AUG 1 1984

Docket No. 50-354

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Mr. R. L. Mittl, General Manager
Nuclear Assurance and Regulation
Public Service Electric and Gas Company
80 Park Plaza T16D
Newark, New Jersey 07101

Dear Mr. Mittl:

Subject: Hope Creek Generating Station Security Plan - Request for Additional Information

This is in response to your letter of May 21, 1984 regarding Revision 2 to the Hope Creek Security Plan.

We have reviewed Revision 2 and have determined that your proposed vital area identification program does not satisfy the criteria contained in Review Guideline No. 17 (enclosed). Your attention is specifically directed to Section C on page 4 which states that "...essentially all safety related equipment must be considered vital." Accordingly, the Hope Creek Physical Security Plan needs to be revised to conform to this guidance.

On another matter, we note that the Guardhouse which forms a part of the protected area barrier is not equipped with an external intrusion alarm as specified in Section 6.3 of NUREG-0908 "Acceptance Criteria for the Evaluation of Nuclear Power Reactor Security Plans". Your plan needs to be amended in this regard also.

Finally, please provide a drawing that shows the details of the intrusion alarms at the intake structure.

In order to avoid delays in the Hope Creek review schedule, we need your response by August 10, 1984.

The enclosures to your May 21, 1984 letter contain Safeguards Information of a type specified in 10 CFR 73.21 and are being withheld from public disclosure.

Sincerely,

A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing

Enclosure:
As stated

cc: See next page
*See previous concurrence sheet

SSPB:DL*	SSPB:DL*	SSPB:DL*	SSPB:DL	LB#2	LB#2	LB#2
CJamerson:ls	JGibson	HBerkow	CThomas	EHylton	DWagner	ASchwencer
7/26/84	7/27/84	7/27/84	7/27/84	8/ /84	8/ /84	8/ /84

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In order for us to complete our review, please respond within 30 days of the date of this letter.

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Sincerely,

"The reporting and/or recordkeeping requirements contained in this letter affect fewer than ten respondents; therefore OMB clearance is not required under P. L. 96-511."

A. Schwencer, Chief
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Division of Licensing

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

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DEFINITION OF
VITAL AREAS AND EQUIPMENT
Revision 1

A. Applicable Sections of 10 CFR 73

73.55 (c)(1):

"The licensee shall locate vital equipment only within a vital area, which in turn, shall be located within a protected area such that access to vital equipment requires passage through at least two physical barriers of sufficient strength to meet the performance requirements of paragraph (a) of this section. More than one vital area may be located within a single protected area."

73.2 (h):

"Vital area means any area which contains vital equipment within a structure, the walls, roof, and floor of which constitute physical barriers of construction at least as substantial as walls as described in paragraph (f)(2)."

73.2 (i):

"Vital equipment means any equipment, system, device, or material failure, destruction, or release of which could directly or indirectly endanger the public health and safety by exposure to radiation. Equipment or systems which would be required to function to protect public health and safety following such failure, destruction or release are also considered to be vital."

B. Assumptions and Definitions

In the application of these regulations to a typical LWR plant, the following considerations and assumptions are made:

1. Paragraph 73.55 (c) requires vital equipment to be enclosed by two barriers. The combination of barriers, in conjunction with other components of the security system, must provide a sufficient delay to an intrusion to meet the performance requirements of 73.55 (a).

2. To "endanger the public health and safety by exposure to radiation" requires a significant off-site release of radioactivity. For LWR's the following sources of significant quantities of radioactivity should be considered:

   a. The reactor core,

   b. Spent fuel,

   c. Radwaste systems, if the total radwaste inventory is greater than nxC, where:

      $\acute{n}$ is the ratio of the applicable dose guideline of 10 CFR 100 to the dose computed for accidental releases in Chapter 15 of the FSAR, and

      C is the release (curies) assumed in the accidental release calculation of the FSAR.

3. Vital Areas fall into two general categories:

   a. Type I vital areas, i.e., those areas wherein successful sabotage can be accomplished by compromising or destroying

the vital systems[1/] or components located within this area. (By definition, an area containing systems or components whose failure or destruction results in a direct release is a Type I vital area.)

b. Type II vital areas, ie., those areas which contain systems or components whose failure or destruction would lead to successful sabotage only in conjunction with additional sabotage activity in at least one other, separate[2/] vital area. (Safety related equipment designed to mitigate the consequences of failures of other systems usually falls into this category.)

4. When classifying vital equipment as Type I or II, the following assumptions apply:

a) The concurrence of violent natural phenomena with a security contingency need not be considered.

b) Random (accidental) failure of equipment concurrent with a security contingency need not be considered. However, a security contingency during routine or planned outages of equipment, as permitted by the technical specifications, must be considered.

1/ "System" refers to all components, mechanical and electrical, including piping, cabling, power supply, and other support systems to carry out the design function provided by the system.

2/ For the purpose of this discussion, a vital area may be considered "separate" if it is separated from the area under consideration by a barrier or distance sufficient to delay the saboteur's access long enough to demonstrate interception and engagement by the security response force.

c) Loss of off-site power must be assumed since it is impractical to protect transmission lines against sabotage.

C. Discussion

The definition of vital equipment, 73.2 (i), includes equipment whose failure would lead to a direct release, as well as equipment required to function for the protection of public health and safety following a postulated sabotage attack. This is analagous to the definition of safety-related equipment, which includes primary fission product barriers, as well as the systems required to mitigate the consequences of a breach of the barrier. Therefore, essentially all safety related equipment must be considered vital. In order to avoid duplication of safety analyses, the systems listed in Reg. Guide 1.29 should be considered vital.

It should be noted that a facility which provides sufficient delay time to permit interruption of the external threat of §(a)(1) at all vital area barriers, and for which adequate protection against the insider threat of §(a)(2) is provided for all vital areas would meet the requirements of 73.55 without the designation of any Type I Vital Areas. In practice, however, it is to the licensee's advantage to segregate vital areas into Type I and II, in order to take credit for the fact that a saboteur could not achieve successful sabotage in Type II vital areas without penetrating additional barriers.

D. Review Guidelines

1. All systems listed in Reg. Guide 1.29 as "Seismic Category I" are considered vital. (A sound technical basis must be provided by the licensee for any deviation from this list.)
2. Type I Vital Areas should be identified by the licensee, using the definitions and assumptions listed in B. If Type I Vital Areas are not identified by the licensee, the list provided in the Appendix may be used as guidance.
3. High assurance protection against the external and internal threat must be provided for all Type I Vital Areas. This requires a demonstration that any external Type I vital barriers provide sufficient delay to the external threat (§(a)(1)) to permit a timely engagement by the armed response force, and appropriately restricted access controls, controls of activity, or other methods of protection against the insider, to meet the internal threat (§(a)(2)). For Type II Vital Areas, a combination of multiple barriers, each of which meets the requirements of 73.2(f)(2) or its equivalent, and the associated individual access controls, provides high assurance protection against the external and internal threat.

Appendix

SAMPLE LIST OF TYPE I VITAL AREAS

1. Primary containment
2. Containment electrical and piping penetration areas
3. Control room
4. Cable spreading room
5. Primary shutdown system (if outside containment)
6. All areas associated with one complete decay heat removal system (including all necessary support systems, e.g., power supply, cooling, and lubricating systems.)
7. Battery rooms (including battery charger areas)

U.S. NUCLEAR REGULATORY COMMISSION

REGULATORY GUIDE

OFFICE OF STANDARDS DEVELOPMENT

Revision 2
February 1976

REGULATORY GUIDE 1.29

SEISMIC DESIGN CLASSIFICATION

A. INTRODUCTION

General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Licensing of Production and Utilization Facilities," requires that nuclear power plant structures, systems, and components important to safety be designed to withstand the effects of earthquakes without loss of capability to perform their safety functions.

Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50 establishes quality assurance requirements for the design, construction, and operation of nuclear power plant structures, systems, and components that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. The pertinent requirements of Appendix B apply to all activities affecting the safety-related functions of those structures, systems, and components.

Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," to 10 CFR Part 100, "Reactor Site Criteria," requires that all nuclear power plants be designed so that, if the Safe Shutdown Earthquake (SSE) occurs, all structures, systems, and components important to safety remain functional. These plant features are those necessary to ensure (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shut down the reactor and maintain it in a safe shutdown condition, or (3) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of 10 CFR Part 100.

This guide describes an acceptable method of identifying and classifying those features of light-water-cooled nuclear power plants that should be designed to withstand the effects of the SSE.

B. DISCUSSION

After reviewing a number of applications for construction permits and operating licenses for boiling and pressurized water nuclear power plants, the NRC staff has developed a seismic design classification system for identifying those plant features that should be designed to withstand the effects of the SSE. Those structures, systems, and components that should be designed to remain functional if the SSE occurs have been designated as Seismic Category I.

C. REGULATORY POSITION

1. The following structures, systems, and components of a nuclear power plant, including their foundations and supports, are designated as Seismic Category I and should be designed to withstand the effects of the SSE and remain functional. The pertinent quality assurance requirements of Appendix B to 10 CFR Part 50 should be applied to all activities affecting the safety-related functions of these structures, systems, and components.

a. The reactor coolant pressure boundary.

b. The reactor core and reactor vessel internals.

c. Systems[1] or portions of systems that are required for (1) emergency core cooling, (2) postaccident containment heat removal, or (3) postaccident

[1] The system boundary includes those portions of the system required to accomplish the specified safety function and connected piping up to and including the first valve (including a safety or relief valve) that is either normally closed or capable of automatic closure when the safety function is required.

containment atmosphere cleanup (e.g., hydrogen removal systems).

d. Systems[1] or portions of systems that are required for (1) reactor shutdown, (2) residual heat removal, or (3) cooling the spent fuel storage pool.

e. Those portions of the steam systems of boiling water reactors extending from the outermost containment isolation valve up to but not including the turbine stop valve, and connected piping of 2-1/2 inches or larger nominal pipe size up to and including the first valve that is either normally closed or capable of automatic closure during all modes of normal reactor operation. The turbine stop valve should be designed to withstand the SSE and maintain its integrity.

f. Those portions of the steam and feedwater systems of pressurized water reactors extending from and including the secondary side of steam generators up to and including the outermost containment isolation valves, and connected piping of 2-1/2 inches or larger nominal pipe size up to and including the first valve (including a safety or relief valve) that is either normally closed or capable of automatic closure during all modes of normal reactor operation.

g. Cooling water, component cooling, and auxiliary feedwater systems[1] or portions of these systems, including the intake structures, that are required for (1) emergency core cooling, (2) postaccident containment heat removal, (3) postaccident containment atmosphere cleanup, (4) residual heat removal from the reactor, or (5) cooling the spent fuel storage pool.

h. Cooling water and seal water systems[1] or portions of these systems that are required for functioning of reactor coolant system components important to safety, such as reactor coolant pumps.

i. Systems[1] or portions of systems that are required to supply fuel for emergency equipment.

j. All electric and mechanical devices and circuitry between the process and the input terminals of the actuator systems involved in generating signals that initiate protective action.

k. Systems[1] or portions of systems that are required for (1) monitoring of systems important to safety and (2) actuation of systems important to safety.

l. The spent fuel storage pool structure, including the fuel racks.

m. The reactivity control systems, e.g., control rods, control rod drives, and boron injection system.

n. The control room, including its associated vital equipment, cooling systems for vital equipment, and life support systems, and any structures or equipment inside or outside of the control room whose failure could result in incapacitating injury to the occupants of the control room.[2]

o. Primary and secondary reactor containment.

p. Systems,[1] other than radioactive waste management systems,[3] not covered by items 1.a through 1.o above that contain or may contain radioactive material and whose postulated failure would result in conservatively calculated potential offsite doses (using meteorology as prescribed by Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors," and Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors") that are more than 0.5 rem to the whole body or its equivalent to any part of the body.

q. The Class 1E electric systems, including the auxiliary systems for the onsite electric power supplies, that provide the emergency electric power needed for functioning of plant features included in items 1.a through 1.p above.

2. Those portions of structures, systems, or components whose continued function is not required but whose failure could reduce the functioning of any plant feature included in items 1.a through 1.q above to an unacceptable safety level should be designed and constructed so that the SSE would not cause such failure.

3. Seismic Category I design requirements should extend to the first seismic restraint beyond the defined boundaries. Those portions of structures, systems, or components that form interfaces between Seismic Category I and non-Seismic Category I features should be designed to Seismic Category I requirements.

4. The pertinent quality assurance requirements of Appendix B to 10 CFR Part 50 should be applied to all activities affecting the safety-related functions of those portions of structures, systems, and components covered under Regulatory Positions 2 and 3 above.

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*Lines indicate substantive changes from previous issue.

[1] See footnote 1, p. 1.29-1.

[2] Wherever practical, structures and equipment whose failure could possibly cause such injuries should be relocated or separated to the extent required to eliminate this possibility.

[3] Specific guidance on seismic requirements for radioactive waste management systems is under development.

D. IMPLEMENTATION

The purpose of this section is to provide information to applicants regarding the NRC staff's plans for using this regulatory guide.

This guide reflects current NRC staff practice. Therefore, except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein is being and will continue to be used in the evaluation of submittals for operating license or construction permit applications until this guide is revised as a result of suggestions from the public or additional staff review.