MAR 16 1983

The Honorable Thomas J. Downey
United States House of Representatives
Washington, D.C. 20515
50-322

Dear Congressman Downey:

This is in response to your letter to Chairman Palladino dated February 14, 1983, concerning Mr. James H. Conran's allegations about the safety of the Shoreham Nuclear Power Plant. The NRC staff has prepared the enclosed point-by-point response to the questions raised in your letter. Please note that we can provide only a partial response to question number 4 at this time. A complete history of the resources assigned to A-17 will be provided as soon as it is prepared.

I hope you find this information responsive to your concerns. Please let me know if I can be of any further assistance.

Sincerely,

(Signed) William J. Dircks

William J. Dircks
Executive Director for Operations

Enclosures:
1. Response to Questions
2. Pages 52 & 53, 1980 Annual Report of the USNRC
3. Page 29, 1981 Annual Report of the USNRC
4. Draft Task Action Plan, Systems Interaction in Nuclear Power Plants (A-17)

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Revised in EDO 5520 - EDO417 - See previous concurrences.

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The Honorable Thomas J. Downey
United States House of Representatives
Washington, DC 20515

Dear Congressman Downey:

This is in response to your letter dated February 14, 1983, concerning Mr. James H. Conran's allegations about the safety of the Shoreham Nuclear Power Plant. You refer to the NRC staff's Unresolved Safety Issue, Systems Interaction in Nuclear Power Plants (A-17). You noted that the US Nuclear Regulatory Commission 1981 Annual Report indicated that reports would be issued under A-17 for Phases I and II. The Phase I report (NUREG/CR-1321) was issued in April 1980. The redirection of the staff's systems interaction program subsequent to the Three Mile Island-2 accident replaced Phase II as it had been initially planned (USNRC 1981 Annual Report, page 29).

You questioned the technical basis to support licensing the Shoreham Plant without a detailed systems interaction study being done. In summary, the technical basis is (a) the NRC staff's current licensing requirements provide reasonable assurance of no undue risk to public health and safety from adverse systems interactions, and (b) the NRC staff's program on systems interactions is confirmatory in nature.

Enclosed is a response to these questions and the numbered questions in your letter. Please note that we can provide only a partial response to question number 4 at this time. A complete history of the resources assigned to A-17 will be provided as soon as it is prepared.

Sincerely,

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NRC FORM 318 (10-80) NRCM 0240 OFFICIAL RECORD COPY USGPO: 1981—335-960

MEMORANDUM FOR: Chairman Palladino

FROM: William J. Dircks
Executive Director for Operations

SUBJECT: PROPOSED REPLY TO CONGRESSMAN DOWNEY CONCERNING SHOREHAM

I enclose, for your approval, a proposed reply to Congressman Downey's February 14, 1983 letter concerning Shoreham and bearing on the issue of systems interactions, Unresolved Safety Issue A-17.

Please note that we can provide only a partial response to question number 4 at this time. A complete history of the resources assigned to USI A-17 will be provided as soon as it is prepared.

William J. Dircks
Executive Director for Operations

Enclosure:
Proposed reply

cc: Commissioner Gilinsky
Commissioner Ahearne
Commissioner Roberts
Commissioner Asselstine
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ENCLOSURE 1

Response to Questions from

Congressman Thomas J. Downey

Transmitted February 14, 1983

QUESTION:

Congressman Downey noted from Table 3 (Schedule for Resolution of Current Unresolved Safety Issues) from the 1981 Annual Report of the U.S. Nuclear Regulatory Commission that reports would be issued under Task Number A-17 for Phases I and II. He was led to understand that the reports were not issued.

RESPONSE:

The NRC staff issued the "Final Report-Phase I Systems Interaction Methodology Applications Program", NUREG/CR-1321, SAND80-0384, April 1980. The staff and the ACRS concluded that the methodology applied in Phase I was unsuccessful because the mathematical model for a broader scoped study would be too large for present computers to manipulate, the model's format was inscrutable for readily indicating important systems interactions, and the application did not discover specific events of interest. (Transcripts of ACRS Subcommittee on Plant Arrangements, February 20, 1980).

The results from Phase I were summarized in the 1980 Annual Report, Enclosure 2 (pages 52 and 53), and we regret that Congressman Downey was misled by Table 3 (page 15) in the 1981 Annual Report. The redirection of the staff's systems interaction program subsequent to Phase I and the Three Mile Island-2 accident is summarized in the 1981 Annual Report on page 29, Enclosure 3. Three other reports were issued during 1981 as part of the redirected program. They are NUREG/CR-1859, NUREG/CR-1896, and NUREG/CR-1901.

QUESTION:

"What technical basis then does the NRC or LILCO suggest supports an operating license without a detailed systems interaction study being done?"

RESPONSE:

The NRC staff support for licensing the Shoreham plant considering systems interactions is provided in the NRC staff Safety Evaluation Report (NUREG-0420) and is summarized as follows:

There is reasonable assurance that the Shoreham plant can be operated without endangering the health and safety of the public. The Shoreham application was evaluated against licensing requirements that were founded on the principle of defense-in-depth. The Shoreham design was reviewed against the

Standard Review Plan (NUREG-0800) which requires interdisciplinary reviews of equipment, and addresses different types of potential systems interactions. Use of NUREG-0800 in the review process results in safety requirements such as physical separation and independence of redundant safety systems, and protection against hazards such as high-energy line ruptures (Section 3.6.1 of NUREG-0800), missiles (Section 3.5.1 & 3.5.2), high winds (Section 3.3), flooding (Sections 3.4, 3.5, & 3.6), seismic events (Section 3.2.1, 3.4, & 3.9.2), and fires (Section 9.5.1). Also, the quality assurance program that is followed during the design, construction, and operational phases for a plant contributes to the prevention of introducing adverse systems interactions. Thus, the existing requirements and licensing review procedures currently provide for an adequate degree of plant safety against potential adverse systems interactions.

Furthermore, the staff's program on Unresolved Safety Issue A-17 was initiated to confirm that present review procedures and safety criteria provide an acceptable level of independence for systems required for safety by evaluating the potential for the more important undesirable interactions between and among systems. To date, the program has provided no indication that present review procedures and criteria do not provide reasonable assurance that the effects of potential systems interactions on plant safety will be within the effects on plant safety previously evaluated (i.e., within the design-basis envelop).

The NRC staff continues to be confident that current regulatory requirements and procedures provide an adequate degree of public health and safety pending the resolution of USI A-17. However, as part of the resolution of this Unresolved Safety Issue, the staff will later determine whether Long Island Lighting Company must have further analyses performed on Shoreham to identify unforeseen significant adverse systems interactions, and whether other plant specific system interaction analyses must be performed.

Below we are providing answers to Congressman Downey's specific questions raised in his February 14, 1983 letter.

QUESTION 1:

Provide "A definition of systems interaction."

RESPONSE:

A rigorous definition of the phrase "systems interaction" does not exist in the sense that the scope of the issue can be derived deductively from the definition. The phrase "systems interaction" denotes the types of events that could occur or have been experienced where an intersystems dependence has jeopardized the designed action of a safety-related system. Examples of such dependencies are provided below in answer to Question 2.

Systems interactions include:

<u>Functionally coupled</u> systems interactions that result either from the sharing of components between systems or through physical connections between systems including electrical, hydraulic, pneumatic and mechanical.

<u>Spatially coupled</u> systems interactions that result from the proximity of systems to one another within the plant.

<u>Induced-humanly coupled</u> systems interactions where a plant malfunction or an error in written procedures induces an operator action.

<u>QUESTION 2</u>:

Provide "...several examples of the type of systems interaction Mr. Conran, the NRC or LILCO think could have adverse effects on Shoreham along with detailed listings of the consequences and probabilities associated with these interactions."

<u>RESPONSE</u>:

Five examples of systems interactions that could occur at a nuclear power plant are:

1. A failure in a power supply causing spurious signals to the control system which in turn can result in opening of relief valves and subsequent loss of primary coolant.

2. A failure in a power supply which could result in failure of control instrumentation leading to a transient resulting in reactor scram.

3. Failure of both vent and drain systems, due to a common discharge, could lead to a partial failure to scram in BWRs.

4. Failure of a turbine could generate a missile which in turn could damage safety related equipment.

5. A fire in some compartments could result in some loss of decay heat removal capability.

The NRC staff believes that the potential occurrence of any significant specific systems interactions at Shoreham is minimized because of the current licensing requirements and procedures that were utilized in reviewing Shoreham. These requirements and procedures have evolved to their current state partially through the NRC's reactions to events that have occurred at other plants like the systems interactions, examples just listed.

The five examples represent adverse systems interactions that are serious safety concerns in themselves, even though their resulting consequences are limited to the systems of the plant. By themselves, their consequences do not extend to a release of radioactive material to the public. These systems interactions would have to occur coincidentally with other independent faults before there would be a release of radioactive material.

In the program to achieve resolution of Task A-17, the staff plans to assess the consequences of each adverse systems interaction which is discovered against the current regulatory requirements which are deterministic. Additionally, the staff plans to assess the consequences and probabilities of adverse systems interactions that have consequences beyond the current regulatory requirements.

QUESTION 3:

Provide "...a copy of the NRC task action plan for the resolution of the systems interaction unresolved safety issue (A-17)."

RESPONSE:

A copy of the draft Task Action Plan is enclosed (Enclosure 4).

QUESTION 4:

Provide "...a complete history of the original schedule of the resolution of A-17 all delays and the budget and person-years assigned to this issue since its identification in 1978."

RESPONSE:

The history of the staff efforts to resolve issue A-17 is summarized in the NRC annual reports. A complete history with the detailed descriptions and budgetary information requested is not readily available. Such a complete history will be provided as soon as it has been prepared. A partial history that describes the most significant delays to A-17 is provided in the response to Question 5.

The more recent history and status of the resolution of A-17 and the budget and person-years assigned is described in the draft Task Action Plan attached in response to Question 3. The plan is in draft form pending approval of a proposed increase in funding.

QUESTION 5:

"...if A-17 has been delayed or the resolution schedule extended as Mr. Conran alleges, when was this decision made, by whom, and for what reason?"

RESPONSE:

There have been schedule slippages that have led to extending the program plans to achieve resolution of A-17. The NRC has not allowed extensions of the schedule for A-17 to compromise the reasonable assurance of public health and safety concerning the Shoreham plant.

The staff's plan to resolve the A-17 issue required completion of the following tasks:

1. Develop methodology for conducting systems interaction analysis.

2. Demonstration of these methods on a small number of LWRs.

3. Review of Industry studies on systems interactions.

4. Develop value and cost of any new regulatory requirement on systems interactions.

In February of 1982 the NRC staff estimated that the efforts in Task 1 were essentially complete and if approval were given by the Director of the Office of Nuclear Reactor Regulation to proceed with selection of LWRs at which demonstration systems interaction analysis could be performed, the NRC staff could come to a decision regarding resolution of A-17 by January 1984. The Director of NRR concluded that the safety value which might result from requiring additional systems interaction analyses had not been demonstrated. In addition, the Director of NRR, taking into consideration that such analyses required concurrent use of other methods (for example, probabilistic risk assessments) to assess the safety significance of any identified adverse interactions and since the systems interaction studies are very expensive, considered other alternatives for Task 2. One alternative considered was to require that systems interaction analyses be performed on the first group of NREP/SEP III (National Reliability Evaluation Program/Systematic Evaluation Program Phase III) plants and conclusions drawn on the basis of results from these studies. Because of delays in NREP/SEP III program, the preferred alternative is to perform systems interaction analyses on one plant, which the utility itself has studied by its own method. This allows the most efficient use of resources for a comparison that is less complicated by plant-wise variations. The staff has recently secured the cooperation of a utility for this phase of the staff's systems interaction program. By October 1984, the staff expects to complete the reviews of all phases of the program and to make a decision on the need for any requirement for plant specific systems interaction analyses. The attached Task Action Plan (in response to Question 3) describes the program's current status and the details of the plan for problem resolution.

In summary, the staff continues to believe that reasonable progress towards a timely resolution of A-17 is being made, and that, pending completion of that effort, there is reasonable assurance that the design, construction and operational practices used for the Shoreham facility provides reasonable assurance that the plant can be operated without endangering public health and safety.

Extracted from

1980 Annual Report of the USNRC

development of useful formulations, advanced material properties and engineering verification is being accomplished by the NRC through several technical assistance contracts with active NRC staff participation. The engineering method will account for radiation-induced material degradation.

Since the publication of the 1979 NRC Annual Report, the following has been accomplished:

(1) The newly developed elastic-plastic fracture test method for routine determination of fracture toughness was employed to provide data from irradiated specimens of pressure vessel steels.

(2) Advanced elastic-plastic fracture mechanics concepts were developed and the results published.

(3) Elastic-plastic fracture mechanics methods were employed to develop formulas for predicting fracture of pressure vessels with both surface and through-wall cracks in the cylindrical shell regions.

(4) A team of recognized experts in the several engineering disciplines involved in Task A-11 was assembled and is working actively under several NRC contracts to evaluate the "J-integral" and "tearing modulus" concepts with respect to reactor pressure vessel applications and revision of existing codes and standards.

Task A-11 is now scheduled to be completed by December 31, 1980, with the issuance of a NUREG report. This delayed completion date remains well in advance of the latest acceptable date to assure that adequate fracture toughness is maintained in those older reactor vessels that will have lower toughness with the passage of time.

Fracture Toughness and Potential for Lamellar Tearing of Component Supports

During the course of the licensing review for a specific pressurized water reactor (PWR), a number of questions were raised as to (1) the adequacy of the fracture toughness properties of the material used to fabricate the reactor coolant pump supports and steam generator supports, and (2) the potential for failure due to lamellar tearing of these same supports. Because materials and designs similar to those of the PWR originally reviewed have been used in other plants, review of this issue was designated as generic Task A-12. This review has recently been expanded to include other PWR supports and the supports of cooling water reactors as well.

Definitive acceptance criteria regarding fracture toughness of all support materials and resistance to stress-corrosion cracking of high-strength support materials were forwarded to licensees and applicants in letters dated May 19 and 20, 1980. Because of negative responses, the NRC staff convened a meeting with licensees, applicants, and other industry representatives in August 1980. The outcome of the meeting was tentative NRC staff acceptance of a program sponsored by industry through the Electric Power Research Institute for resolution of issues regarding fracture toughness and stress corrosion. The NRC staff established the following specific criteria for the industry-sponsored program to be acceptable:

(1) Fracture toughness values must be confirmed.

(2) Plant-specific geometries must be included in the calculations.

(3) Residual stresses must be included.

(4) Methods of determining initial flow size must be clearly defined, and mockup or modeling must be used to demonstrate reliability of non-destructive examination methods.

(5) A probability of failure argument as the sole means of proving acceptability of high strength materials will not be accepted.

In addition, the NRC staff required that the proposed alternative program be presented to the staff by the end of 1980. This program, if found acceptable by the NRC staff, may then be utilized by licensees and applicants. Failure to do this will result in the staff's imposition of its original criteria, modified to incorporate comments deemed applicable.

Lamellar tearing, the second aspect of the problem, is a cracking phenomenon which occurs beneath welds and is principally found in rolled steel plate fabrications. The results of an extensive survey by a consultant to the staff revealed that, although lamellar tearing is a common occurrence in structural steel construction, virtually no inservice failures attributable to lamellar tearing are known. Nonetheless, additional research is being planned to provide a more definitive and complete evaluation of the importance of lamellar tearing to the structural integrity of nuclear power plant support systems. This research will be a follow-on effort to Generic Task A-12. The Electric Research Institute has been asked to fund and manage the desired research.

Systems Interaction In Nuclear Power Plants

In November 1974, the Advisory Committee on Reactor Safeguards requested that the staff give attention to the evaluation of safety systems from a multidisciplinary point of view in order to identify potentially undesirable interactions between plant systems. The concern arises because the design and analysis of systems is frequently assigned to specialists whose focus could lead them to overlook adverse interactions between systems. Task A-17 was initiated to provide an independent investigation of systems required to perform safety functions in order to assess the degree to which the current review procedures take potential systems interactions into account. This investigation has been conducted by Sandia Laboratories under contract assistance to the NRC.

The contractor effort on Phase I of the task began in May 1978 and was completed in March 1980, seeking to identify areas where interactions are possible between systems which could negate or seriously degrade the performance of safety functions. The investigation, conducted by means of "fault tree" analyses, identified the way in which NRC review procedures account for these interactions; it was completed during 1979.

A contractor report was published under the title, "Final Report - Phase I: Systems Interaction Methodology Applications Program" (NUREG/CR-1321, April 1980). Another report providing the NRC staff's conclusions based on the contractor's work was scheduled to be issued in April 1980. However, the Three Mile Island Unit 2 accident caused the NRC staff to consider reorienting the Task A-17 Phase I effort so as to include improved treatment of such matters as operator actions, design errors, and maintenance procedures. It was decided not to disrupt the Phase I effort, which was nearing completion, but rather to consider expanding the Phase II effort to include treatment of TMI-2 related issues.

On February 20, 1980, the NRC staff and its contractor presented the results of the Phase I investigation to the Subcommittee on Plant Arrangements of the Advisory Committee on Reactor Safeguards. While the subcommittee encouraged the NRC staff to continue its investigation using the more disciplined and formal methods of analyses, it nevertheless recommended that the NRC staff provide a demonstration of the efficacy of the "fault tree" method of analysis used in Phase I before extending the investigation to include the treatment of other matters. The NRC staff has been unsuccessful in attempting to demonstrate the efficacy of the fault tree method of analysis for revealing potential systems interactions. Whether the fault tree method of Phase I is practical by itself or needs to be supplemented, or perhaps replaced, by alternative methods needs to be determined. For this reason, the NRC staff's conclusions based on the contractor's work and the scoping of Phase II follow-on work have both been delayed from the forecasted completion date of April 1980. The NRC staff now plans to define a way to demonstrate the analytical method and issue a report on the demonstration by November 1981, and from that base the NRC staff plans to define the scope of Phase II follow-on studies by March 1982.

Concurrent with this effort on Task A-17, the NRC staff and utility applicants and licensees are performing investigations of systems interaction using alternative methods. One method, which will be conducted at the Indian Point Unit 3 plant, employs "failure modes and effects analyses" together with a compartment-by-compartment examination of a plant. Another method which has been performed by the applicant at the Diablo Canyon plant evaluates the overall effect on the plant safety system function of failure of nonseismic equipment, components and structures because of earthquake. This study is now being reviewed by the NRC staff and the ACRS. The staff concluded that there is reasonable assurance that there are no systems interactions from a seismic initiator that can adversely affect safety.

Following the accident at Three Mile Island and as a consequence of the recommendations of the President's Commission on the Accident at Three Mile Island, the NRC Office of Nuclear Reactor Regulation was reorganized to give greater emphasis to integrated review of plant systems.

Environmental Qualification of Safety-Related Electrical Equipment

Safety systems are installed at nuclear plants to mitigate the consequences of postulated accidents. Certain of these postulated accidents could create severe environmental conditions inside the containment, such as high temperature, humidity, pressure, and radiation levels. The most serious such accident would be a high-energy pipe break in the reactor coolant system piping or in a main steam line. In order to assure that electrical equipment in safety systems will perform its function under accident conditions, the NRC requires that such equipment be qualified to perform in the environment associated with the accident. The process of clarifying the criteria has given rise to certain questions regarding the adequacy of qualification tests and analyses. Generic Task A-24 was established to address this question for those plants which received a Construction Permit Safety Evaluation Report after July 1974.

IEEE Standard No. 323 for Qualifying Class IE Equipment for Nuclear Power Generating Stations and its ancillary standards have provided the focal point for the development of environmental qualification requirements in recent years. These standards set forth basic requirements for environmental qualification of electrical equipment and provide varying degrees of detail for implementation of these requirements.

The staff requires in part that, for newer plants (specifically those for which a construction permit

ENCLOSURE 3

Extracted from
1981 Annual Report of the USNRC

quirements. The approved fire protection program is a condition for licensing.

Several operating plants have requested exemptions for specific areas of their plants from certain Appendix R requirements. Evaluation of these requests will be completed by 1983.

Several licensees petitioned the Commission to stay the backfit requirement of 10 CFR 50.48 until a judicial review of these requirements could be made. The Commission denied the petition. The Circuit Court was then requested by the same licensees to review the Appendix R requirements. The Circuit Court has the case under review.

An audit program to review the fire protection at operating plants at three year intervals has been developed by the Office of Inspection and Enforcement.

Reliability and Risk Assessment

The integration of reliability and risk assessment into the regulatory process on a broad scale will be accomplished by the National Reliability Evaluation Program (NREP), to be implemented on a phased schedule on all operating reactors starting in fiscal year 1983. During fiscal year 1981, the NRC staff has participated in two separate efforts to develop procedures guides for performing these probabilistic risk assessments in a comprehensive and scrutable fashion. The methodology development effort is expected to be completed in fiscal year 1982.

As part of the proposed Interim Rule on Construction Permit and Manufacturing License Applications, the staff required applicants to develop programs for performing probabilistic risk studies within two years of issuance of a construction permit, with the goal of improving the reliability of core and containment cooling functions. Guidelines were issued on potential areas where reliability improvements would be considered based on the result of the risk study. Risk/reliability programs were reviewed and approved for four license applications. In a separate action, the staff identified Millstone Unit Three as a plant under construction in a high-population-density site and required the applicant to perform a risk study which would be reviewed as part of the consideration of an operating license several years hence. The staff has been routinely reviewing reliability studies for auxiliary feedwater systems of pressurized water reactors, as submitted by applicants for operating licenses.

An independent generic evaluation was made of a concern that stemmed from an abnormal occurrence in one of the boiling water reactors at the Browns Ferry nuclear plant in June 1980, when about half of the control rods failed to insert fully during a scram (reactor shutdown). During scram, the control rods are hydraulically inserted in the reactor core. Hydraulic discharge lines from the control rods penetrate the primary containment and come together in the Scram Discharge Volume (SDV) in the reactor building. A large unisolated pipe break in the SDV could result in continuous loss of reactor coolant and melting of the reactor core if left unattended for an extended period of time. To prevent this, decisions would have to be made in the control room to reset the scram signal or to follow a depressurization procedure. Decision "trees" (step-by-step diagrams) were devised to quantify the probability of operator failure to carry out these actions. The result of the study regarding the estimated frequency of such an event, combined with the estimated probability of corrective operator action, provided an important basis for the judgment that this type of an event is not a significant contributor to the probability of core melt. Consequently, the only action taken by the staff was to assure that adequate procedures and instrumentation are available to cope with such an event.

Systems Interaction

The staff program of systems interaction was initiated in May 1978 with the definition of Unresolved Safety Issue A-17 and was intensified by Item II.C.3 of the Three Mile Island Action Plan. The concern arises because the design, analysis and installation of systems are frequently the responsibility of teams of engineers with functional specialties—such as civil, electrical, mechanical or nuclear. Experience at operating plants has led to questions of whether the work of these functional specialists is sufficiently integrated to enable them to minimize adverse interactions among systems.

Staff efforts on systems interaction during fiscal year 1981 were directed principally toward surveying available methods and developing preliminary guidance for the performance of comprehensive analyses and reviewing the results of a recent analysis of the Diablo Canyon and San Onofre facilities in California for potential seismic-initiated interactions. The staff also completed the acceptance review of a program for a comprehensive analysis of systems interaction to be performed at Indian Point Unit 3 in New York.

During the coming year, the staff will complete development of regulatory guidance for application in pilot analyses of systems interaction planned at some new plants nearing completion of construction. The staff will also be evaluating the conduct of the Indian Point-3 analysis scheduled to begin in November 1981 and will be reviewing the results of that effort.

SPECIFIC CONCERNS

Occupational Radiation Exposures

An analysis of the occupational radiation exposures at operating light water reactors (LWR's) for