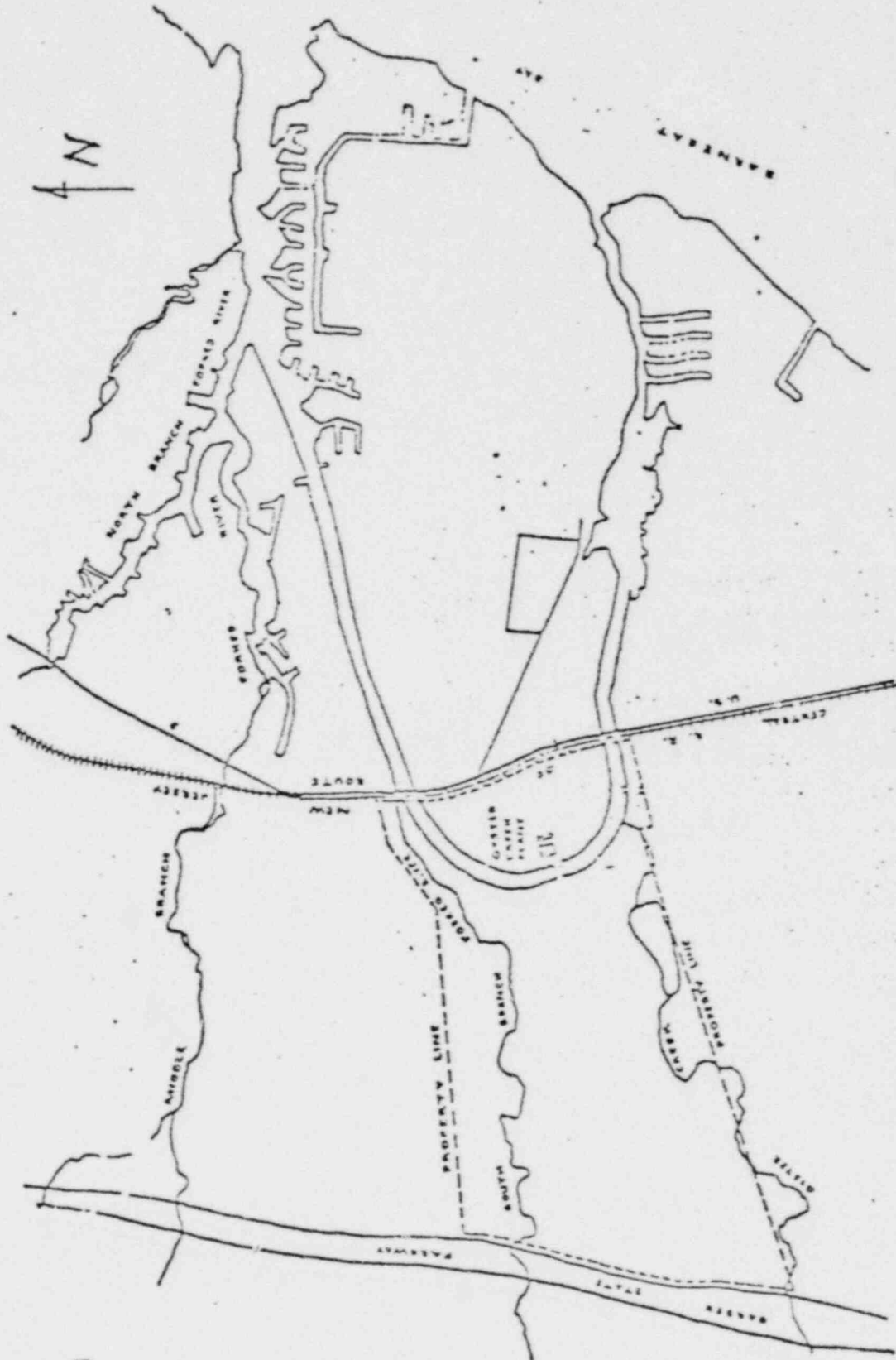
1. Standard Review Plans, NUREG-75/087, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation.
 - a. 2.2.1 - 2.2.2, Identification of Potential Hazards in Site Vicinity.
 - b. 2.2.3, Evaluation of Potential Accidents.
 - c. 3.5.1.5, Site Proximity Missiles (Except Aircraft).
 - d. 3.5.1.6, Aircraft Hazards.
 - e. 3.5.2, Structures, Systems, and Components to be Protected from Externally Generated Missiles.
 - f. 3.5.3, Barrier Design Procedures.

2. Regulatory Guides, U.S. Nuclear Regulatory Commission, Office of Standards Development.
 - a. 1.70, Standard Format and Content of Safety Analysis Reports For Nuclear Power Plants.
 - b. 1.91, Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants.
3. Rules and Regulations, U.S. Nuclear Regulatory Commission.
 - a. 10CFR Part 50.
 - b. 10CFR Part 100.
4. Oyster Creek Facility Description and Safety Analysis Report.
5. Amendment 31, Corrections to Oyster Creek FSAR and Amendments.
6. PSAR, Volume 1, Forked River Nuclear Station Unit 1.

Community	Distance From The Site (Mi.)	Direction From The Site	Name of Company	Industry	Number of Employ
Barnegat	4	South	Research Products Corp Weatherproof Aluminum, Inc.	Dental Supplies Storm Windows	10 6
Bayville	7	NNE	Berkeley Machine Shop, Inc. Denzer-Schafer X-Ray Co., Inc. New Jersey Pulverizing Co. Rainbow Sportswear Corp. Woodland Manufacturing Co.	Machine Shop Silver Recovery Sand Products Sportswear Wrought Iron	12 9 66 30 30
Pine Beach	9	North	Castle Woodcraft	Kitchen Cabinets	8
Toms River	9	North	Best Block of Toms River	Concrete Blocks	15
	9	North	New Jersey Glovriar, Inc. Trilco Terminal	Newspaper Building Materials	15 37
	10	North	Acme Cabinet Corp. Delta Lumber Co., Inc. Fischer's Machine Works Glover, H. Clay Co., Inc. Marban Construction Co. Ocean County Sun, Inc. Quality Aluminum Products Co. Reardon Company Rochelle Novelty Co. Toms River Boat Works Toms River Chemical Corp. Towne Fabrics, Inc.	Cabinets Lumber Machinists Pet Shop Partition Walls Newspaper Aluminum Products Paints Shoe Bags Ship Construction Dyes Fabrics	77 15 6 21 6 25 10 22 10 20 1126 23

Source: 1971 New Jersey State Industrial Directory

EXHIBIT III-4.D-2



OYSTER CREEK

TOPIC II-1.A EXCLUSION AREA AUTHORITY AND CONTROL

The objective of this topic is to assure that appropriate exclusion area authority and control are maintained for Oyster Creek Nuclear Generating Station (OCNGS), as required by 10 CFR Part 100.

The OCNGS is located in Lacey Township, Ocean County, New Jersey, approximately nine miles south of Toms River, New Jersey, 35 miles north of Atlantic City, New Jersey, 45 miles east of Philadelphia, Pennsylvania, and 60 miles south of Newark, New Jersey. The station is located on a 1416 acre property owned by Jersey Central Power and Light Company. The site is bounded on the west by the Garden State Parkway and on the east by the bay. This corresponds to a width of approximately 3 1/2 miles from east to west. The maximum width in the north-south direction is almost 1 mile. U. S. Route 9 divides the property, with 755 acres lying west of the highway and 661 acres lying east of the highway. The station is about 1/4 mile west of the U. S. Route 9 and 1 1/4 miles east of the Garden State Parkway.

The reactor (center line) is located 1358 feet west of the east boundary of U. S. Route 9 which is the minimum exclusion distance, as defined in 10 CFR Part 100.3. All land areas, including mineral rights within the exclusion area, are owned by the licensee.

Parts of the exclusion area are traversed by U. S. Route 9 and Central Railroad of New Jersey. Arrangements have been made to control traffic on U. S. Route 9 in the event of a plant emergency, as part of the OCNGS Emergency Plan. Similar arrangements have not been made with the railroad

line to control traffic under emergency conditions, however, the need no longer exists since the railroad tracks have been removed.

The only waterway traversing the exclusion area is the OCNGS intake and discharge canal. Station security measures are enforced to ensure unauthorized activity does not occur in this waterway.

Based on requirements defined in 10 CFR 100, OCNGS is under proper authority to determine all activities within the exclusion area.

OYSTER CREEK

TOPIC II-1.B POPULATION DISTRIBUTION

The objective of this topic is to verify that the low population zone and population center distance specified for the plant site are compatible with current population distribution and are in conformance with the guidelines of 10 CFR Part 100.

The Oyster Creek Nuclear Generating Station (OCNGS) is located in Ocean County, New Jersey, approximately nine miles south of the town of Toms River. The site is bounded on the north and south by undeveloped land and on the west by the Garden State Parkway. From the east, the site property extends about 3 1/2 miles in land from the bay. The maximum width in the north-south direction is almost 1 mile. The plant is located on a 1,416 acre property owned by Jersey Central Power and Light Company. U. S. Route 9 divides the property, with 755 acres lying west of the highway, and 661 acres lying east of the highway.

The region surrounding the plant is characterized by flat terrain, sandy soils, and numerous freshwater and saltwater marshlands. Two barrier beaches, Seaside Peninsula and Long Beach Island, extend the length of the county providing extensive recreational opportunities on their beaches and bays. These attract a large transient seasonal population.

The licensee has recently completed a survey of the population within a ten mile radius of the plant in conjunction with emergency planning activities¹. Based on preliminary 1980 census figures the permanent resident population within a ten mile radius of the OCNGS is 66,815. The permanent resident population distribution within ten miles of the site is shown in Table 1. The average permanent resident household size within this area is

estimated to be 2.68 based on county-wide averages from the 1980 preliminary census figures.

The low population zone for the OCNGS has a distance of 0.75 miles to its outer boundary. The permanent population residing within one mile of the site and encompassing the low population zone is estimated to be 440 based on the recent survey. The majority of these residences are located northeast and southeast of the site. Previous population estimates identified 226 people residing within one mile of the plant in 1970 and projected a population of 1,228 for the year 2010². There are no schools or recreational areas within the 0.75 mile low population zone. The nearest population center containing more than 25,000 residents is Dover Township, which is 9.5 miles north of the site. Dover Township is made up of Toms River, Gilford Park, and several smaller communities. The 1980 population was approximately 64,445. Table 2 shows the population distribution from 10 to 50 miles from OCNGS.

The transient population within ten miles of the OCNGS is overwhelmingly a seasonal tourist and tourist-related population. There are numerous popular New Jersey shore communities in this area. As a result, there is a seasonal shift in populations within a 5--10 mile radius of the OCNGS which is concentrated in the northeast and southeast sectors due to the proximity to the Atlantic Ocean and Barnegat Bay. The peak seasonal population within a ten mile radius of the plant is expected to be 179,840.

Other than the tourist industry, there are no other significant employment opportunities within a ten mile radius of the plant. Thirty-four percent of the labor force commutes outside of Ocean county and it is estimated

that at least another forty percent of the labor force residing within the ten mile radius is employed outside this area³.

Based on our review, the low population zone and population center distances specified for the OCNGS are in conformance with the distance requirements of 10 CFR Part 100 in that the population center distance is more than one and one-third times the distance from the reactor to the outer boundary of the low population zone. An emergency plan for OCNGS has been submitted to the NRC to assure appropriate protective measures could be taken in behalf of the population within the low population zone and out to ten miles from the site in the event of a serious accident.

REFERENCES

- ¹Oyster Creek Nuclear Generating Station--Emergency Plan, 1981.
- ²Oyster Creek Nuclear Generating Station--Environmental Report, March 1972.
- ³Dresdner Associates, confirmed by William Hayes, Ocean County Emergency Management Coordinator.

TABLE 1
POPULATION BY SECTOR AND DISTANCE
(0 - 10 MILES)

<u>SECTOR</u>	<u>*DISTANCE</u> <u>0-1</u>	<u>*DISTANCE</u> <u>1-2</u>	<u>*DISTANCE</u> <u>2-3</u>	<u>*DISTANCE</u> <u>3-4</u>	<u>*DISTANCE</u> <u>4-5</u>	<u>*DISTANCE</u> <u>5-6</u>	<u>*DISTANCE</u> <u>6-7</u>	<u>*DISTANCE</u> <u>7-8</u>	<u>*DISTANCE</u> <u>8-9</u>	<u>*DISTANCE</u> <u>9-10</u>
N	0	522	2,188	665	93	337	459	3,232	8,369	3,572
ENE	151	551	1,198	884	743	855	1,252	4,130	3,428	3,214
NE	151	368	465	747	1,772	435	353	120	15	1,050
E	0	641	135	856	16	0	0	4	0	0
ESE	0	595	135	0	0	0	6	0	0	0
ESE	69	484	0	0	0	0	4	0	0	0
SE	69	394	0	0	0	58	706	79	0	0
SSF	0	325	344	118	0	0	0	145	316	204
S	0	325	609	1,402	1,125	166	0	28	28	227
SSW	0	0	65	393	806	3,394	0	0	2,847	1,038
SW	0	0	0	22	714	1,884	1,896	895	2	58
WSW	0	0	0	9	21	17	272	158	110	74
W	0	0	0	0	0	5	0	10	22	3
WNW	0	0	0	0	0	0	0	0	0	0
NW	0	0	0	0	0	0	30	129	160	95
NNW	0	80	693	0	0	0	1	0	0	0
TOTAL	440	4,285	5,832	5,096	5,290	7,157	4,979	8,930	15,297	9,515

*Miles From Facility

TABLE 2
POPULATION BY SECTOR AND DISTANCE

(10 - 50 MILES)

SECTOR	*DISTANCE 10-15	*DISTANCE 15-20	*DISTANCE 20-25	*DISTANCE 25-30	*DISTANCE 30-35	*DISTANCE 35-40	*DISTANCE 40-45	*DISTANCE 45-50	TOTALS
N	23,937	51,847	21,735	28,824	40,507	73,080	131,688	166,853	538,471
NNE	19,212	46,014	32,362	63,061	58,542	41,785	7,789	-0-	268,765
NE	-0-	-0-	-0-	-0-	-0-	-0-	-0-	-0-	-0-
ENE	-0-	-0-	-0-	-0-	-0-	-0-	-0-	-0-	-0-
E	-0-	-0-	-0-	-0-	-0-	-0-	-0-	-0-	-0-
ESE	-0-	-0-	-0-	-0-	-0-	-0-	-0-	-0-	-0-
SE	-0-	-0-	-0-	-0-	-0-	-0-	-0-	-0-	-0-
SSE	-0-	-0-	-0-	-0-	-0-	-0-	-0-	-0-	-0-
S	4,704	3,052	172	-0-	-0-	-0-	-0-	-0-	7,928
SSW	1,837	6,247	2,835	7,462	50,066	49,141	21,132	9,622	148,942
SW	1,794	884	401	7,418	9,288	3,555	1,181	12,503	37,024
WSW	355	475	779	4,938	11,872	16,394	25,805	46,024	106,642
W	457	457	3,274	6,382	22,272	73,096	163,192	189,464	458,594
WNW	5,496	9,077	20,087	19,345	27,955	115,414	108,152	558,300	863,826
NW	5,382	6,642	9,639	10,425	47,364	103,222	172,102	62,456	417,232
NNW	10,349	11,245	8,962	16,070	31,378	51,085	75,549	153,553	362,191
TOTAL	73,523	135,940	100,246	163,925	299,844	526,772	710,590	1,198,775	3,209,615

* Miles From Facility

OYSTER CREEK

TOPIC II - I.C POTENTIAL HAZARDS DUE TO NEARBY INDUSTRIAL, TRANSPORTATION, AND MILITARY FACILITIES

The objective of this topic is to assure that Oyster Creek Nuclear Generating Station (OCNGS) is adequately protected and can be operated with an acceptable degree of safety with regard to potential accidents which may occur as the result of activities at nearby industrial, transportation, and military facilities.

Ocean County's industrial base is small, but diversified. Boat building and marine equipment manufacturing were once the dominant industrial activities, but today industrial activity also includes chemical manufacturing, mining of ilmenite, quarrying of industrial sands, garment manufacturing, food processing, and concrete production. Table 1 identifies industrial locations within ten miles of the plant site.

The nearest transportation route to the station is U.S. Route 9 which is located approximately 0.25 miles east of the reactor building. It is uncertain if shipments of explosives travel on Route 9 past the plant. However, with the guidance provided in Regulatory Guide 1.91, Revision 1, the plant is at a marginal safe distance when analyzed for a postulated explosive accident on the highway for truck-size shipments. Further analyses should be performed to establish whether critical plant structures are beyond the adverse effects from an explosion on this transportation route.

The Garden State Parkway is about 1.25 miles west of the plant and forms the western boundary of the site. The separation distance between the highway and the plant exceeds the minimum distance criteria given in the regulatory

guide for truck-size shipments of explosive materials. It is expected that an explosive accident on the highway will not affect safe operation of the plant.

The Licensee has reviewed the present status of transport of hazardous chemicals in the vicinity of OCNGS and has identified no frequent shipments as defined by Regulatory Guide 1.78.

The nearest railroad corridor is approximately 0.25 miles east of the reactor building. Rail traffic through this corridor is discontinued and the railroad tracks have been removed.

There are no large commercial harbors within ten miles of the site. Public marinas are the chief recreational facilities in the immediate site area. The Intracoastal Waterway is the only inland waterway used for shipping in the area, however, it is not heavily used. It is approximately two miles east of the plant at its closest point. Major shipping lanes in the Atlantic Ocean are located well off-shore. Thus shipping is not considered to be a hazard to the OCNGS.

The nearest pipelines to the plant lie in a corridor along U.S. Route 9 approximately 0.25 miles from the plant. These consist of an 8" and 6" diameter natural gas pipelines. No evaluation has been performed to assess the effects of pipeline accidents on the safe operation of the plant. There are no gas or oil production fields, underground storage facilities or refineries in the vicinity of the plant.

There are no missile sites within a ten mile radius of the OCNGS site. Nine airfields are located within 20 miles of the plant (Figure 1). Two of the airfields are military installations: McGuire Air Force Base, also used by the U. S. Air Force, U. S. Air National Guard, and the Military Air Transport Service (MATS) 25 miles to the NW, and Lakehurst Naval Air Station, 20 miles NNW. Other airports listed by the Federal Aviation Administration (FAA) are Breton Woods, 17 miles north; Eagle's Nest, 12 miles SSW; Coyle Tower, 10 miles west; Ocean County, 9 miles NNW; Manahawkin, 9 miles SSW; and Beechwood, 8 miles NNE. In addition, there is a sod strip 2 miles NE at Forked River.

The FAA lists three restricted areas in the vicinity of the plant. Two of these areas, R5001A and R5001B are contiguous near Fort Dix, 15 miles to the NNW. These restricted areas are used mainly as firing ranges for small arms, artillery, and mortars. The third area, R5002, at Warren Grove is a low-level aerial target range used by the Air National Guard. Its closest boundary to the plant is 7.5 miles. Bombs, rockets, and 20 mm gun fire are used in the target range. The bombs are dummies that give off a flash, but no explosive charge. The rockets do not have explosive charges, only a propellant to deliver the rocket on target and the 20 mm shells have solid heads without explosives.

Two air corridors pass in the vicinity of the plant. One is used by the Air National Guard and is known as "New York 838" which is a low-level, high-speed military training route, and goes to the Warren Grove Aerial Target Range (R5002). The route is used only to make approaches to the range. After the practice run over the target has been completed, the planes climb to higher

altitudes to return to their bases by routes other than "New York 838."

The other is a civilian corridor marked "Victor Air Lane 312", which is aligned east-west and passes over the site. The airline can be used by all types of aircraft, but the FAA - which controls all civilian aviation - specifies minimum safe altitudes at which planes can be flown in the corridor.

The licensee has not performed an analyses using analytical models to calculate the probability of an aircraft crash at the plant, however, the site was judged to be acceptable when examined for a companion unit.¹

¹ Safety Evaluation of the Jersey Central Power & Light Company Forked River Nuclear Generating Station Unit 1, Docket No. 50-363, U. S. Atomic Energy Commission, 1972.