NRCPDR

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

DEC 27 1978

Project No. P-657

Mr. A. E. Kintigh
Vice President - Generation
New York State Electric and Gas Corporation
4500 Vestal Parkway East
Binghamton, New York 13902

Dear Mr. Kintigh:

SUBJECT: ACCEPTANCE REVIEW OF APPLICATION FOR CONSTRUCTION PERMITS AND OPERATING LICENSES FOR NYSE&G 1 AND 2

On November 22, 1978, you tendered an application for construction permits and operating licenses for the NYSE&G 1 and 2. Your application included the general information required by 10 CFR 50.33, the Preliminary Safety Analysis Report (PSAR), the Preliminary Physical Security Plan and the Environmental Report.

We have completed our review of your tendered application and have concluded it is acceptable for docketing, with the exception of the Environmental Report. The acceptance review of the Environmental Report is still in progress and will be the subject of separate correspondence.

Accordingly, you should provide for docketing, three (3) originals signed under oath or affirmation by a duly authorized officer of your organization. In addition, the submittal should include forty (40) copies of the Preliminary Safety Analysis Report and (15) copies of the General Information section. As required by Section 50.30 of 10 CFR Part 50, you should retain an additional thirty (30) copies of the Preliminary Safety Analysis Report and ten (10) copies of the General Information section for direct distribution in accordance with Enclosure 1 to this letter and further instructions which might be provided later. Within 10 days after docketing, you must provide an affidavit that distribution has been made in accordance with Enclosure 1. All subsequent amendments to the Preliminary Safety Analysis Report will require sixty (60) copies for distribution. The distribution affidavit requirement also applies to any subsequent amendments to your application.

Our conclusion that the PSAR is sufficiently complete is based on our evaluation of all the information filed taken as a whole, with the recognition that substantive deficiencies exist that need to be corrected during the review.

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During the course of our acceptance review we encountered difficulties with the information provided in the areas of hydrology and instrumentation and control. We met with your representatives to discuss these matters on December 15, 1978. Our concerns in the hydrology area are contained in Enclosure 2, Request for Additional Information. We are continuing to develop our need for information in the area of instrumentation and control and will forward those questions in the near future. Our concerns in other areas are also contained in Enclosure 2. You should provide a schedule for responding to our Request for Additional Information. We will then prepare our schedule for the review of NYSE&G. You will be advised of key milestones as soon as the schedule is developed.

If during the course of our review, you believe there is a need to appeal a staff position because of disagreement, this need should be brought to the staff's attention as early as possible so that the appropriate meeting can be arranged on a timely basis. A written request is not necessary and all such requests should be initiated through our staff project manager assigned to the review of your application, Robert Capra. This procedure is an informal one, designed to allow opportunity for applicants to discuss, with management, areas of disagreement in the case review.

Sincerely,

Roger S. Boyd, Director
Division of Project Management
Office of Nuclear Reactor Regulation

Enclosures:
As Stated

cc w/enclosures:
See page 3

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ENCLOSURE 1

NYSE&G 1 & 2
PSAR & CP APPLICATION DISTRIBUTION

	PSAR	CP APPLICATI
Mr. Gordon Schipper Supervisor, Town of New Haven Box 82 New Haven, New York 13121	(1)	
Mr. Herbert Van Schaack Chairman, Oswego County Legislature 46 East Bridge Street Oswego, New York 13126	(1)	
T. K. DeBoer, Director Technological Development Program State of New York Energy Office Agency Building #2 Empire State Plaza Albany, New York 12223	(1)	(1)
Ms. Barbara Metzger U. S. Environmental Protection Agency Region 2 Office 26 Federal Plaza New York, New York 10007	(1)	
Director of Nuclear Reactor Regulation Nuclear Regulatory Commission 1717 H Street, N. W. Washington, D. C.	(15)	(15)
Huber Magill Lawrence & Farrell 99 Park Avenue New York, New York 10016	(2)	(2)
Charles A. Zielinski Chairman, NYS Public Service Commission Empire State Plaza Agency Building Three Albany, New York 12223	(5)	

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ENCLOSURE 2

REQUEST FOR ADDITIONAL INFORMATION

NEW HAVEN NUCLEAR STATION, UNITS 1 & 2

PROJECT NO. P-557

010.0 AUXILIARY SYSTEMS BRANCH

010.1 (10.4.5) Your evaluation of the effects of an expansion joint failure at the main condenser is incomplete. Expand your evaluation to include the following:

(1) The maximum flow rate through a completely failed expansion joint.

(2) For the postulated failure, give the rate of rise of water in the associated spaces and total height of the water when the circulating water flow has been stopped or overflows to site grade (assuming a 30 minute operator delay time before isolation occurs).

(3) For each flooded space provide information, with the aid of drawings if necessary, to demonstrate the adequacy of the protection provided for all safety related systems that could become affected as a result of flooding. Include a discussion of the consideration given to passageways, pipe chases and/or the cableways joining the flooded space to the spaces in the annulus building containing safety related system components.

022.0 CONTAINMENT SYSTEMS BRANCH

022.1 (9.4.5.2.6) In the Safety Evaluation Report for SWESSAR-P1/CESSAR (NUREG-0096), Section 6.2.5 contains a discussion of containment purging. It is our position that the subject of containment purging during normal plant operation is an interface matter that must be addressed by the utility-applicant referencing SWESSAR-P1. Two options are provided, namely, (1) declare that containment purging dur[illegible] operation is not necessary, in which case an on-line purge system need not be provided; or (2) provide an on-line purge system, using the two spare containment penetrations included in the SWESSAR-P1 containment design for that purpose. The latter option would require conformance with our Branch Technical Position CSB-4, "Containment Purging During Normal Plant Operations."

In Section 6.2.5 of the SER, "normal operation" is defined as the reactor operating modes of hot standby, reactor startup, and power operation, and containment purging would be allowed during the hot shutdown, cold shutdown and refueling modes. However, it should be noted that Standard Review Plan 6.2.4 has been revised to include the hot shutdown mode in the definition of normal operation. In essence the on-line purge system must conform to BTP CSB 6-4 if containment purging is anticipated during any reactor operating mode requiring containment integrity.

Section 9.4.5.2.6 of the NYSE&G PSAR states that containment purging during normal operations will not be required. However in light of our position stated above, discuss your position with regard to the installation of an on-line containment purge system.

If you find that use of the purge system will be needed during normal operation, then you must propose an on-line purge system design and provide the information and analyses identified in Branch Technical Position CSB 6-4, for our review. (A copy of Standard Review Plan 6.2.4 - Rev. 1, which includes the Branch Technical Position, is enclosed).

022.2 (6.2.1, 6.2.5.7.1) Clarify the following:

(1) In Figure 6.2-1 of the NYSE&G PSAR, the abscissa of the graph is not numbered and the curves are not identified.

(2) Section 6.2.5.7.1 of the NYSE&G PSAR states that the hydrogen recombiner specification will be in accordance with Table 6.2-4 of SWESSAR-P1. It appears that Table 6.2.5-1 should have been referenced.

U.S. NUCLEAR REGULATORY COMMISSION
STANDARD REVIEW PLAN
OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 6.2.4 CONTAINMENT ISOLATION SYSTEM

REVIEW RESPONSIBILITIES

Primary - Containment Systems Branch (CSB)

Secondary - Accident Analysis Branch (AAB)
Instrumentation and Control System Branch (ICSB)
Mechanical Engineering Branch (MEB)
Structural Engineering Branch (SEB)
Reactor Systems Branch (RSB)
Power Systems Branch (PSB)

I. AREAS OF REVIEW

The design objective of the containment isolation system is to allow the normal or emergency passage of fluids through the containment boundary while preserving the ability of the boundary to prevent or limit the escape of fission products that may result from postulated accidents. This SRP section, therefore, is concerned with the isolation of fluid systems which penetrate the containment boundary, including the design and testing requirements for isolation barriers and actuators. Isolation barriers include valves, closed piping systems, and blind flanges.

The CSB reviews the information presented in the applicant's safety analysis report (SAR) regarding containment isolation provisions to assure conformance with the requirements of General Design Criteria 54, 55, 56 and 57. The CSB review covers the following aspects of containment isolation:

1. The design of containment isolation provisions, including:

   a. The number and location of isolation valves, i.e., the isolation valve arrangements and the physical location of isolation valves with respect to the containment.

   b. The actuation and control features for isolation valves.

   c. The positions of isolation valves for normal plant operating conditions (including shutdown), post-accident conditions, and in the event of valve operator power failures.

   d. The valve actuation signals.

   e. The basis for selection of closure times of isolation valves.

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f. The mechanical redundancy of isolation devices.

g. The acceptability of closed piping systems inside containment as isolation barriers.

2. The protection provided for containment isolation provisions against loss of function from missiles, pipe whip, and earthquakes.

3. The environmental conditions inside and outside the containment that were considered in the design of isolation barriers.

4. The design criteria applied to isolation barriers and piping.

5. The provisions for detecting a possible need to isolate remote-manual-controlled systems, such as engineered safety features systems.

6. The design provisions for and technical specifications pertaining to operability and leakage rate testing of the isolation barriers.

7. The calculation of containment atmosphere released prior to isolation valve closure for lines that provide a direct path to the environs.

PSB has primary responsibility for the qualification test program for electric valve operators, and the ICSB has primary responsibility for the qualification test program for the sensing and actuation instrumentation of the plant protection system located both inside and outside of containment. The MEB has review responsibility for the qualification test program to demonstrate the performance and reliability of containment isolation valves. The MEB and SEB have review responsibility for mechanical and structural design of the containment isolation provisions to ensure adequate protection against missiles, pipe whip, and earthquakes. The AAB reviews the radiological dose consequence analysis for the release of containment atmosphere prior to closure of containment isolation valves in lines that provide a direct path to the environs. The RSB reviews the closure time for containment isolation valves in lines that provide a direct path to the environs, with respect to the prediction of onset of accident induced fuel failure.

II. ACCEPTANCE CRITERIA

The general design criteria establish requirements for isolation barriers in lines penetrating the primary containment boundary. In general, two isolation barriers in series are required to assure that the isolation function is satisfied assuming any single active failure in the containment isolation provisions.

The design of the containment isolation provisions will be acceptable to CSB if the following criteria are satisfied:

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1. General Design Criteria 55 and 56 require that lines that penetrate the primary containment boundary and either are part of the reactor coolant pressure boundary or

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connect directly to the containment atmosphere should be provided with isolation valves as follows:

a. One locked closed isolation valve[1/] inside and one locked closed isolation valve outside containment, or

b. One automatic isolation valve inside and one locked closed isolation valve outside containment; or

c. One locked closed isolation valve inside and one automatic isolation valve[2/] outside containment; or

d. One automatic isolation valve inside and one automatic isolation valve[2/] outside containment.

2. General Design Criterion 57 requires that lines that penetrate the primary containment boundary and are neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere should be provided with at least one locked closed, remote-manual, or automatic isolation valve[2/] outside containment.

3. The general design criteria permit containment isolation provisions for lines penetrating the primary containment boundary that differ from the explicit requirements of General Design Criteria 55 and 56 if the basis for acceptability is defined. Following are guidelines for acceptable alternate containment isolation provisions for certain classes of lines:

a. Regulatory Guide 1.11 describes acceptable containment isolation provisions for instrument lines. In addition, instrument lines that are closed both inside and outside containment, are designed to withstand the pressure and temperature conditions following a loss-of-coolant accident, and are designed to withstand dynamic effects, are acceptable without isolation valves.

b. Containment isolation provisions for lines in engineered safety features or engineered safety feature-related systems may include remote-manual valves, but provisions should be made to detect possible leakage from these lines outside containment.

c. Containment isolation provisions for lines in systems needed for safe shutdown of the plant (e.g., liquid poison system, reactor core isolation cooling system, and isolation condenser system) may include remote-manual valves, but provision should be made to detect possible leakage from these lines outside containment.

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1/Locked closed isolation valves are defined as sealed closed barriers (see item II.3.f).

2/A simple check valve is not normally an acceptable automatic isolation valve for this application.

d. Containment isolation provisions for lines in the systems identified in items b and c normally consist of one isolation valve inside and one isolation valve outside containment. If it is not practical to locate a valve inside containment (for example, the valve may be under water as a result of an accident), both valves may be located outside containment. For this type of isolation valve arrangement, the valve nearest the containment and the piping between the containment and the valve should be enclosed in a leak-tight or controlled leakage housing. If, in lieu of a housing, conservative design of the piping and valve is assumed to preclude a breach of piping integrity, the design should conform to the requirements of SRP section 3.6.2. Design of the valve and/or the piping compartment should provide the capability to detect leakage from the valve shaft and/or bonnet seals and terminate the leakage.

e. Containment isolation provisions for lines in engineered safety feature or engineered safety feature-related systems normally consist of two isolation valves in series. A single isolation valve will be acceptable if it can be shown that the system reliability is greater with only one isolation valve in the line, the system is closed outside containment, and a single active failure can be accommodated with only one isolation valve in the line. The closed system outside containmant should be protected from missiles, designed to seismic Category I standards, classified Safety Class 2 (Ref. 5), and should have a design temperature and pressure rating at least equal to that for the containment. The closed system outside containment should be leak tested, unless it can be shown that the system integrity is being maintained during normal plant operations. For this type of isolation valve arrangement the valve is located outside containment, and the piping between the containment and the valve should be enclosed in a leak tight or controlled leakage housing. If, in lieu of a housing, conservative design of the piping and valve is assumed to preclude a breach of piping integrity, the design should conform to the requirements of SRP section 3.6.2. Design of the valve and/or the piping compartment should provide the capability to detect leakage from the valve shaft and/or bonnet seals and terminate the leakage.

f. Sealed closed barriers may be used in place of automatic isolation valves. Sealed closed barriers include blind flanges and sealed closed isolation valves which may be closed manual valves, closed remote-manual valves, and closed automatic valves which remain closed after a loss-of-coolant accident. Sealed closed isolation valves should be under administrative control to assure that they cannot be inadvertently opened. Administrative control includes mechanical devices to seal or lock the valve closed, or to prevent power from being supplied to the valve operator.

g. Relief valves may be used as isolation valves provided the relief set point is greater than 1.5 times the containment design pressure.

4. Isolation valves outside containment should be located as close to the containment as practical, as required by General Design Criteria 55, 56, and 57.

5. The position of an isolation valve for normal and shutdown plant operating conditions and post-accident conditions depends on the fluid system function. If a fluid system does not have a post-accident function, the isolation valves in the lines should be automatically closed. For engineered safety feature or engineered safety feature-related systems, isolation valves in the lines may remain open or be opened. The position of an isolation valve in the event of power failure to the valve operator should be the "safe" position. Normally this position would be the post-accident valve position. All power-operated isolation valves should have position indication in the main control room.

6. There should be diversity in the parameters sensed for the initiation of containment isolation.

7. System lines which provide an open path from the containment to the environs should be equipped with radiation monitors that are capable of isolating these lines upon a high radiation signal. A high radiation signal should not be considered one of the diverse containment isolation parameters.

8. Containment isolation valve closure times should be selected to assure rapid isolation of the containment following postulated accidents. The valve closure time is the time it takes for a power operated valve to be in the fully closed position after the actuator power has reached the operator assembly; it does not include the time to reach actuation signal setpoints or instrument delay times, which should be considered in determining the overall time to close a valve. System design capabilities should be considered in establishing valve closure times. For lines which provide an open path from the containment to the environs; e.g., the containment purge and vent lines, isolation valve closure times on the order of 5 seconds or less may be necessary. The closure times of these valves should be established on the basis of minimizing the release of containment atmosphere to the environs, to mitigate the offsite radiological consequences, and assure that emergency core cooling system (ECCS) effectiveness is not degraded by a reduction in the containment backpressure. Analyses of the radiological consequences and the effect on the containment backpressure due to the release of containment atmosphere should be provided to justify the selected valve closure time. Additional guidance on the design and use of containment purge systems which may be used during the normal plant operating modes (i.e., startup, power operation, hot standby and hot shutdown) is provided in Branch Technical Position CSB 6-4 (Ref. 9). For plants under review for operating licenses or plants for which the Safety Evaluation Report for construction permit application was issued prior to July 1, 1975, the methods described in Section B, Items B.1., a, b, d, e, f, and g, B.2 through B.4, and B.5.b, c, and d of Branch Technical Position 6-4 should be implemented. For these plants, BTP Items B.1.c and B.5.a, regarding the size of the purge system used during normal plant operation and the justification by acceptable dose consequence analysis, may be

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waived if the applicant commits to limit the use of the purge system to less than 90 hours per year while the plant is in the startup, power, hot standby and hot shutdown modes of operations. This commitment should be incorporated into the Technical Specifications used in the operation of the plant.

9. The use of a closed system inside containment as one of the isolation barriers will be acceptable if the design of the closed system satisfies the following requirements:

   a. The system does not communicate with either the reactor coolant system or the containment atmosphere.

   b. The system is protected against missiles and pipe whip.

   c. The system is designated seismic Category I.

   d. The system is classified Safety Class 2 (Ref. 5).

   e. The system is designed to withstand temperatures at least equal to the containment design temperature.

   f. The system is designed to withstand the external pressure from the containment structural acceptance test.

   g. The system is designed to withstand the loss-of-coolant accident transient and environment.

   Insofar as CSB is concerned with the structural design of containment internal structures and piping systems, the protection of isolation barriers against loss of function from missiles, pipe whip, and earthquakes will be acceptable if isolation barriers are located behind missile barriers, pipe whip was considered in the design of pipe restraints and the location of piping penetrating the containment, and the isolation barriers, including the piping between isolation valves, are designated seismic Category I, i.e., designed to withstand the effects of the safe shutdown earthquake, as recommended by Regulatory Guide 1.29.

10. The design criteria applied to components performing a containment isolation function, including the isolation barriers and the piping between them, or the piping between the containment and the outermost isolation barrier, are acceptable if:

   a. Group B quality standards, as defined in Regulatory Guide 1.26 are applied to the components, unless the service function dictates that Group A quality standards be applied.

   b. The components are designated seismic Category I, in accordance with Regulatory Guide 1.29.

11. The design of the containment isolation system is acceptable if provisions are made to allow the operator in the main control room to know when to isolate fluid systems that are equipped with remote manual isolation valves. Such provisions may include instruments to measure flow rate, sump water level, temperature, pressure, and radiation level.

12. Provisions should be made in the design of the containment isolation system for operability testing of the containment isolation valves and leakage rate testing of the isolation barriers. The isolation valve testing program should be consistent with that proposed for other engineered safety features. The acceptance criteria for the leakage rate testing program for containment isolation barriers are presented in SRP section 6.2.6.

For those areas of review identified in subsection I of this SRP section as being the responsibility of other branches, the acceptance criteria and their methods of application are contained in the SRP sections corresponding to those branches.

III REVIEW PROCEDURES

The procedures described below provide guidance on review of the containment isolation system. The reviewer selects and emphasizes material from the review procedures as may be appropriate for a particular case. Portions of the review may be done on a generic basis for aspects of containment isolation common to a class of containments, or by adopting the results of previous reviews of plants with essentially the same containment isolation provisions.

Upon request from the primary reviewer, the secondary review branches will provide input for the areas of review stated in subsection I. The primary reviewer obtains and uses such input as required to assure that this review procedure is complete.

The CSB determines the acceptability of the containment isolation system by comparing the system design criteria to the design requirements for an engineered safety feature. The quality standards and the seismic design classification of the containment isolation provisions, including the piping penetrating the containment, are compared to Regulatory Guides 1.26 and 1.29, respectively.

The CSB also ascertains that no single fault can prevent isolation of the containment. This is accomplished by reviewing the containment isolation provisions for each line penetrating the containment to determine that two isolation barriers in series are provided, and in conjunction with the PSB by reviewing the power sources to the valve operators.

The CSB reviews the information in the SAR justifying containment isolation provisions which differ from the explicit requirements of General Design Criteria 55, 56 and 57. The CSB judges the acceptability of these containment isolation provisions based on a comparison with the acceptance criteria given in subsection II.

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The CSB reviews the position of isolation valves for normal and shutdown plant operating conditions, post-accident conditions, and valve operator power failure conditions as listed in the SAR. The position of an isolation valve for each of the above conditions depends on the system function. In general, power-operated valves in fluid systems which do not have a post-accident safety function should close automatically. In the event of power failure to a valve operator, the valve position should be the position of greater safety, which is normally the post-accident position. However, special cases may arise and these will be considered on an individual basis in determining the acceptability of the prescribed valve positions. The CSB also ascertains from the SAR that all power-operated isolation valves have position indication capability in the main control room.

The CSB reviews the signals obtained from the plant protection system to initiate containment isolation. In general, there should be a diversity of parameters sensed; e.g., abnormal conditions in the reactor coolant system, the secondary coolant system, and the containment, which generate containment isolation signals. Since plant designs differ in this regard and many different combinations of signals from the plant protection system are used to initiate containment isolation, the CSB considers the arrangement proposed on an individual basis in determining the overall acceptability of the containment isolation signals.

The CSB reviews isolation valve closure times. In general, valve closure times should be less than one minute, regardless of valve size. (See the acceptance criteria for valve closure times in subsection II.) Valves in lines that provide a direct path to the environs, e.g., the containment purge and ventilation system lines and main steam lines for direct cycle plants, may have to close in times much shorter than one minute. Closure times for these valves may be dictated by radiological dose analyses or ECCS performance considerations. The CSB will request the AAB or RSB to review analyses justifying valve closure times for these valves as necessary.

The CSB determines the acceptability of the use of closed systems inside containment as isolation barriers by comparing the system designs to the acceptance criteria specified in subsection II.

The MEB and SEB have review responsibility for the structural design of the containment internal structures and piping systems, including restraints, to assure that the containment isolation provisons are adequately protected against missiles, pipe whip, and earthquakes. The CSB determines that for all containment isolation provisions, missile protection and protection against loss of function from pipe whip and earthquakes were design considerations. The CSB reviews the system drawings (which should show the locations of missile barriers relative to the containment isolation provisions) to determine that the isolation provisions are protected from missiles. The CSB also reviews the design criteria applied to the containment isolation provisions to determine that protection against dynamic effects, such as pipe whip and earthquakes, was considered in the design. The CSB will request the MEB to review the design adequacy of piping and valves for which conservative design is assumed to preclude possible breach of system integrity in lieu of providing a leak tight housing.

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Systems having a post-accident safety function may have remote-manual isolation valves in the lines penetrating the containment. The CSB reviews the provisions made to detect leakage from these lines outside containment and to allow the operator in the main control room to isolate the system train should leakage occur. Leakage detection provisions may include instrumentation for measuring system flow rates, or the pressure, temperature, radiation, or water level in areas outside the containment such as valve rooms or engineered safeguards areas. The CSB bases its acceptance of the leakage detection provisions described in the SAR on the capability to detect leakage and identify the lines that should be isolated.

The CSB determines that the containment isolation provisions are designed to allow the isolation barriers to be individually leak tested. This information should be tabulated in the safety analysis report to facilitate the CSB review.

The CSB determines from the descriptive information in the SAR that provisions have been made in the design of the containment isolation system to allow periodic operability testing of the power-operated isolation valves and the containment isolation system. At the operating license stage of review, the CSB determines that the content and intent of proposed technical specifications pertaining to operability and leak testing of containment isolation equipment is in agreement with requirements developed by the staff.

IV. EVALUATION FINDINGS

The information provided and the CSB review should support concluding statements similar to the following, to be included in the staff's safety evaluation report:

"The scope of review of the containment isolation system for the (plant name) has included schematic drawings and descriptive information for the isolation provisions for fluid systems which penetrate the containment boundary. The review has also included the applicant's proposed design bases for the containment isolation provisions, and analyses of the functional capability of the containment isolation system.

"The basis for the staff's acceptance has been the conformance of the containment isolation provisions to the Commission's regulations as set forth in the General Design Criteria, and to applicable regulatory guides, staff technical positions, and industry codes and standards. (Special problems or exceptions that the staff takes to specific containment isolation provisions or the functional capability of the containment isolation system should be discussed.)

"The staff concludes that the containment isolation system design conforms to all applicable regulations, guides, staff positions, and industry codes and standards, and is acceptable."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 54, "Piping Systems Penetrating Containment."

2. 10 CFR Part 50, Appendix A, General Design Criterion 55, "Reactor Coolant Pressure Boundary Penetrating Containment."

3. 10 CFR Part 50, Appendix A, General Design Criterion 56, "Primary Containment Isolation."

4. 10 CFR Part 50, Appendix A, General Design Criterion 57, "Closed System Isolation Valves."

5. Regulatory Guide 1.141, "Containment Isolation Provisions For Fluid Systems."

6. Regulatory Guide 1.11, "Instrument Lines Penetrating Primary Reactor Containment."

7. Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants."

8. Regulatory Guide 1.29, "Seismic Design Classification."

9. Branch Technical Position CSB 6-4, "Containment Purging During Normal Plant Operations," attached to this SRP section.

Branch Technical Position CSB 6-4

CONTAINMENT PURGING DURING NORMAL PLANT OPERATIONS

A. BACKGROUND

This branch technical position pertains to system lines which can provide an open path from the containment to the environs during normal plant operation; e.g., the purge and vent lines of the containment purge system. It supplements the position taken in SRP section 6.2.4.

While the containment purge system provides plant operational flexibility, its design must consider the importance of minimizing the release of containment atmosphere to the environs following a postulated loss-of-coolant accident. Therefore, plant designs must not rely on its use on a routine basis.

The need for purging has not always been anticipated in the design of plants, and therefore, design criteria for the containment purge system have not been fully developed. The purging experience at operating plants varies considerably from plant to plant. Some plants do not purge during reactor operation, some purge intermittently for short periods and some purge continuously.

The containment purge system has been used in a variety of ways, for example, to alleviate certain operational problems, such as excess air leakage into the containment from pneumatic controllers, for reducing the airborne activity within the containment to facilitate personnel access during reactor power operation, and for controlling the containment pressure, temperature and relative humidity. However, the purge and vent lines provide an open path from the containment to the environs. Should a LOCA occur during containment purging when the reactor is at power, the calculated accident doses should be within 10 CFR 100 guideline values.

The sizing of the purge and vent lines in most plants has been based on the need to control the containment atmosphere during refueling operations. This need has resulted in very large lines penetrating the containment (about 42 inches in diameter). Since these lines are normally the only ones provided that will permit some degree of control over the containment atmosphere to facilitate personnel access, some plants have used them for containment purging during normal plant operation. Under such conditions, calculated accident doses could be significant. Therefore, the use of these large containment purge and vent lines should be restricted to cold shutdown conditions and refueling operations.

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The design and use of the purge and vent lines should be based on the premise of achieving acceptable calculated offsite radiological consequences and assuring that emergency core cooling (ECCS) effectiveness is not degraded by a reduction in the containment backpressure.

Purge system designs that are acceptable for use on non-routine basis during normal plant operation can be achieved by providing additional purge and vent lines. The size of these lines should be limited such that in the event of a loss-of-coolant accident, assuming the purge and vent valves are open and subsequently close, the radiological consequences calculated in accordance with Regulatory Guides 1.3 and 1.4 would not exceed the 10 CFR 100 guideline values. Also, the maximum time for valve closure should not exceed five seconds to assure that the purge and vent valves would be closed before the onset of fuel failures following a LOCA.

The size of the purge and vent lines should be about eight inches in diameter for PWR plants. This line size may be overly conservative from a radiological viewpoint for the Mark III BWR plants and the HTGR plants because of containment and/or core design features. Therefore, larger line sizes may be justified. However, for any proposed line size, the applicant must demonstrate that the radiological consequences following a loss-of-coolant accident would be within 10 CFR 100 guideline values. In summary, the acceptability of a specific line size is a function of the site meteorology, containment design, and radiological source term for the reactor type, e.g., BWR, PWR or HTGR.

B. BRANCH TECHNICAL POSITION

The system used to purge the containment for the reactor operational modes of power operation, startup, hot standby and hot shutdown; i.e., the on-line purge system, should be independent of the purge system used for the reactor operational modes of cold shutdown and refueling.

1. The on-line purge system should be designed in accordance with the following criteria:

    a. The performance and reliability of the purge system isolation valves should be consistent with the operability assurance program outlined in MEB Branch Technical Position MEB-2, Pump and Valve Operability Assurance Program. (Also see SRP Section 3.9.3.) The design basis for the valves and actuators should include the buildup of containment pressure for the LOCA break spectrum, and the purge line and vent line flows as a function of time up to and during valve closure.

    b. The number of purge and vent lines that may be used should be limited to one purge line and one vent line.

    c. The size of the purge and vent lines should not exceed about eight inches in diameter unless detailed justification for larger line sizes is provided.

d. The containment isolation provisions for the purge system lines should meet the standards appropriate to engineered safety features; i.e., quality, redundancy, testability and other appropriate criteria.

e. Instrumentation and control systems provided to isolate the purge system lines should be independent and actuated by diverse parameters; e.g., containment pressure, safety injection actuation, and containment radiation level. If energy is required to close the valves, at least two diverse sources of energy shall be provided, either of which can affect the isolation function.

f. Purge system isolation valve closure times, including instrumentati elays, should not exceed five seconds.

g. Provisions should be made to ensure that isolation valve closure w not be prevented by debris which could potentially become entrained in the ng air and steam.

2. The purge system should not be relied on for temperature and humidity control within the containment.

3. Provisions should be made to minimize the need for purging of the containment by providing containment atmosphere cleanup systems within the containment.

4. Provisions should be made for testing the availability of the isolation function and the leakage rate of the isolation valves, individually, during reactor operation.

5. The following analyses should be performed to justify the containment purge system design:

a. An analysis of the radiological consequences of a loss-of-coolant accident. The analysis should be done for a spectrum of break sizes, and the instrumentation and setpoints that will actuate the vent and purge valves closed should be identified. The source term used in the radiological calculations should be based on a calculation under the terms of Appendix K to determine the extent of fuel failure and the concomitant release of fission products, and the fission product activity in the primary coolant. A pre-existing iodine spike should be considered in determining primary coolant activity. The volume of containment in which fission products are mixed should be justified, and the fission products from the above sources should be assumed to be released through the open purge valves during the maximum interval required for valve closure. The radiological consequences should be within 10 CFR 100 guideline values.

b. An analysis which demonstrates the acceptability of the provisions made to protect structures and safety-related equipment; e.g., fans, filters and ductwork, located beyond the purge system isolation valves against loss of function from the environment created by the escaping air and steam.

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c. An analysis of the reduction in the containment pressure resulting from the partial loss of containment atmosphere during the accident for ECCS backpressure determination.

d. The allowable leak rates of the purge and vent isolation valves should be specified for the spectrum of design basis pressures and flows against which the valves must close.

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040.0 POWER SYSTEMS BRANCH

040.1 (3.11) In order to ensure that your environmental qualification program conforms with General Design Criteria 1, 2, 4 and 23 of Appendix A and Sections III and XI of Appendix B to 10 CFR Part 50, and to the national standards mentioned in Part II "Acceptance Criteria" (which includes IEEE Std 323) contained in Standard Review Plan Section 3.11, the following information on the qualification program is required for all Class 1E equipment.

*(1) Identify all Class 1E Equipment, and provide the following:

a. Type (functional designation)
b. Manufacturer
c. Manufacturer's type number and model number
d. The equipment should include the following, as applicable:

1) Switchgear
2) Motor control centers
3) Valve operators
4) Motors
5) Logic equipment
6) Cable
7) Diesel generator control equipment
8) Sensors (pressure, pressure differential, temperature and neutron)
9) Limit Switches
10) Heaters
11) Fans
12) Control Boards
13) Instrument racks and panels
14) Connectors
15) Electrical penetrations
16) Splices
17) Terminal blocks

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*(2) Categorize the equipment identified in (1) above into one of the following categories:

(a) Equipment that will experience the environmental conditions of design basis accidents for which it must function to mitigate said accidents, and that will be qualified to demonstrate operability in the accident environment for the time required for accident mitigation with safety margin to failure.

(b) Equipment that will experience environmental conditions of design basis accidents through which it need not function for mitigation of said accidents, but through which it must not fail in a manner detrimental to plant safety or accident mitigation, and that will be qualified to demonstrate the capability to withstand any accident environment for the time during which it must not fail with safety margin to failure.

(c) Equipment that will experience environmental conditions of design basis accidents through which it need not function for mitigation of said accidents, and whose failure (in any mode) is deemed not detrimental to plant safety or accident mitigation, and need not be qualified for any accident environment, but will be qualified for its non-accident service environment.

(d) Equipment that will not experience environmental conditions of design basis accidents and that will be qualified to demonstrate operability under its normal or abnormal service environment. This equipment would normally be located outside the reactor containment.

*(3) For each type of equipment in the categories of equipment listed in (2) above provide separately the equipment design specification requirements, including:

(a) The system safety function requirements.

(b) An environmental envelope as a function of time which includes all extreme parameters, both maximum and minimum values, expected to occur during plant shutdown, normal operation, abnormal operation, and any design basis event (including LOCA and MSLB), including post event conditions.

(c) Time required to fulfill its safety function when subjected to any of the extremes of the environmental envelope specified above.

(d) Technical bases should be provided to justify the placement of each type equipment in the categories 2.b and 2.c listed above.

*(4) Provide the qualification test plan, test set-up, test procedures, and acceptance criteria for at least one of each group of equipment

of (1.d) as appropriate to the category identified in (2) above. If any method other than type testing was used for qualification (operating experience, analysis, combined qualification, or on-going qualification), describe the method in sufficient detail to permit evaluation of its adequacy.

**(5) For each category of equipment identified in (2) above, state the actual qualification envelope simulated during testing (defining the duration of the hostile environment and the margin in excess of the design requirements). If any method other than type testing was used for qualification, identify the method and define the equivalent "qualification envelope" so derived.

**(6) A summary of test results that demonstrates the adequacy of the qualification program. If analysis is used for qualification, justification of all analysis assumptions must be provided.

**(7) Identification of the qualification documents which contain detailed supporting information, including test data, for items 4, 5 and 6.

In addition, in accordance with the requirements of Appendix B of 10 CFR 50, the staff requires a statement verifying: 1) that all Class 1E equipment has been (OL) or will be (CP) qualified to the program described above, and 2) that the detailed qualification information and test results are (or will be) available for an NRC audit.

*For Items 1 through 4. Provide a commitment that this information will be supplied prior to tendering your FSAR.
**For Items 5 through 7. Provide a commitment that this information will be supplied in your FSAR.

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040.2 (8.2) (RSP) Recent operating experience has shown that adverse effects on the safety-related power system and safety related equipment and loads can be caused by sustained low or high grid voltage conditions. We therefore require that your design of the safety related electrical system meet the following staff positions. In this regard, supplement the description of your design so as to show how it meets these positions or provide appropriate analyses to justify non-conformance with these positions.

(1) In addition to the undervoltage scheme provided to detect loss of offsite power at the safety busses, we require that an additional level of voltage protection for the onsite power system be provided with a time delay and that this additional level of voltage protection shall satisfy the following criteria:

(a) The selection of voltage and time set points shall be determined from an analysis of the voltage requirements of the safety-related loads at all onsite system distribution levels;

(b) The voltage protection shall include coincidence logic on a per bus basis to preclude spurious trips of the offsite power source;

(c) The time delay selected shall be based on the following conditions:

(i) The allowable time delay, including margin, shall not exceed the maximum time delay that is assumed in the PSAR accident analyses;

(ii) The time delay shall minimize the effect of short duration disturbances from reducing the availability of the offsite power source(s); and

(iii) The allowable time duration of a degraded voltage condition at all distribution system levels shall not result in failure of safety systems or components;

(d) The voltage sensors shall automatically initiate the disconnection of offsite power sources whenever the voltage set point and time delay limits have been exceeded;

(e) The voltage sensors shall be designed to satisfy the applicable requirements of IEEE Std. 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations"; and

(f) The Technical Specifications provided during the OL stage shall include limiting condition for operation, surveillance requirements, trip set points with minimum and maximum limits, and allowable values for the second-level voltage protection sensors and associated time delay devices.

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(2) We require that the current system designs automatically prevent load shedding of the emergency buses once the onsite sources are supplying power to all sequenced loads on the emergency buses. The design shall also include the capability of the load shedding feature to be automatically reinstated if the onsite source supply breakers are tripped. The automatic bypass and reinstatement feature shall be verified during the periodic testing identified in Position 3.

In the event an adequate basis can be provided for retaining the load shed feature when loads are energized by the onsite power system, we will require that the setpoint value in the Technical Specifications, which is currently specific as "...equal to or greater than..." be amended to specify a value having maximum and minimum limits. The licensee' bases for the setpoints and limits selected must be documented.

(3) We require that the Technical Specifications provided during the OL stage include a test requirement to demonstrate the full functional operability and independence of the onsite power sources at least once per 18 months during shutdown. The Technical Specifications shall include a requirement for tests: (1) simulating loss of offsite power; (2) simulating loss of offsite power in conjunction with a safety feature actuation signal; and (3) simulating interruption and subsequent reconnection of onsite power sources to their respective buses. Proper operation shall be determined by:

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(a) Verifying that on loss of offsite power the emergency buses have been de-energized and that the loads have been shed from the emergency buses in accordance with design requirements.

(b) Verifying that on loss of offsite power the diesel generators start on the autostart signal, the emergency buses are energized with permanently connected loads, the auto-connected shutdown loads are energized through the load sequencer, and the system operates for five minutes while the generators are loaded with the shutdown loads.

(c) Verifying that on a safety features actuation signal (without loss of offsite power) the diesel generators start on the autostart signal and operate on standby for five minutes.

(d) Verifying that on loss of offsite power in conjunction with a safety features actuation signal the diesel generators start on the autostart signal, the emergency buses are energized with permanently connected loads, the auto-connected emergency (accident) loads are energized through the load sequencer, and the system operates for five minutes while the generators are loaded with the emergency loads.

(e) Verifying that on interruption of the onsite sources the loads are shed from the emergency buses in accordance with design requirements and that subsequent loading of the onsite sources is through the load sequencer.

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(4) The voltage levels [illegible] the safety-related buses should be optimized for the full load and minimum load conditions that are expected throughout the anticipated range of voltage variations of the offsite power source by appropriate adjustment of the voltage tap settings of the intervening transformers. We require that the adequacy of the design in this regard be verified by actual measurement and by correlation of measured values with analysis results. Provide a description of the method for making this verification; before initial reactor power operation, provide the documentation required to establish that this verification has been accomplished.

040.3 (Appendix A) Appendix 3A of the New Haven PSAR does not address selected regulatory guides which relate to the electrical power system. Of primary concern in this regard are Regulatory Guides 1.41 (Preoperational Testing of Redundant On-Site Electric Power Systems to Verify Proper Load Group Assignments) and 1.108 (Periodic Testing of Diesel Generator Units Used As On-Site Electric Power Systems at Nuclear Power Plants). Accordingly, clearly state your intentions with regard to conforming to these regulatory guides and explicitly identify and justify any exceptions which are to be taken relating to these guides.

040.4 (8.3) Diesel generator alarms in the control room: A review of malfunction reports of diesel generators at operating nuclear plants has uncovered that in some cases the information available to the control room operator to indicate the operational status of the diesel generator may be imprecise and could lead to misinterpretation. This can be caused by the sharing of a single annunciator station to alarm conditions that render a diesel generator unable to respond to an automatic emergency start signal and to also alarm abnormal, but not disabling, conditions. Another cause can be the use of wording of an annunciator window that does not specifically say that a diesel generator is inoperable (i.e., unable at the time to respond to an automatic emergency start signal) when in fact it is inoperable for that purpose.

Review and evaluate the alarm and control circuitry for the diesel generators at your facility to determine how each condition that renders a diesel generator unable to respond to an automatic emergency start signal is alarmed in the control room. These conditions include not only the trips that lock out the diesel generator start and require manual reset, but also control switch or mode switch positions that block automatic start, loss of control voltage, insufficient starting air pressure or battery voltage, etc. This review should consider all aspects of possible diesel generator operational conditions, for example test conditions and operation from local control stations. One area of particular concern is the unreset condition following a manual stop

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at the local station which terminates a diesel generator test and prior to reseting the diesel generator controls for enabling subsequent automatic operation.

Provide the details of your evaluation, the results and conclusions, and a tabulation of the following information:

(1) all conditions that render the diesel generator incapable of responding to an automatic emergency start signal for each operating mode as discussed above;

(2) the wording on the annunciator window in the control room that is alarmed for each of the conditions identified in (1);

(3) any other alarm signals not included in (1) above that also cause the same annunciator to alarm;

(4) any condition that renders the diesel generator incapable of responding to an automatic emergency start signal which is not alarmed in the control room; and

(5) any proposed modifications resulting from this evaluation.

040.5 (8.4) Concerning fire stops and seals for cable systems expand the information provided in Section 8.4.2 of the New Haven PSAR so as to address each of the following.

(1) State the design criteria for each type of fire stop and seal installation.

(2) Provide the interval (physical distance) at which the fire stops are installed in vertical cable trays, and in horizontal cable trays (if any).

(3) The QA and test procedures used to verify that penetration fire stops and seals have been properly installed.

(4) The qualification testing of the fire stops and seals to demonstrate adequacy of the <u>life</u> of the plant.

(5) The administrative procedures and controls that will be followed when it becomes necessary to breach a completed fire stop or seal to add or remove cables.

(6) The periodic inspections performed to identify open or deteriorated fire stops and seals.

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040.6 (9.5.5) (9.5.6) (9.5.7) (RSP) You state in the PSAR that the descriptions for the following systems will be provided in the FSAR: (1) diesel engine cooling water system, (2) diesel engine air starting system, and (3) diesel engine lubrication system.

We require information on these systems in the PSAR in order to make evaluations according to the acceptance criteria in Sections 9.5.5, 9.5.6 and 9.5.7 of the Standard Review Plan. Revise sections 9.5.5, 9.5.6, and 9.5.7 in your PSAR to conform to Regulatory Guide 1.70 and the criteria in the SRP's.

040.7 (9.5.x) In accordance with the "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants", Regulatory Guide 1.70, Revision 2, there should be a section in the PSAR for the diesel generator combustion air intake and exhaust system. Provide such a section which we can evaluate according to the criteria of Section 9.5.8 of the Standard Review Plan.

040.8 (10.2) (10.3) (10.4) You state in the PSAR that the information for the following systems are provided in SWESSAR-P1: 1) Section 10.2, Turbine Generator Steam System, 2) Section 10.3, Main Steam System, 3) Section 10.4.1 Main Condensers, and 4) Section 10.4.4, Turbine Bypass System.

However, the information in SWESSAR-P1 for the above is given for both the General Electric and Westinghouse designs. We will need to know specifically which one of these two manufacturers will be selected in order that we can evaluate the systems according to the criteria in Sections 10.2, 10.3, 10.4.1 and 10.4.4 of the Standard Review Plan.

Revise your PSAR sections 10.2, 10.3, 10.4.1 and 10.4.4 to conform to Regulatory Guide 1.70 and the designated SRP's.

112.0 MECHANICAL ENGINEERING BRANCH

112.1 (3.9.4.1) The information presented in Sections 3.9.4.1, 3.9.4.2, 3A.1-1.68 and 14 of the PSAR concerning the piping preoperational testing program is not completely acceptable. Section 3.9.1.1 of Table 1.8-1 in the PSAR states that this program is the responsibility of the applicant. The staff requires a commitment to the program outlined in Paragraphs I.1 and II.1a through II.1f of NRC Standard Review Plan 3.9.2, "Dynamic Testing and Analysis of Systems, Components and Equipment", Revision 1. Provide such a commitment in Section 3.9.4 of the PSAR.

130.0 STRUCTURAL ENGINEERING BRANCH

130.1
3.7.1 Compare the results of the confirmatory dynamic analysis performed for the New Haven site with the SWESSAR-P1 dynamic analysis.

130.2
3.7.2.1.2 Discuss which of the three "amplified response spectra" methods ("Umbrella" spectrum, maximum modal response, maximum site response) is used for the New Haven Site. If more than one is used, specify which components were analyzed by each method.

130.3
3.7.2.6 The SWESSAR-P1 SSAR states that design envelopes for seismic response of Category I structures, systems, and components incorporate subgrades with shear modulus values ranging from 6,000 psi to 1,000,000 psi. However, the New Haven PSAR Table 2.5-4 lists shear modulus values (at all elevations given) in excess of 1,000,000 psi. Please explain the effect these high shear modulus values have on the results of the SWESSAR-P1 analyses.

221.0 ANALYSIS BRANCH - REACTOR ANALYSIS SECTION

221.1 (4.4) It is our position that loose parts monitoring systems (LPMS) shall be described in each utility's PSAR. The description should include the design criteria for the system and a description of the location of all the intended sensors in the LPMS along with a description of how they are to be mounted. In addition, information should be provided to address the capability of the system components within containment to remain operational following seismic events up to and including the operating basis earthquake. A discussion should also be provided of any analysis and/or tests to demonstrate that the LPMS will be adequate for the normal operating radiation, vibration, temperature and humidity environment of the reactor system.

231.0 CORE PERFORMANCE BRANCH - REACTOR FUELS SECTION

231.1 The need for routine surveillance is discussed in Revision 1 of Section 4.2 of the Standard Review Plan. The New Haven PSAR does not provide for such a program. Therefore, you should propose a surveillance program that includes a description of (a) the on-line fuel rod failure detection method, (b) CEA integrity assurance, and (c) a post-irradiation fuel surveillance plan.

312.0 ACCIDENT ANALYSIS BRANCH

312.1 (2.1) With respect to the requirements of 10 CFR Part 100, Section 100.3(a), the PSAR states (2.1.2.1) that none of the property within the site boundary (including the whole of the exclusion area) is currently owned by NSYE&G. The PSAR further states that you intend to purchase or otherwise obtain control of the site property subsequent to the completion of the New York State power plant siting proceeding. Discuss your acquisition of the site property in relation to the scheduled construction permit date. Include in your discussion your intentions regarding acquisition of the mineral rights for all properties within the exclusion area. If any lands within the exclusion area are public owned, state how you will obtain the authority required under Part 100.3(a). For any properties within the exclusion area which you will not own, describe the bases on which you will conclude that the authority required under Part 100.3(a) will be acquired.

321.0 <u>EFFLUENT TREATMENT SYSTEMS BRANCH</u> - APPLICATION SECTION

321.1 (11.3, 9.4) Indicate conformance with Regulatory Guide 1.140, "Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System, Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants." List any exceptions and alternatives, with their justifications, to the regulatory positions in this guide.

321.2 (10.4.2) Indicate the quality group classification for the main condenser evacuation system and address the potential for explosive gaseous mixtures that may exist in the system.

321.3 (11.2 11.3) Perform a cost-benefit analysis on the liquid and gaseous radwaste management systems to show conformance with the design objectives of 10 CFR Part 50, Appendix I, Section II.D.

321.4 (15.7.3) Perform an analysis on the radiological consequences of a postulated component failure that could result in the release of radioactive liquids to the site related potable water supplies and nearby surface water, such as rupture of a liquid-containing tank outside containment.

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331.0 RADIOLOGICAL ASSESSMENT BRANCH - RADIATION PROTECTION SECTION

331.1
(12.1)
With regard to participation by NYSE&G in the radiation protection review of plant changes made during the final design and construction processes, from the point of view of maintaining occupational radiation exposures as low as is reasonably achievable:

(1) Identify by title the individual(s) who have been and will be responsible for this radiation protection design review, and describe how she or he relates to the individual responsible for the overall design.

(2) Provide a breakdown by title of radiation protection personnel who have been and who will be participating in such reviews, tabulating the health physics education and experience required of such.

(3) Describe formal arrangements and procedures for assuring that adequate radiation protection reviews are performed throughout the design and construction processes and adequate records are kept to document the completion of each such review.

(4) Describe specific examples of actual dose-reducing changes in design that have resulted from these radiation protection design reviews.

331.2
(12.1.7.1)
Provide a dose assessment in accordance with Regulatory Guide 8.19. Develop your best estimates of occupational radiation exposures from all significant contributors to such exposures, averaged over at least five years of operation. Take into account duties and occupancy requirements for plant, utility, and contractor personnel, as NYSE&G expects to operate the plant, and improvements made in this plant relative to those plants from which current exposure data have been denied. Describe the assumptions and calculations used in the dose assessment process to demonstrate a systematic process for considering and evaluating possible dose reducing design features and associated operating procedures.

331.3
(12.1.7.3)
Taking into account facilities described in CESSAR System 80, SWESSAR-P1, and in the PSAR for New Haven, describe the projected traffic flow for male and female workers proceeding

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to work stations in restricted areas at the beginning of a shift, and on the way out at the end of the shift. Demonstrate the adequacy of space to be provided for such workers to change from street to work clothing, facilities provided for lockers, shower, and decontamination, if necessary, and monitors and other precautions to be provided to assure that radioactive contamination is not transplanted to uncontrolled areas. Provide layout diagrams of how the health physical areas and facilities will be used to assure worker protection, and indicate traffic patterns to and from potentially contaminated areas.

372.0 HYDROLOGY - METEOROLOGY BRANCH - METEOROLOGY

372.1 (2.3.1) The discussion of the frequency of lightning (page 2.3-2) provides estimates of lightning flashes for a 1 square mile area in the site region. Provide the probability of a lightning strike to safety-related structures utilizing the estimates of lightning flashes to ground per unit area and considering the "attractive area" of the structures. Identify the bases for the estimates and the assumptions that were considered.

322.2 (2.3.1) Provide the probability of a tornado occurrence in the region of the site and its associated mean recurrence interval. Identify the bases for the estimates and the assumptions that were considered.

372.3 (2.3.1) It is stated that "the saturated snowpack on flat roofs of safety related structures is 55 psf" (Page 2.3-5). Explain how this value is obtained considering the 48-hour PMWP is equivalent to 82.3 psf and "the design snowload for safety related structures is 50 psf".

372.4 (2.3.1) Design values for extreme snow loads and the 100 year fastest mile wind speed were calculated using data from the site vicinity and the Hancock International Airport in Syracuse, respectively. Clarify if these values are to be used in the design of the plant or if the values contained in Section 2.3.1 of SWESSAR-P1 are to be used.

372.5 (2.3.2) Summary tables of wind speed, wind direction and stability for onsite measurements contain a variable wind direction category. Provide the criteria used to define this category.

372.6 (2.3.3) It is stated that the meteorological tower is located "in an area representative of the site". (page 2.3-31). Provide additional information about the tower location such as; the distance to the proposed plant structures; the distance to the nearest obstructions; the heights of the nearest obstructions; and a description of the ground around the base of the tower.

372.7 (2.3.3) Discuss in greater detail how the hourly averaged values are obtained. (i.e., Is it the average of 3 readings a minute for at least 45 minutes of each hour?) What are the criteria for "invalid" data?

372.8 (2.3.3) Provide the dates and times of significant instrument outage, the causes of the outage, and the corrective action taken.

372.9 (2.3.3) Data from the strip charts were "used when the computer data were not available or when subjective data evaluation was appropriate". How many hours (or what percentage) of the hourly averaged onsite data from April 1, 1977 through March 31, 1978 were obtained in this manner?

372.10 (2.3.3) In the event of an accidental release at the New Haven site, how will meteorological data be received in the plant control room if: (1) the "control room remote readout system" becomes inoperable; (2) data from the meteorological tower become unavailable?

372.11 (2.3.3) In what form will the meteorological data be received on the remote readout system in the control room. (e.g. instantaneous, hourly averages, etc.)

372.12 (2.3.3) As discussed in Regulatory Guide 1.70, onsite meteorological data should be available on magnetic tape. Having access to onsite meteorological data on magnetic tape would facilitate the review of atmospheric dispersion characteristics. If available, provide onsite meteorological data for the period April 1, 1977 through March 31, 1978 in the form of hour-by-hour averages on magnetic tape using the enclosed format.

372.13 (2.3.4) Provide the dimensions of the containment structure used to calculate the minimum cross-sectional area of 4,239 m^2.

372.14 (2.3.4) The atmospheric dispersion model and procedures used to evaluate dispersion conditions to be used in an assessment of the consequences of design basis accidents described in your Section 2.3.4 are based on draft Regulatory Guide 1.XXX, "Atmospheric Despersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants", (September 1977). Enclosed is an Interim Branch Technical Position

No. 2 (Revision 1) (August 2, 1978) that is directly pertinent to the use of draft R. G. 1.XXX. Provide your response to this position.

372.15 (9.2.5) Wet bulb temperatures for the ultimate heat sink were provided for the worst 1-day and worst 29-day periods. Provide the criteria used to determine these worst-case periods of wet bulb temperature.

441.0 OPERATING LICENSE BRANCH - TRAINING SECTION

441.1 (13.2.1.1.3) It is highly desirable that experienced individuals participate in courses utilizing a nuclear plant simulator similar to the facility for which the individual will be seeking a license. You should provide this training for all candidates for an SRO or RO license under categories 2 and 3 of Section 13.2.1.1.3.

441.2 (13.2) Provide a commitment to conduct an initial fire protection training program, including:

(1) Periodic drills during construction.

(2) Provisions for indoctrination of construction personnel, as necessary.

The initial training shall be completed prior to receipt of fuel at the site.

411.3 (13.2) Provide a detailed description of the training program for the individual(s) responsible for the formulation and assurance of the implementation of the fire protection program. The training program should address those items listed in Section A.1 of Appendix A to Branch Technical Position ASB 9.5-1, Guidelines for Fire Protection for Nuclear Power Plants.

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442.0 OPERATING LICENSE BRANCH - PROCEDURES SECTION

442.1 (13.0) Provide a commitment to conduct all safety-related operations by detailed written and approved procedures.

442.2 (13.5) Provide a commitment to provide written and approved fire protection procedures and include them on the preliminary schedule for completion.

421.0 QUALITY ASSURANCE BRANCH - QUALITY ASSURANCE SECTION

421.1 In the area of design verification, it appears that 17.1.1.3.4 on page 17.1-12 of the PSAR does not meet Regulatory Guide 1.64 regarding design verification. Please clarify this matter. The following commitment has been found acceptable by the staff in the past:

"Procedures require that design reviews be in accordance with Regulatory Guide 1.64, Revision 2. If in an exceptional circumstance the designer's immediate supervisor is the only technically qualified individual available, this review can be conducted by the supervisor, providing that:

(1) The other provisions of the Regulatory Guide are satisfied,

(2) The justification is individually documented and approved in advance by the supervisor's management, and

(3) Quality assurance audits cover frequency and effectiveness of use of supervisors as design verifier to guard against abuse."

421.2 Describe measures which assure that audits include an objective evaluation of quality-related practices, procedures, and instructions.

421.3 Please discuss the authority and independence of the Manager-Quality Assurance considering that the Nuclear Quality Assurance Procedures require approval by the Manager-Nuclear Projects as shown in Table 17.1-4.

421.4 Reference 1 for Section 17.1 should be Revision C of Stone & Webster Engineering Corporation topical report SWSQAP 1-74A, "Standard Nuclear Quality Assurance Program," dated December 1978 since it supersedes earlier revisions.

423.0 QUALITY ASSURANCE BRANCH - INITIAL TEST AND OPERATIONS SECTION

423.1 A detailed review of the initial test program has not been and will not be conducted in the PSAR review. Therefore, the commitments to conduct tests in accordance with specific regulatory guides has not been reviewed. However, since additional regulatory guidance applicable to the initial test program may be developed prior to development of your initial test program, the PSAR should indicate that all Regulatory Guides will be reviewed for applicability at the time of development of your initial test program. Include a commitment that the specific applicability of Regulatory Guides to your initial test program will be provided in the FSAR.

361.0 GEOSCIENCES BRANCH - GEOLOGY - SEISMOLOGY, SECTION

361.1 (2.5) The New Haven and CESSAR PSAR's do not contain sufficient data to make comparisons between the CESSAR seismic design and Regulatory Guide 1.60 spectra. Please provide plots (on tripartite graph paper) which permit comparison of the New Haven/CESSAR spectra and the Regulatory Guide 1.60 spectra using 0.2g as the high frequency input for the SSE and 0.1g for the OBE.

361.2 (2.5) Until the New Haven PSAR Appendix 2.5.I, "Regional Fault Investigations" is submitted and reviewed, Round One Questions cannot be issued. Provide a schedule for submitting this information.

36[illegible] Appendix 2.5A. Section 2.5A.2 discusses several pop-up features in the Alexandria Bay, New York area in order to improve the "understanding of the cause/origin and significance of the small-scale anticlinal features which exist at Nine Mile Point and Fitzpatrick Nuclear Stations." Provide the following information with regards to pop-up features in general and as discussed in Appendix 2.5A, Section 2.5A.2.

(1) Provide a discussion and description of the pop-up features found at Nine Mile Point and Fitzpatrick Nuclear Stations as they relate to the proposed New Haven site.

(2) In Appendix 2.5H a compression buckle (i.e. pop-up) is described in Rock Pit II. Due to the presence of this feature under the site, provide a detailed discussion of the potential for the development of further pop-up features in the excavation due to unloading. Also discuss the potential for extension of excavation side walls into the excavation due to horizontal compression. These discussions should include but not be limited to such information as the in-situ stresses in the site area, proposed monitoring methods for

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such features developing during and after excavation while in an unloaded situation, and methods of repairing the excavation should such features develop.

361.4 (2.5) The discussion of Minor Geologic Structures in PSAR Section 2.5.1.2.3.3 supposedly discusses 3 structures (PSAR, pg. 2.5-43, paragraph 2, line 1). From the discussion provided in Section 2.5.1.2.3.3, pages 2.5-43 and 44, it would appear to actually discuss the following five features:

(1) A N78 W fault in the Fitzpatrick excavation,
(2) N N78 W normal fault in the Fitzpatrick intake discharge tunnel area,
(3) the barge slip fault,
(4) the cooling tower fault, and
(5) the drainage ditch fault

Please clarify the discussion and provide the location of these labeled faults on PSAR Figure 2.5-9.

361.5 (2.5) Provide a copy of the Site Confirmation Reports cited as PSAR Reference Numbers 81 and 82.

361.6 (2.5) Provide an analysis of all available LANDSAT and other air photographs and a complete detailed discussion of all lineaments in the New Haven site area describing their relationship to regional geology. The regional lineaments described by Saunders and Hicks in their 1976 paper entitled "Regional Geomorphic Lineaments on Satellite Imagery-Their Origin and Applications" should be included in the discussion.

361.7 (2.5) Appendix 2.5B discusses a geophysical survey of Lake George, New York to investigate an anomalous north-south trending feature in the lake bottom. While the staff realizes the features lies within 200 miles of the proposed New Haven site, the significance of a detailed report on this features is unclear with regard to the safety of the proposed New Haven site. Please clarify.

361.8 (2.5) Please provide a copy of the photographs taken of the trench walls and floor, if available.

361.9 (2.5) With regard to excavation and geologic mapping, our position is as follows:

(1) Site excavations made for Category I structures including Category I buried pipelines are to be geologically mapped in detail. The mapped surfaces are to include the excavation walls and floors. In addition, mapping is to include geologic units and features exposed as a result of excavations made for other than Category I structures and pipelines if such mapping is required to adequately interpret the site geology.

(2) In order to complete Regulatory staff arrangements for a site visit for the purpose of observing the mapped surfaces, we are to be notified shortly before the applicant completely maps the geologic units in Category I structure excavations.

(3) All features that could pose potential hazard to the site safety are to be reported to us immediately.

Please describe your intentions to satisfy the requirements of this position.

370.0 HYDROLOGY - METEOROLOGY BRANCH - HYDROLOGY SECTION

371.1 (2.4.1.1) Provide topographic maps of the site and vicinity of sufficient scale to show their relationship of the site to the nearby streams and rivers, both pre- and post-construction. Note that neither Figure 2.4-12 nor 2.4-14 are sufficient. Provide a large size topographic map of the site. Denote site boundaries and major plant features on all maps.

371.2 (2.4.1.1) Describe the site and all safety-related elevations, structures, exterior accesses, equipment and systems from the standpoint of hydrologic considerations, as requested in section 2.4 of Regulatory Guide 1.70. What is the Probable Maximum Flood elevation at the site?

371.3 (2.4.1.2) Indicate the locations of the intake and discharge structures, the pumphouse and the connecting pipelines on figure 2.4-3.

371.4 (2.4.1.2) Provide the descriptions and discussions requested in section 2.4 Regulatory Guide 1.70 for the streams and rivers near the site.

371.5 (2.4.2.2) What are the Probable Maximum Flood flows and elevations at the site for the potential sources of flooding identified as having been investigated?

371.6 Locate the site boundary and major plant structures in figure 2.4-14.

371.7 (2.4.2.3) The statement that the maximum water level at the site due to the Probable Maximum Precipitation is 5.3 inches has not been justified. Provide details of your calculations. Describe your proposed site drainage system. Include on the topographic map of the site requested in question 371.1, the drainage areas and postulated flow paths.

371.8 (2.4.3.1) To what stream does the time distribution of PMP shown in figure 2.4-16 apply? The total rainfall for the 48 hours is only 22.46 inches. Table 2.4-4, derived from Hydrometeorological Report No. 33, lists the 48 hour PMP for Butterfly Creek and Tributary FW (drainage areas less than 10 square miles) as 28.0 inches and for Catfish Creek (36 square mile drainage area) as 25.2 inches. We have calculated the 48 hour PMP's to be 31.0 inches and 27.6 inches, respectively, using the more recent Hydrometeorological Report No. 51. Justify the PMP values you use as conservative and revise your calculations as necessary.

371.9 (2.4.3.2) Why is Tributary FE not listed in Table 2.4-4? Was a flood analysis performed for this stream?

371.10 (2.4.3) How were the backwater effects of tributaries FE and FW combined in the calculations of flood levels at the site?

371.11 (2.4.3.3) The determination of time of concentration and storage coefficient has not been shown to be conservative. The regression equations used in table 2.4-6 do not result in values for Sterling Creek and Beaverdam Brook as conservative as the values obtained using stream gage data (table 2.4-5). Justify that the parameters used for the flood analysis of Butterfly and Catfish Creeks and Tributaries FW and FE are conservative.

371.12 (2.4.3.4) Provide the PMF hydrographs for all four of the streams listed in table 2.4-7. Does the hydrograph of Tributary FE, shown in figure 2.4-22, correspond to conditions prior to (as suggested by table 2.4-6) or after diversion?

371.13 (2.4.3.5) Provide the following information, including figures and maps as necessary, used in your analysis of water levels near the site:

(1) Water surface starting conditions with justification,

(2) Cross sections used for each calculation,

(3) Procedures used to peak the hydrographs and the results thereof,

(4) Discuss how the effects of roadways were considered in the analysis, and

(5) Input data for each of the computer runs made.

371.14 (2.4.3.5) Provide water surface profiles of the Probable Maximum Flood upstream and downstream of the plant for all nearby streams.

371.15 (2.4.3.5) Discuss the erosion potential of the nearby streams. Include, as a minimum, the following information:

(1) Procedures used to determine stream velocities;

(2) Values of the parameters used in the analysis and justification of their conservatism,

(3) The design of protective riprap, if necessary,

371.16 (2.4.3.5) Discuss the potential of spillage from one stream to another during the PMF. If such potential exists, discuss the consequences in terms of revised flows, water levels and erosion capability of the streams near the site.

371.17 (2.4.3.5) Discuss the potential of alteration of stream courses, including the capture of headwaters of one stream by another. If such potential exists, discuss the consequences in terms of revised flows water levels and erosion capability of the streams near the site.

371.18 (2.4.5) What is the design flood level of the pumphouse, including equipment therein, and how does this compare with the Probable Maximum Surge level and the historical maximum water level?

371.19 (2.4.7) Discuss the potential of lake ice blocking the intake or discharge. Include a discussion of the estimated frequency of such occurrences.

371.20 (2.4.11) What are the design low water levels of the intake and discharge systems and their bas

371.21 (2.4.12) Your analysis of the dispersion of liquid effluents in Lake Ontario is incomplete. Please provide the following:

(1) Identify and justify the conservatism of all the parameters used in your analysis, and

(2) Discuss the results of your analysis, including their sensitivity to variation of the input parameters. Ref-renced section 11.2.8 only contains a table of dilution factors and travel times; the discussion refers back to this section. In addition, the minimum shoreline dilution factor listed in table 11.2-2 is 78 with a travel time of 12 hours. In this section, however, you state that the minimum shoreline dilution factor is 22. Include an explanation of this apparent discrepancy in your discussions of results.

371.22 (2.4.12) Provide an evaluation of an accidental failure of an outside tank that could result in contaminated liquid being released directly to one of the nearby streams. If outside tanks are to be protected by dikes, consider their possible failure modes or leakage rates. Discuss the models used, justify the conservatism of all parameters and discuss the results (including their sensitivity to the assumptions and input parameters chosen).

371.23 (2.4.13) Identify the water users of Butterfly Creek, Catfish Creek and nearby springs.

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371.24 (2.4.13) Identify those wells listed in table 2.4-10 and shown in figure 2.4-24 that will be purchased by the applicant and the proposed use of those wells.

371.25 (2.4.13) Provide justification for the use of a distribution coefficient (k_d) equal to 100 for all ions subject to sorption. Since the value of the distribution function is dependent upon the properties of the ions, the groundwater and the aquifer, a single value of k_d must be shown to be conservative for all situations in which it is used. Alternately, you may use a different distribution coefficient for each ion, provided you justify the conservatism of each one.

371.26 (2.4.13) (RSP) Justify the conservatism of the design basis ground water level for dynamic loading. It is our position that the design basis should be the maximum level expected during the life of the plant. The piezometric data shown in figures 2.5-49 through 2.5-62 indicate that groundwater levels can be at the ground surface throughout the site area. Therefore, it is our position, that the design basis ground water level for both static and dynamic loading be taken as plant grade.

371.27 (3.4) Identify the design ground water level and the maximum surface water level due to site runoff.

371.28 (9.2.5) Provide analyses to substantiate that the cooling towers will meet the criteria of Regulatory Guide 1.27. This information should include actual performance data for similar cooling towers operating under load near the design conditions or justification that conservative drift loss and heat transfer values have been used. If your cooling tower design and specifications have not progressed to the stage where a predictive model can be developed and verified by high-quality performance data from existing towers of similar size and type, then commit to furnish the required analyses and data to NRC, for review and approval, prior to construction of the cooling towers.