22/A-17

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October 21, 1977

Mr. Victor Stello, Director
Division of Operating Reactors
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Zion Station Units 1 and 2
<u>NRC Docket Nos. 50-295 and 50-304</u>

Dear Mr. Stello:

The NRC Staff has requested that Commonwealth Edison Company provide them with an outline of the proposed review of Zion Station relative to the ACRS concerns about Backfit and Systems Interaction. These proposals were discussed with the NRC Staff at meetings on August 17 and September 29, 1977.

The outline of our proposed reviews are attached to this letter.

Very truly yours,

for R. L. Bolger
R. L. Bolger
Assistant Vice President

Attachments

Commonwealth Edison

Zion Station Units 1 and 2

Proposed Zion System Interaction Study

NRC Docket Nos. 50-295 and 50-304

Proposed Zion System Interaction Study

The ACRS, in it's letter of June 17, 1977, recommended that Commonwealth Edison conduct a "study of systems interaction relating to the possibility that failure of safety and non-safety systems will interfere with the plant operator's ability to accomplish shutdown heat removal." The Committee cited general lack of understanding of the term "systems interaction" as the reason little progress had been made toward resolution of their concerns. The ACRS letter to L. Manning Muntzing of November 8, 1974 was referenced as a source of guidance in this matter. Since the receipt of the June 17 letter, Commonwealth Edison has reviewed the 1974 letter, and has discussed possible approaches to system interaction studies with several consultants and the NRC staff. A satisfactory plan has been devised and will be described here.

The 1974 ACRS letter expresses basically one concern regarding the design of nuclear plants: the design effort should be co-ordinated so that identifiable systems interactions are reviewed. The Committee emphasized their interest in a multidisciplinary review approach which incorporates all the relevant design specialties. They went on to discuss specific examples. In Commonwealth Edison's various presentations to the ACRS and the NRC staff, both the general and the specific concerns have been addressed. The multidisciplinary nature of the design process has been described. Similar reviews of operational event records and facility changes have been detailed. The specific examples cited in the 1974 letter have all been addressed.

In the intervening years new modes of system interaction have been identified, primarily as a byproduct of plant operation. Some of these events have had undesirable consequences, but none have had significant safety impact. The ACRS has cited the Zion diesel fire as an example of such interactions. They feel that an additional study should be performed.

Commonwealth Edison has reviewed ACRS letters and transcripts and has contacted several consultants on the subject of systems interaction. Several ideas have been discussed with the NRC staff. Physical surveys, electrical logic and interdependency reviews have been considered. All seem to be limited by the imagination of the individuals performing the study. None is completely rigorous. It is anticipated that new types of system interactions will continue to appear as long as the plants continue to operate. Hopefully, most events will be innocuous. The important thing is that each new event should be reviewed in detail to assure that similar events having undesirable consequences are not likely.

For the present, plans for physical surveys and logic reviews have been set aside in favor of detailed reviews of those events which have occurred that involve undesirable systems interaction. Both physical and electrical interactions would be covered in the event review but they would be approached on a case-by-case basis rather than from a more general standpoint. The event review would be straightforward in its scope. The review would attempt to encompass all those systems interaction modes which have been identified, and none of those which have not.

Licensee Event Reports generated since 1969 would be the basis for the event review. There are approximately 15,000 such LER's. Simply reviewing the LER's for those involving systems interaction would be a major project. The events identified in the first phase of the project would be subjected to detailed review in the second phase. The list is expected to contain between 100 and 1,000 events.

It is expected that the first phase of the review (LER Scanning) could be complete by December 1, 1977. The list generated would be submitted to the NRC staff for review and approval. Without knowing the number of items on the list, it is impossible to project the completion of the second phase (LER Study). If the list is small, the review could certainly be complete by May 1, 1978. A report would list the events reviewed and summarize the findings.

The LER Scanning phase of the project would be conducted by engineering personnel with broad knowledge of plant systems. They would be directed to identify those events which involved systems necessary to accomplish shutdown heat removal under non-accident conditions. Specifically, these systems are:

Reactor Coolant	Auxilary Power
Residual Heat Removal	Instrumentation Power
Component Cooling	Chemical & Volume Control
Service Water	Auxilary Feedwater

System interaction events would be identified wherever the action of any one system threatened or resulted in the loss of effectiveness of any of the systems identified above, whether the initiating action was a normal control action, a malfunction, or operator-initiated. There is no plan to extend the single failure criterion; rather the plan is to extend it's applicability by the detailed review.

The LER Study phase would involve multi-disciplinary review of each event. Responsibility for review of an event would be assigned to an individual with a background in the area to be studied. The diesel generator fire, for example, would be assigned to someone with a background in power plant electrical distribution systems. If an event has implications for design aspects other than this individual's particular specialty, individuals with other technical backgrounds would be drawn into the review of that event. Results of the review would be documented and used in preparation of summaries for inclusion in the final report. If the occurrence of undesirable system interactions are found to be likely, specific recommendations would be put forth in that report.

At this point in time copies of the computer listings of the 15,000 LER's are being obtained. It is expected that detailed review of the LER's would require that the original LER form be obtained. Once the preparer's name has been acquired, further details could be requested. Field trips might also be needed to complete the review. The assistance of NRC staff personnel might also be needed.

Commonwealth Edison

Zion Station Units 1 and 2

Proposed Zion Backfit Study

NRC Docket Nos. 50-295 and 50-304

Proposed Zion Backfit Study

The Zion plant would be reviewed against current design criteria to identify differences which could have a major impact on safety. Regulatory Guides and Standard Review Plan Branch Technical Positions identified on the attached list would be included in the review. Features of the Byron/Braidwood design would be referenced as examples of design details which are compatible with the current criteria. Concurrent with the design review, an attempt would be made to assess feasibility and cost of backfits.

The list of Guides and Positions to be included would be finalized by November 1, 1977. A report of the review could be ready for submittal to the NRC staff by about June 1, 1978. The report would describe the approach used and document the results.

DIVISION 1 REGULATORY GUIDES
POWER REACTORS

Number	Title
1.1	Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps (Safety Guide 1, 11/2/70)
1.2	Thermal Shock to Reactor Pressure Vessels (Safety Guide 2, 11/2/70)
1.4	Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors (Revision 2, 6/74)
1.6	Independence Between Redundant Standby (Onsite) Power Sources Between Their Distribution Systems (Safety Guide 6, 3/10/71)
1.7	Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident (Revision 1, 9/76)
1.8	Personnel Selection and Training (Revision 1-R, 9/75; reissued 5/77)
1.9	Selection of Diesel Generator Set Capacity for Standby Power Supplies (Safety Guide 9, 3/10/71)
1.11	Instrument Lines Penetrating Primary Reactor Containment (Safety Guide 11, 3/10/71)
	Supplement to Safety Guide 11, Backfitting Considerations (2/17/72)
1.12	Instrumentation for Earthquakes (Revision 1, 4/74)
1.13	Spent Fuel Storage Facility Design Basis (Revision 1, 12/75)
1.14	Reactor Coolant Pump Flywheel Integrity (Revision 1, 8/75)
1.16	Reporting of Operating Information--Apendix A Technical Specifications (Revision 4, 8/75)
1.17	Protection of Nuclear Power Plants Against Industrial Sabotage (Revision 1, 6/73)
1.21	Measuring, Evaluating and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents for Light-Water-Cooled Nuclear Power Plants (Revision 1, 6/74)
1.22	Periodic Testing of Protection System Actuation Functions (Safety Guide 22, 2/17/72)
1.23	Onsite Meteorological Programs (Safety Guide 23, 2/17/72)
1.24	Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure (Safety Guide 24, 3/23/72)
1.25	Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors (Safety Guide 25, 3/23/72)
1.27	Ultimate Heat Sink for Nuclear Power Plants (Revision 2, 1/76)
1.28	Quality Assurance Program Requirements (Design and Construction) (Safety Guide 28, 6/7/72)
1.32	Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants (Revision 2, 2/77)
1.33	Quality Assurance Program Requirements (Operation) (Revision 1, 1/77)
1.37	Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants (3/16/73)

Number	Title
1.39	Housekeeping Requirements for Water-Cooled Nuclear Power Plants (Revision 1, 10/76)
1.40	Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants (3/16/73)
1.42	(Withdrawn--See 41 FR 11891, 3/22/76)
1.45	Reactor Coolant Pressure Boundary Leakage Detection Systems (5/73)
1.46	Protection Against Pipe Whip Inside Containment (5/73)
1.47	Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems (5/73)
1.49	Power Levels of Nuclear Power Plants (Revision 1, 12/73)
1.51	(Withdrawn--See 40 FR 30510, 7/21/75)
1.52	Design, Testing, and Maintenance Criteria for Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants (Revision 1, 7/76)
1.53	Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems (6/73)
1.56	Maintenance of Water Purity in Boiling Water Reactors (6/73)
1.58	Qualification of Nuclear Power Plant Inspection, Examination and Testing Personnel (8/73)
1.59	Design Basis Floods for Nuclear Power Plants (Rev. 2, 8/77)
1.62	Manual Initiation of Protective Actions (10/73)
1.65	Materials and Inspections for Reactor Vessel Closure Studs (10/73)
1.67	Installation of Overpressure Protection Devices (10/73)
1.73	Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants (1/74)
1.74	Quality Assurance Terms and Definitions (2/74)
1.75	Physical Independence of Electric Systems (Revision 1, 1/75)
1.76	Design Basis Tornado for Nuclear Power Plants (4/74)
1.77	Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors (5/74)
1.78	Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release (6/74)
1.81	Shared Emergency and Shutdown Electric Systems for Mul nit Nuclear Power Plants (Revision 1, 1/75)
1.82	Sumps for Emergency Core Cooling and Containment Spray Systems (6/74)
1.83	Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes (Revision 1, 7/75)
1.86	Termination of Operating Licenses for Nuclear Reactors (6/74)
1.88	Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records (Revision 2, 10/76)
1.89	Qualification of Class IE Equipmen for Nuclear Power Plants (11/74)
1.91	Evaluation of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plant Sites (1/75)
1.93	Availability of Electric Power Sources (12/74)
1.95	Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release (Revision 1, 1/77)
1.97	Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident (Rev. 1, 8/77)
1.99	Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials (Revision 1, 4/77)

Number	Title
1.100	Seismic Qualification of Electric Equipment for Nuclear Power Plants (Revision 1, 8/77)
1.101	Emergency Planning for Nuclear Power Plants (Rev. 1, 3/77)
1.102	Flood Protection for Nuclear Power Plants (Revision 1, 9/76)
1.105	Instrument Setpoints (Revision 1, 11/76)
1.106	Thermal Overload Protection for Electric Motors on Motor-Operated Valves (Revision 1, 3/77)
1.108	Periodic Testing of Diesel Generators Used As Onsite Electric Power Systems at Nuclear Power Plants (Rev. 1, 8/77)
1.109	Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I (3/76)
1.110	Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors (3/76)
1.111	Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors (Revision 1, 7/77)
1.112	Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors (Revision 0-R, 4/76; reissued 5/77)
1.113	Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I (Revision 1, 4/77)
1.114	Guidance on Being Operator at the Controls of a Nuclear Power Plant (Revision 1, 11/76)
1.115	Protection Against Low-Trajectory Turbine Missiles (Revision 1, 7/77)
1.117	Tornado Design Classification (6/76)
1.118	Periodic Testing of Electric Power and Protection Systems (6/76)
1.120	Fire Protection Guidelines for Nuclear Power Plants (6/76)
1.121	Bases for Plugging Degraded PWR Steam Generator Tubes (8/76)
1.126	An Acceptable Model and Related Statistical Methods for the Analysis of Fuel Densification (3/77)
1.127	Inspection of Water-Control Structures Associated with Nuclear Power Plants (4/77)
1.128	Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants (4/77)
1.129	Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Nuclear Power Plants (4/77)
1.134	Medical Certification and Monitoring of Personnel Requiring Operator Licenses (9/77)

STANDARD REVIEW PLANS

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3.4.1	Flood Protection
3.4.2	Analysis Procedures
3.5.1.1	Internally Generated Missiles (Outside Containment)
3.5.1.2	Internally Generated Missiles (Inside Containment)
3.5.1.3	Turbine Missiles
3.5.1.4	Missiles Generated by Natural Phenomena
3.5.1.5	Site Proximity Missiles (Except Aircraft)
3.5.1.6	Aircraft Hazards
3.5.2	Structures, Systems, and Components to be Protected from Externally Generated Missiles
3.6.1	Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment
3.6.2	Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping
3.7.4	Seismic Instrumentation
3.9.1	Special Topics for Mechanical Components
3.9.2	Dynamic Testing and Analysis of Mechanical Systems and Components
3.9.3	ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures
3.9.4	Control Rod Drive Systems
3.9.6	Enservice Testing of Pumps and Valves
3.10	Seismic Qualification of Category I Instrumentation and Electric Equipment
3.11	Environmental Design of Mechanical and Electrical Equipment
5.2.1.1	Compliance with 10 CFR 50.55a
5.2.1.2	Applicable Code Cases
5.2.2	Overpressurization Protection
5.2.4	RCPB Inservice Inspection & Testing
5.2.5	RCPB Leakage Detection
5.3.2	Pressure-Temperature Limits
5.3.3	Reactor Vessel Integrity
5.4.1.1	Pump Flywheel Integrity (PWR)
5.4.2.2	Steam Generator Inservice Inspection
5.4.7	Residual Heat Removal (RHR) System
5.4.11	Pressurizer Relief Tank System
6.2.1	Containment Functional Design
6.2.1.1.A	PWR Dry Containments, Including Subatmospheric Containments
6.2.1.2	Subcompartment Analysis
6.2.1.3	Mass and Energy Release Analysis for Postualted Loss-of-Coolant Accidents
6.2.1.4	Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures
6.2.1.5	Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies
6.2.2	Containment Heat Removal Systems
6.2.4	Containment Isolation System
6.2.6	Containment Leakage Testing
6.3	Emergency Core Cooling System
6.4	Habitability Systems

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6.5.1	ESF Filter Systems
6.5.2	Containment Spray as a Fission Product Cleanup System
6.6	Inservice Inspection of Class 2 and 3 Components
7.1	Introduction
7.2	Reactor Trip System
7.3	Engineered Safety Feature Systems
7.4	Systems Required for Safe Shutdown
7.5	Safety-Related Display Instrumentation
7.6	All Other Instrumentation Systems Required for Safety
7.7	Control Systems Not Required for Safety
Appendix 7-A	Branch Technical Positions (EICSB)
Appendix 7-B	General Agenda, Station Site Visits
Table 7-1	Acceptance Criteria for Controls
8.1	Introduction
8.3.1	A-C Power Systems (Onsite)
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9.1.1	New Fuel Storage
9.1.2	Spent Fuel Storage
9.1.3	Spent Fuel Pool Cooling and Cleanup System
9.1.4	Fuel Handling System
9.2.1	Station Service Water System
9.2.2	Reactor Auxiliary Cooling Water Systems
9.2.3	Demineralized Water Makeup System
9.2.4	Potable and Sanitary Water Systems
9.2.5	Ultimate Heat Sink
9.2.6	Condensate Storage Facilities
9.3.1	Compressed Air System
9.3.2	Process Sampling System
9.3.3	Equipment and Floor Drainage System
9.3.4	Chemical and Volume Control System (PWR) (Including Boron Recovery System)
9.4.1	Control Room Area Ventilation System
9.4.2	Spent Fuel Pool Area Ventilation System
9.4.3	Auxiliary and Radwaste Area Ventilation System
9.4.4	Turbine Area Ventilation System
9.4.5	Engineered Safety Feature Ventilation System
9.5.1	Fire Protection System
9.5.2	Communications Systems
9.5.3	Lighting Systems
9.5.4	Emergency Diesel Engine Fuel Oil Storage and Transfer System
9.5.5	Emergency Diesel Engine Cooling Water System
9.5.6	Emergency Diesel Engine Starting System
9.5.7	Emergency Diesel Engine Lubrication System
9.5.8	Emergency Diesel Engine Combustion Air Intake and Exhaust System
10.2	Turbine Generator
10.2.3	Turbine Disk Integrity
10.3	Main Steam Supply System
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Standard Review Plans (Cont'd)

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10.4.3	Turbine Gland Sealing System
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10.4.7	Condensate and Feedwater System
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10.4.9	Auxiliary Feedwater System (PWR)
11.1	Source Terms
11.2	Liquid Waste Management Systems
11.3	Gaseous Waste Management Systems
11.4	Solid Waste Management Systems
11.5	Process and Effluent Radiological Monitoring and Sampling Systems
12.1	Assuring That Occupational Radiation Exposures are As Low As Is Reasonably Achievable
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12.3	Radiation Protection Design Features
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13.1.1	Management and Technical Support Organization
13.1.2	Operating Organization
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15.1.5	Spectrum of Stream System Piping Failures Inside and Outside of Containment (PWR)
15.2.1-15.2.5	Loss of External Load, Turbine Trip, Loss of Condenser Vacuum, Closure of Main Steam Isolation Valve (BWR), and Steam Pressure Regulator Failure (Closed)
15.2.6	Loss of Non-Emergency A-C Power to the Station Auxiliaries
15.2.7	Loss of Normal Feedwater Flow
15.2.8	Feedwater System Pipe Breaks Inside and Outside Containment (PWR)
15.3.1-15.3.2	Loss of Forced Reactor Coolant Flow Including Trip of Pump and Flow Controller Malfunctions
15.3.3-15.3.4	Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break
15.4.1	Uncontrolled Control Rod Assembly Withdrawal From a Subcritical or Low Power Startup Condition

Standard Review Plans (Cont'd)

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15.4.3	Control Rod M-soperation (System Malfunction or Operator Error)
15.4.4-15.4.5	Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR CORE FLOW RATE
15.4.6	Chemical and Volume Control System Malfunction That Results in a Decrease in the Boron Concentration in the Reactor Coolant (PWR)
15.4.7	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position
15.4.8	Spectrum of Rod Ejection Accidents (PWR)
15.4.9	Spectrum of Rod Drop Accidents (BWR)
15.5.1-15.5.2	Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction That Increases Reactor Coolant Inventory
15.6.1	Inadvertent Opening of a PWR Pressurizer Safety/Relief Valve or a BWR Safety/Relief Valve
15.6.2	Failure of Small Lines Carrying Primary Coolant Outside Containment
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15.6.5	Loss-of-Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary
15.7.1	Waste Gas System Failure
15.7.2	Radioactive Liquid Waste System Leak or Failure (Release to Atmosphere)
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15.8	Anticipated Transients Without Scram
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