Exelon

OCNGS UFSAR

Revision 12, April 2001

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1.4 IDENTIFICATION OF AGENTS AND CONTRACTORS

The Oyster Creek Nuclear Generating Station (OCNGS) was designed and constructed by the General Electric Company Atomic Power Equipment Department as a turnkey project. Burns and Roe Inc. is the Architect-Engineer of record.

The plant is owned **and operated by AmerGen Energy Company, LLC, who is the licensee** for the facility.

The suppliers of major components and equipment were as follows:

Item	Supplier
Reactor Vessel	Combustion Engineering Co.
Reactor Vessel Internals	P. F. Avery Co.
Steel Containment	Chicago Bridge and Iron Co.
Isolation Condensers	Foster Wheeler Corp.
Reactor & Turbine Buildings Structural Steel	American Bridge Co.
Reactor Recirculation Pumps	Byron Jackson
Reactor Recirculation System - Piping & Supports	Grinnell
Reactor Recirculation System – Valves	Chapman Div. Crane Co.
Main Condenser	Worthington

(6) Provide suitable checks to assure that conflicting or contradictory information is not conveyed to operators and other personnel until resolution is reached; and,

(7) Provide periodic internal audits to assure that the feedback program functions effectively at all levels.

Response

The review and assessment of information previously reviewed by Operating Experience Assessment Implementation (OEAI) is now done by several departments including: Training, Engineering, and Regulatory Affairs.

1.9.12 Item I.C.6 - Guidance on Procedures for Verifying Correct Performance of Operating Activities

Position

It is required (from NUREG-0660) that licensees' procedures be reviewed and revised, as necessary, to assure that an effective system of verifying the correct performance of operating activities is provided as a means of reducing human errors and improving the quality of normal operations. This will reduce the frequency of occurrence of situations that could result in or contribute to accidents. Such a verification system may include automatic system status monitoring, human verification of operations and maintenance activities independent of the people performing the activity (see NUREG-0585, Recommendation 5), or both.

Implementation of automatic status monitoring, if required, will reduce the extent of human verification of operations and maintenance activities but will not eliminate the need for such verification in all instances. The procedures adopted by the licensees may consist of two phases - one before and one after installation of automatic status monitoring equipment, if required, in accordance with Item I.D.3.

Response

The requirements for verification of correct performance of operating activities have been specified in the plants' administrative procedures. Verification of system alignment is required during installation and removal of safety tags and electrical or mechanical jumpers, and during lifting and reinstallation of electrical leads. Instrumentation valve lineups are verified following surveillance tests or maintenance evolutions requiring valve manipulation. **Plant management will determine when to use Concurrent Verification or Independent Verification.**

NRC has found this action acceptable to satisfy the requirements of this item.

1.9.13 Item I.D.1 - Control Room Design Reviews

Position

In accordance with Task Action Plan I.D.1, Control Room Design Reviews (NUREG-0660), all licensees and applicants for operating licenses will be required to conduct a detailed Control Room design review to identify and correct design deficiencies. This detailed Control Room design review is expected to take about a year. Therefore, the Office of Nuclear Reactor Regulation (NRR) requires that those applicants for operating licenses who are unable to complete this review prior to issuance of a license make preliminary assessments of their Control Rooms to identify significant human factors and instrumentation problems and establish a schedule approved by NRC for correcting deficiencies. These applicants will be required to complete the more detailed Control Room reviews on the same schedule as licensees with operating plants.

Response

In December 1980 a detailed Control Room Design Review (CRDR) was initiated for the OCNGS Control Room. Human factors design reviews were conducted, and the CRDR was completed in September 1982. The program plan was submitted on July 1, 1983 and the "Summary Report on the Oyster Creek Control Room Design Review" was submitted on April 30, 1984.

On January 17 and 18, 1990, the NRC staff conducted an onsite audit of Oyster Creek's DCRDR and SPDS. By SER, dated June 28, 1990, the staff has concluded that GPUN meets all of the nine NUREG-0737 Supplement 1 DCRDR requirements and the issue is considered resolved.

1.9.14 Item I.D.2 - Plant Safety Parameter Display Console

Position

In accordance with Task Action Plan I.D.2, Plant Safety Parameter Display Console (NUREG-0660), each applicant and licensee shall install a safety parameter display system (SPDS) that will display to operating personnel a minimum set of parameters which define the safety status of the plant. This can be attained through continuous indication of direct and derived variables as necessary to assess plant safety status.

Response

A Critical Safety Functions (CSF) approach was used as the basis for a Safety Parameter Display System and five CSFs were chosen that correlate with the basic objectives of Emergency Operating Procedures (EOPs). A parameter set was then selected largely based on EOPs parameters and the Emergency Plan radiation monitoring **requirements**.

Groups of parameters were then assigned to the different CSFs based on plant operating and emergency procedures logic.

periodic integrated leak tests at intervals not to exceed each refueling cycle.

Response

Plant systems outside containment which would or could contain highly radioactive fluids during a serious transient or accident include: shutdown cooling, isolation condenser, core spray, containment spray, reactor water cleanup, standby gas treatment, reactor coolant sampling, reactor coolant system instrumentation, drywell equipment and floor drains, reactor equipment and floor drains and the scram dump volume and associated vents and drains. An initial leakage reduction test/inspection was performed on these systems. A total of 1925 inspections were made on 1642 components, considering both internal and external leakage paths. Where leaks existed, repairs were made, if possible, and post repair inspections were made to document the reduced leakage. Results were tabulated and reported.

Following performance of this initial leakage reduction program, an ongoing leak reduction program was implemented which includes various leak tests/inspections on a 24-month (refueling interval) basis. The systems inspected under the ongoing preventive maintenance program include those portions of shutdown cooling, isolation condensers, core spray, containment spray and reactor water cleanup systems that can be inspected as permitted by system design and radiological conditions. In addition, the ongoing program includes identification of leakage from visual surveillance by plant personnel (operator rounds/tours) and responses of area and effluent monitors.

In response to IE Circular 79-21, a leak reduction program has been implemented wherein flow paths by which radioactivity can leave the plant internal environs to the outside environment have been identified and examined. Flow paths to the outside environment were reviewed with the objective of minimizing the potential of unplanned radioactivity releases.

1.9.50 Item III.D.3.3 - Improved Inplant Iodine Instrumentation Under Accident Conditions

Position

(1) Each licensee shall provide equipment and associated training and procedures for accurately determining the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident.

(2) Each applicant for a fuel loading license to be issued prior to January 1, 1981 shall provide the equipment, training, and procedures necessary to accurately determine the presence of airborne radioiodine in areas within the plant where plant personnel may be present during an accident.

Response

Air samplers containing Silver Zeolite radioiodine sampling cartridges are contained in Emergency kits in the Control Room to collect and analyze air samples for radioiodine during an accident.

Oyster Creek Emergency Procedures incorporate ALARA concepts for inplant radioiodine sampling during accident conditions. Personnel training in the Oyster Creek Emergency Procedures is required to be performed on a periodic basis.

1.9.51 Item III.D.3.4 - Control Room Habitability Requirements

Position

In accordance with Task Action Plan Item III.D.3.4 and Control Room habitability, licensees shall assure that Control Room operators will be adequately protected against the effects of accidental release of toxic and radioactive gases and that the nuclear power plant can be safely operated or shutdown under design basis accident conditions (Criterion 19, "Control Room," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50).

Response

Interim modifications for Control Room Habitability were implemented during the Cycle 11 refueling outage. NRC approved Amendments No. 115 and 139 to the OCNGS Facility Operating License which completes the interim and final actions in response to this topic. The final modifications were implemented during the Cycle 12 refueling outage (see Section 6.4).

1.9.52 References

(1) Safety Evaluation by the Office of NRR Relating to High Point Vents for the Isolation Condenser, dated April 24, 1986.

CHAPTER 2

SITE CHARACTERISTICS

2.1 GEOGRAPHY AND DEMOGRAPHY

2.1.1 Site Location and Description

The Oyster Creek Nuclear Generating Station (OCNGS) is located on the coastal pine barrens of New Jersey in Lacey and Ocean Townships, Ocean County. The station site owned by **AmerGen Energy Company, LLC.** U.S. Route 9 divides the property. There are 152 acres west of Route 9 and 708 acres to the east. The plant site is located to the west of Route 9, and is bounded on the north by the South Branch of Forked River and on the south by Oyster Creek. Barnegat Bay forms the eastern site boundary and the Garden State Parkway the western site boundary. Figure 2.1-1 is an aerial photograph of the OCNGS site and environs.

The power island of the OCNGS is situated approximately midway between Oyster Creek and the South Branch of Forked River and about 1400 feet west of Route 9. Route 9 provides access to the site.

Approximately 352 acres of land onsite were used during station construction. This includes 22 acres for the generating station and auxiliary facilities, 8.5 acres for the emergency fire pond, and 33.5 acres for railroad, transmission right-of-way, and spoil areas due to dredging of the South Branch of Forked River and Oyster Creek.

The site is approximately 35 miles north of Atlantic City, New Jersey and 45 miles east of Philadelphia, Pennsylvania. Approximately 9.5 miles north of the site are several small residential communities; Toms River, South Toms River, Beachwood, Pine Beach, Ocean Gate, Island Heights and Gilford Park.

West of the Garden State Parkway the land is undeveloped woodland, and wooded wetlands are found along the banks of small creeks to the north, south, and west of the site. East of the station along the shoreline of Barnegat Bay, the land is residentially developed for year round and seasonal use. The terrain surrounding the site is relatively flat along the shoreline to gently rolling inland. About 4 miles inland just west of the Garden State Parkway, the terrain rises to heights in the range of 65 to 90 feet above mean sea level.

A state game farm on which quail and pheasant are raised is located approximately 2.25 miles northeast of the station.

Effluent Release Limits

Some radioactive material is released from the plant under controlled conditions as part of the normal operation of the facility. Other radioactive material not normally intended for release could be inadvertently released in the event of an accident. Therefore, limits have been placed on the above types of radioactive materials to ensure that the limits of 10CFR20 which apply to releases during normal operation, and the limits of 10CFR100 which apply to accidental releases, are not exceeded. This applies to the restricted area, **exclusion area boundary and low population zone.** Effluent release limits are established in the Technical Specifications.

Radiation dose from liquid effluents may be received through the ingestion of fish, shellfish, and from direct exposure. Personal radiation exposure via other aqueous pathways is negligible. Barnegat Bay, which is approximately 2.5 miles east of station, is not a source for drinking water supplies. Radioactive material in liquid effluents are kept as low as is reasonably achievable, in compliance with 10CFR50, Appendix I.

2.1.2 Exclusion Area Authority and Control

The reactor (centerline) is located 1358 feet (approximately 0.25 mile) west of the **eastern** boundary of Route 9. **The** exclusion **area boundary** for the OCNGS **is defined as a 1358 foot (414 meter) radius extending from the reactor centerline as illustrated in Figure 2.1-7. A perimeter security fence encompasses the area immediately around the plant. Access to this area is controlled by a security force, in accordance with the requirements of 10 CFR 73. On December 29, 1981 the NRC issued Amendment 59 to the POL-DPR-16 for this facility, pursuant to which GPU Nuclear Inc. was added as a licensee authorized to possess, use and operate Oyster Creek, and Jersey Central Power and Light d/b/a GPU Energy remained as a licensee, authorized to possess the facility.** This area meets the requirements of 10CFR100 for exclusion area determination.

2.1.3 Population Distribution

The 1980 final census figures were used to determine the permanent

In 1990, two natural gas pipelines were installed in the vicinity of Oyster Creek Nuclear Generating Station (OCNGS). One pipeline is a 16 inch diameter line which runs parallel to Route 9 on the east side of the plant. This pipeline is outside the plant exclusion zone except where it crosses the intake and discharge canals. The other pipeline is a 16 inch line that branches off the main line at a point North of the plant. The branch line runs roughly adjacent to the north side of the intake canal to the combustion turbines owned by **GPUE**.

USNRC Systematic Evaluation Program (SEP) Topic II-1C for Oyster Creek evaluated the impact of a 6 inch and 8 inch diameter natural gas pipeline also located along Route 9 at one quarter mile from the plant and concluded that the pipelines do not pose a significant hazard to the plant due to the distance involved. While the newly installed gas lines are larger in diameter and were pressurized to higher pressures than those analyzed by NRC, it is reasonable to conclude that the primary factors which influenced NRC's conclusion of no hazard (i.e., distance from the plant and low probability of failure) would result in a similar conclusion for the new installation.

Also, NUREG 0014 comprising the USNRC safety assessment for the Construction of TVA Hartsville Nuclear Plants concluded that the existence of a pipeline in the vicinity represents no undue threat to the safe operation of the proposed facility and that accidents occurring to that pipeline need not be considered in the design of the plant.

This conclusion was based on the extensive research study performed for TVA, "Mechanic's Research Inc. Nuclear Power Plant Risks from a Natural Gas Pipeline, a Research Study Performed for TVA, August 1974".

The differences between the gasline at the TVA plant and the one at Oyster Creek are as follows:

- The distance from the gas line to the plant is approximately one half mile versus approximately one quarter mile at Oyster Creek.

- The diameter of the pipeline is 22 inches versus 16 inches at Oyster Creek.

- The working pressure is 720 psi versus 350 psi (up to 550 psi in the future, in the main line) at Oyster Creek.

As shown in "Mechanic's Research Inc. Nuclear Power Plant Risks from a Natural Gas Pipeline, a Research Study Performed for TVA, August 1974", (Risk Assessment Summary Table) the probability of a pipeline accident affecting the TVA facilities is of an order of magnitude of 10^{-7} or less. Taking the above differences into consideration, a qualitative judgement can be made that the USNRC conclusions listed in NUREG-0014 are applicable for the Oyster Creek facility.

OCNGS
FSAR UPDATE
Table 2.3-33
(Sheet 1 of 2)

STACK DISPERSION PARAMETERS FOR GROUND RELEASES
FORKED RIVER METEOROLOGICAL TOWER DATA
FOR THE YEARS 1989, 1990
(X/Q IN S/M^3)

Sector	805m 0.5 miles	3218m 2 miles	4827m 3 miles	6436m 4 miles	8045m 5 miles	16090m 10 miles	38120m 20 miles	48270m 30 miles	64360m 40 miles	80450m 50 miles
N	1.07E-05	9.75E-07	5.87E-07	4.03E-07	2.84E-07	1.13E-07	4.09E-08	3.09E-08	2.17E-08	1.74E-08
NNE	1.35E-05	1.28E-06	7.65E-07	5.25E-07	3.64E-07	1.44E-07	5.12E-08	3.83E-08	2.68E-08	2.14E-08
NE	1.46E-05	1.40E-06	8.38E-07	5.76E-07	4.01E-07	1.59E-07	5.68E-08	4.24E-08	2.96E-08	2.37E-08
ENE	1.32E-05	1.24E-06	7.44E-07	5.12E-07	3.57E-07	1.42E-07	5.07E-08	3.80E-08	2.66E-08	2.12E-08
E	1.36E-05	1.27E-06	7.62E-07	5.25E-07	3.68E-07	1.47E-07	5.28E-08	3.95E-08	2.77E-08	2.21E-08
ESE	1.28E-05	1.17E-06	7.02E-07	4.83E-07	3.38E-07	1.35E-07	4.85E-08	3.64E-08	2.56E-08	2.05E-08
SE	1.39E-05	1.27E-06	7.63E-07	5.26E-07	3.70E-07	1.48E-07	5.36E-08	4.03E-08	2.84E-08	2.27E-08
SSE	1.17E-05	1.06E-06	6.41E-07	4.42E-07	3.13E-07	1.25E-07	4.55E-08	3.43E-08	2.41E-08	1.93E-08
S	1.03E-05	9.29E-07	5.62E-07	3.88E-07	2.75E-07	1.10E-07	4.02E-08	3.05E-08	2.15E-08	1.72E-08
SSW	6.20E-06	5.66E-07	3.39E-07	2.32E-07	1.62E-07	6.39E-08	2.30E-08	1.73E-08	1.21E-08	9.71E-09
SW	7.64E-06	6.95E-07	4.15E-07	2.84E-07	1.95E-07	7.68E-08	2.75E-08	2.06E-08	1.44E-08	1.15E-08
WSW	5.64E-06	5.29E-07	3.13E-07	2.13E-07	1.45E-07	5.65E-08	1.99E-08	1.48E-08	1.03E-08	8.20E-09
W	5.14E-06	4.66E-07	2.80E-07	1.92E-07	1.34E-07	5.30E-08	1.91E-08	1.44E-08	1.01E-08	8.09E-09
WNW	4.47E-06	4.02E-07	2.41E-07	1.66E-07	1.16E-07	4.60E-08	1.66E-08	1.25E-08	8.82E-09	7.05E-09
NW	5.46E-06	4.93E-07	2.95E-07	2.03E-07	1.41E-07	5.59E-08	2.02E-08	1.52E-08	1.07E-08	8.54E-09
NNW	7.92E-06	7.14E-07	4.29E-07	2.95E-07	2.07E-07	8.23E-08	2.99E-08	2.26E-08	1.59E-08	1.28E-08

Table 2.3-33

(Sheet 2 of 2)

STACK DISPERSION PARAMETERS FOR GROUND RELEASES
FORKED RIVER METEOROLOGICAL TOWER DATA
FOR THE YEARS 1989, 1990
(D/Q IN M-2)

Sector	805m 0.5 miles	3218m 2 miles	4827m 3 miles	6436m 4 miles	8045m 5 miles	16090m 10 miles	38120m 20 miles	48270m 30 miles	64360m 40 miles	80450m 50 miles
N	1.23E-08	1.12E-09	5.10E-10	3.27E-10	2.29E-10	6.75E-11	1.54E-11	9.98E-12	5.74E-12	3.74E-12
NNE	1.98E-08	1.80E-09	8.23E-10	5.28E-10	3.70E-10	1.06E-10	2.48E-11	1.61E-11	9.26E-12	6.03E-12
NE	1.67E-08	1.52E-09	6.95E-10	4.45E-10	3.12E-10	8.94E-11	2.09E-11	1.36E-11	7.82E-12	5.09E-12
ENE	1.59E-08	1.45E-09	6.64E-10	4.26E-10	2.98E-10	8.54E-11	2.00E-11	1.30E-11	7.47E-12	4.86E-12
E	1.60E-08	1.46E-09	6.66E-10	4.27E-10	2.99E-10	8.57E-11	2.00E-11	1.30E-11	7.49E-12	4.88E-12
ESE	1.97E-08	1.80E-09	8.22E-10	5.27E-10	3.70E-10	1.06E-10	2.47E-11	1.61E-11	9.25E-12	6.03E-12
SE	1.89E-08	1.72E-09	7.88E-10	5.05E-10	3.54E-10	1.01E-10	2.37E-11	1.54E-11	8.87E-12	5.77E-12
SSE	1.25E-08	1.14E-09	5.22E-10	3.35E-10	2.35E-10	6.71E-11	1.57E-11	1.02E-11	5.87E-12	3.82E-12
S	9.02E-09	8.21E-10	3.76E-10	2.41E-10	1.69E-10	4.83E-11	1.13E-11	7.35E-12	4.23E-12	2.75E-12
SSW	6.80E-09	6.20E-10	2.83E-10	1.82E-10	1.27E-10	3.65E-11	8.52E-12	5.54E-12	3.19E-12	2.08E-12
SW	1.08E-08	9.80E-10	4.48E-10	2.87E-10	2.02E-10	5.77E-11	1.35E-11	8.77E-12	5.04E-12	3.28E-12
WSW	9.66E-09	8.80E-10	4.02E-10	2.58E-10	1.81E-10	5.18E-11	1.21E-11	7.87E-12	4.53E-12	2.95E-12
W	6.53E-09	5.94E-10	2.72E-10	1.74E-10	1.22E-10	3.50E-11	8.18E-12	5.32E-12	3.06E-12	1.99E-12
WNW	5.30E-09	4.83E-10	2.21E-10	1.42E-10	9.93E-11	2.84E-11	6.64E-12	4.32E-12	2.48E-12	1.62E-12
NW	7.76E-09	7.06E-10	3.23E-10	2.07E-10	1.45E-10	4.16E-11	9.72E-12	6.32E-12	3.63E-12	2.37E-12
NNW	8.63E-09	7.86E-10	3.59E-10	2.30E-10	1.62E-10	4.62E-11	1.08E-11	7.03E-12	4.04E-12	2.63E-12

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Table 2.3-34

(Sheet 1 of 2)

STACK DISPERSION PARAMETERS FOR ELEVATED RELEASES
FORKED RIVER METEOROLOGICAL TOWER DATA
FOR THE YEARS 1989, 1990
(X/Q IN S/M^3)

Sector	805m 0.5 miles	3218m 2 miles	4827m 3 miles	6436m 4 miles	8045m 5 miles	16090m 10 miles	38120m 20 miles	48270m 30 miles	64360m 40 miles	80450m 50 miles
N	1.01E-08	2.05E-08	1.77E-08	1.45E-08	1.18E-08	5.88E-09	2.27E-09	1.73E-09	1.24E-09	9.52E-10
NNE	9.69E-09	2.51E-08	2.28E-08	2.08E-08	1.69E-08	8.32E-09	3.13E-09	2.37E-09	1.69E-09	1.29E-09
NE	7.52E-09	1.38E-08	1.40E-08	1.26E-08	1.09E-08	6.07E-09	2.54E-09	1.97E-09	1.44E-09	1.13E-09
ENE	1.42E-08	1.42E-08	1.34E-08	1.16E-08	9.91E-09	5.34E-09	2.22E-09	1.72E-09	1.26E-09	9.86E-10
E	1.53E-08	1.51E-08	1.38E-08	1.19E-08	1.00E-08	5.36E-09	2.23E-09	1.74E-09	1.28E-09	1.00E-09
ESE	2.27E-08	1.85E-08	1.61E-08	1.35E-08	1.12E-08	5.65E-09	2.25E-09	1.73E-09	1.26E-09	9.77E-10
SE	2.56E-09	1.87E-08	1.64E-08	1.38E-08	1.14E-08	5.89E-09	2.37E-09	1.83E-09	1.34E-09	1.04E-09
SSE	1.35E-08	1.29E-08	1.18E-08	1.01E-08	8.44E-08	4.42E-09	1.79E-09	1.39E-09	1.01E-09	7.90E-10
S	5.36E-09	1.27E-08	1.13E-08	9.44E-09	7.83E-09	4.05E-09	1.65E-09	1.27E-09	9.33E-10	7.27E-10
SSW	5.44E-09	1.85E-08	2.04E-08	1.51E-08	1.18E-08	5.50E-09	2.03E-09	1.53E-09	1.09E-09	8.35E-10
SW	1.19E-08	2.48E-08	2.45E-08	1.78E-08	1.36E-08	6.27E-09	2.25E-09	1.69E-09	1.20E-09	9.09E-10
WSW	1.56E-08	1.91E-08	1.93E-08	1.44E-08	1.12E-08	5.66E-09	1.97E-09	1.47E-09	1.03E-09	7.80E-10
W	1.30E-08	1.58E-08	1.59E-08	1.33E-08	1.03E-08	5.38E-09	1.88E-09	1.40E-09	9.90E-10	7.50E-10
WNW	1.13E-08	1.89E-08	1.51E-08	1.27E-08	9.76E-09	5.04E-09	1.78E-09	1.34E-09	9.44E-10	7.16E-10
NW	1.30E-08	2.39E-08	1.92E-08	1.44E-08	1.13E-09	5.74E-09	2.01E-09	1.50E-09	1.06E-09	8.02E-10
NNW	9.43E-09	1.82E-08	1.75E-08	1.36E-08	1.08E-08	5.06E-09	1.87E-09	1.41E-09	1.01E-09	7.67E-10

Table 2.3-34
(Sheet 2 of 2)

STACK DISPERSION PARAMETERS FOR ELEVATED RELEASES
FORKED RIVER METEOROLOGICAL TOWER DATA
FOR THE YEARS 1989, 1990
(D/Q IN M-2)

Sector	805m 0.5 miles	3218m 2 miles	4827m 3 miles	6436m 4 miles	8045m 5 miles	16090m 10 miles	38120m 20 miles	48270m 30 miles	64360m 40 miles	80450m 50 miles
N	5.18E-09	5.54E-10	2.78E-10	1.86E-10	1.39E-10	6.81E-11	2.06E-11	1.31E-11	7.62E-12	4.86E-12
NNE	7.06E-09	7.90E-10	3.99E-10	2.76E-10	2.13E-10	1.14E-10	3.50E-11	2.21E-11	1.26E-11	7.87E-12
NE	3.80E-09	4.19E-10	2.25E-10	1.78E-10	1.56E-10	1.05E-10	3.36E-11	2.10E-11	1.18E-11	7.25E-12
ENE	4.80E-09	4.98E-10	2.61E-10	1.93E-10	1.60E-10	9.70E-11	3.07E-11	1.94E-11	1.10E-11	6.90E-12
E	5.40E-09	5.52E-10	2.87E-10	2.06E-10	1.65E-10	9.53E-11	2.99E-11	1.90E-11	1.09E-11	6.88E-12
ESE	8.52E-09	8.56E-10	4.35E-10	2.94E-10	2.22E-10	1.11E-10	3.42E-11	2.20E-11	1.29E-11	8.28E-12
SE	8.25E-09	8.26E-10	4.20E-10	2.83E-10	2.13E-10	1.06E-10	3.26E-11	2.10E-11	1.23E-11	7.94E-12
SSE	4.26E-09	4.45E-10	2.30E-10	1.64E-10	1.31E-10	7.43E-11	2.32E-11	1.47E-11	8.43E-12	5.31E-12
S	2.90E-09	3.23E-10	1.65E-10	1.18E-10	9.38E-11	5.35E-11	1.66E-11	1.04E-11	5.91E-12	3.68E-12
SSW	2.98E-09	3.38E-10	1.67E-10	1.09E-10	7.96E-11	3.70E-11	1.09E-11	6.94E-12	3.99E-12	2.53E-12
SW	5.83E-09	6.45E-10	3.15E-10	1.98E-10	1.37E-10	5.45E-11	1.56E-11	1.00E-11	5.91E-12	3.84E-12
WSW	5.65E-09	6.01E-10	2.94E-10	1.83E-10	1.26E-10	4.79E-11	1.37E-11	8.90E-12	5.35E-12	3.53E-12
W	3.59E-09	3.75E-10	1.85E-10	1.17E-10	8.20E-11	3.35E-11	9.78E-12	6.36E-12	3.80E-12	2.50E-12
WNW	2.88E-09	3.05E-10	1.50E-10	9.52E-11	6.66E-11	2.72E-11	7.90E-12	5.12E-12	3.05E-12	1.99E-12
NW	4.07E-09	4.24E-10	2.10E-10	1.35E-10	9.56E-11	4.07E-11	1.20E-11	7.78E-12	4.63E-12	3.02E-12
NNW	4.07E-09	4.34E-10	2.16E-10	1.41E-10	1.02E-10	4.67E-11	1.39E-11	8.93E-12	5.23E-12	3.37E-12

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3.1.42 Criterion 46 - Testing of Cooling Water System

Criterion

The cooling water system shall be designed to permit appropriate periodic pressure and functional testing. **This testing** assures:

a. the structural and leaktight integrity of its components,

b. the operability and the performance of the active components of the system, and,

c. the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation for **both** reactor shutdown and Loss-of-Coolant Accidents. **This** includes operation of applicable portions of the protection system and the transfer between normal and emergency power sources.

Discussion

The Reactor Building Closed Cooling Water System is normally in operation. **Testing of the system includes the closure timing of containment isolation valves, inservice testing of the pumps and valves, and calibration of the local pump discharge pressure gauges.** The Shutdown Cooling System has been designed to permit periodic testing. The testing provisions of the Containment Spray and Emergency Service Water System are covered under GDC 40.

The Reactor Building Closed Cooling Water System and the Shutdown Cooling System are described in Sections 9.2 and 5.4, respectively.

3.1.43 Criterion 50 - Containment Design Basis

Criterion

The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and, with sufficient margin, the calculated pressure and temperature conditions resulting from any Loss-of-Coolant Accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and energy from metal-water and other chemical reactions that may result from degraded emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.

Discussion

The Primary Containment structure, including access openings and penetration, was designed to withstand the peak accident pressure and temperatures that could occur during the postulated design basis Loss-of-Coolant Accident. The Containment design includes considerable allowance for energy addition from metal-water or other chemical reactions beyond those that could occur during the accident. The integrity of the complete Containment was designed, and will be maintained, to limit offsite doses from postulated design basis accidents to a value below the guideline values stated in 10CFR100.

The Primary Containment structural design is discussed in detail in Section 3.8. The basis for defining the functional capability of the Containment Systems are discussed in Section 6.2. Table 3.1-1 summarizes the SEP topics which address GDC 50. Section 1.10 provides a summary of the program.

3.1.44 Criterion - Fracture Prevention of Containment Pressure Boundary

Criterion

The reactor containment boundary shall be designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions (1) its ferritic materials behave in a non-brittle manner, (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining (1) material properties, (2) residual, steady stage and transient stresses, and (3) size of flaws.

Discussion

The Primary Containment was designed to the version of the ASME Boiler and Pressure Vessel Code Section VIII, plus nuclear code cases, in effect at the time of design (C.1965). The containment vessel is designed to be at a minimum temperature of NDT + 30°F during any plant condition at which the Primary Containment could be pressurized.

For further discussion, refer to Sections 3.8 and 6.2.

A Turbine Missile Hazard will not affect the safety function of the intake canal due to the following:

a) As delineated in Section 3.5.1.3 of the FSAR, the failure of turbines to the extent of producing missiles is highly unlikely.

b) The CT's have metal enclosures which will reduce the velocity of the missiles, if produced.

c) If a partial blockage should occur due to the effects of a turbine missile, a lesser percentage of the total canal area can supply the cooling water requirements. A total blockage due to a turbine missile is not considered possible. Also, due to the existence of redundant Emergency Service Water pumps on the intake structure, it is not considered credible for more than one set of pumps to be affected by a missile.

3.5.1.6 <u>Aircraft Hazards</u>

An evaluation of aircraft accidents for the OCNGS is presented in Reference . Further evaluation of aircraft hazards is presented as part of the Systematic Evaluation Program in Section 1.10.

3.5.2 <u>Structures, Systems and Components to be Protected from Externally Generated Missiles</u>

The Class I (Seismic) Structures for the OCNGS include the following:

a. Reactor Building exterior concrete walls

b. Reactor Building insulated metal siding

c. Reactor Building roof decking

d. Reactor Building steel for craneway enclosure

e. Control Room walls

f. Ventilation Stack

g. Battery Room (interior room)

h. Diesel Generator and Fuel Oil Tank Vaults

Generally, safety related equipment is enclosed in the listed Class I structures. An analysis of missile protection for safety related equipment was performed as part of the SEP. This program is discussed in detail in Section 1.10.

3.5.3 References

(1) Thullen - "Loads on Spherical Shells," Chicago Bridge & Iron Co.., Oak Brook, Illinois, August 1964.

(2) Not Used.

(3) Not Used.

(4) Application for License, Quad Cities Units 1 and 2, Docket No. 50-254/265, Amendment 3.

(5) Ibid, Amendment 4.

(6) Application for License, Browns Ferry, Amendment 6, Question C-8.

(7) "An Analysis of Turbine Missiles Resulting from Last Stage Wheel Failure," GE Topical Report TR67SL211.

(8) NUREG-0822, Integrated Plant Safety Assessment, Systematic Evaluation Program. Oyster Creek Nuclear Generating Station, Docket No. 50-219, Final Report. January 1983.

(9) OCNGS Procedure 2000-ABN-3200.33, "Toxic Material/Flammable Gas Release - No Radiation Involved".

(10) Letter from G. E. (Martin O'Connor) to GPUN (Frank Collado), dated June 7, 1996.

(11) LES Calculation No. 72-01-01, Turbine Missile Analysis for New Monoblock Rotor and Blades," October, 1996. Revision 3.

(12) Journal of the Structural Division, "Assessment of Empirical Concrete Impact Formulas," By George E. Sliter, Dated May 1980, page 1035-1036.

(13) Letter from GE Power Generation (George Reluzco) to General Electric Company (J. Hess), Dated March 28, 1996.

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4

Two of the corner rooms have pump suction pipes running through the lower area of the entrance doorways. These doorways are sealed by encircling the suction pipe in a steel plate that extends across the doorway and by water tight, bulkhead doors above these steelplates. The doors are designed for a head of 20 feet. For simplicity in design and procurement, the remaining two corner room doorways are similarly sealed with the steel plate, water tight door system outlined above.

The sealed piping penetrations are the pump suction lines: two run through the aforementioned corner room doorways and the other two penetrate through the corner room walls. The penetration seal in all cases will be of 1/8 inch steel fabrication with a flexible "U" joint in the horizontal direction to allow for expansion. The "U" joint is welded on one end to a ring attached to the suction piping and on the other end to a sleeve through the wall or door plate. The penetrations leaving the corner rooms through their outer walls are already sealed as part of secondary containment requirements.

Check valves are provided in the sumps and floor drains of both the corner rooms and the inner torus room the prevent backflow into any of these rooms.

3.8.4.1.2 Control Room (and Supporting Part of Turbine Building)

The Control Room is located on the northeast corner of the Turbine Building Operating Floor at El. 46'-6". The Cable Rooms serving the controls are located immediately above and below the Control Room at El. 36'-0" and El. 63'-9". The mechanical equipment (HVAC) for the Control Room is housed immediately above the Control Room at elevation 63'-9", in the Upper Cable Room.

The rooms are supported by heavy reinforced concrete members which make up part of the Turbine Building structure. Walls for the rooms are reinforced concrete not less than 18 inches thick. The roof of the Upper Cable Room is reinforced concrete. An overhead crane supported on structural steel framework traverses the roof above this room.

General arrangement drawings of the Control Room are shown on the Turbine Building drawings 3E-151-02-006 through 3E-151-02-008.

3.8.4.1.3 Spent Fuel Pool

The Spent Fuel Pool is a rectangular tank like structure, lined with a stainless steel sheet and constructed of massive reinforced concrete walls and bottom slab. **The pool liner is ¼" thick on the bottom of the pool, the north wall, and at the opening to the reactor cavity. The liner is a nominal 1/8" thick on the remaining walls.** The pool is located immediately north of the drywell with its top being at the refueling floor at El. 119'-3". The massive walls are build integral with the drywell concrete walls. The plan dimensions of the pool are normally 39 feet in the east-west direction and 27 feet in the north-south direction. The bottom of the pool is at El. 80'-6" making the pool 38 feet 9 inches deep. In addition to being attached to the drywell the pool structure is built monolythic with the remainder of the Reactor Building concrete structure and is supported above the foundation mat, at El. (-) 19'-6", by concrete columns.

General arrangement drawings of the Spent Fuel Pool are shown on Drawings 3E-153-02-006 through 3E-153-02-008.

3.8.4.1.4 Ventilation Stack

The 394 foot reinforced concrete stack (368 feet above grade) is linked by tunnels to the Reactor Building, Turbine Building, and both Radwaste Buildings. These tunnels contain piping and air ducts between the buildings and the stack. The off-gas piping is buried; it is not run through the tunnels.

The top of the stack foundation mat is at El.(-)3'-0". Floors at El. 23'-6" and El. 35'-0" are reached through the heating boiler house next to the stack. Off-gas piping enters the stack at El. 0'-0", and penetrates both floors. Absolute filters in the larger of the two pipes are housed below grade in a controlled area accessible only through the tunnel. The differential pressure indicator and isolating valve are located above the floor at El. 23'-6". Exhaust fans for the Reactor Building ventilating ducts are located outdoors at grade level and discharge to the stack above the second floor level. The tunnels are accessible through a stairway to a concrete block enclosed entrance at grade level directly west of the stack.

3.8.4.1.5 Emergency Diesel Generator Building

The Emergency Diesel Generator and Diesel Oil Storage Tank Building is located southwest of the Turbine and Pretreatment Buildings, at the bend in the plant road and east of the discharge canal.

The building is constructed of reinforced concrete with a 16 foot 0 inch by 16 foot 9 inch appendage for the diesel oil storage tank and two 17 foot 6 inch by 72 foot 6 inch compartments for housing the diesel generators. The walls are 18 inches thick with additional 4 inch by 4 inch welded wire fabric reinforcement of the inside face of all exterior walls.

The foundation slab for the diesel generator compartments is 24 inches thick with the top coinciding with the finished grade at El. 23'-0". The tank compartment foundation slab is 24 inches thick with the top located at El. 18' 4", 4 feet 8 inches below finished grade. Sumps and drain lines are included in the foundation slab for drainage purposes. Conduit exits through the north end of the diesel generator foundation slab into buried conduit banks.

3.11 ENVIRONMENTAL DESIGN OF INSTRUMENTATION AND ELECTRICAL EQUIPMENT

The Environmental Qualification (EQ) Program provides assurance that specific electrical equipment will perform its intended safety function. Specifically, the objectives of the EQ Program are:

a. Document the qualification of electrical equipment important to safety as required by 10CFR50.49.

b. Establish the maintenance/surveillance required to maintain the qualification of this electrical equipment over the life of the plant.

3.11.1 Equipment Identification and Environmental Conditions

3.11.1.1 Identification of Electrical Equipment

3.11.1.1.1 Criteria for Selection of Equipment

The EQ program addresses all electrical equipment important to safety as defined in 10CFR50.49(b)(1),(2), and (3). The EQ Master List, GPUN document No. 990-1464, identifies electrical equipment or components which must be environmentally qualified for use in a harsh environment. Equipment important to safety which is exposed only to a mild environment during postulated accident conditions is not included in the EQ Program.

3.11.1.1.2 Class 1E Equipment and Interfaces

A detailed review of each system (Figure 3.11-1) was performed to identify all of the systems' electrical components and instrumentation in accordance with 10CFR50.49. This was performed by listing each major electrical component (motor operator, pump motor, instrument, etc.) and then identifying the auxiliary electrical equipment within the circuit to the component (cable, terminal blocks, splices, etc.). The identification of the major electrical equipment was carried out through a review of the electrical one lines, elementary wiring diagrams, Technical Specifications, Emergency Operating Procedures, (as needed to support 10CFR50.49 Program requirements) and Process and Instrumentation Diagrams.

Certain Class 1E equipment was classified as commodity items. A variety of plant walkdowns were performed to provide reasonable assurance that qualification attributes were accurate.

3.11.1.1.3 NUREG-0737 and Regulatory Guide 1.97 Equipment

Supplement 1 to NUREG 0737 requires that certain post accident monitoring instrumentation be provided to enable operators to assess plant and environmental conditions during and following an accident. The post accident monitoring instrumentation is selected using the guidance provided by ANSI/ANS 4.5-1980 as endorsed by Regulatory Guide 1.97.

3.11.1.1.4 EQ Position on DOR Guidelines, NUREG-0588 and R. G. 1.89

All equipment within the scope of this program has been evaluated for compliance with either the DOR Guidelines NUREG-0588, Category I, or 10CFR50.49 with guidance from Regulatory Guide 1.89.

Oyster Creek was an operating plant when the DOR Guidelines were issued in November 1979. Therefore, installed equipment was required to meet the requirement of the DOR Guidelines.

Replacement parts must be qualified to 10CFR50.49 except where there are determined acceptable "Sound Reasons to the Contrary" as defined in Regulatory Guide 1.89, R1, Section C, paragraph 6, subparagraphs "a" through "g".

3.11.1.2 Environmental Conditions

The environmental parameters for each plant area were determined for both normal and accident service conditions. These parameters are documented in Reference 3.11.6.3.

The plant environmental conditions applicable to a specific piece of equipment can be obtained **from the EQ files identified on the EQ Master List**.

3.11.1.2.1 Normal Service Conditions

The plant normal service conditions include all aspects of normal operation, including all levels of power operation, shutdown condition, cold shutdown, or refuel mode as defined by the Technical Specifications and any other normally anticipated operational occurrence (which includes a loss of HVAC).

The normal service conditions for a specific component are given in Reference 3.11.6.3 and encompass the applicable temperature, pressure humidity, and radiation conditions postulated to occur at the specific equipment location during normal operation of the plant. The methodology used to define the normal service conditions are described below.

a. Temperature/Pressure

The temperatures inside the drywell were obtained from measurements taken during normal operation. The temperatures in the various rooms in the reactor and turbine buildings were based on calculated average annual temperatures for those areas of the plant which are ventilated by outside air under normal operating conditions. Atmospheric pressure (14.7 psia) is assumed for all areas outside the drywell and drywell pressure is assumed to be 16 psia.

b. Humidity

The humidity inside the drywell was obtained from measurements taken during normal operation (use of 50% is deemed conservative). For areas outside the drywell, the humidity can vary reaching a maximum of 100%. Thus, a value of 100% RH has been used unless a lower value has been justified and documented. **Humidity variations during normal operation will not be evaluated in accordance with 10CFR50.49(e)(2).**

c. Radiation

Normal power operation general area radiation dose rates for the drywell, reactor and turbine buildings are taken from health physics dose rate surveys as well as from predicted normal operating dose rates.

The radiation dose is integrated over a 40 year plant operating time.

3.11.1.2.2 Accident Service Conditions

The development of accident service conditions considered the environmental conditions resulting from a postulated Loss-of-Coolant-Accident (LOCA), Main Steam Line Break (MSLB) inside the drywell, and High Energy Line Break (HELBS) outside of the drywell in the reactor and turbine buildings.

The analyses of these postulated accidents address the following environmental parameters:

a. Temperature/Pressure

b. Humidity

c. Chemical Spray

d. Submergence

e. Radiation

The specific analyses performed and their results are discussed in greater detail below.

a. Temperature/Pressure

A description of the analyses performed to determine the containment temperature and pressure response to a LOCA is found in Reference 3.11.6.3. The main steam line break provides the most severe containment temperature and the DBA LOCA provides the most severe containment pressure response. The resulting time dependent temperature and pressure profiles which are used in the Oyster Creek environmental qualification program are shown in Figures 3.11-2 and 3.11-3 (Reference 3.11.6.3).

A High Energy Line Break (HELB) can produce a harsh environment in the reactor and turbine buildings. The peak temperatures and pressures are based on the worst case HELB affecting a specific EQ zone.

Temperature profiles were generated for five double ended guillotine pipe breaks in the areas where the environmental response for electrical equipment would be most severe for equipment qualification (Reference 3.11.6.3). The pipe breaks are as follows:

	System	Break Size and Location
1.	Main Steam	24" main steam pipe within the steam tunnel at elevation 23'-6" in Reactor Building
2.	Cleanup and Demineralizer	6" pipe to Cleanup Auxiliary Pump at Elevation 51'-3" in Reactor Building.
3.	Emergency Condenser	16" emergency condenser pipe break at elevation 75'-3" in Reactor Building.
4.	Reactor Feedwater	14" reactor feed line break in the feedwater pump room elevation 14'-11" in Turbine Building basement floor.
5.	Main Steam	24" main steam pipe at elevation 23'-6" in Turbine Building mezzanine floor.

b. Humidity

Environmental qualification of equipment during the above mentioned postulated accidents is based on 100 percent RH for the drywell, reactor and turbine buildings, and the standby gas treatment system tunnel. A lesser value may be used, when justification is developed for a given component and included in the EQ file for that component.

c. Chemical Spray

The demineralized water spray inside the drywell is considered to have an insignificant effect on metallic or non-metallic components. However, the moisture intrusion effects of a water spray have been considered in the qualification process.

d. Submergence

The plant areas which could be submerged during postulated accident conditions are in the reactor building, **drywell** and the steam tunnel. For all other plant areas, the design of floor drains will prevent submergence of electrical equipment. There is no electrical equipment in the EQ Program which is submerged.

e. Radiation

For equipment qualification purposes, the accident radiation conditions postulated to occur result from a non isolable pipe break in the reactor coolant system inside of containment and were developed based upon the bounding requirements of the DOR Guidelines of NUREG-0588, as applicable to the component being qualified.

The accident radiation doses, gamma and beta (if applicable) were integrated over the duration of the accident (which is usually taken as 1 year). This is a conservative approach if the required component operating time is appreciably less than the radiation integration time (e.g., hours versus months, respectively).

3.11.1.2.3 Equipment Operability Time

The operability duration is the length of time during and following an accident that equipment must maintain its ability to perform its intended safety function. The safety function includes:

1. The ability to initiate short term protective action.

2. The ability to place the plant in a controlled condition.

3. The ability to keep the plant in a stable condition after the accident until personnel are able to enter the plant to inspect, repair, or replace equipment.

Operating time may be defined on a case by case basis. However, most components requiring environmental qualification were those devices required to perform one of the functions delineated in Table 3.11-1 suitable for the equipment to accomplish its intended safety function (Reference 3.11.6.2 as modified by the Technical Specifications Section No. 5.2.)

3.11.2 Qualification Tests and Analyses

3.11.2.1 Acceptance Criteria

Electrical equipment was evaluated to ensure that it will function as required after exposure to its normal and postulated accident environments. All qualification conforms to the requirements of either the DOR Guidelines, or NUREG 0588-Category 1 (IEEE 323-1974).

3.11.2.1.1 Accident Environments

Each piece of equipment entered into the Oyster Creek EQ Program was evaluated to determine if it would function as required during exposure to postulated accident conditions. The components need only be qualified to the accident parameters of the accidents the component is required to mitigate, for the time period it is required to function. The accident parameters are specified in individual EQ files which are identified on the EQ Master List.

3.11.2.1.2 Margins

Equipment within the scope of the Oyster Creek EQ Program was qualified to accident environmental profiles which enveloped the plant parameters. The conservatisms included in these profiles are judged to be sufficient to account for uncertainties associated with the analytical techniques, definitions of performance requirements, and variations in commercial production.

3.11.2.1.3 Connection Interfaces

Equipment exposed to steam conditions coincident with pressure is provided with a seal **where required.**

3.11.2.1.4 Performance Specifications

The EQ Program includes an evaluation of equipment to ensure that performance specifications are achieved under conditions existing during and following postulated accidents. This evaluation reviews functional requirements, and/or loop accuracy based upon function, location, environment and performance requirement.

3.11.2.1.5 Voltage and Frequency

Safety related electrical equipment is subject to variations in power supply characteristics such as voltage and frequency. For the AC distribution system, these are comprised of the expected off-site power supply variations, including degraded grid conditions, and the expected variations of the diesel generator if off-site power has been lost. These conditions are addressed in the **EQ files for the specific equipment**.

3.11.2.1.6 Synergistic Effects/Phase Changes

The equipment qualification effort did consider synergisms/phase changes to the extent identified as follows:

a. If the vendor identified a synergistic effect/phase change, it was evaluated.

b. If the reviewer was aware of a synergistic effect/phase change, it was evaluated. As additional synergistic effect/phase change data became available it was evaluated and factored into the program.

c. If neither a. nor b. existed, then no further actions were taken to determine if any synergistic effect/phase change were known (e.g. literature research).

3.11.2.1.7 Field Verification

a. Walkdowns

Adequate field inspection of as-installed Class 1E equipment was performed by walkdowns. The field inspection, provided a reasonable assurance that there is: (a) a traceable link between the equipment installed at the plant and the equipment that was qualified, (b) a direct verification that any special installation requirements identified in the qualification program were applied, and (c) a verification that external gaskets, seals, protective covers, etc. have been installed.

3.11.3 Qualification Test Results

3.11.3.1 Documentation

The qualification documentation for electrical equipment are assembled into EQ files. An EQ file is prepared for a group of components having the same manufacturer but different plant tag numbers. The EQ file contains all necessary reports, analyses, and correspondence submitted by the vendor to satisfy Purchase Order (P.O.) requirements, thereby establishing a direct link between plant installed equipment and providing the basis for environmental qualification in accordance with 10CFR50.49.

3.11.3.2 Independent Verification

Each EQ file is **prepared and verified** to ensure the completeness and accuracy of the data presented. The results of this review are documented in Environmental Qualification files, as necessary, based upon the requirements of the DOR Guidelines or NUREG 0588-Category I.

3.11.4 Loss of Ventilation

A loss of ventilation is considered a possible normal operational occurrence and, therefore, does not establish a harsh environmental condition. Loss of HVAC is not considered a design basis accident for environmental qualification purposes, unless the loss of HVAC is a direct consequence of a design basis accident. Therefore, the interpretation of 10CFR50.49, as it applies to loss of HVAC is that, "an area whose environment results from normal plant operation or an anticipated operational occurrence is still considered, and should be classified, as a mild environment". (Reference 3.11.6.3)

3.11.5 Estimated Chemical and Radiation Environment

3.11.5.1 Chemical Spray

The Containment Spray System is described in OCFSAR Section 6.2.2.

3.11.5.2 Radiation

The accident radiation conditions postulated are from a non-isolable pipe break in the reactor coolant system inside of containment, and were developed based upon the bounding requirements of the DOR Guidelines of NUREG-0588.

The accident radiation doses, gamma and beta (if applicable) were integrated over the duration of the accident (which is usually taken as 1 year). This is a conservative approach if the required component operating time is appreciably less than the radiation integration time (e.g., hours versus months, respectively).

3.11.5.2.1 Inside Drywell - General Areas

The percent of core inventory assumed to be released from the fuel for a LOCA meets the NUREG-0588 requirements of:

100 percent of the Noble Gas Core Inventory; 50 percent of the Iodine Core Inventory; and, 1 percent of the other nuclides in the Core Inventory.

a. Gamma Dose

The source term, basic assumptions and model used to develop the total gamma dose radiation service condition for Class 1E equipment located in general areas inside the drywell are described in Reference 3.11.6.3. An estimated value of 5.0 x 10E7 RADs for the total airborne gamma dose in the drywell (1 year integrated dose from airborne iodine and airborne noble gases).

b. Beta Dose

The source term, basic assumptions and model used to develop the total beta radiation dose for Class 1E equipment located in general areas inside of the drywell are described in Reference 3.11.6.3.

An estimated value of 9.6 x 10E8 RADs for the total airborne beta dose in the drywell (1 year integrated dose from airborne iodine and airborne noble gases has been utilized). Per the DOR Guidelines, electrical cable is considered to be the most vulnerable to damage from beta radiation from the general classes of electrical equipment. The beta surface dose was reduced by a factor of 10 within 30 mils of the surface of electrical cable insulation, therefore, for cables with 30 mils or greater insulation, beta radiation is 9.6 x 10E7 RADs.

The above radiation service conditions inside the drywell are applicable to equipment in the drywell vapor space. A calculation was not performed for equipment submerged in the torus fluids as none exist.

3.11.5.2.2 Outside Primary Containment

Radiation values outside containment conservatively assumed a 100% fuel failure even though for HELB's outside containment, the radiation exposure will be significantly less due to isolation of the line break and expected minimum fuel damage.

Reference 3.11.6.3 defines the calculated general area radiation maps and radiation exposures at specified equipment locations. The analysis included the effects of radiation shine from the drywell, airborne radioactivity due to leakage from the primary containment and gamma doses due to recirculation piping and components. The analysis calculated beta radiation and gamma radiation.

3.11.6 References

NRC Letter LS05-85-05-031 to P.B. Fiedler, dated May 28, 1985, "Safety Evaluation for Final Resolution of Environmental Qualification of Electric Equipment Important to Safety".

(1) NRC Letter LS05-85-05-031 to P.B. Fedler, dated May 28, 1985, "Safety Evaluation for Final Resolution of Environmental Qualification of Electric Equipment Important to Safety."

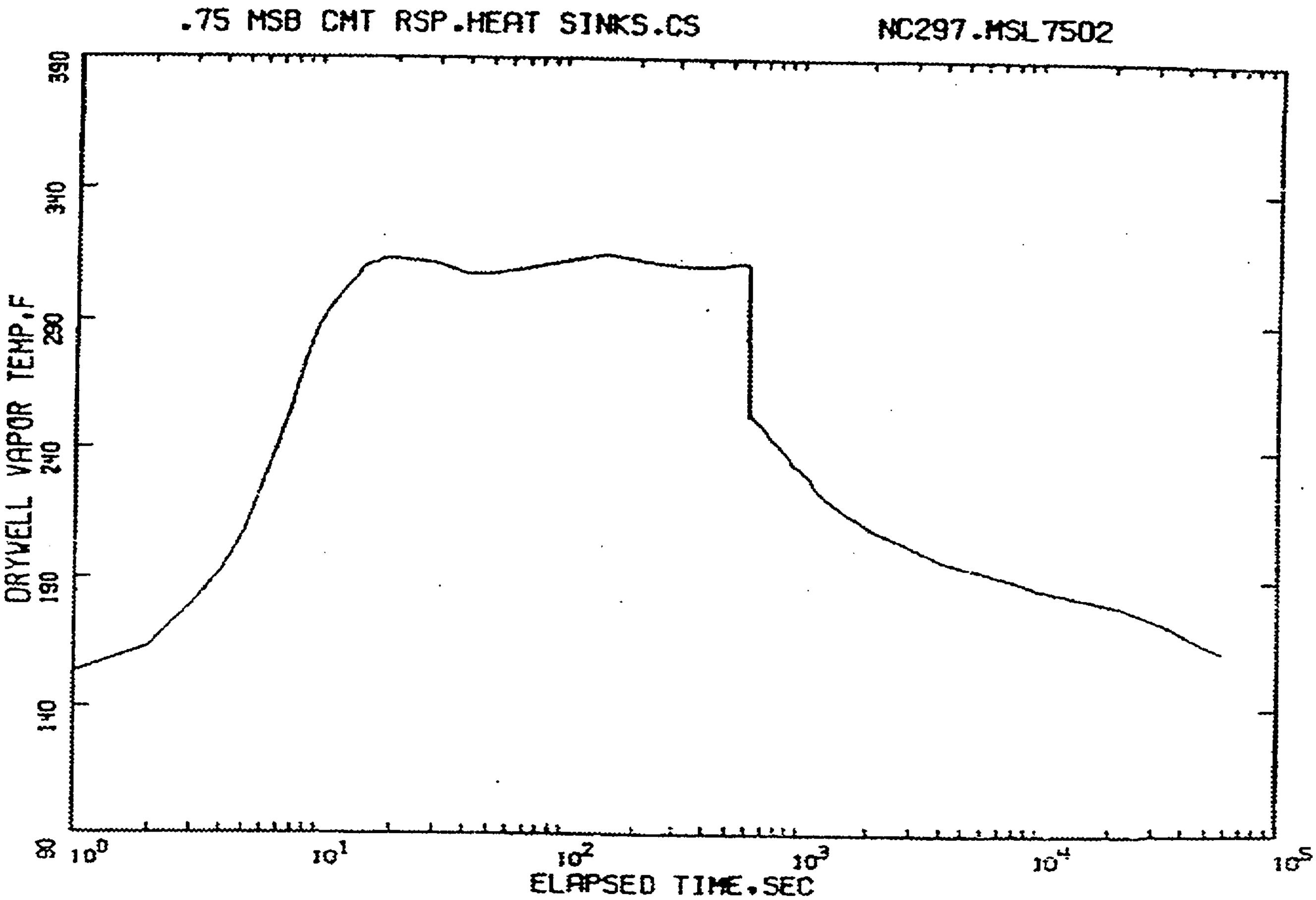
(2) Impell Report 02-0370-1293 Rev. 0, dated December 14, 1984, "Identification of Safety Related Equipment".

(3) ES-027, GPUN Technical Functions Standard "Environmental Parameters - Oyster Creek NGS".

(4) USNRC Letter to GPUN, dated August 8, 1986, "Inspection Report No. 05000219/86-08.

(5) TR-028 Rev. 3, dated October 12, 1992, "OC Response to USNRC R.G. 1.97".

DBA TEMPERATURE PROFILE, DRYWELL



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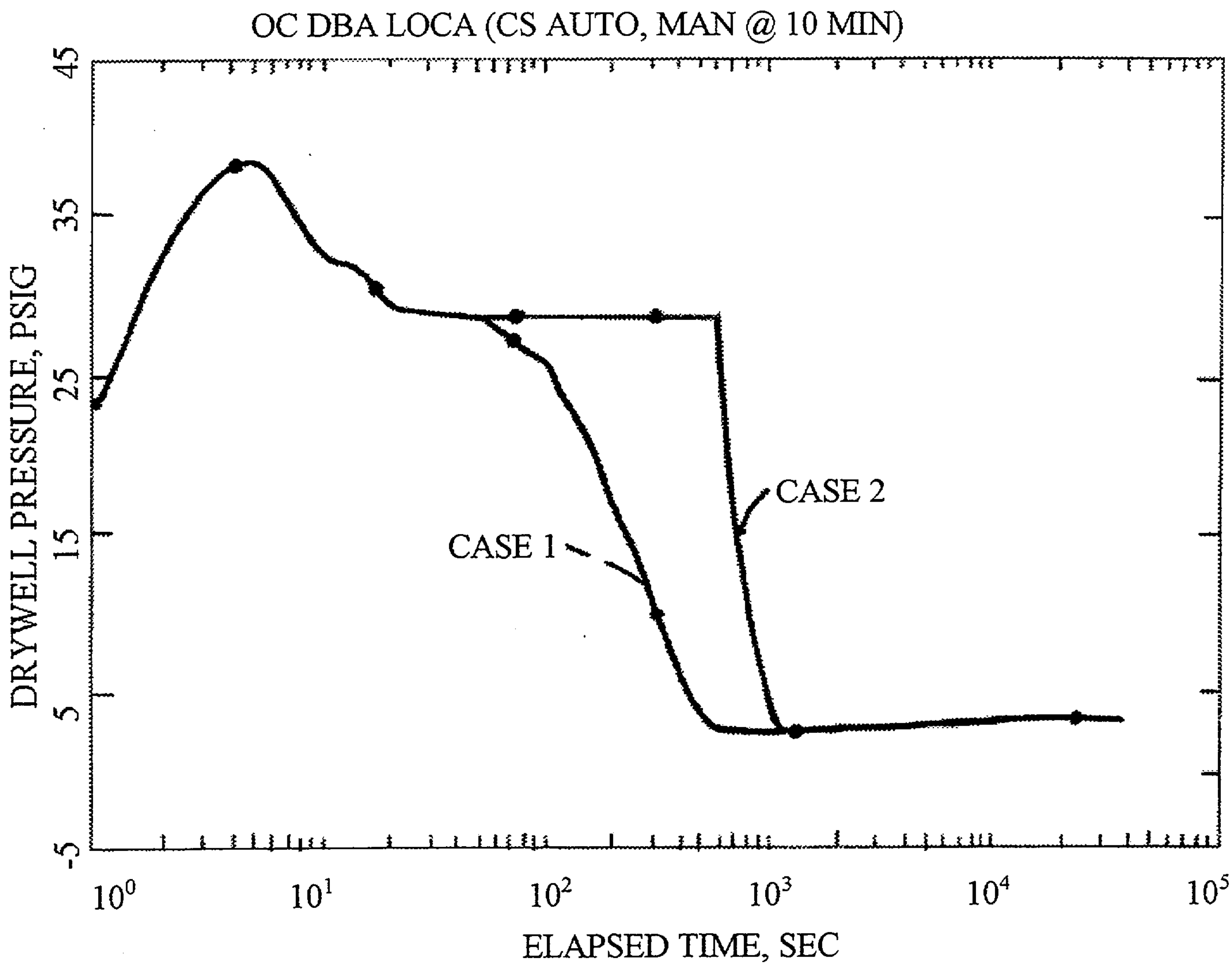
OYSTER CREEK NUCLEAR GENERATING STATION

EQ Temperature Profile
Main Steam Line Break

FIGURE 3.11-2

DBA PRESSURE PROFILE, DRYWELL



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OYSTER CREEK NUCLEAR GENERATING STATION

EQ Pressure Profile
Containment Spray Line Break
C-1302-241-E610-082

FIGURE 3.11-3

Figure 3.11-3A

(Deleted)

Figure 3.11-3B

(Deleted)

Figure 3.11-3C

(Deleted)

TABLE 4.3-1
(Sheet 1 of 3)

TYPE OF FUEL UTILIZED IN EACH FUEL CYCLE

Fuel Cycle	Fuel Type						
	GE I (7x7)	GE II (7x7)	ENC III (7x7)	ENC III-E (7x7)	ENC III-F (7x7)	ENC V (8x8)	ENC V-B (8x8)
Cycle 1A 5/3/69 - 9/18/71	560	--	--	--	--	--	--
Refueling Change	(-24)	(+24)					
Cycle 1B 11/11/71 - 5/1/72	536	24	--	--	--	--	--
Refueling Change	(-136)	(+132)	(+4)				
Cycle 2 6/20/72 - 4/13/73	400	156	4	--	--	--	--
Refueling Change	(-148)			(+148)			
Cycle 3 6/4/73 - 4/13/74	252	156	4	148	--	--	--
Refueling Change	(-68)	(-4)			(+72)		
Cycle 4 7/1/74 - 3/29/75	184	152	4	148	72	--	--
Refueling Change	(-112)				(+36)		
Cycle 5 5/25/75 - 12/27/75	72	152	4	148	108	4	72
Refueling Change	(-56)						(+56)
Cycle 6 3/10/76 - 4/23/77	16	152	4	148	108	4	128
Refueling Change	(-16)	(-108)	(-4)				(+128)
Cycle 7 8/1/77 - 9/16/78	--	44	--	148	108	4	256
Refueling Change		(-44)		(-84)	(-40)		(+168)
Cycle 8 12/5/78 - 1/5/80	--	--	--	64	68	4	424
Refueling Change				(-64)	(-44)		(-152) (+160)

TABLE 4.3-1
(Sheet 2 of 3)

TYPE OF FUEL UTILIZED IN EACH FUEL CYCLE

Fuel Type

Fuel Cycle	ENC III-F (7x7)	ENC V (8x8)	ENC V-B (8x8)	GE P8DRB239 (8x8)	GE P8DRB265H(8x8)	GE P8DRB299ZA (8x8)	GE P8DR299Z (8x8)
Cycle 9 7/15/80 - 2/12/83	24	4	532	--	--	--	--
Refueling Change	(-24)	(-4)	(-144)	(+112)	(+60)		
Cycle 10	--	--	388	112	60	--	--
Refueling Change			(-188)	--	(+4)	(+48)	(+136)

Fuel Cycle	ENC V-B (8x8)	GE P8DRB239 (8x8)	GE P8DRB265H (8x8)	GE P8DRB299ZA (8x8)	GE P8DR299Z (8x8)	GE P8DRB321(EB) (8x8)	GE P8DQB338-12Gz (8x8)	GE P8DQB338-11GZ (8x8)
Cycle 11	200	112	64	48	136	--	--	--
Refueling Change	(-171)	(-1)	--	--	(+20)	(+152)	--	--
Cycle 12	29	111	64	48	156	152	--	--
Ref2ueling Change 2	(17)	(-87)	(-36)	--	(-4)	(+68)	(+60)	(+16)
C2ycle 13	12	24	28	48	152	220	60	16
Re2fueling Change	(-12)	(-24)	(-28)	(-48)	(-56)	--	(+132)	(+36)
Cycle 14	--	--	--	--	96	220	192	52
Refeuling Change					(-96)	(076)	(+4)	(+8)

Fuel Cycle	GE P8DRB321(EB) (8x8)	GE P8DQB338-12GZ (8x8)	GE P8DQB338-11GZ (8x8)	GE P8DWB338-11GZ (8x8)	GE P8DWB348-12GZ (8x8)
Cycle 15	144	196	60	48	112
Refueling Change	(-144)	(-36)	(-8)	(+40)	(+148)
Cycle 16	--	160	52	88	260
Refueling Change		(-140)	(-44)	(+40)	(+144)

TABLE 4.3-1
(Sheet 3 of 3)

TYPE OF FUEL UTILIZED IN EACH FUEL CYCLE

Fuel Cycle	GE P8DQB338-12GZ (8x8)	GE P8DQB338-11GZ (8x8)	GE P8DWB338-11GZ (8x8)	GE P8DWB348-12GZ (8x8)
Cycle 17	**20**	**8**	**128**	**404**
Refueling Change	**(-20)**	**(-8)**	**(-48)** **(+48)**	**(-108)** **(+136)**
Cycle 18	--	--	**128**	**432**

TABLE 4.3-2
(Sheet 1 of 1)

REPRESENTATIVE REACTIVITY COEFFICIENTS AND NUCLEAR PARAMETERS

Coefficient		Reload Cycle 18
Void Coefficient at Core Average Voids, (Delta k/k)/(percent void)		
	BOC	**-1.15×10^{-3}**
	EOC	**-1.26×10^{-3}**
Fuel Temperature (Doppler) Coefficient (Delta k/k)/F		
	BOC	**-1.84×10^{-5}**
	EOC	**-2.25×10^{-5}**
Parameter		
Delayed Neutron Fraction		
	BOC	**0.00633**
	EOC	**0.00540**
*, Neutron Lifetime, Microseconds		
	BOC	**35.4**
	EOC	**38.1**

TABLE 4.3-3
(Sheet 1 of 1)

CALCULATED CORE EFFECTIVE MULTIPLICATION FACTOR, CONTROL SYSTEM WORTH AND SHUTDOWN MARGIN

	Cycle 18
^{k}critical (with uncertainty)	**0.9910**
k_{eff} (cold-68 F) Beginning of Cycle	
Uncontrolled*	**1.1071**
Fully Controlled	**0.9578**
Strongest Control Rod Withdrawn	**0.9792**
R, Maximum Increase in Cold Core Reactivity With Exposure Into Cycle, Delta k	**0.0000**
Minimum Shutdown Margin, Delta k	**0.0118**

* All controls rods out of the core.

Details of considerations when determining cladding surface temperature are given in Reference 3. The model used to calculate the fuel cladding temperature is documented in Reference 1. Fuel cladding temperature as a function of heat flux for the P8x8R design is shown in Figure 4.4-1 for the beginning of life conditions and in Figure 4.4-2 for late in life conditions.

4.4.2.2 Critical Power Ratio

There are three different types of boiling heat transfer for water in a forced convection system: nucleate boiling, transition boiling, and film boiling. Nucleate boiling, at lower heat fluxes, is an extremely efficient mode of heat transfer, allowing large quantities of heat to be transferred with a very small temperature rise at the heated wall. As power is increased, the boiling heat transfer surface alternates between film and nucleate boiling, leading to fluctuations in heated wall temperatures. The bundle power at the point of departure from the nucleate boiling region into the transition boiling region anywhere in the bundle is called the critical power. Transition boiling begins at the critical power, and is characterized by fluctuations in cladding surface temperature. Film boiling occurs at the highest heat fluxes; it begins as transition boiling comes to an end. Film boiling heat transfer is characterized by stable wall temperatures which are higher than those experienced during nucleate boiling.

The objective for normal operation and transient events is to maintain nucleate boiling and thus avoid a transition to film boiling. Operating limits are specified to maintain adequate margin to the onset of the boiling transition. The parameter used for core design and monitoring is the critical power ratio. This is defined as the ratio of the critical power (bundle power at which some point within the assembly experiences onset of boiling transition) to the operating bundle power. The critical power is determined at the same mass flux, inlet temperature and pressure which exists at the specified reactor condition. Thermal margin is stated in terms of the minimum value of the critical power ratio (MCPR), which corresponds to the most limiting fuel assembly in the core. To ensure that adequate margin is maintained, a design requirement based on a statistical analysis was selected. This requirement states that moderate frequency transients caused by a single operator error or equipment malfunction shall be limited such that, considering uncertainties in manufacturing and monitoring the core operating state, more than 99.9 percent of the fuel rods would be expected to avoid boiling transition (Reference 12). Both the transient (safety) thermal limit and normal operating thermal limit for the fuel in terms of MCPR are derived from this basis.

The MCPR operating limit is the minimum CPR allowed by the Technical Specifications for a given bundle type. The CPR is a function of several parameters, the most important of which are bundle power, bundle flow and local power distribution. The operating limit of CPR is selected for each bundle type such that during the most limiting event of moderate frequency, the CPR in that bundle will not be less than the safety limit CPR. The MCPR is attained when the bundle power, bundle flow and other relevant parameters combine to yield the Technical Specifications value.

For historical perspective, it is to be noted that the design criteria for evaluating the thermal design margin was revised from one using critical heat flux (CHF) to one using critical power (CP) at the time of the OCNGS cycle 5 reload. This revision in criteria was established in Supplement 4 of Reference 7 which provides an introduction to the evaluation of critical power and a critical power criteria to determine the operating margin to boiling transition.

A fuel cladding integrity safety limit of 1.07 was established for General Electric (GE) P8x8R fuel for the Oyster Creek Cycle 10 reload. The safety limit has been conservatively applied to the GE 8x8EB and GE 8x8NB fuel designs based on an NRC approved generic safety limit of 1.06 for these fuel designs (Reference 3).

During Oyster Creek's Cycle 15 operation, GE issued a 10 CFR Part 21 notification stating that the generic safety limit may be non-conservative for GE fuel designs. GE performed a cycle specific safety limit evaluation and determined that a 1.07 safety limit is acceptable for the GE fuel designs in the Cycle 15 core loading. Since Oyster Creek was using a 1.07 safety limit, no action was necessary. However, a cycle specific safety limit must be determined for each reload.

A 1.09 safety limit was calculated for the Oyster Creek Cycle 16 core loading and a 1.08 safety limit was calculated for **both** cycle 17 **and 18** although the 1.09 safety limit was retained in the technical specifications. The new safety limit is calculated using NRC approved methods (Reference 3) and includes interim procedures for application to plant and cycle specific evaluations which were reviewed by the NRC. These interim procedures have been revised to incorporate cycle specific parameters which include: 1) actual core loading; 2) conservative variations of projected control blade patterns; 3) the actual bundle parameters (e.g., local peaking); and 4) the fuel cycle exposure range. The cycle specific parameters were calculated using GPU Nuclear approved methods and provided to GE for the safety limit calculation.

The GEXL plus (Reference 3) critical power correlation is used to calculate transient delta CPR values for any design basis transient or accident where CPR is the limiting parameter. The minimum CPR operating limit ensures that the fuel cladding integrity safety limit is not violated for these transients or accidents. The largest delta CPR transient for Oyster Creek is the Turbine Trip without Bypass Transient and this transient delta CPR establishes the operating limit minimum CPR. Representative delta CPR values for different transients are given in Table 4.4-3.