RAS-1604

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

'00

In the Matter of:	:	**Docket No. 50-423-LA-3**
	:	
Northeast Nuclear Energy Company	:	
	:	
(Millstone Nuclear Power Station, Unit No. 3)	:	**ASLBP No. 00-771-01-LA**

CONNECTICUT COALITION AGAINST MILLSTONE AND LONG ISLAND COALITION AGAINST MILLSTONE RESPONSE TO NORTHEAST NUCLEAR ENERGY COMPANY'S FIRST REQUEST FOR PRODUCTION

The Connecticut Coalition Against Millstone ("CCAM") and Long Island Coalition Against Millstone ("CAM") (collectively, "Intervenors") herewith provide documents responsive to the Northeast Nuclear Energy Company's First Request for Production, as follows:

IV. GENERAL DOCUMENT PRODUCTION REQUESTS

Request No. G-1 and G-2: Documents identified in Exhibit A and Exhibit B in Intervenors' Reply to NNECO's First Set of Interrogatories, in addition to the following:

a. U.S. Nuclear Regulatory Commission Reactor Event No. 36828 on March 23, 2000 (Farley Unit 1)

b. NUREG-1431 (available on N.R.C. website at: http://www.nrc.gov/NRR/sts/sts.htm#1431)

V. SPECIFIC DOCUMENT PRODUCTION REQUESTS

Request Nos. 4-1 - 4-6 and 5-1 - 5-3:

Documents identified in Exhibit A in Intervenors' Reply to NNECO's First Set of Interrogatories, in addition to the following:

a. U.S. Nuclear Regulatory Commission Reactor Event No. 36828 on March 23, 2000 (Farley Unit 1)

b. NUREG-1431 (available on N.R.C. website at: http://www.nrc.gov/NRR/sts/sts.htm#1431)

Template = SECY-035

SECY-02

Request Nos. 6-1 - 6-5:

Documents identified in Intervenors' Reply to NNECO's First Set of Interrogatories as Exhibit B.

Respectfully submitted,

Nancy Burton, Esq.
147 Cross Highway
Redding Ridge CT 06876
Tel. 203-938-3952
ATTORNEY FOR
CT COALITION AGAINST MILLSTONE
LI COALITION AGAINST MILLSTONE

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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Northeast Nuclear Energy Company	**:**
	:
(Millstone Nuclear Power Station, Unit No. 3)	**:**
	:ASLBP No. 00-771-01-LA

CERTIFICATE OF SERVICE

I hereby certify that copies of "Connecticut Coalition Against Millstone and Long Island Coalition Against Millstone Response to Northeast Nuclear Energy Company's First Request for Production" and the documents identified therein in the above-captioned proceeding have been served on the following by deposit in the United States Mail, first class, this 29th day of March, 2000.

David A. Repka, Esq.
Winston & Strawn
1400 L Street NW
Washington DC 20005
(served by U.S. Express Mail)

Office of the Secretary
U.S. Nuclear Regulatory Commission
Washington DC 20555

Adjudicatory File
Atomic Safety and Licensing Board Panel
U.S. Nuclear Regulatory Commission
Washington DC 20555

Office of Commission Appellate Adjudication
U.S. Nuclear Regulatory Commission
Washington DC 20555

Charles Bechoefer
Chairman
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington DC 20555-0001

Dr. Richard F. Cole
Administrative Judge
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
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Dr. Charles N. Kelber
Administrative Judge
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Nancy Burton

Nancy Burton, Esq.
147 Cross Highway
Redding Ridge CT 06876

org Daily Events Report
U.S. Nuclear Regulatory Commission

Operations Center

Event Reports For

03/23/2000 - 03/24/2000

** EVENT NUMBERS **36825 36826 36827 36828 36829Power
Reactor Event Number: 36826FACILITY:
FARLEY NOTIFICATION DATE:
03/23/2000UNIT:
1
NOTIFICATION TIME: 13:17[EST]|

EVENT TEXTTHREE FUEL ASSEMBLIES DETERMINED TO BE IN WRONG POSITIONSIN THE SPENT FUEL POOLUnit 1 personnel determined that three fuel assemblies, in the Unit 1 SpentFuel Pool (SFP), were in positions inconsistent with the Technical Specifications (TS). Preliminary assessment indicates that the Keff limit of 0.95 (TS 3.7.15) for the SFP would have still been met. This condition existed for ten days, since the last Unit 1 core offload. The assemblies have been returned to the positions allowed by TS. An investigation is in progress to determine the cause of this event and to determine if the Keff limit was exceeded. There will be a 30 day written report submitted.

LIST OF EXHIBITS TO ORANGE COUNTY'S SUMMARY AND SWORN SUBMISSION REGARDING CONTENTION TC-2

1. Declaration of Dr. Gordon Thompson in Support of Orange County's Summary and Sworn Statement Regarding Contention TC-2 (January 4, 2000)

2. Letter from Brian K. Grimes of the NRC Staff to All Power Reactor Licensees (April 14, 1978)

3. Draft 1, Regulatory Guide 1.13, Revision 2, "Spent Fuel Storage Facility Design Basis (December 1981)

4. Memorandum from Laurence Kopp, NRC, to Timothy Collins, NRC, re: Guidance On The Regulatory Requirements For Criticality Analysis Of Fuel Storage At Light-Water Reactor Power Plants (August 19, 1998)

5. Letter from Donna B. Alexander, CP&L, to U.S. NRC, enclosing response to April 29, 199, RAI (June 14, 1999)

6. Transcript of Deposition of Michael J. DeVoe, P.E. (October 20, 1999)

7. AEC Press Release entitled "AEC seeking public comment on proposed design criteria for nuclear power plant construction permits" (November 22, 1965)

8. Internal AEC memorandum from G.A. Arlotto to J.J. DiNunno and Robert H. Bryan (October 7, 1966), and attached Revised Draft of General Design Criteria for Nuclear Power Plant Construction Permits (October 6, 1966) (relevant excerpts)

9. Letter from J J DiNunno, AEC, to David Okrent, ACRS (October 25, 1966), and attached October 20, 1966 draft of General Design Criteria (relevant excerpts)

10. Letter from J. J. DiNunno, AEC, to Nunzio J. Palladino, ACRS (February 8, 1967), and attached draft of General Design Criteria (relevant excerpts)

11. Note by the Secretary, W.B. McCool, to AEC Commissioners re: Proposed Amendment to 10 CFR 50: General Design Criteria for Nuclear Power Plant Construction Permits (June 16, 1967) (relevant excerpts)

12. Notice of proposed rulemaking for General Design Criteria, 32 Fed. Reg. 10,213 (July 11, 1967)

13. Letter from William B. Cottrell, ORNL, to H. L. Price, AEC (September 6, 1967) ane enclosed ORNL comments on proposed GDC.

14. Letter from Edson G. Case, AEC, to Dr. Stephen H. Hanauer, ACRS (July 23, 1969), enclosing General Design Criteria for Nuclear Power Units (July 15, 1969) (relevant excerpts)

15. Memorandum from Edson G. Case, NRC, to Harold L. Price, et al., AEC, re: Revised General Design Criteria (October 12, 1970), and enclosed letter from Edward A. Wiggin, AIF, to Edson G. Case, NRC (October 6, 1970)

16. Final Rule, General Design Criteria for Nuclear Power Plants, 36 Fed. Reg. 3,255 (February 20, 1971)

17. Letter from Donna B. Alexander, CP&L, to U.S. NRC (October 15, 199), enclosing letter from Scott H. Pellet, Holtec International, to Steven Edwards, CP&L (October 11, 1999)

CONTENTION TC-2: EXHIBIT 1

Declaration of Dr. Gordon Thompson in Support of Orange County's Summary and Sworn Statement Regarding Contention TC-2 (January 4, 2000)

January 4, 2000

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
CAROLINA POWER & LIGHT	)	Docket No. 50-400
(Shearon Harris Nuclear	)	
Power Plant)	)	
	)	

DECLARATION OF DR. GORDON THOMPSON IN SUPPORT OF ORANGE COUNTY'S SUMMARY AND SWORN SUBMISSION REGARDING CONTENTION TC-2 (INADEQUATE PREVENTION OF CRITICALITY)

I, Gordon Thompson, declare as follows:

1. I am the executive director of the Institute for Resource and Security Studies (IRSS), a nonprofit, tax-exempt corporation based in Massachusetts. Our office is located at 27 Ellsworth Avenue, Cambridge, MA 02139. IRSS was founded in 1984 to conduct technical and policy analysis and public education, with the objective of promoting peace and international security, efficient use of natural resources, and protection of the environment.

2. I received an undergraduate education in science and mechanical engineering at the University of New South Wales, in Australia. Subsequently, I pursued graduate studies at Oxford University and received from that institution a Doctorate of Philosophy in mathematics in 1973, for analyses of plasmas undergoing thermonuclear fusion. During my graduate studies I was associated with the fusion research program of the UK Atomic Energy Authority.

3. During my professional career, I have performed technical and policy analyses on a range of issues related to international security, energy supply, environmental protection, and sustainable use of natural resources. Since 1977, a significant part of my work has consisted of technical analyses of safety and environmental issues related to nuclear facilities. These analyses have been sponsored by a variety of nongovernmental organizations and local, state and national governments, predominantly in North America and Western Europe. Drawing upon these analyses, I have provided expert testimony in legal and regulatory proceedings, and have served on committees advising US government agencies. A copy of my resume is appended as Attachment A to the Declaration of Dr. Gordon Thompson (February 12, 1999), which is attached as Exhibit 2 to Orange County's Supplemental Petition to Intervene (April 5, 1999).

4. I have reviewed the December 23, 1998, license amendment application filed by Carolina Power and Light (CP&L) for an amendment to Facility Operating License No. NPF-63, which seeks permission to activate spent fuel storage pools C and D at the Shearon Harris nuclear power plant. I have also reviewed the NRC's Federal Register notice for the proposed license amendment, the Final Safety Analysis Report for the Shearon Harris Nuclear Power Plant, and the Final Environmental Statement related to the operation of Shearon Harris Nuclear Power Plant, Units 1 and 2 (NUREG-0972, October 1983). In addition, I reviewed various correspondence and technical documents relating to the proposed license amendment and to risks of spent fuel storage, which are identified in Orange County's contentions.

5. I participated in the preparation of Orange County's contentions regarding the proposed license amendment. Following admission of Contention TC-2, Inadequate Criticality Prevention, I was principally responsible for evaluating whether CP&L's License Amendment Application conforms to the requirements of General Design Criterion 62 and applicable NRC Staff guidance.

6. In making my evaluation, I conducted an extensive review of documents related to criticality prevention at Harris and in general, including correspondence between CP&L and the NRC Staff, criticality studies performed by or for CP&L, NRC Staff and licensee documents regarding proposed spent fuel storage pool expansion applications, Licensee Event Reports of criticality-related occurrences, NRC Staff and industry guidance documents and related correspondence, the rulemaking history of GDC 62, and other publicly available information regarding spent fuel storage and criticality prevention. I also participated in preparing for depositions of CP&L and NRC Staff witnesses regarding contention TC-2, and in reviewing the deposition testimony of these witnesses. In addition, I was deposed by both CP&L and the NRC Staff.

7. I am responsible for all of the technical factual assertions contained in Orange County's Detailed Summary Of Facts, Data And Arguments On Which Orange County Intends To Rely At Oral Argument To Demonstrate The Existence Of A Genuine And Substantial Dispute Of Fact With The Licensee Regarding The Proposed Expansion Of Spent Fuel Storage Capacity At The Harris Nuclear Power Plant, With Respect To Criticality Prevention Issues (Contention TC-2), including Appendices A, B, and C, submitted to the Licensing Board on January 4, 2000 (hereinafter "Summary"). As I have attested in signing the Summary, the technical factual assertions therein are true and correct to the best of my knowledge, and all expressions of technical opinion therein are based on my best professional judgment.

I declare, under penalty of perjury, that the foregoing is true and correct.

Executed on January 4, 2000.

G. R. Thompson

Gordon Thompson

CONTENTION TC-2: EXHIBIT 2

Letter from Brian K. Grimes of the NRC Staff to All Power Reactor Licensees (April 14, 1978)

ENCLOSURE 2

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON D. C. 20555

April 14, 1978

To All Power Reactor Licensees

Gentlemen:

Enclosed for your information and possible future use is the NRC guidance on spent fuel pool modifications, entitled "Review and Acceptance of Spent Fuel Storage and Handling Applications". This document provides (1) additional guidance for the type and extent of information needed by the NRC Staff to perform the review of licensee proposed modifications of an operating reactor spent fuel storage pool and (2) the acceptance criteria to be used by the NRC Staff in authorizing such modifications. This includes the information needed to make the findings called for by the Commission in the Federal Register Notice dated September 16, 1975 (copy enclosed) with regard to authorization of fuel pool modifications prior to the completion of the Generic Environmental Impact Statement, "Handling and Storage of Spent Fuel from Light Water Nuclear Power Reactors".

The overall design objectives of a fuel storage facility at a reactor complex are governed by various Regulatory Guides, the Standard Review Plan (NUREG-75/087), and various industry standards. This guidance provides a compilation in a single document of the pertinent portions of these applicable references that are needed in addressing spent fuel pool modifications. No additional regulatory requirements are imposed or implied by this document.

Based on a review of license applications to date requesting authorization to increase spent fuel storage capacity, the staff has had to request additional information that could have been included in an adequately documented initial submittal. If in the future you find it necessary to apply for authorization to modify onsite spent fuel storage capacity, the enclosed guidance provides the necessary information and acceptance criteria utilized by the NRC staff in evaluating these applications. Providing the information needed to evaluate the matters covered by this document would likely avoid the necessity for NRC questions and thus significantly shorten the time required to process a fuel pool modification amendment.

Sincerely,

Brian Grimes

Brian K. Grimes, Assistant Director
for Engineering and Projects
Division of Operating Reactors

Enclosures:
1. NRC Guidance
2. Notice

ENCLOSURE NO. 1

OT POSITION FOR REVIEW AND ACCEPTANCE OF SPENT FUEL STORAGE AND HANDLING APPLICATIONS

I. <u>BACKGROUND</u>

Prior to 1975, low density spent fuel storage racks were designed with a large pitch, to prevent fuel pool criticality even if the pool contained the highest enrichment uranium in the light water reactor fuel assemblies. Due to an increased demand on storage space for spent fuel assemblies, the more recent approach is to use high density storage racks and to better utilize available space. In the case of operating plants the new rack system interfaces with the old fuel pool structure. A proposal for installation of high density storage racks may involve a plant in the licensing stage or an operating plant. The requirements of this position do not apply to spent fuel storage and handling facilities away from the nuclear reactor complex.

On September 16, 1975, the Commission announced (40 F. R. 42801) its intent to prepare a generic environmental impact statement on handling and storage of spent fuel from light water power reactors. In this notice, the Commission also announced its conclusion that it would not be in the public interest to defer all licensing actions intended to ameliorate a possible shortage of spent fuel storage capacity pending completion of the generic environmental impact statement.

The Commission directed that in the consideration of any such proposed licensing action, an environmental impact statement or environmental impact appraisal shall be prepared in which five specific factors in addition to the normal cost/benefit balance and environmental stresses should be applied, balanced and weighed.

The overall design objectives of a fuel storage facility at the reactor complex are governed by various Regulatory Guides, the Standard Review Plan, and industry standards which are listed in the reference section. Based on the reviews of such applications to date it is obvious that the staff had to request additional information that could be easily included in an adequately documented initial submittal. It is the intent of this document to provide guidance for the type and extent of information needed to perform the review, and to indicate the acceptance criteria where applicable.

II. REVIEW DISCIPLINES

The objective of the staff review is to prepare (1) Safety Evaluation Report, and (2) Environmental Impact Appraisal. The broad staff disciplines involved are nuclear, mechanical, material, structural, and environmental.

Nuclear and thermal-hydraulic aspects of the review include the potential for inadvertant criticality in the normal storage and handling of the spent fuel, and the consequences of credible accidents with respect to criticality and the ability of the heat removal system to maintain sufficient cooling.

Mechanical, material and structural aspects of the review concern the capability of the fuel assembly, storage racks, and spent fuel pool system to withstand the effects of natural phenomena such as earthquakes, tornadoes, flood, effects of external and internal missiles, thermal loading, and also other service loading conditions.

The environmental aspects of the review concern the increased thermal and radiological releases from the facility under normal as well as accident conditions, the occupational radiation exposures, the generation of radioactive waste, the need for expansion, the commitment of material and nonmaterial resources, realistic accidents, alternatives to the proposed action and the cost-benefit balance.

The information related to nuclear and thermal-hydraulic type of analyses is discussed in Section III.

The mechanical, material, and structural related aspects of information are discussed in Section IV.

The information required to complete an environmental impact assessment, including the five factors specified by the Commission, is provided in Section V.

III. NUCLEAR AND THERMAL-HYDRAULIC CONSIDERATIONS

1. Neutron Multiplication Factor

To include all credible conditions, the licensee shall calculate the effective neutron multiplication factor, k_{eff}, in the fuel storage pool under the following sets of assumed conditions:

1.1 Normal Storage

a. The racks shall be designed to contain the most reactive fuel authorized to be stored in the facility without any control rods or any noncontained* burnable poison and the fuel shall be assumed to be at the most reactive point in its life.

b. The moderator shall be assumed to be pure water at the temperature within the fuel pool limits which yields the largest reactivity.

c. The array shall be assumed to be infinite in lateral extent or to be surrounded by an infinitely thick water reflector and thick concrete,** as appropriate to the design.

d. Mechanical uncertainties may be treated by assuming "worst case" conditions or by performing sensitivity studies and obtaining appropriate uncertainties.

e. Credit may be taken for the neutron absorption in structural materials and in solid materials added specifically for neutron absorption, provided a means of inspection is established (refer to Section 1.5).

1.2 Postulated Accidents

The double contingency principle of ANSI N 16.1-1975 shall be applied. It shall require two unlikely, independent, concurrent events to produce a criticality accident.

Realistic initial conditions (e.g., the presence of soluble boron) may be assumed for the fuel pool and fuel assemblies. The

*"Noncontained" burnable poison is that which is not an integral part of the fuel assembly.

**It should be noted that under certain conditions concrete may be a more effective reflector than water.

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postulated accidents shall include: (1) dropping of a fuel element on top of the racks and any other achievable abnormal location of a fuel assembly in the pool; (2) a dropping or tipping of the fuel cask or other heavy objects into the fuel pool; (3) effect of tornado or earthquake on the deformation and relative position of the fuel racks; and (4) loss of all cooling systems or flow under the accident conditions, unless the cooling system is single failure proof.

1.3 Calculation Methods

The calculation method and cross-section values shall be verified by comparison with critical experiment data for assemblies similar to those for which the racks are designed. Sufficiently diverse configurations shall be calculated to render improbable the "cancellation of error" in the calculations. So far as practicable the ability to correctly account for heterogeneities (e.g., thin slabs of absorber between storage locations) shall be demonstrated.

A calculational bias, including the effect of wide spacing between assemblies shall be determined from the comparison between calculation and experiment. A calculation uncertainity shall be determined such that the true multiplication factor will be less than the calculated value with a 95 percent probability at a 95 percent confidence level. The total uncertainity factor on k_{eff} shall be obtained by a statistical combination of the calculational and mechanical uncertainties. The k_{eff} value for the racks shall be obtained by summing the calculated value, the calculational bias, and the total uncertainty.

1.4 Rack Modification

For modification to existing racks in operating reactors, the following information should be provided in order to expedite the review:

(a) The overall size of the fuel assembly which is to be stored in the racks and the fraction of the total cell area which represents the overall fuel assembly in the model of the nominal storage lattice cell;

(b) For H_2O + stainless steel flux trap lattices; the nominal thickness and type of stainless steel used in the storage racks and the thermal (.025 ev) macroscopic neutron absorption cross section that is used in the calculation method for this stainless steel;

(c) Also, for the H_2O + stainless steel flux trap lattices, the change of the calculated neutron multiplication factor of infinitely long fuel assemblies in infinitely large arrays in the storage rack (i.e., the k_∞ of the nominal fuel storage lattice cell and the changed k_∞) for:

(1) A change in fuel loading in grams of U^{235}, or equivalent, per axial centimeter of fuel assembly where it is assumed that this change is made by increasing the enrichment of the U^{235}; and,

(2) A change in the thickness of stainless steel in the storage racks assuming that a decrease in stainless steel thickness is taken up by an increase in water thickness and vice versa;

(d) For lattices which use boron or other strong neutron absorbers provide:

(1) The effective areal density of the boron-ten atoms (i.e., B^{10} atoms/cm^2 or the equivalent number of boron-ten atoms for other neutron absorbers) between fuel assemblies.

(2) Similar to Item C, above, provide the sensitivity of the storage lattice cell k_∞ to:

(a) The fuel loading in grams of U^{235}, or equivalent, per axial centimeter of fuel assembly,

(b) The storage lattice pitch; and,

(c) The areal density of the boron-ten atoms between fuel assemblies.

1.5 Acceptance Criteria for Criticality

The neutron multiplication factor in spent fuel pools shall be less than or equal to 0.95, <u>including all uncertainties</u>, under all conditions

(1) For those facilities which employ a strong neutron absorbing material to reduce the neutron multiplication factor for the storage pool, the licensee shall provide the description of onsite tests which will be performed to confirm the presence and retention of the strong absorber in the racks. The results of an initial, onsite verification test shall show within 95 percent confidence limits that there is a sufficient amount of neutron absorber in the racks to maintain the neutron multiplication factor at or below 0.95. In addition, coupon or other type of surveillance testing shall be performed on a statistically acceptable sample size on a

periodic basis throughout the life of the racks to verify the continued presence of a sufficient amount of neutron absorber in the racks to maintain the neutron multiplication factor at or below 0.95.

(2) Decay Heat Calculations for the Spent Fuel

The calculations for the amount of thermal energy that will have to be removed by the spent fuel pool cooling system shall be made in accordance with Branch Technical Position APCSB 9-2 entitled, "Residual Decay Energy for Light Water Reactors for Long Term Cooling." This Branch Technical Position is part of the Standard Review Plan (NUREG 75/087).

(3) Thermal-Hydraulic Analyses for Spent Fuel Cooling

Conservative methods should be used to calculate the maximum fuel temperature and the increase in temperature of the water in the pool. The maximum void fraction in the fuel assembly and between fuel assemblies should also be calculated.

Ordinarily, in order not to exceed the design heat load for the spent fuel cooling system it will be necessary to do a certain amount of cooling in the reactor vessel after reactor shutdown prior to moving fuel assemblies into the spent fuel pool. The bases for the analyses should include the established cooling times for both the usual refueling case and the full core off load case.

A potential for a large increase in the reactivity in an H_2O flux trap storage lattice exists if, somehow, the water is kept out or forced out of the space between the fuel assemblies, conceivably by trapped air or steam. For this reason, it is necessary to show that the design of the storage rack is such that this will not occur and that these spaces will always have water in them. Also, in some cases, direct gamma heating of the fuel storage cell walls and of the intercell water may be significant. It is necessary to consider direct gamma heating of the fuel storage cell walls and of the intercell water to show that boiling will not occur in the water channels between the fuel assemblies. Under postulated accident conditions where all non-Category I spent fuel pool cooling systems become inoperative, it is necessary to show that there is an alternate method for cooling the spent pool water. When this alternative method requires the installation of alternate components or significant physical alteration of the cooling system, the detailed steps shall be described, along with the time required for each. Also, the average amount of water in the fuel pool and the expected heat up rate of this water assuming loss of all cooling systems shall be specified.

(4) Potential Fuel and Rack Handling Accidents

The method for moving the racks to and from and into and out of the fuel pool, should be described. Also, for plants where the spent fuel pool modification requires different fuel handling procedures than that described in the Final Safety Analysis Report, the differences should be discussed. If potential fuel and rack handling accidents occur, the neutron multiplication factor in the fuel pool shall not exceed 0.95. These postulated accidents shall not be the cause of the loss of cooling for either the spent fuel or the reactor.

(5) Technical Specifications

To insure against criticality, the following technical specifications are needed on fuel storage in high density racks:

1. The neutron multiplication factor in the fuel pool shall be less than or equal to 0.95 at all times.

2. The fuel loading (i.e., grams of uranium-235, or equivalent, per axial centimeter of assembly) in fuel assemblies that are to be loaded into the high density racks should be limited. The number of grams of uranium-235, or equivalent, put in the plant's technical specifications shall preclude criticality in the fuel pool.

Excessive pool water temperatures may lead to excessive loss of water due to evaporation and/or cause fogging. Analyses of thermal load should consider loss of all pool cooling systems. To avoid exceeding the specified spent fuel pool temperatures, consideration shall be given to incorporating a technical specification limit on the pool water temperature that would resolve the concerns described above. For limiting values of pool water temperatures refer to ANSI-N210-1976 entitled, "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations," except that the requirements of the Section 9.1.3.III.1.d of the Standard Review Plan is applicable for the maximum heat load with normal cooling systems in operation.

IV. MECHANICAL, MATERIAL, AND STRUCTURAL CONSIDERATIONS

(1) Description of the Spent Fuel Pool and Racks

Descriptive information including plans and sections showing the spent fuel pool in relation to other plant structures shall be provided in order to define the primary structural aspects and elements relied upon to perform the safety-related functions of the pool and the racks. The main safety function of the spent fuel pool and the racks is to maintain the spent fuel assemblies in a safe configuration through all environmental and abnormal loadings, such as earthquake, and impact due to spent fuel cask drop, drop of a spent fuel assembly, or drop of any other heavy object during routine spent fuel handling.

The major structural elements reviewed and the extent of the descriptive information required are indicated below.

(a) Support of the Spent Fuel Racks: The general arrangements and principal features of the horizontal and the vertical supports to the spent fuel racks should be provided indicating the methods of transferring the loads on the racks to the fuel pool wall and the foundation slab. All gaps (clearance or expansion allowance) and sliding contacts should be indicated. The extent of interfacing between the new rack system and the old fuel pool walls and base slab should be discussed, i.e., interface loads, response spectra, etc.

If connections of the racks are made to the base and to the side walls of the pool such that the pool liner may be perforated, the provisions for avoiding leakage of radioactive water of the pool should be indicated.

(b) Fuel Handling: Postulation of a drop accident, and quantification of the drop parameters are reviewed under the environmental discipline. Postulated drop accidents must include a straight drop on the top of a rack, a straight drop through an individual cell all the way to the bottom of the rack, and an inclined drop on the top of a rack. Integrity of the racks and the fuel pool due to a postulated fuel handling accident is reviewed under the mechanical, material, and structural disciplines. Sketches and sufficient details of the fuel handling system should be provided to facilitate this review.

(2) Applicable Codes, Standards and Specifications

Construction materials should conform to Section III, Subsection NF of the ASME* Code. All Materials should be selected to be compatible with the fuel pool environment to minimize corrosion and galvanic effects.

Design, fabrication, and installation of spent fuel racks of stainless steel material may be performed based upon the AISC** specification or Subsection NF requirements of Section III of the ASME B&PV Code for Class 3 component supports. Once a code is chosen its provisions must be followed in entirety. When the AISC specification procedures are adopted, the yield stress values for stainless steel base metal may be obtained from the Section III of the ASME B&PV Code, and the design stresses defined in the AISC specifications as percentages of the yield stress may be used. Permissible stresses for stainless steel welds used in accordance with the AISC Code may be obtained from Table NF-3292.1-1 of ASME Section III Code.

Other materials, design procedures, and fabrication techniques will be reviewed on a case by case basis.

(3) Seismic and Impact Loads

For plants where dynamic input data such as floor response spectra or ground response spectra are not available, necessary dynamic analyses may be performed using the criteria described in Section 3.7 of the Standard Review Plan. The ground response spectra and damping values should correspond to Regulatory Guide 1.60 and 1.61 respectively. For plants where dynamic data are available, e.g., ground response spectra for a fuel pool supported by the ground, floor response spectra for fuel pools supported on soil where soil-structure interaction was considered in the pool design or a floor response spectra for a fuel pool supported by the reactor building, the design and analysis of the new rack system may be performed by using either the existing input parameters including the old damping values or new parameters in accordance with Regulatory Guide 1.60 and 1.61. The use of existing input with new damping values in Regulatory Guide 1.61 is not acceptable.

Seismic excitation along three orthogonal directions should be imposed simultaneously for the design of the new rack system.

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*American Society of Mechanical Engineers Boiler and Pressure Vessel Codes, Latest Edition.

**American Institute of Steel Construction, Latest Edition.

The peak response from each direction should be combined by square root of the sum of the squares. If response spectra are available for a vertical and horizontal directions only, the same horizontal response spectra may be applied along the other horizontal direction.

The effect of submergence of the rack system on the damping and the mass of the fuel racks has been under study by the NRC. Submergence in water may introduce damping from two sources, (a) viscous drag, and (b) radiation of energy away from the submerged body in those cases where the confining boundaries are far enough away to prevent reflection of waves at the boundaries. Viscous damping is generally negligible. Based upon the findings of this current study for a typical high density rack configuration, wave reflections occur at the boundaries so that no additional damping should be taken into account.

A report on the NRC study is to be published shortly under the title "Effective Mass and Damping of Submerged Structures (UCRL-52342)," by R. G. Dong. The recommendations provided in this report on the added mass effect provide an acceptable basis for the staff review. Increased damping due to submergence in water is not acceptable without applicable test data and/or detailed analytical results.

Due to gaps between fuel assemblies and the walls of the guide tubes, additional loads will be generated by the impact of fuel assemblies during a postulated seismic excitation. Additional loads due to this impact effect may be determined by estimating the kinetic energy of the fuel assembly. The maximum velocity of the fuel assembly may be estimated to be the spectral velocity associated with the natural frequency of the submerged fuel assembly. Loads thus generated should be considered for local as well as overall effects on the walls of the rack and the supporting framework. It should be demonstrated that the consequent loads on the fuel assembly do not lead to a damage of the fuel.

Loads generated from other postulated impact events may be acceptable, if the following parameters are described in the report: the total mass of the impacting missile, the maximum velocity at the time of impact, and the ductility ratio of the target material utilized to absorb the kinetic energy.

(4) Loads and Load Combinations:

Any change in the temperature distribution due to the proposed modification should be identified. Information pertaining to the applicable design loads and various combinations thereof should be provided indicating the thermal load due to the effect of the maximum temperature distribution through the pool walls and base slab. Temperature gradient across the rack structure due to differential heating effect between a full and an empty cell should be indicated and incorporated in the design of the rack structure. Maximum uplift forces available from the crane should be indicated including the consideration of these forces in the design of the racks and the analysis of the existing pool floor, if applicable.

The specific loads and load combinations are acceptable if they are in conformity with the applicable portions of Section 3.8.4-II.3 of the Standard Review Plan.

(5) Design and Analysis Procedures

Details of the mathematical model including a description of how the important parameters are obtained should be provided including the following: the methods used to incorporate any gaps between the support systems and gaps between the fuel bundles and the guide tubes; the methods used to lump the masses of the fuel bundles and the guide tubes; the methods used to account for the effect of sloshing water on the pool walls; and, the effect of submergence on the mass, the mass distribution and the effective damping of the fuel bundle and the fuel racks.

The design and analysis procedures in accordance with Section 3.8.4-II.4 of the Standard Review Plan are acceptable. The effect on gaps, sloshing water, and increase of effective mass and damping due to submergence in water should be quantified.

When pool walls are utilized to provide lateral restraint at higher elevations, a determination of the flexibility of the pool walls and the capability of the walls to sustain such loads should be provided. If the pool walls are flexible (having a fundamental frequency less than 33 Hertz), the floor response spectra corresponding to the lateral restraint point at the higher elevation are likely to be greater than those at the base of the pool. In such a case using the response spectrum approach, two separate analyses should be performed as indicated below:

(a) A spectrum analysis of the rack system using response spectra corresponding to the highest support elevation provided that there is not significant peak frequency shift between the response spectra at the lower and higher elevations; and,

(b) A static analysis of the rack system by subjecting it to the maximum relative support displacement.

The resulting stresses from the two analyses above should be combined by the absolute sum method.

In order to determine the flexibility of the pool wall it is acceptable for the licensee to use equivalent mass and stiffness properties obtained from calculations similar to those described "Introduction to Structural Dynamics" by J. M. Biggs published by McGraw Hill Book Company. Should the fundamental frequency of the pool wall model be higher than or equal to 33 Hertz, it may be assumed that the response of the pool wall and the corresponding lateral support to the new rack system are identical to those of the base slab, for which appropriate floor response spectra or ground response spectra may already exist.

(6) Structural Acceptance Criteria

When AISC Code procedures are adopted, the structural acceptance criteria are those given in Section 3.8.4.II.5 of the Standard Review Plan for steel and concrete structures. For stainless steel the acceptance criteria expressed as a percentage of yield stress should satisfy Section 3.8.4.II.5 of the Standard Review Plan. When subsection NF, Section III, of the ASME B&PV Code is used for the racks, the structural acceptance criteria are those given in the Table below.

For impact loading the ductility ratios utilized to absorb kinetic energy in the tensile, flexural, compressive, and shearing modes should be quantified. When considering the effects of seismic loads, factors of safety against gross sliding and overturning of racks and rack modules under all probable service conditions shall be in accordance with the Section 3.8.5.II-5 of the Standard Review Plan. This position on factors of safety against sliding and tilting need not be met provided any one of the following conditions is met:

(a) it can be shown by detailed nonlinear dynamic analyses that the amplitudes of sliding motion are minimal, and impact between adjacent rack modules or between a rack module and the pool walls is prevented provided that the factors of safety against tilting are within the values permitted by Section 3.8.5.II.5 of the Standard Review Plan.

(b) it can be shown that any sliding and tilting motion will be contained within suitable geometric constraints such as thermal clearances, and that any impact due to the clearances is incorporated.

(7) Materials, Quality Control, and Special Construction Techniques:

The materials, quality control procedures, and any special construction techniques should be described. The sequence of installation of the new fuel racks, and a description of the precautions to be taken to prevent damage to the stored fuel during

TABLE

Load Combination	
Elastic Analysis	Acceptance Limit
D + L	Normal limits of NF 3231.1a
D + L + E	Normal limits of NF 3231.1a
D + L + To	1.5 times normal limits or the lesser of 2 Sy and Su
D + L + To + E	1.5 times normal limits or the leser of 2 Sy and Su
D + L + Ta + E	1.6 times normal limits or the lesser of 2 Sy or Su
D + L + Ta + E[1]	Faulted condition limits of NF 3231.1c
Limit Analysis	
1.7 (D + L)	Limits of XVII-4000 of Appendix XVII of ASME Code Section III
1.7 (D + L + E)	
1.3 (D + L + To)	
1.3 (D + L + E + To)	
1.1 (D + L + Ta + E)	

Notes:

1. The abbreviations in the table above are those used in Section 3.8.4 of the Standard Review Plan where each term is defined except for Ta which is defined as the highest temperature associated with the postulated abnormal design conditions.

2. Deformation limits specified by the Design Specification limits shall be satisfied, and such deformation limits should preclude damage to the fuel assemblies.

3. The provisions of NF 3231.1 shall be ammended by the requirements of the paragraphs c.2, 3, and 4 of the Regulatory Guide 1.124 entitled "Design Limits and Load Combinations for Class 1 Linear-Type Component Supports."

the construction phase should be provided. Methods for structural qualification of special poison materials utilized to absorb neutron radiation should be described. The material for the fuel rack is reviewed for compatibility inside the fuel pool environment. The quality of the fuel pool water in terms of the pH value and the available chlorides, fluorides, boron, heavy metals should be indicated so that the long-term integrity of the rack structure, fuel assembly, and the pool liner can be evaluated.

Acceptance criteria for special materials such as poison materials should be based upon the results of the qualification program supported by test data and/or analytical procedures.

If connections between the rack and the pool liner are made by welding, the welder as well as the welding procedure for the welding assembly shall be qualified in accordance with the applicable code.

If precipitation hardened stainless steel material is used for the construction of the spent fuel pool racks, hardness testing should be performed on each rack component of the subject material to verify that each part is heat treated properly. In addition, the surface film resulting from the heat treatment should be removed from each piece to assure adequate corrosion resistance.

(8) Testing and Inservice Surveillance

Methods for verification of long-term material stability and mechanical integrity of special poison material utilized for neutron absorption should include actual tests.

Inservice surveillance requirements for the fuel racks and the poison material, if applicable, are dependent on specific design features. These features will be reviewed on a case by case basis to determine the type and the extent of inservice surveillance necessary to assure long-term safety and integrity of the pool and the fuel rack system.

V. COST/BENEFIT ASSESSMENT

1. Following is a list of information needed for the environmental Cost/Benefit Assessment:

1.1 What are the specific needs that require increased storage capacity in the spent fuel pool (SFP)? Include in the response:

   (a) status of contractual arrangements, if any, with fuel-storage or fuel-reprocessing facilities,

   (b) proposed refueling schedule, including the expected number of fuel assemblies that will be transferred into the SFP at each refueling until the total existing capacity is reached,

   (c) number of spent fuel assemblies presently stored in the SFP,

   (d) control rod assemblies or other components stored in the SFP, and

   (e) the additional time period that spent fuel assemblies would be stored onsite as a result of the proposed expansion, and

   (f) the estimated date that the SFP will be filled with the proposed increase in storage capacity.

1.2 Discuss the total construction associated with the proposed modification, including engineering, capital costs (direct and indirect) and allowances for funds used during construction.

1.3 Discuss the alternative to increasing the storage capacity of the SFP. The alternatives considered should include:

   (a) shipment to a fuel reprocessing facility (if available),

   (b) shipment to an independent spent fuel storage facility,

   (c) shipment to another reactor site,

   (d) shutting down the reactor.

   The discussion of options (a), (b) and (c) should include a cost comparison in terms of dollars per KgU stored or cost per assembly. The discussion of (d) should include the cost for providing replacement power either from within or outside the licensee's generating system.

1.4 Discuss whether the commitment of material resources (e.g., stainless steel, boral, B_4C, etc.) would tend to significantly foreclose the alternatives available with respect to any other licensing actions designed to ameliorate a possible shortage of spent fuel storage capacity. Describe the material resources that would be consumed by the proposed modification.

1.5 Discuss the additional heat load and the anticipated maximum temperature of water in the SFP which would result from the proposed expansion, the resulting increase in evaporation rates, the additional heat load on component and/or plant cooling water systems and whether there will be any significant increase in the amount of heat released to the environment.

V.2. RADIOLOGICAL EVALUATION

2. Following is a list of information needed for radiological evaluation:

2.1 The present annual quantity of solid radioactive wastes generated by the SFP purification system. Discuss the expected increase in solid wastes which will result from the expansion of the capacity of the SFP.

2.2. Data regarding krypton-85 measured from the fuel building ventilation system by year for the last two years. If data are not available from the fuel building ventilation system, provide this data for the ventilation release which includes this system.

2.3 The increases in the doses to personnel from radionuclide concentrations in the SFP due to the expansion of the capacity of the SFP, including the following:

(a) Provide a table showing the most recent gamma isotopic analysis of SFP water identifying the principal radionuclides and their respective concentrations.

(b) The models used to determine the external dose equivalent rate from these radionuclides. Consider the dose equivalent rate at some distance above the center and edge of the pool respectively. (Use relevant experience if necessary).

(c) A table of recent analysis performed to determine the principal airborne radionuclides and their respective concentrations in the SFP area.

(d) The model and assumptions used to determine the increase, if any, in dose rate from the radionuclides identified in (c) above in the SFP area and at the site boundary.

(e) An estimate of the increase in the annual man-rem burden from more frequent changing of the demineralizer resin and filter media.

(f) The buildup of crud (e.g., ^{58}Co, ^{60}Co) along the sides of the pool and the removal methods that will be used to reduce radiation levels at the pool edge to as low as reasonably achievable.

(g) The expected total man-rem to be received by personnel occupying the fuel pool area based on all operations in that area including the doses resulting from (e) and (f) above.

A discussion of the radiation protection program as it affects (a) through (g) should be provided.

2.4 Indicate the weight of the present spent fuel racks that will be removed from the SFP due to the modification and discuss what will be done with these racks.

V.3 ACCIDENT EVALUATION

3.1 The accident review shall consider:

(a) cask drop/tip analysis, and

(b) evaluation of the overhead handling system with respect to Regulatory Guide 1.104.

3.2 If the accident aspects of review do not establish acceptability with respect to either (a) or (b) above, then technical specifications may be required that prohibit cask movement in the spent fuel building.

3.3 If the accident review does not establish acceptability with respect to (b) above, then technical specifications may be required that:

(1) define cask transfer path including control of

(a) cask height during transfer, and

(b) cask lateral position during transfer

(2) indicate the minimum age of fuel in pool sections during movement of heavy loads near the pool. In special cases evaluation of consequences-limiting engineered safety features such as isolation systems and filter systems may be required.

3.4 If the cask drop/tip analysis as in 3.1(a) above is promised for future submittal, the staff evaluation will include a conclusion on the feasibility of a specification of minimum age of fuel based on previous evaluations.

3.5 The maximum weight of loads which may be transported over spent fuel may not be substantially in excess of that of a single fuel assembly. A technical specification will be required to this effect.

3.6 Conclusions that determination of previous Safety Evaluation Reports and Final Environmental Statements have not changed significantly or impacts are not significant are made so that a negative declaration with an Environmental Impact Appraisal (rather than a Draft and Final Environmental Statement) can be issued. This will involve checking realistic as well as conservative accident analyses.

VI. REFERENCES

1. Regulatory Guides

   1.13 - Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations

   1.29 - Seismic Design Classification

   1.60 - Design Response Spectra for Seismic Design of Nuclear Power Plants

   1.61 - Damping Values for Seismic Design of Nuclear Power Plants

   1.76 - Design Basis Tornado for Nuclear Power Plants

   1.92 - Combining Modal Responses and Spatial Components in Seismic Response Analysis

   1.104 - Overhead Crane Handling Systems for Nuclear Power Plants

   1.124 - Design Limits and Loading Combinations for Class 1 Linear-Type Components Supports

2. Standard Review Plan

   3.7 - Seismic Design

   3.8.4 - Other Category I Structures

   9.1 - Fuel Storage and Handling

   9.5.1 - Fire Protection System

3. Industry Codes and Standards

   1. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code Section III, Division 1

   2. American Institute of Steel Construction Specifications

   3. American National Standards Institute, N210-76

   4. American Society of Civil Engineers, Suggested Specification for Structures of Aluminium Alloys 6061-T6 and 6067-T6

5. The Aluminium Association, Specification for Aluminium Structures

CONTENTION TC-2: EXHIBIT 3

Draft 1, Regulatory Guide 1.13, Revision 2, "Spent Fuel Storage Facility Design Basis (December 1981)

U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REGULATORY RESEARCH

DRAFT REGULATORY GUIDE AND VALUE/IMPACT STATEMENT

December 1981
Division 1
Task CE 913-5

Contact: C. Schulten (301)443-5910

PROPOSED REVISION 2* TO REGULATORY GUIDE 1.13

SPENT FUEL STORAGE FACILITY DESIGN BASIS

A. INTRODUCTION

General Design Criterion 61, "Fuel Storage and Handling and Radioactivity Control," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," requires that fuel storage and handling systems be designed to ensure adequate safety under normal and postulated accident conditions. It also requires that these systems be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions. This guide describes a method acceptable to the NRC staff for implementing Criterion 61.

B. DISCUSSION

Working Group ANS-57.2 of the American Nuclear Society Subcommittee ANS-50 has developed a standard that details minimum design requirements for

*The substantial number of changes in this proposed revision has made it impractical to indicate the changes with lines in the margin.

This regulatory guide and the associated value/impact statement are being issued in draft form to involve the public in the early stages of the development of a regulatory position in this area. They have not received complete staff review and do not represent an official NRC staff position.

Public comments are being solicited on both drafts, the guide (including any implementation schedule) and the value/impact statement. Comments on the value/impact statement should be accompanied by supporting data. Comments on both drafts should be sent to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Branch, by MAR 5 1982

Requests for single copies of draft guides (which may be reproduced) or for placement on an automatic distribution list for single copies of future draft guides in specific divisions should be made in writing to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Technical Information and Document Control.

spent fuel storage facilities at nuclear power stations. This standard was approved by the American National Standards Committee N18, Nuclear Design Criteria. It was subsequently approved and designated ANSI N210-1976/ANS-57.2, "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations," by the American National Standards Institute on April 12, 1976.

Primary facility design objectives are:

a. To prevent loss of water from the fuel pool that would uncover fuel,

b. To protect the spent fuel from mechanical damage, and

c. To provide the capability for limiting the potential offsite exposures in the event of significant release of radioactivity from the fuel.

If spent fuel storage facilities are not provided with adequate protective features, radioactive materials could be released to the environment as a result of either loss of water from the storage pool or mechanical damage to fuel within the pool.

1. LOSS OF WATER FROM STORAGE POOL

Unless protective measures are taken to prevent the loss of water from a fuel storage pool, the spent fuel could overheat and cause damage to fuel cladding integrity, which could result in the release of radioactive materials to the environment. Equipment failures in systems connected to the pool could also result in the loss of pool water. A permanent coolant makeup system designed with suitable redundancy or backup would prevent the fuel from being uncovered should pool leaks occur. Further, early detection of pool leakage and fuel damage can be made using pool-water-level monitors and pool radiation monitors that alarm locally and also at a continuously manned location to ensure timely operation of building filtration systems. Natural events such as earthquakes or high winds can damage the fuel pool either directly or by the generation of missiles. Earthquakes or high winds could also cause structures or cranes to fall into the pool. Designing the facility to withstand these occurrences without significant loss of watertight integrity will alleviate these concerns.

2. MECHANICAL DAMAGE TO FUEL

The release of radioactive material from fuel may occur as a result of fuel-cladding failures or mechanical damage caused by the dropping of fuel elements or objects onto fuel elements during the refueling process and at other times.

Plant arrangements consider low-probability accidents such as the dropping of heavy loads (e.g., a 100-ton fuel cask) where such loads are positioned or moved in or over the spent fuel pool. It is desirable that cranes capable of carrying heavy loads be prevented from moving into the vicinity of the stored fuel.

Missiles generated by high winds also are a potential cause of mechanical damage to fuel. This concern can be eliminated by designing the fuel storage facility to preclude the possibility of the fuel being struck by missiles generated by high winds.

3. LIMITING POTENTIAL OFFSITE EXPOSURES

Mechanical damage to the fuel might cause significant offsite doses unless dose reduction features are provided. Dose reduction designs such as negative pressure in the fuel handling building during movement of spent fuel would prevent exfiltration and ensure that any activity released to the fuel handling building will be treated by an engineered safety feature (ESF) grade filtration system before release to the environment. Even if measures not described are used to maintain the desired negative pressure, small leaks from the building may still occur as a result of structural failure or other unforeseen events.

The staff considers Seismic Category I design assumptions acceptable for the spent fuel pool cooling, makeup, and cleanup systems. Tornado protection requirements are acceptable for the water makeup source and its delivery system, the pool structure, the building housing the pool, and the filtration-ventilation system. Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," and 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," provide guidelines to limit potential offsite exposures through the filtration-ventilation system of the pool building.

Occupational radiation exposure is kept as low as is reasonably achievable (ALARA) in all activities involving personnel, and efforts toward maintaining exposures ALARA are considered in the design, construction, and operational phases. Guidance on maintaining exposures ALARA is provided in Regulatory Guide 8.8, "Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable."

C. REGULATORY POSITION

The requirements in ANSI N210-1976/ANS-57.2, "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations,"* are generally acceptable to the NRC staff as a means for complying with the requirements of General Design Criterion 61, "Fuel Storage and Handling and Radioactivity Control," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 as related to light-water reactors (LWRs), subject to the following clarifications and modifications:

1. In lieu of the example inventory in Section 4.2.4.3(1), the example inventory should be that inventory of radioactive materials that are predicted to leak under the postulated maximum damage conditions resulting from the dropping of a single spent fuel assembly onto a fully loaded spent fuel pool storage rack. Other assumptions in the analysis should be consistent with those given in Regulatory Guide 1.25 (Safety Guide 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."

2. In addition to meeting the requirements of Section 5.1.3, boiling of the pool water may be permitted only when the resulting thermal loads are properly accounted for in the design of the pool structure, the storage racks, and other safety-related structures, equipment, and systems.

*Copies may be obtained from the American Nuclear Society, 555 North Kensington Avenue, La Grange Park, Illinois 60525

3. In addition to meeting the requirements of Section 5.1.3, the fuel storage pool should be designed (a) to prevent tornado winds and missiles generated by these winds from causing significant loss of watertight integrity of the fuel storage pool and (b) to prevent missiles generated by tornado winds from striking the fuel. These requirements are discussed in Regulatory Guide 1.117, "Tornado Design Classification." The fuel storage building, including walls and roof, should be designed to prevent penetration by tornado-generated missiles or from seismic damage to ensure that nothing bypasses the ESF-grade filtration system in the containment building.

4. In addition to meeting the requirements of Section 5.1.5.1, provisions should be made to ensure that nonfuel components in fuel pools are handled below the minimum water shielding depth. A system should be provided that, either through the design of the system or through administrative procedures, would prohibit unknowing retrieval of these components.

5. In addition to meeting the requirements of Section 5.1.12.10, the maximum potential kinetic energy capable of being developed by any object handled above stored spent fuel, if dropped, should not exceed the kinetic energy of one fuel assembly and its associated handling tool when dropped from the height at which it is normally handled above the spent fuel pool storage racks.

6. In addition to meeting the requirements of Section 5.2.3.1, an interface should be provided between the cask venting system and the building ventilation system to minimize personnel exposure to the "vent-gas" generated from filling a dry loaded cask with water.

7. In addition to meeting the requirements of Section 5.3.3, radioactivity released during a Condition IV fuel handling accident should be either contained or removed by filtration so that the dose to an individual is less than the guidelines of 10 CFR Part 100. The calculated offsite dose to an individual from such an event should be well within the exposure guidelines of 10 CFR Part 100 using appropriately conservative analytical methods and assumptions. In order to ensure that released activity does not bypass the

filtration system, the ESF fuel storage building ventilation should provide and maintain a negative pressure of at least 3.2 mm (0.125 in.) water gauge within the fuel storage building.

8. In addition to the requirements of Section 6.3.1, overhead handling systems used to handle the spent fuel cask should be designed so that travel directly over the spent fuel storage pool or safety-related equipment is not possible. This should be verified by analysis to show that the physical structure under all cask handling pathways will be adequately designed so that unacceptable damage to the spent fuel storage facility or safety-related equipment will not occur in the event of a load drop.

9. In addition to the references listed in Section 6.4.4, Safety Class 3, Seismic Category I, and safety-related structures and equipment should be subjected to quality assurance programs that meet the applicable provisions of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50. Further, these programs should obtain guidance from Regulatory Guide 1.28, "Quality Assurance Program Requirements (Design and Construction)," endorsing ANSI N45.2, and from the applicable provisions of the ANSI N45.2-series standards endorsed by the following regulatory guides:

1.30 (Safety Guide 30) "Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment" (N45.2.4).

1.38 "Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants" (N45.2.2).

1.58 "Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel" (N45.2.6).

1.64 "Quality Assurance Requirements for the Design of Nuclear Power Plants" (N45.2.11).

1.74 "Quality Assurance Terms and Definitions" (N45.2.10).

1.88 "Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records" (N45.2.9).

1.94 "Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants" (N45.2.5).

1.116 "Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems" (N45.2.8).

1.123 "Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants" (N45.2.13).

10. The spent fuel pool water temperatures stated in Section 6.6.1(2) exceed the limits recommended by the NRC staff. For the maximum heat load during Condition I occurrences with normal cooling systems in operation and assuming a single active failure, the pool water temperature should be kept at or below 60°C (140°F). Under abnormal maximum heat load conditions (full core unload) and also for Condition IV occurrences, the pool water temperature should be kept below boiling.

11. A nuclear criticality safety analysis should be performed in accordance with Appendix A to this guide for each system that involves the handling, transfer, or storage of spent fuel assemblies at LWR spent fuel storage facilities.

12. The spent fuel storage facility should be equipped with both electrical interlocks and mechanical stops to keep casks from being transported over the spent fuel pool.

13. Sections 6.4 and 9 of ANS-57.2 list those codes and standards referenced in ANS-57.2. Although this regulatory guide endorses with clarifications and modifications ANS-57.2, a blanket endorsement of those referenced codes and

standards is not intended. (Other regulatory guides may contain some such endorsements.)

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 proposed revision has been released to encourage public participation in its development. Except in those cases in which an applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method to be described in the active guide reflecting public comments will be used in the evaluation of applications for construction permits and operating licenses docketed after the implementation date to be specified in the active guide. Implementation by the staff will in no case be earlier than June 30, 1982.

APPENDIX A

NUCLEAR CRITICALITY SAFETY

1. SCOPE OF NUCLEAR CRITICALITY SAFETY ANALYSIS

1.1 A nuclear criticality safety analysis should be performed for each system that involves the handling, transfer, or storage of spent fuel assemblies at light-water reactor (LWR) spent fuel storage facilities.

1.2 The nuclear criticality safety analysis should demonstrate that each LWR spent fuel storage facility system is subcritical (k_{eff} not to exceed 0.95).

1.3 The nuclear criticality safety analysis should include consideration of all credible normal and abnormal operating occurrences, including:

a. Accidental tipping or falling of a spent fuel assembly,
b. Accidental tipping or falling of a storage rack during transfer,
c. Misplacement of a spent fuel assembly,
d. Accumulation of solids containing fissile materials on the pool floor or at locations in the cooling water system,
e. Fuel drop accidents,
f. Stuck fuel assembly/crane uplifting forces,
g. Horizontal motion of fuel before complete removal from rack,
h. Placing a fuel assembly along the outside of rack, and
i. Objects that may fall onto the stored spent fuel assemblies.

1.4 At all locations in the LWR spent fuel storage facility where spent fuel is handled or stored, the nuclear criticality safety analysis should demonstrate that criticality could <u>not</u> occur without at least two unlikely, independent, and concurrent failures or operating limit violations.

1.5 The nuclear criticality safety analysis should explicitly identify spent fuel assembly characteristics upon which subcriticality in the LWR spent fuel storage facility depends.

1.6 The nuclear criticality safety analysis should explicitly identify design limits upon which subcriticality depends that require physical verification at the completion of fabrication or construction.

1.7 The nuclear criticality safety analysis should explicitly identify operating limits upon which subcriticality depends that require implementation in operating procedures.

2. CALCULATION METHODS AND CODES

Methods used to calculate subcriticality should be validated in accordance with Regulatory Guide 3.41, "Validation of Calculational Methods for Nuclear Criticality Safety," which endorses ANSI N16.9-1975.

3. METHOD TO ESTABLISH SUBCRITICALITY

3.1 The evaluated multiplication factor of fuel in the spent fuel storage racks, k_s, under normal and credible abnormal conditions should be equal to or less than an established maximum allowable multiplication factor, k_a; i.e.,

$$k_s \leq k_a$$

The factor, k_s, should be evaluated from the expression:

$$k_s = k_{sn} + \Delta k_{sb} + \Delta k_u + \Delta k_{sc}$$

where

k_{sn} = the computed effective multiplication factor; k_{sn} is calculated by the same methods used for benchmark experiments for design storage parameters when the racks are loaded with the most reactive fuel to be stored,

Δk_{sb} = the bias in the calculation procedure as obtained from the comparisons with experiments and including any extrapolation to storage pool conditions,

Δk_{u} = the uncertainty in the benchmark experiments, and

Δk_{sc} = the combined uncertainties in the parameters listed in paragraph 3.2 below.

3.2 The combined uncertainties, Δk_{sc}, include:

a. Statistical uncertainty in the calculated result if a Monte Carlo calculation is used,

b. Uncertainty resulting from comparison with calculational and experimental results,

c. Uncertainty in the extrapolation from experiment to storage rack conditions, and

d. Uncertainties introduced by the considerations enumerated in paragraphs 4.3 and 4.4 below.

3.3 The various uncertainties may be combined statistically if they are independent. Correlated uncertainties should be combined additively.

3.4 All uncertainty values should be at the 95 percent probability level with a 95 percent confidence value.

3.5 For spent fuel storage pool, the value of k_a should be no greater than 0.95.

4. STORAGE RACK ANALYSIS ASSUMPTIONS

4.1 The spent fuel storage rack module design should be based on one of the following assumptions for the fuel:

a. The most reactive fuel assembly to be stored at the most reactive point in the assembly life, or

b. The most reactive fuel assembly to be stored based on a minimum confirmed burnup (see Section 6 of this appendix).

Both types of rack modules may be present in the same storage pool.

4.2 Determination of the most reactive spent fuel assembly includes consideration of the following parameters:

a. Maximum fissile fuel loading,
b. Fuel rod diameter,
c. Fuel rod cladding material and thickness,
d. Fuel pellet density,
e. Fuel rod pitch and total number of fuel rods within assembly,
f. Absence of fuel rods in certain locations, and
g. Burnable poison content.

4.3 The fuel assembly arrangement assumed in storage rack design should be the arrangement that results in the highest value of k_s considering:

a. Spacing between assemblies,
b. Moderation between assemblies, and
c. Fixed neutron absorbers between assemblies.

4.4 Determination of the spent fuel assembly arrangement with the highest value of k_s shall include consideration of the following:

a. Eccentricity of fuel bundle location within the racks and variations in spacing among adjacent bundles,
b. Dimensional tolerances,
c. Construction materials,
d. Fuel and moderator density (allowance for void formations and temperature of water between and within assemblies),

e. Presence of the remaining amount of fixed neutron absorbers in fuel assembly, and

f. Presence of structural material and fixed neutron absorber in cell walls between assemblies.

4.5 Fuel burnup determination should be made for fuel stored in racks where credit is taken for burnup. The following methods are acceptable:

a. A minimum allowable fuel assembly reactivity should be established, and a reactivity measurement should be performed to ensure that each assembly meets this criterion; or

b. A minimum fuel assembly burnup value should be established as determined by initial fuel assembly enrichment or other correlative parameters, and a measurement should be performed to ensure that each fuel assembly meets the established criterion; or

c. A minimum fuel assembly burnup value should be established as determined by initial fuel assembly enrichment or other correlative parameters, and an analysis of each fuel assembly's exposure history should be performed to determine its burnup. The analyses should be performed under strict administrative control using approved written procedures. These procedures should provide for independent checks of each step of the analysis by a second qualified person using nuclear criticality safety assessment criteria described in paragraph 1.4 above.

The uncertainties in determining fuel assembly storage acceptance criteria should be considered in establishing storage rack reactivity, and auditable records should be kept of the method used to determine the fuel assembly storage acceptance criterion for as long as the fuel assemblies are stored in the racks.

Consideration should be given to the axial distribution of burnup in the fuel assembly, and a limit should be set on the length of the fuel assembly that is permitted to have a lower average burnup than the fuel assembly average.

5. USE OF NEUTRON ABSORBERS IN STORAGE RACK DESIGN

5.1 Fixed neutron absorbers may be used for criticality control under the following conditions:

a. The effect of neutron-absorbing materials of construction or added fixed neutron-absorbers may be included in the evaluation if they are designed and fabricated so as to preclude inadvertent removal by mechanical or chemical action.

b. Fixed neutron absorbers should be an integral, nonremovable part of the storage rack.

c. When a fixed neutron absorber is used as the primary nuclear criticality safety control, there should be provision to:

(1) Initially confirm absorber presence in the storage rack, and

(2) Periodically verify continued presence of absorber.

5.2 The presence of a soluble neutron absorber in the pool water should not normally be used in the evaluation of k_s. However, when calculating the effects of Condition IV faults, realistic initial conditions (e.g., the presence of soluble boron) may be assumed for the fuel pool and fuel assemblies.

6. CREDIT FOR BURNUP IN STORAGE RACK DESIGN

6.1 Consideration should be given to the fact that the reactivity of any given spent fuel assembly will depend on initial enrichment, ^{235}U depletion, amount of burnable poison, plutonium buildup and fission product burnable poison depletion, and the fact that the rates of depletion and plutonium and fission product buildup are not necessarily the same.

6.2 Consideration should be given to the practical implementation of the spent fuel screening process. Factors to be considered in choosing the screening method should include:

a. Accuracy of the method used to determine storage rack reactivity;
b. Reproducibility of the result, i.e., what is the uncertainty in the result?
c. Simplicity of the procedure; i.e., how much disturbance to other operations is involved?
d. Accountability, i.e., ease and completeness of recordkeeping; and
e. Auditability.

DRAFT VALUE/IMPACT STATEMENT

1. PROPOSED ACTION

1.1 Description

Each nuclear power plant has a spent fuel storage facility. General Design Criterion 61, "Fuel Storage and Handling and Radioactivity Control," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," requires that fuel storage and handling systems be designed to ensure adequate safety under normal and postulated accident conditions. The proposed action would provide an acceptable method for implementing this criterion. This action would be an update of Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis."

1.2 Need for Proposed Action

Since Regulatory Guide 1.13 was last published in December of 1975, additional guidance has been provided in the form of ANSI standards and NUREG reports. The Office of Nuclear Reactor Regulation has requested that this guide be updated.

1.3 Value/Impact of Proposed Action

1.3.1 NRC

The applicants' basis for the design of the spent fuel storage facility will be the same as that used by the staff in its review of a construction permit or operating license application. Therefore, there should be a minimum number of cases where the applicant and the staff radically disagree on the design criteria.

1.3.2 Government Agencies

Applicable only if the agency, such as TVA, is an applicant.

1.3.3 Industry

The value/impact on the applicant will be the same as for the NRC staff.

1.3.4 Public

No major impact on the public can be foreseen.

1.4 Decision on Proposed Action

The guidance furnished on the design basis for the spent fuel storage facility should be updated.

2. TECHNICAL APPROACH

The American Nuclear Society published ANS-57.2 (ANSI N210), "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations." Part of the update of Regulatory Guide 1.13 would be an evaluation of this standard and possible endorsement by the NRC. Also, recommendations made by Task A-36, which were published in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," would be included.

3. PROCEDURAL APPROACH

Since Regulatory Guide 1.13 already deals with the proposed action, logic dictates that this guide be updated.

4. STATUTORY CONSIDERATIONS

4.1 NRC AUTHORITY

Authority for this regulatory guide is derived from the safety requirements of the Atomic Energy Act of 1954, as amended, through the Commission's regulations, in particular, General Design Criterion 61 of Appendix A to 10 CFR Part 50.

4.2 Need for NEPA Assessment

The proposed action is not a major action as defined by paragraph 51.5(a)(10) of 10 CFR Part 51 and does not require an environmental impact statement.

5. CONCLUSION

Regulatory Guide 1.13 should be updated.

CONTENTION TC-2: EXHIBIT 4

Memorandum from Laurence Kopp, NRC, to Timothy Collins, NRC, re: Guidance On The Regulatory Requirements For Criticality Analysis Of Fuel Storage At Light-Water Reactor Power Plants (August 19, 1998)

August 19, 1998

MEMORANDUM TO: Timothy Collins, Chief
Reactor Systems Branch
Division of Systems Safety and Analysis

FROM: Laurence Kopp, Sr. Reactor Engineer /s/
Reactor Systems Branch
Division of Systems Safety and Analysis

SUBJECT: GUIDANCE ON THE REGULATORY REQUIREMENTS FOR CRITICALITY ANALYSIS OF FUEL STORAGE AT LIGHT-WATER REACTOR POWER PLANTS

Attached is a copy of guidance concerning regulatory requirements for criticality analysis of new and spent fuel storage at light-water reactor power plants used by the Reactor Systems Branch. The principal objective of this guidance is to clarify and document current and past NRC staff positions that may have been incompletely or ambiguously stated in safety evaluation reports or other NRC documents. It also describes and compiles, in a single document, NRC staff positions on more recently proposed storage configurations and characteristics in spent fuel rerack or enrichment upgrade requests. This guidance is not applicable to fuel storage in casks, nor does it consider the mechanical, chemical, thermal, radiological, and other aspects of the storage of new and spent fuel.

Attachment:
As stated

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

GUIDANCE ON THE REGULATORY REQUIREMENTS FOR CRITICALITY ANALYSIS OF FUEL STORAGE AT LIGHT-WATER REACTOR POWER PLANTS

1. INTRODUCTION

This document defines the NRC Reactor Systems Branch guidance for the assurance of criticality safety in the storage of new (unirradiated or fresh) and spent (irradiated) fuel at light-water reactor (LWR) power stations. Safety analyses submitted in support of licensing actions should consider, among other things, normal operation, incidents, and postulated accidents that may occur in the course of handling, transferring, and storing fuel assemblies and should establish that an acceptable margin exists for the prevention of criticality under all credible conditions.

This guidance is not applicable to fuel storage in casks, nor does it consider the mechanical, chemical, thermal, radiological, and other aspects of the storage of new and spent fuel. The guidance considers only the criticality safety aspects of new and spent LWR fuel assemblies and of fuel that has been consolidated; that is, fuel with fuel rods reassembled in a more closely packed array.

The guidance stated here is based, in part, on (a) the criticality positions of Standard Review Plan (SRP) Section 9.1.1 (Ref. 1) and SRP 9.1.2 (Ref. 2), (b) a previous NRC position paper sent to all licensees (Ref. 3), and (c) past and present practices of the staff in its safety evaluation reports (SERs). The guidance also meets General Design Criterion 62 (Ref. 4), which states:

> Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

The principal objective of this guidance is to clarify and document current and past staff positions that may have been incompletely or ambiguously stated in SERs or other staff documents. A second purpose is to state staff positions on recently proposed storage configurations and characteristics in spent fuel rerack or enrichment upgrade requests (for example, multiple-region spent fuel storage racks, checkerboard loading patterns for new and spent fuel storage, credit for burnup in the spent fuel to be stored, and credit for non-removable poison inserts). Although these statements are not new staff positions, this document compiles them in a single paper. In addition, a recently approved staff position for pressurized-water reactors (PWRs) would allow partial credit for soluble boron in the pool water (Ref 5)

The guidance stated here is applicable to both PWRs and boiling-water reactors (BWRs). The most notable difference between PWR and BWR fuel storage facilities is the larger size of the fuel assemblies and the presence of soluble boron in the spent fuel pool water of PWRs

The determination of the effective multiplication factor, k_{eff}, for the new or spent fuel storage racks should consider and clearly identify the following:

a. fuel rod parameters, including:

   1. rod diameter

   2. cladding material and cladding thickness

   3. fuel rod pellet or stack density and initial uranium-235 (U-235) enrichment of each fuel rod in the assembly (a bounding enrichment is acceptable)

b. fuel assembly parameters, including:

   1. assembly length and planar dimensions

   2. fuel rod pitch

   3. total number of fuel rods in the assembly

   4. locations in the fuel assembly lattice that are empty or contain nonfuel material

   5. integral neutron absorber (burnable poison) content of various fuel rods and locations in fuel assembly

   6. structural materials (e.g., grids) that are an integral part of the fuel assembly

The criticality safety analysis should explicitly address the treatment of axial and planar variations of fuel assembly characteristics such as fuel enrichment and integral neutron absorber (burnable poison), if present (e.g., gadolinia in certain fuel rods of BWR and PWR assemblies or integral fuel burnable absorber (IFBA) coatings in certain fuel rods of PWR assemblies).

Whenever reactivity equivalencing (i.e., burnup credit or credit for imbedded burnable absorbers) is employed, or if a correlation with the reactivity of assemblies in a standard core geometry is used (k_{∞}), such as is typically done for BWR racks, the equivalent reactivities must be evaluated in the storage rack configuration. In this latter approach, sufficient uncertainty should be incorporated into the k_{∞} limit to account for the reactivity effects of (1) nonuniform enrichment variation in the assembly, (2) uncertainty in the calculation of k_{∞}, and (3) uncertainty in average assembly enrichment.

If various locations in a storage rack are prohibited from containing any fuel, they should be physically or administratively blocked or restricted to non-fuel material. If the criticality safety of the storage racks relies on administrative procedures, these procedures should be explicitly identified and implemented in operating procedures and/or technical specification limits

2. CRITICALITY ANALYSIS METHODS AND COMPUTER CODES

A variety of methods may be used for criticality analyses provided the cross-section data and geometric capability of the analytical model accurately represent all important neutronic and geometrical aspects of the storage racks. In general, transport methods of analysis are necessary for acceptable results. Storage rack characteristics such as boron carbide (B_4C) particle size and thin layers of structural and neutron absorbing material (poisons) need to be carefully considered and accurately described in the analytical model. Where possible, the primary method of analysis should be verified by a second, independent method of analysis. Acceptable computer codes include, but are not necessarily limited to, the following:

- CASMO - a multigroup transport theory code in two dimensions
- NITAWL-KENO5a - a multigroup transport theory code in three dimensions, using the Monte Carlo technique
- PHOENIX-P - a multigroup transport theory code in two dimensions, using discrete ordinates
- MONK6B - a multigroup transport theory code in three dimensions, using the Monte Carlo technique
- DOT - a multigroup transport theory code in two dimensions, using discrete ordinates

Similarly, a variety of cross-section libraries is available. Acceptable cross-section libraries include the 27-group, 123-group, and 218-group libraries from the SCALE system developed by the Oak Ridge National Laboratory and the 8220-group United Kingdom Nuclear Data Library (UKNDL). However, empirical cross-section compilations, such as the Hansen-Roach library, are not acceptable for criticality safety analyses (see NRC Information Notice No. 91-26). Other computer codes and cross-section libraries may be acceptable provided they conform to the requirements of this position statement and are adequately benchmarked.

The proposed analysis methods and neutron cross-section data should be benchmarked, by the analyst or organization performing the analysis, by comparison with critical experiments. This qualifies both the ability of the analyst and the computer environment. The critical experiments used for benchmarking should include, to the extent possible, configurations having neutronic and geometric characteristics as nearly comparable to those of the proposed storage facility as possible. The Babcock & Wilcox series of critical experiments (Ref 6) provides an acceptable basis for benchmarking storage racks with thin strong absorber panels for reactivity control. Similarly, the Babcock & Wilcox critical experiments on close-packed arrays of fuel (Ref. 7) provide an acceptable experimental basis for benchmark analyses for consolidated fuel arrays. A comparison with methods of analysis of similar sophistication (e g . transport theory) may be used to augment or extend the range of applicable critical experiment data

The benchmarking analyses should establish both a bias (defined as the mean difference between experiment and calculation) and an uncertainty of the mean with a one-sided tolerance factor for 95-percent probability at the 95-percent confidence level (Ref 8)

The maximum k_{eff} shall be evaluated from the following expression:

$$k_{eff} = k(calc) + \delta k(bias) + \delta k(uncert) + \delta k(burnup),$$

where

k(calc)	= calculated nominal value of k_{eff},
δk(bias)	= bias in criticality analysis methods,
δk(uncert)	= manufacturing and calculational uncertainties, and
δk(burnup)	= correction for the effect of the axial distribution in burnup, when credit for burnup is taken.

A bias that reduces the calculated value of k_{eff} should not be applied. Uncertainties should be determined for the proposed storage facilities and fuel assemblies to account for tolerances in the mechanical and material specifications. An acceptable method for determining the maximum reactivity may be either (1) a worst-case combination with mechanical and material conditions set to maximize k_{eff} or (2) a sensitivity study of the reactivity effects of tolerance variations. If used, a sensitivity study should include all possible significant variations (tolerances) in the material and mechanical specifications of the racks; the results may be combined statistically provided they are independent variations. Combinations of the two methods may also be used.

3. ABNORMAL CONDITIONS AND THE DOUBLE-CONTINGENCY PRINCIPLE

The criticality safety analysis should consider all credible incidents and postulated accidents. However, by virtue of the double-contingency principle, two unlikely independent and concurrent incidents or postulated accidents are beyond the scope of the required analysis. The double-contingency principle means that a realistic condition may be assumed for the criticality analysis in calculating the effects of incidents or postulated accidents. For example, if soluble boron is normally present in the spent fuel pool water, the loss of soluble boron is considered as one accident condition and a second concurrent accident need not be assumed. Therefore, credit for the presence of the soluble boron may be assumed in evaluating other accident conditions.

4. NEW FUEL STORAGE FACILITY (VAULT)

Normally, fresh fuel is stored temporarily in racks in a dry environment (new fuel storage vault) pending transfer into the spent fuel pool and then into the reactor core. However, moderator may be introduced into the vault under abnormal situations, such as flooding or the introduction of foam or water mist (for example, as a result of fire fighting operations). Foam or mist affects the neutron moderation in the array and can result in a peak in reactivity at low moderator density (called "optimum" moderation, Ref. 9). Therefore, criticality safety analyses must address two independent accident conditions, which should be incorporated into plant technical specifications:

a. With the new fuel storage racks loaded with fuel of the maximum permissible reactivity and flooded with pure water, the maximum k_{eff} shall be no greater than 0.95, including mechanical and calculational uncertainties, with a 95-percent probability at a 95-percent confidence level.

b. With the new fuel storage racks loaded with fuel of the maximum permissible reactivity and flooded with moderator at the (low) density corresponding to optimum moderation, the maximum k_{eff} shall be no greater than than 0.98, including mechanical and calculational uncertainties, with a 95-percent probability at a 95-percent confidence level.

An evaluation need not be performed for the new fuel storage facility for racks flooded with low-density or full-density water if it can be clearly demonstrated that design features and/or administrative controls prevent such flooding.

Under the double-contingency principle, the accident conditions identified above are the principle conditions that require evaluation. The simultaneous occurrence of other accident conditions need not be considered.

Usually, the storage racks in the new fuel vault are designed with large lattice spacing sufficient to maintain a low reactivity under the accident condition of flooding. Specific calculations, however, are necessary to assure the limiting k_{eff} is maintained no greater than 0.95.

At low moderator density, the presence of relatively weak absorber material (for example, stainless steel plates or angle brackets) is often sufficient to preclude neutronic coupling between assemblies, and to significantly reduce the reactivity. For this reason, the phenomenon of low-density (optimum) moderation is not significant in racks in the *spent fuel pool* under the initial conditions before the pool is flooded.

Under low-density moderator conditions, neutron leakage is a very important consideration. The new fuel storage racks should be designed to contain the highest enrichment fuel assembly to be stored without taking credit for any nonintegral neutron absorber. In the evaluation of the new fuel vaults, fuel assembly and rack characteristics upon which subcriticality depends should be explicitly identified (e.g., fuel enrichment and the presence of steel plates or braces).

5. SPENT FUEL STORAGE RACKS

A. Reference Criticality Safety Analysis

1. For BWR pools or for PWR pools where no credit for soluble boron is taken, the criticality safety analyses must address the following condition, which should be incorporated into the plant technical specifications:

a With the spent fuel storage racks loaded with fuel of the maximum permissible reactivity and flooded with full-density unborated water, the maximum k_{eff} shall be less than or equal to 0.95, including mechanical and calculational uncertainties, with a 95-percent probability at a 95-percent confidence level

2. If partial credit for soluble boron is taken, the criticality safety analyses for PWRs must address two independent conditions, which should be incorporated into the plant technical specifications:

   a. With the spent fuel storage racks loaded with fuel of the maximum permissible reactivity and flooded with full-density unborated water, the maximum k_{eff} shall be less than 1.0, including mechanical and calculational uncertainties, with a 95-percent probability at a 95-percent confidence level.

   b. With the spent fuel storage racks loaded with fuel of the maximum permissible reactivity and flooded with full density water borated to [*] ppm, the maximum k_{eff} shall be no greater than 0.95, including mechanical and calculational uncertainties, with a 95-percent probability at a 95-percent confidence level.[1]

3. The reference criticality safety analysis should also include, as a minimum, the following:

   a. If axial and planar variations of fuel assembly characteristics are present, they should be explicitly addressed, including the locations of burnable poison rods.

   b. For fuel assemblies containing burnable poison, the maximum reactivity should be the peak reactivity over burnup, usually when the burnable poison is nearly depleted.

   c. The spent fuel storage racks should be assumed to be infinite in the lateral dimension or to be surrounded by a water reflector and concrete or structural material as appropriate to the design. The fuel may be assumed to be infinite in the axial dimension, or the effect of a reflector on the top and bottom of the fuel may be evaluated.

   d. The evaluation of normal storage should be done at the temperature (water density) corresponding to the highest reactivity. In poisoned racks, the highest reactivity will usually occur at a water density of 1 0 (i.e., at 4°C). However, if the temperature coefficient of reactivity is positive, the evaluation should be done at the highest temperature expected during normal operations: i.e., equilibrium temperature under normal refueling conditions (including full-core offload), with one coolant train out of service and the pool filled with spent fuel from previous reloads.

4. The fuel assembly arrangement assumed in the criticality safety analysis of the spent fuel storage racks should also consider the following

[1] [*] is the boron concentration required to maintain the 0 95k_{eff} limit without consideration of accidents

a. the effect of eccentric positioning of fuel assemblies within the storage cells

b. the reactivity consequence of including the flow channel in BWR fuel assemblies

5. If one or more separate regions are designated for the storage of spent fuel, with credit for the reactivity depletion due to fuel burnup, the following applies.

a. The minimum required fuel burnup should be defined as a function of the initial nominal enrichment.

b. The spent fuel storage rack should be evaluated with spent fuel at the highest reactivity following removal from the reactor (usually after the decay of xenon-135). Operating procedures should include provision for independent confirmation of the fuel burnup, either administratively or experimentally, before the fuel is placed in storage cells of the designated region(s).

c. Subsequent decay of longer-life nuclides, such as Pu-241, over the rack storage time may be accounted for to reduce the minimum burnup required to meet the reactivity requirements.

d. A reactivity uncertainty due to uncertainty in the fuel depletion calculations should be developed and combined with other calculational uncertainties. In the absence of any other determination of the depletion uncertainty, an uncertainty equal to 5 percent of the reactivity decrement to the burnup of interest is an acceptable assumption.

e. A correction for the effect of the axial distribution in burnup should be determined and, if positive, added to the reactivity calculated for uniform axial burnup distribution.

B. Additional Considerations

1. The reactivity consequences of incidents and accidents such as (1) a fuel assembly drop and (2) placement of a fuel assembly on the outside and immediately adjacent to a rack must be evaluated. Under the double-contingency principle, credit for soluble boron, if present, is acceptable for these postulated accident conditions

2. If either credit for burnup is assumed or racks of different enrichment capability are in the same fuel pool, fuel assembly misloadings must be considered Normally, a misloading error involving only a single assembly need be considered unless there are circumstances that make multiple loading errors credible. Under the double-contingency principle, credit for soluble boron if present, is acceptable for these postulated accident conditions

3. The analysis must also consider the effect on criticality of natural events (e.g., earthquakes) that may deform, and change in the relative position of, the storage racks and fuel in the spent fuel pool.

4. Abnormal temperatures (above those normally expected) and the reactivity consequences of void formation (boiling) should be evaluated to consider the effect on criticality of loss of all cooling systems or coolant flow, unless the cooling system meets the single-failure criterion. Under the double-contingency principle, credit for soluble boron, if present, is acceptable for these abnormally elevated temperature conditions.

5. Normally, credit may only be taken for neutron absorbers that are an integral (nonremovable) part of a fuel assembly or the storage racks. Credit for added absorber (rods, plates, or other configurations) will be considered on a case-by-case basis, provided it can be clearly demonstrated that design features prevent the absorbers from being removed, either inadvertently or intentionally without unusual effort such as the necessity for special equipment maintained under positive administrative control.

6. If credit for soluble boron is taken, the minimum required pool boron concentration (typically, the refueling boron concentration) should be incorporated into the plant technical specifications or operating procedures. A boron dilution analysis should be performed to ensure that sufficient time is available to detect and suppress the worst dilution event that can occur from the minimum technical specification boron concentration to the boron concentration required to maintain the $0.95k_{eff}$ design basis limit. The analysis should consider all possible dilution initiating events (including operator error), dilution sources, dilution flow rates, boration sources, instrumentation, administrative procedures, and piping. This analysis should justify the surveillance interval for verifying the technical specification minimum pool boron concentration.

7. Consolidated fuel assemblies usually result in low values of reactivity (undermoderated lattice). Nevertheless, criticality calculations, using an explicit geometric description (usually triangular pitch) or as near an explicit description as possible, should be performed to assure a k_{eff} less than 0.95.

6. REFERENCES

1. NRC, "Standard Review Plan" NUREG-0800, Rev.2, Section 9.1.1, "New Fuel Storage," July 1981.

2. NRC, "Standard Review Plan" NUREG-0800, Rev. 2, Section 9.1.2, "Spent Fuel Storage," July 1981.

3. Brian K. Grimes, NRC, letter to all power reactor licensees, with enclosure, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," April 14, 1978.

4. *Code of Federal Regulations*, Title 10, Part 50, Appendix A, General Design Criterion 62, "Prevention of Criticality in Fuel Storage and Handling."

5. Westingnouse Electric Corporation, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology," WCAP-14416-NP-A, November 1996.

6. Babcock & Wilcox Company, "Critical Experiments Supporting Close Proximity Water Storage of Power Reactor Fuel," BAW-1484-7, July 1979.

7. Babcock & Wilcox Company, "Critical Experiments Supporting Underwater Storage of Tightly Packed Configurations of Spent Fuel Pins," BAW-1645-4, 1981.

8. National Bureau of Standards, *Experimental Statistics*, Handbook 91, August 1963.

9. J. M. Cano, R. Caro, and J. M. Martinez Val, "Supercriticality Through Optimum Moderation in Nuclear Fuel Storage," *Nuclear Technology*, Volume 48, May 1980.

CONTENTION TC-2: EXHIBIT 5

Letter from Donna B. Alexander, CP&L, to U.S. NRC, enclosing response to April 29, 199, RAI (June 14, 1999)

Carolina Power & Light Company
Harris Nuclear Plant
P.O. Box 165
New Hill NC 27562

SERIAL: HNP-99-094

JUN 14 1999

United States Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, DC 20555

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION
REGARDING THE LICENSE AMENDMENT REQUEST TO PLACE
HNP SPENT FUEL POOLS 'C' & 'D' IN SERVICE

Dear Sir or Madam:

By letter dated April 29, 1999, the NRC issued a request for additional information (RAI) regarding the Harris Nuclear Plant (HNP) license amendment request, submitted by CP&L letter Serial: HNP-98-188, dated December 23, 1998, to place spent fuel pools C and D in service. The HNP response to the NRC RAI is enclosed. The enclosed information is provided as a supplement to our December 23, 1998 license amendment request and does not change our initial determination that the proposed license amendment represents a no significant hazards consideration.

Please refer any questions regarding the enclosed information to Mr. Steven Edwards at (919) 362-2498.

Sincerely,

Donna B. Alexander

Donna B. Alexander
Manager, Regulatory Affairs
Harris Nuclear Plant

KWS/kws

Enclosure

9906210117 990614
PDR ADOCK 05000400
P PDR

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Mr. J. B. Brady, NRC Senior Resident Inspector
Mr. Mel Fry, N.C. DEHNR
Mr. R. J. Laufer, NRC Project Manager
Mr. L. A. Reyes, NRC Regional Administrator - Region II

bc:

Mr. K. B. Altman
Mr. G. E. Attarian
Mr. R. H. Bazemore
Mr. C. L. Burton
Mr. S. R. Carr
Mr. J. R. Caves
Mr. H. K. Chernoff (RNP)
Mr. B. H. Clark
Mr. W. F. Conway
Mr. G. W. Davis
Mr. W. J. Dorman (BNP)
Mr. R. S. Edwards
Mr. R. J. Field
Mr. K. N. Harris
Ms. L. N. Hartz
Mr. W J. Hindman
Mr. C. S. Hinnant
Mr. W. D. Johnson
Mr. G. J. Kline
Ms. W. C. Langston (PE&RAS File)
Mr. R. D. Martin
Mr. T. C. Morton
Mr. J. H. O'Neill, Jr.
Mr. J. S. Scarola
Mr. J. M. Taylor
Nuclear Records
Harris Licensing File
Files: H-X-0511
H-X-0642

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION REGARDING THE LICENSE AMENDMENT REQUEST TO PLACE HNP SPENT FUEL POOLS 'C' & 'D' IN SERVICE

Requested Item 1

Although the burnup criteria for storage in Pools C or D will be implemented by administrative procedures to ensure verified burnup prior to fuel transfer into these pools, an administrative failure should be assumed and evaluation of a fuel assembly misloading event (i.e., a fresh pressurized-water reactor (PWR) assembly inadvertently placed in a location restricted to a burned assembly as per Technical Specifications (TS) Figure 5.6.1) should be analyzed.

Response to Requested Item 1

The presence of soluble boron in the spent fuel pool water will assure that the reactivity is maintained substantially less than the design limitation in the event of a misloading event as described above. The Double Contingency Principle provides that neither the utility nor the staff is required to assume two unlikely, independent, concurrent events. Therefore, a failure of the administrative controls related to fuel assembly placement and the inadvertent dilution of the spent fuel pool water need not be considered to occur simultaneously. As a result, credit for the presence of soluble boron in the spent fuel pool water may be taken for an assembly misloading event as described. A minimum spent fuel pool boron concentration of 2000 ppm is maintained in accordance with HNP chemistry procedure CRC-001. This minimum boron concentration is more than adequate to offset the reactivity addition from a postulated fuel assembly misloading event. Based on analysis performed by Holtec International, it has been determined that a soluble boron concentration of 400 ppm would be sufficient to maintain k_{eff} less than 0.95 in the event of a fuel assembly misloading event (i.e., a fresh pressurized-water reactor (PWR) assembly inadvertently placed in a location restricted to a burned assembly as per TS Figure 5.6.1).

Requested Item 2

How will the burnup requirements needed to meet TS Figure 5.6.1 be ascertained for fuel assemblies shipped from other PWR plants (Robinson)?

Response to Requested Item 2

The burnup curve (proposed TS Figure 5.6.1) applies to the Robinson 15 x 15 fuel assembly types identified in Table 4.3.1 of Enclosure 6 to CP&L's license amendment request, dated 12/23/98.

The selection of spent fuel for shipment to Harris is made in accordance with procedure NFP-NGGC-0003, entitled "Procedure for Selection of Irradiated Fuel for Shipment in the IF-300 Spent Fuel Cask." The purpose of this procedure is to assure that the requirements of the IF-300

Cask Certificate of Compliance No. 9001 are met with regard to the selection of irradiated fuel to be shipped and that the fuel selected for shipment is acceptable for storage at CP&L's Harris plant. This procedure has been in use since 1990 for Robinson spent fuel shipments.

A computer program, which has also been in use since 1990 for Robinson spent fuel shipments, is used in conjunction with the above-referenced fuel selection procedure. For candidate assemblies to be shipped, the program retrieves the fuel type, enrichment, burnup, and decay heat from the special nuclear materials database. The initial enrichment data for each fuel assembly is contained in this database along with the other fuel data, and this data is based on manufacturing records. The burnup data for each fuel assembly is also included in the database along with the other isotopic inventories, and this data is obtained from the core monitoring software used for the Robinson plant. The special nuclear material database and core monitoring software have also been in use since 1990 for Robinson shipments.

The burnup curve proposed as TS Fig. 5.6.1 for pools C and D has already been programmed into the software for use in conjunction with fuel selection procedure NFP-NGGC-0003; however, this version is not yet in production as testing and documentation per CP&L's computer code quality assurance requirements are in progress. This new version will screen candidate PWR (Robinson) fuel against the burnup curve.

Revision to fuel selection procedure NFP-NGGC-0003 to reflect criticality screening requirements for fuel to be stored in Harris pools C or D has begun, but will not be completed until after: (1) the software changes identified above have been tested and the revised software placed in production status, and (2) the NRC has approved CP&L's license amendment application to place spent fuel pools C and D in service.

Requested Item 3

The fuel enrichment tolerance is specified in Section 4.5.2.5 as +0.0/-0.05. Why isn't a positive tolerance of +0.05 assumed (i.e., 5.0+0.05 weight percent U-235)?

Response to Requested Item 3

A maximum U-235 enrichment of 5.0 weight percent was specified, because it is the maximum enrichment allowed by both the Robinson and Harris Technical Specifications. Robinson TS 4.3.1.1.a states that the spent fuel racks shall be maintained with fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent. Robinson TS 4.3.1.2.a states that the new fuel racks shall be maintained with fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent. Harris TS 5.3.1 states that the initial core loading shall have a maximum enrichment of 3.5 weight percent U-235 and that reload fuel shall have a maximum enrichment of 5.0 weight percent U-235.

Also, the manufacturing facility of Siemens Power Corporation (SPC), the current fuel supplier for both the Robinson and Harris plants, is limited by license to a maximum U-235 enrichment of 5.0 weight percent. The SPC manufacturing tolerance is 0.05 weight percent U-235. Therefore, for enrichments with a tolerance of +/- 0.05%, the nominal design enrichment may not exceed

4.95 weight percent U-235 to ensure that the nominal plus the tolerance does not exceed 5.0 weight percent. The fuel enrichment and density tolerances specified in Section 4.5.2.5 appropriately supports a maximum allowable enrichment of 5.0 weight percent U-235.

Requested Item 4

Justify that the allowance that was assumed for possible differences between the fuel vendor and the Holtec calculations is sufficient to also encompass burnup calculational uncertainties.

Response to Requested Item 4

The Criticality Safety Calculations for the BWR Fuel Racks are summarized in Table 4.2.2 of Enclosure 6 to CP&L's license amendment request, dated 12/23/98. An uncertainty on depletion was not explicitly included in the uncertainties summarized in Table 4.2.2. Instead, the 0.01 additive allowance for comparisons to vendor calculations discussed in Section 4.4.2.2 also accounts for burnup uncertainty. This practice is acceptable for the following two reasons:

First, the BWR calculations consider the peak reactivity during burnup. The k_{inf} in the rack corresponding to a peak k_{inf} in the Standard Cold Core Geometry (SCCG) of 1.32 was calculated in the analysis. The burnup corresponding to this peak reactivity value is simply a by-product of this calculation and, in contrast to PWR analysis, burnup is not used as a criteria for establishing acceptability for fuel storage. Any uncertainty in the burnup calculation would simply decrease or increase, with burnup, the location of the peak reactivity. However, the k_{inf} in the SCCG and the k_{inf} in the rack would remain the same at the peak in reactivity. As a result, an additional uncertainty on depletion is not necessary.

Second, the fuel vendor performs similar depletion calculations to those discussed in Section 4. Therefore any uncertainty in depletion is an inherent part of the comparison between those calculations in Section 4 and those performed by the vendor to determine the peak k_{inf} in SCCG as a function of burnup. Again, it is noted that the actual burnup at which the peak occurs is not used in the BWR acceptable fuel storage criteria.

Requested Item 5

The summary of criticality safety calculations shown in Tables 4.2.1 and 4.2.2 indicates that the total uncertainty is a statistical combination of the manufacturing tolerances but do not indicate methodology biases and uncertainties. Were these included?

Response to Requested Item 5

Section 4.4.1 of Enclosure 6 to CP&L's license amendment request, dated 12/23/98, discusses the fact that CASMO-3, because it is a two-dimensional code, can not be directly compared to critical experiments and as a result a calculational/methodology bias is not available for CASMO-3. This section also discusses MCNP, which is a full three-dimensional Monte Carlo code, which has been benchmarked against critical experiments. CASMO-3 was used as the primary method of calculation and the results from CASMO-3 were compared to the regulatory limit of $k_{eff} \leq 0.95$ in Tables 4.2.1 and 4.2.2. As noted, the methodology bias and uncertainty were not included in these tables. However, these factors were implicitly included in a code-to-code comparison between CASMO-3 and MCNP shown in Table 4.5.1.

As discussed above, a methodology bias can not be developed for CASMO-3. Therefore, CASMO-3 results were compared to MCNP results to either verify that it produces conservative results relative to the benchmarked MCNP, or to determine a code-to-code bias. This comparison is discussed in Sections 4.5.1 and 4.6.1 with the results presented in Table 4.5.1. In the comparison between MCNP and CASMO-3, the methodology bias, uncertainty on the bias, calculational statistics, and a correction from 20°C to 4°C were added to the MCNP results. These results indicate that CASMO-3 is conservative relative to the benchmarked code MCNP and therefore the code-to-code bias was 0.0 for CASMO-3. Since the code-to-code bias was 0.0, it was not included in Tables 4.2.1 and 4.2.2. In conclusion, it can be stated that even though a methodology bias and uncertainty were not directly included in the final results shown in Tables 4.2.1 and 4.2.2, they were implicitly included through comparison of CASMO-3 and the benchmarked MCNP, provided in Table 4.5.1.

CONTENTION TC-2: EXHIBIT 6

Transcript of Deposition of Michael J. DeVoe, P.E. (October 20, 1999)

COPY

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
DOCKET NO. 50-400-LA
ASLBP NO. 99-762-02-LA

In the Matter of:)
)
CAROLINA POWER AND LIGHT COMPANY)
)
(Shearon Harris Nuclear Power Plant))
)
- - - - - - - - - - - - - - - - - - -

DEPOSITION

OF

MICHAEL J. DEVOE, P.E.

At the Offices of Carolina Power & Light Company
411 Fayetteville Street Mall
Raleigh, North Carolina

October 20, 1999
9:40 a.m.

A P P E A R A N C E S

JOHN H. O'NEILL, JR., ESQ.
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Also Present

DR. GORDON THOMPSON

JOHN R. CAVES, P.E.

LAWRENCE KOPP

C O N T E N T S

E X H I B I T S

calculations done by Holtec for its criticality analysis.

A I reviewed the report, the documents, the results of those calculations.

Q And what aspect of the report are you competent to evaluate?

A To insure that they use the appropriate fuel data that was provided, that they analyzed all the fuel designs that we intend to store in those pools, and that the results appear to be reasonable and in accordance with applicable requirements.

Q Have you provided Holtec with any information other than information about the characteristics of the spent fuel itself?

MR. HOLLAWAY: Objection. Could you clarify--I think I know what you're talking about--but whether you're speaking about this particular license amendment application and these analyses.

Q In providing Holtec with information necessary for the criticality analysis that it performed with respect to this particular license amendment application, have you provided Holtec with any information other than information about the

MR. DEVOE

characteristics of the spent fuel itself?

A No, I don't believe so.

Q Have you provided Holtec with any information about the boron concentrations in the spent fuel pools?

A No.

Q Have you provided Holtec with any information about CP&L's system for tracking spent fuel movements at Harris?

A No.

Q I'm going to ask you some questions about CP&L's measures for identifying and keeping track of the spent fuel assemblies that come into the Harris plant and reside there.

Would I be correct in saying that there are three basic steps involved here? One would be the cataloguing or describing of the characteristics of each spent fuel assembly.

Another would be tracking the specific location of each spent fuel assembly, and--I'm sorry. I have to strike the word "spent"--each fuel assembly. And the last would be to verify that steps 1 and 2 have been taken appropriately.

MR. DEVOE

Is that a correct description of the steps involved?

A I believe so.

Q I'd like to ask you about the first step, which would be describing or cataloguing the nature of the fuel.

Can you tell me what are the characteristics that are catalogued or described when you make a record of fuel assemblies coming into the plant?

A The assembly ID, their serial number, the amount of fissile material contained in that bundle in terms of grams and the uranium enrichment.

Q There's a record of that information that's made when the assembly enters the plant; is that correct?

A Are you talking about fuel coming fresh to the Harris plant or being brought from other plants to Harris?

Q I'm talking about both.

A Well, there's also a record of the burn-up and there's also a record of all of its previous locations.

And it's not clear to me when you ask that question as to what time you're talking about. I think I know, but--

MS. CURRAN: Well, I'm talking about when the fuel arrives at Harris, where have you gotten the information.

MR. HOLLAWAY: Regardless of where it's coming from.

MS. CURRAN: Regardless of where it's coming from.

Q So answer the question with respect to both fresh and spent fuel if it's different. So for the amount of fissile material.

A For the fresh, that comes from the fuel vendors. For the exposed, it comes from our reactor records, our special nuclear material accountability.

Q Now, when Harris gets spent fuel assemblies, at the moment they are coming from other CP&L plants; is that correct?

A (Witness nods affirmatively.)

Q So when you say "our" you mean other CP&L--that's the CP&L organization.

A Correct. I mean the Brunswick plant or the

Robinson plant.

Q Has CP&L ever accepted fuel assemblies from any other facility other than--at Harris from any facility other than a CP&L facility?

A Not to my knowledge.

Q You had mentioned a special nuclear material, accountability something. What did you say it was?

A It's just a program, computer program.

Q All right. I'll come back to that. Let's finish going through this list of the information that's included in the first record that's made.

So the amount of fissile material in a spent fuel assembly that's coming from a CP&L plant or from the Harris plant is recorded--when you get it at the Harris plant, you get information from CP&L's special nuclear material accountability program; is that right?

A (Witness nods affirmatively.)

Q Okay. What about with respect to burn-up? Where does that information come from for fresh and spent fuel assemblies?

A Well, for the fresh fuel assemblies the burn-up is zero. For the exposed fuel assemblies, that's part of the core monitoring program output.

Q Is that another computer program?

A Yes.

Q Okay. And then the last item was previous locations. Who provides that information to you with respect to the fresh fuel?

A With the fresh fuel, that would not have-- it's not a valid record for fresh fuel.

Q And how about for the spent fuel?

A It's a history of the approved and executed fuel movement instructions.

Q And who provides that information? Where does that information come from?

A That comes from the completed fuel movement procedure instructions.

Q Is that also a computer program?

A This information is stored on a database, computer database, yes. But the locations are not necessarily computer-generated. The movement instructions can be written by hand.

Q Could you explain that? Is it a piece of paper?

A Yes.
Q It's a piece of paper.
A Uh-huh (affirmative).
Q And that has a list of the previous locations.
A I'm not sure of that, of how far back it goes.
Q Okay. So when this information is received, what happens to it? How is it recorded?
A It's maintained on its database.
Q Does CP&L have a procedure for verifying all of the information that's provided when an assembly is received?
A I don't know.
Q Okay. Could you describe for me the special nuclear material accountability program?
A I could describe my understanding of it. It's not one of my specialty areas.
Q Okay.
A As a licensee, we're required to keep track of certain information associated with what we call special nuclear material, which in this case is uranium. And we keep track of how much we have and

where it's at.

And also, when we, if you will, generate special nuclear material by irradiating a fuel in a reactor, there are other isotopes besides uranium 235 that we are required to account for. And that program is more of a--not necessarily a computer program, but a program that we have in place to keep track of that material.

Q So is this the program that you go to when you want to ask where is a specific fuel assembly?

A Yes.

Q Is this program universal to all of the CP&L plants or is it just at Harris?

A All three sites use the program.

Q Using this special nuclear material accountability program, can you track the history of each spent fuel assembly?

A Yes.

Q So is this database a unified database that covers the complete history for every fuel assembly that's used at any CP&L plant?

A I'm not sure.

Q I'm sorry. Were you about to say something more?

A No.

Q Is it a realtime database?

A I'm not sure what you mean by that question.

Q If I want to know right now what is the complete history of each fuel assembly from the database, can I get that information immediately?

A Yes.

Q That's what I mean by realtime.

A Yes.

Q Yes. And that covers all of the information up to the present.

A Provided the records have been loaded into the system.

Q Okay. Why don't you tell me about that?

A I'm not sure what you mean by that question.

Q Well, is the information not recorded immediately into the database when it's received?

A Obviously, between the physical movement of the fuel and the recording on the paper and the transferring of the paper to the person responsible for the database and entering that information into the database, there is a certain time lapse.

Q Are there any requirements for the maximum amount of time that is allowed to pass?

A I don't know.

Q Do you know what the backup is for this database?

A No, I don't.

Q What is the physical location of the database?

A I'm not sure.

Q How is the database accessible to a Harris plant operator?

A The database is maintained on our computer network. And if a person has been granted access to that database, they would be able to access it from a desktop personal computer.

Q Is there a paper copy of the database that's kept?

A I'm not sure.

Q You also mentioned a core monitoring program. That's correct?

A Correct.

MR. DEVOE

Q Could you describe that for me?

A Basically, that's the on-line reactor core monitoring system that monitors power shapes, power levels, and what we use to demonstrate conformance with technical specification requirements.

Q Does the core monitoring program have any relationship to the SNM accountability program?

A Yes.

Q How are they related?

A The core monitoring program tracks the assembly burn-up and the isotopes, the generation of the isotopes as a result of that burn-up. And then that information is transferred to the special nuclear material program.

Q When new information is received or generated that needs to be input into the special nuclear material accountability program, but it hasn't been input yet--I just want to take that situation--is any notation made in the program at that point?

A I don't know.

Q You don't know. Do you know who does know?

A Yes.

Q Can you tell me that?

A The names of the people?

Q Uh-huh (affirmative).

A Bob Kunita and Linda Young.

Q And why are they people that know? What are their responsibilities?

A One of their responsibilities is the special nuclear material program.

Q This program, the special nuclear material accountability program, deals with fuel that is in a CP&L system. That's correct?

A Yes.

Q If CP&L takes fuel from other plants that are not currently in the system, are there any other measures that would be added to this program?

MR. HOLLAWAY: I object to that question. There's no foundation.

Q It's my understanding that CP&L has just purchased a nuclear plant in Florida. If CP&L were to take fuel from the plant in Florida for which the information about the fuel characteristics is not currently in the SNM accountability program, what measures would be taken?

MR. HOLLAWAY: Well, I object again.

There's no foundation for that scenario occurring.

Q You may answer the question.

A I mean, that's highly speculative. You know, the purchase has not been finalized. And you're asking me to predict what would happen in the future.

My best estimate or expectation of what would happen is that it would be treated the same way, that that information would be added. But I'm not aware of any thoughts of even doing that.

Q I'd like to go to the step which would be the tracking of the fuel. Suppose that you need to know where a specific fuel assembly is in a plant. How do you find out?

A One method would be to go to the special nuclear material database.

Q And assuming the information has been input into the computer, the database would tell you where it was.

A Correct.

Q What would another method be?

A Depending upon where the fuel was, if it was in the core, in the reactor, there's other--you can look at the loading pattern. The core map describes the loading pattern.

Q The core map? Is that what you said?

A Yes.

Q And where is that kept?

A I'm not sure what--

Q What kind of a map is it? Is it a paper map?

A Yes. It's a paper map.

Q And where is the paper map located?

A It would be in the reload modification package, what we call an engineering service request.

Q Are there other methods for tracking the fuel?

A You could--yes. You could go through the completed fuel-handling procedures.

Q And is that a paper document or a computer file?

A It would be a paper document.

Q And where is that kept?

A In the vault, the quality assurance vault.

Q Suppose that a fuel assembly is moving down a canal and I want the tracking system to tell me where it is and where it was and where it's going.

MR. DEVOE

Would the tracking system give me that information?
A I don't know.
Q Do you know if Federal Express would do that for you if it was a package?
MR. HOLLAWAY: Object. That's not relevant to the proceeding. The witness may answer.
A I've seen it on advertisements.
Q If I wanted to get the history of a fuel assembly that was in a particular rack position, could I get that by going to the SNM database?
A I believe so, yes.
Q And I could get the complete history of that assembly?
MR. HOLLAWAY: I'm going to object as ambiguous. When you say "history," I'm not sure what you mean.
Q The question is that for each location can you--first of all, can you identify each location where the fuel has been?
A Yes.
Q And for each of those locations, can you get information about the fuel characteristics, the burn-up, enrichment, amount of fissile material?

A Yes.

Q Can I get this information that we just discussed--when I'm looking for the previous history of locations and the characteristics at each location, can I get this information immediately?

A What do you mean by "immediately"?

Q Within a minute.

A I believe so.

Q If I knew the characteristics of the fuel and the previous location of the fuel, could I use that information to find the current location using this program?

A Yes.

Q So it will search--even if you don't have all the information, you can put in some input and get an ID?

MR. HOLLAWAY: Object. When you say "some input"--

Q Well, if you put in some of the characteristics of the fuel.

A Yes. It's a database and it's searchable, as a database would be.

Q Supposing that I were trying to verify whether the information regarding, say, the burn-up level that's recorded in the database for a particular fuel assembly is correct. How would I go about doing that? What measures does CP&L have for doing that?

A I'm not sure.

Q You don't know anything about measures for validating the data that's been input into the program.

A I'm not involved in those activities or familiar with the activities.

Q Can you tell me who would be?

A Yes. Bob Kunita and Linda Young.

MR. HOLLAWAY: At some point I'd like to take a break just as a regular break to talk to him.

MS. CURRAN: Okay.

MR. HOLLAWAY: I don't know if this would be a good time.

MS. CURRAN: Maybe in just a few minutes.

MR. HOLLAWAY: Okay.

Q If you look at a specific rack position, how can you be sure that the fuel in that position in the rack has the same ID as the database says it does?

A How can you be sure? One way is the surveys of the pool, the video camera.

Q Now, tell me, why would that tell you?

A You could lower the video camera into the rack location of interest and actually read the serial number on the bundle and then compare that to your records.

Q Is that done?

MR. HOLLAWAY: Object as ambiguous. You said, "Is that done?" Is that done when, by who? I don't know what that means.

Q Before a fuel assembly is moved, are any steps taken to verify the identity of the fuel assembly that's being moved?

A I don't know.

MS. CURRAN: This is a good time for a break.

(Whereupon, a brief recess was taken.)

Q Mr. DeVoe, have you searched the database that we've been speaking of?

A No.

Q You haven't?

A No.

Q How do you know about this database?

A I request it to be queried to provide data that I might need.

Q So someone else uses the database, but you provide the request for information?

A Yes.

Q You've provided some information to me about the characteristics of the database today. How did you get that information?

A From being aware of what is in the database by looking at the results of the data queries and by providing the information that gets put into the database and by being familiar with the special nuclear materials plant.

Q You're saying you provide information that gets input to the database; is that correct?

A When you say "I," do you mean myself personally or in my work functions?

Q Well, you were the one that said it, so I guess--

A Okay. Could you repeat the question?

Q You said that you provide input to the database, I believe; is that correct?

A Yes.

Q What kind of information have you provided to the database?

A Information on fresh fuel when it's first delivered to the reactor site.

Q Can you tell me how you got the information that went into the database? Did you read a piece of paper?

A It's provided as part of the QA documentation by the fuel vendor when the fuel is manufactured. And I participate in surveillances of the vendor while he's manufacturing our fuel. And one of the things that I review is the documentation.

Q And what is the documentation of?

A In reference to our discussions here, it's the assembly ID and amount of uranium as manufactured.

Q When you review it, are you reviewing a piece of paper?

A Yes.

Q And then do you hand that piece of paper to a person who's inputting the information to the computer?

MR. DEVOE

A Yes.

Q So you don't transcribe or create a new document. You take the information that's been given to you and you hand it to the person who is inputting it to the computer.

A It's provided that way, yes.

Q It's provided--

A Well, I physically don't give the sheet of paper to the person.

Q But you make sure that person gets it.

A Yes.

Q In the course of providing information to Holtec regarding spent fuel characteristics for purposes of Holtec's criticality analysis in this particular proceeding, did you make any attempt to validate the data that you were providing?

A Yes.

Q And how did you do that?

A First off, the fuel-related data describes the fresher or unexposed fuel and its, primarily, dimensions and enrichments and the physical characteristics of the fuel assembly.

And we worked with the respective fuel

vendors to obtain that data, and they provided it under their quality assurance plans.

And then when it was received by CP&L, we perform a review of it to insure that it's the information that Holtec has requested and it's appropriate for use in this project.

Q What do you mean by "appropriate"?

A That we provided information on all the fuel types present as opposed to maybe just the most current, to make sure that we covered all the fuel types that were planned to be loaded, that it was in fact describing our fuel.

The vendors make fuel for many customers, and we insure that it describes our fuel, the CP&L fuel.

Q It sounds like the information you provided was from the vendors, not from the CP&L database; is that correct?

A Correct. The fuel information describes the fresh fuel.

Q Let me make sure I understand what you've told me. It's my understanding that you have attempted to validate data that you obtained from

MR. DEVOE

vendors in order to give to Holtec. Is that correct?

A Yes, we do validate it.

Q Have you ever attempted to validate data that you got from the CP&L SNM accountability database?

A I haven't.

Q In your current position, have you searched the SNM accountability database?

A Yes. I requested a query.

Q For what purpose?

A I requested a listing of the fuel assemblies that were, in this case, presently in the Robinson spent fuel pool.

Q And did you attempt--scratch that. And for what purpose did you request that information?

A To support a criticality evaluation.

Q And did you attempt to validate the information that you had obtained from the database, this listing?

A No.

Q Would you know how to do that?

A Would I know how to validate it?

Q Right.

MR. DEVOE

A Yes.

Q How would you do it?

A I would compare the database records against the reactor records.

Q What reactor records?

A As I mentioned earlier, there's an on-line process computer that tracks fuel assembly burn-up. And I would--the records produced by that are the records that are intended to be transferred into the special nuclear material database.

Q Is that a separate source of information? If it gets transferred into the special nuclear material database, how do you know that's not the source of information you originally queried?

A I'm not sure I understand the question.

Q You had said that the on-line process computer puts information into the SNM accountability database; is that correct?

A The information generated is transferred to the SNM. It's not automatically put there, but it's transferred. In this case, we're talking about a limited set of the information, just the burn-up and the isotopics at this point.

MR. DEVOE

Q Are you competent to physically validate the burn-up level of the spent fuel assembly?

A I'm not sure what you mean by "physically."

Q What I'm talking about is, if you have a specific fuel assembly in a rack and you're required to experimentally verify the characteristics, the burn-up characteristics that are described in the database, would you be able to that?

MR. HOLLAWAY: I object to that for a foundation. I don't know that there's any foundation for a requirement, as you stated, for experimental validation of fuel assembly burn-up.

Q Can you answer the question?

A I am not.

Q You would not know how to do that?

A I know of ways it could be done, but I have never done that myself.

Q What are the ways that it could be done?

A One technique is called gamma scan.

Q A gamma what?

A Scan.

Q How does that work?

A The assembly is--measurements are taken of the assembly for a particular isotope that is representative of the power and burn-up distribution.

Q Can that operation be done reliably for an assembly that is in a rack with other assemblies nearby?

A I don't know.

Q Did you participate in the design of the SNM accountability database?

A No.

Q Have you ever been involved in any changes to the program?

A Yes.

Q Can you describe that for me?

A We are in the process of making and implementing changes to support adding to the database the information required to facilitate implementation of pools C and D. And I'm providing the input. I'm not doing any manipulation of coding.

Q And what kind of input are you providing?

A The maximum planar average enrichment for the fuel assemblies.

Q Who is responsible for actually changing the program?

A Bob Kunita.

Q The same person you mentioned before?

A Yes.

Q Do you know what kinds of changes are being made?

A In general, I don't know the specific coding changes. But the functionality changes I'm aware of.

Q Would you please describe those for me?

A It's to add a data field that records the maximum planar average enrichment for the PWR fuel assemblies and for the boiling water reactor, BWR fuel, the maximum lattice planar average enrichment, and the standard cold core geometry K-infinity.

Q Mr. DeVoe, are you familiar with the physical process for introducing boron into the spent fuel pools at Harris?

A No.

Q You know nothing about it?

A (No response.)

Q Do you know how the boron gets into the pool?

A I believe I do.

Q Can you explain it to me?

A I believe it's added by the operators.

Q How do the operators do it?

A I don't know those details.

Q Do you know whether boron concentrations are monitored in the pools?

A Yes.

Q Do you know how frequently they're monitored?

A No, I don't.

Q Are records maintained of boron measurements?

A Yes.

Q For how long are they maintained?

A I do not know.

Q Has CP&L prepared any boron dilution analyses that you know of?

A Could you clarify that? We're restricting this to the pool.

Q Yes.

A And what do you mean by "boron dilution"?

Q I'm actually using a term that's provided in an NRC guidance document, which isn't defined further than that.

A I'm not aware of any calculations.

Q Do you know if the physical correspondence between spent fuel pools makes any difference in maintaining the boron concentration in the pools?

A I do not know that.

Q Do you know if CP&L has ever done any studies or analyses of its own experience with maintaining boron concentrations in its spent fuel pools?

A No, I do not.

Q Do you know if there are any industry studies that have been done on industry experience with controlling boron that was in spent fuel pools?

A No, I do not.

Q Do you know of any studies prepared by CP&L or any other entity that would describe the probability or consequences of accidents resulting from errors in boron concentration levels?

A No.

Q Do you know of any studies or analyses by CP&L or any other entity of the probability or consequences of criticality accidents in spent fuel pools?

A I don't recall seeing any.

Q Do you know of any analyses or studies done by CP&L or any other entity regarding errors or accidents caused by the mishandling or misplacement of fuel assemblies?

A No.

Q Can you tell me what regulations or guidance documents are followed by CP&L in attempting to maintain criticality control at the Harris plant?

A Yes.

Q And what are they?

A GDC-62 and Reg Guide 1.13.

Q Any others?

A Not that I'm aware of.

MS. CURRAN: We're going to take a short break.

(Whereupon, a brief recess was taken.)

(Mr. Caves and Mr. O'Neill exit.)

MS. CURRAN: I have no further questions.

MS. UTTAL: I don't have any questions.

MR. DEVOE

MR. HOLLAWAY: I don't have any questions.

AND FURTHER DEPONENT SAITH NOT.

(Deposition concluded at 11:30 a.m.)

SIGNATURE OF WITNESS

I have read the foregoing pages numbered 4 through 39, inclusive, which contain a correct transcription of answers made by me to the questions therein recorded, with the exceptions and/or additions reflected on the errata sheet attached hereto, if any.

Signed this 9th day of December, 1999.

Michael J. DeVoe

MICHAEL J. DEVOE, P.E.

****

STATE OF ______________________

COUNTY OF ______________________

Subscribed and sworn to before me this _______ day of ______________________, 1999.

____

NOTARY PUBLIC

(SEAL)

My Commission Expires:

____

ERRATA

I, Michael J. DeVoe, the witness herein, have read my deposition and request that the following changes be made:

PAGE	LINE	CHANGE	REASON FOR CHANGE
8	5	", the documents," to "that documents"	Typographical
8	9	"use" to "used"	Typographical
14	10	"have - -" to "have significance,"	Clarification
15	12	"its database" to "the database"	Typographical
16	5	"irradiating a fuel" to "irradiating fuel"	Typographical
27	17	"materials plant" to "material plan"	Typographical
29	22	"fresher" to "fresh"	Typographical

CONTENTION TC-2: EXHIBIT 8

Internal AEC memorandum from G.A. Arlotto to J.J. DiNunno and Robert H. Bryan (October 7, 1966), and attached Revised Draft of General Design Criteria for Nuclear Power Plant Construction Permits (October 6, 1966)
(relevant excerpt)

Those Listed Below

October 7, 1966

G. A. Arlotto
Facilities Standards Branch, SS

REVISED DRAFT - GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANT CONSTRUCTION PERMITS

Attached is a revised draft of the General Design Criteria for Nuclear Power Plant Construction Permits dated October 6, 1966, which I developed for your consideration. In comparison with the previous draft, which was dated July 25, 1966, the attached version reflects the following:

1. Changes suggested by ACRS Subcommittee members at meetings of August 10 and September 21, 1966.

2. Changes suggested in the Backup Document dated August 9, 1966.

3. Changes suggested in memorandum from Robert H. Bryan to J. J. DiNunno dated October 3, 1966.

4. Changes resulting from discussions among the addressees and myself.

5. My suggestions which time did not permit resolution of with the addressees.

Attachment:
As Stated Above

Addressees:
J. J. DiNunno, Assistant Director for Reactor Standards, SS
Robert H. Bryan, Chief, Facilities Standards Branch, SS

OFFICE ▶	SS:FSB					
SURNAME ▶	Arlotto:jjb					
DATE ▶	10-7-66					

Revised Draft
10/6/66

GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANT CONSTRUCTION PERMITS

The purpose of these criteria is to define or describe the basic safety objectives to be met in the design of a nuclear power plant. They are intended: (1) to serve as guidance to the applicant in preparing an application for an AEC construction permit and (2) to aid the AEC staff in reviewing that application.

The application of these criteria to a specific design involves a considerable amount of engineering judgment. There may be instances in which one or more of these criteria are unnecessary or are insufficient. It is not intended that the criteria be used as a check list of design objectives for all proposed plants, and the applicant is free to establish the safety of his design by alternative criteria. The criteria will be modified if, or as, future technological developments and experience warrant.

An applicant for a construction permit is expected to present a design approach together with data and analyses sufficient to give assurance that the design can reasonably be expected to fulfill all applicable criteria. It is recognized that the nature and detail of technical information and analysis required at the construction permit stage to provide such assurance may vary, depending on the particular criterion under consideration. Category A criteria encompass critical safety areas so fundamental in the design, procurement, fabrication, and construction of the plant that modification for reasons of safety at the operating license review stage would be exceedingly difficult and costly; in essence, for practical purposes, decisions made at the construction permit stage in these areas are irrevocable. Where novel features are associated with criteria which are site-sensitive or are directly related to limiting the accidental release of radioactivity into the public domain, they must be dealt with in a relatively complete way at the construction permit stage even if the "irrevocable" condition is not met. Category B criteria encompass safety areas where the modifications can be made for reasons of safety at the operating license review stage without placing an undue burden on the parties concerned. These criteria principally concerned with protecting the operational capability of the reactor may be dealt with in relatively less detail at the construction permit stage if more detailed information and analysis are not available at that time.

All applicable safety criteria must, of course, be fulfilled as a condition for issuance of a license to operate the plant.

CRITERION 1 (Category A) QUALITY AND PERFORMANCE STANDARDS

Those features of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences shall be designed, fabricated, and erected to:

(a) Quality standards* that reflect the importance of the safety function to be performed. Where generally recognized codes or standards on design, materials, fabrication, and inspection are applicable, they shall be used. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety function, they shall be supplemented as necessary.

* A showing of sufficiency and applicability of standards used shall be required.

(2) Active components, such as pumps and valves, can be tested periodically for operability and required functional performance.

(3) A capability is provided to test periodically the delivery capability at a position as close to the spray nozzles as is practical.

(4) A capability is provided to test under conditions as close to the design as practical the full operational sequence that would bring the systems into action, including the transfer to alternate power sources.

CRITERION 10 (Category B) FUEL AND WASTE STORAGE SYSTEMS

Storage and handling systems for fuel and waste shall be designed on the basis that:

1. Possibilities for inadvertent criticality must be prevented by engineered systems or processes toevery extent practicable. Such means as geometric safe spacing limits shall be emphasized over procedural controls.

2. Reliable decay heat removal means must be provided as necessary to prevent fuel or storage volume damage that could result in radioactivity release to plant operating areas or the public environs. Such means must be assured for all anticipated normal and abnormal conditions as well as those accident situations whereby normal cooling could credibly become lost.

CONTENTION TC-2: EXHIBIT 7

AEC Press Release entitled "AEC seeking public comment on proposed design criteria for nuclear power plant construction permits"
(November 22, 1965)

AEC
UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

No. H-252
Tel. 973-3335 or
973-3446

FOR IMMEDIATE RELEASE
(Monday, November 22, 1965)

AEC SEEKING PUBLIC COMMENT ON PROPOSED DESIGN CRITERIA
FOR NUCLEAR POWER PLANT CONSTRUCTION PERMITS

The Atomic Energy Commission is seeking comment from the nuclear industry and other interested persons on proposed general design criteria which have been developed to assist in the evaluation of applications for nuclear power plant construction permits.

The proposed criteria have been developed by the AEC regulatory staff and discussed with the Commission's Advisory Committee on Reactor Safeguards (ACRS). They represent an effort to set forth design and performance criteria for reactor systems, components and structures which have evolved over the years in licensing of nuclear power plants by the AEC. As such, they reflect the predominating experience to date with water reactors but most of them are generally applicable to other reactors as well.

It is recognized that further efforts by the AEC regulatory staff and the ACRS will be necessary to fully develop these criteria. However, the criteria as now proposed are sufficiently advanced to submit for public comment. Also, they are intended to give interim guidance to applicants and reactor equipment manufacturers.

The development and publication of criteria for nuclear power plants was one of the key recommendations of the special Regulatory Review Panel which studied ways of streamlining the Commission's reactor licensing procedures.

In the further development of these criteria, the AEC intends to hold discussions with organizations in the nuclear industry and to issue from time to time explanatory information on each criterion. Following such discussions with industry and receipt of other public comment, the AEC expects to develop and publish criteria that will serve as a basis for evaluation of applications for nuclear power plant construction permits.

(more)

GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANT CONSTRUCTION PERMITS

Attached hereto are general design criteria used by the AEC in judging whether a proposed nuclear power facility can be built and operated without undue risk to the health and safety of the public. They represent design and performance criteria for reactor systems, components and structures which have evolved over the years in licensing of nuclear power plants by the AEC. As such they reflect the predominating experience to date with water reactors but most of them are generally applicable to other reactors as well.

It should be recognized that additional criteria will be needed for evaluation of a detailed design, particularly for unusual sites and environmental conditions, and for new and advanced types of reactors. Moreover, there may be instances in which it can be demonstrated that one or more of the criteria need not be fulfilled. It should also be recognized that the application of these criteria to a specific design involves a considerable amount of engineering judgment.

An applicant for a construction permit should present a design approach together with data and analysis sufficient to give assurance that the design can reasonably be expected to fulfill the criteria.

FACILITY

CRITERION 1

Those features of reactor facilities which are essential to the prevention of accidents or to the mitigation of their consequences must be designed, fabricated, and erected to:

(a) Quality standards that reflect the importance of the safety function to be performed. It should be recognized, in this respect, that design codes commonly used for nonnuclear applications may not be adequate.

CRITERION 6

Clad fuel must be designed to accommodate throughout its design lifetime all normal and abnormal modes of anticipated reactor operation, including the design overpower condition, without experiencing significant cladding failures. Unclad or vented fuels must be designed with the similar objective of providing control over fission products. For unclad and vented solid fuels, normal and abnormal modes of anticipated reactor operation must be achieved without exceeding design release rates of fission products from the fuel over core lifetime.

CRITERION 7

The maximum reactivity worth of control rods or elements and the rates with which reactivity can be inserted must be held to values such that no single credible mechanical or electrical control system malfunction could cause a reactivity transient capable of damaging the primary system or causing significant fuel failure.

CRITERION 8

Reactivity shutdown capability must be provided to make and hold the core subcritical from any credible operating condition with any one control element at its position of highest reactivity.

CRITERION 9

Backup reactivity shutdown capability must be provided that is independent of normal reactivity control provisions. This system must have the capability to shut down the reactor from any operating condition.

CRITERION 14

Means must be included in the control room to show the relative reactivity status of the reactor such as position indication of mechanical rods or concentrations of chemical poisons.

CRITERION 15

A reliable reactor protection system must be provided to automatically initiate appropriate action to prevent safety limits from being exceeded. Capability must be provided for testing functional operability of the system and for determining that no component or circuit failure has occurred. For instruments and control systems in vital areas where the potential consequences of failure require redundancy, the redundant channels must be independent and must be capable of being tested to determine that they remain independent. Sufficient redundancy must be provided that failure or removal from service of a single component or channel will not inhibit necessary safety action when required. These criteria should, where applicable, be satisfied by the instrumentation associated with containment closure and isolation systems, afterheat removal and core cooling systems, systems to prevent cold-slug accidents, and other vital systems, as well as the reactor nuclear and process safety system.

CRITERION 16

The vital instrumentation systems of Criterion 15 must be designed so that no credible combination of circumstances can intefere with the performance of a safety function when it is needed. In particular, the effect of influences common to redundant channels which are intended to

CRITERION 19

The maximum integrated leakage from the containment structure under the conditions described in Criterion 17 above must meet the site exposure criteria set forth in 10 CFR 100. The containment structure must be designed so that the containment can be leak tested at least to design pressure conditions after completion and installation of all penetrations, and the leakage rate measured over a suitable period to verify its conformance with required performance. The plant must be designed for later tests at suitable pressures.

CRITERION 20

All containment structure penetrations subject to failure such as resilient seals and expansion bellows must be designed and constructed so that leak-tightness can be demonstrated at design pressure at any time throughout operating life of the reactor.

CRITERION 21

Sufficient normal and emergency sources of electrical power must be provided to assure a capability for prompt shutdown and continued maintenance of the reactor facility in a safe condition under all credible circumstances.

CRITERION 22

Valves and their associated apparatus that are essential to the containment function must be redundant and so arranged that no credible combination of circumstances can interfere with their necessary functioning. Such redundant valves and associated apparatus must be independent

CRITERION 26

Where unfavorable environmental conditions can be expected to require limitations upon the release of operational radioactive effluents to the environment, appropriate hold-up capacity must be provided for retention of gaseous, liquid, or solid effluents.

CRITERION 27

The plant must be provided with systems capable of monitoring the release of radioactivity under accident conditions.

CONTENTION TC-2: EXHIBIT 9

Letter from J J DiNunno, AEC, to David Okrent, ACRS (October 25, 1966), and attached October 20, 1966 draft of General Design Criteria (relevant excerpt)

October 25, 1966

Dr. David Okrent, Chairman
Advisory Committee on Reactor Safeguards
U. S. Atomic Energy Commission
Washington, D.C. 20545

Dear Dr. Okrent:

Enclosed for consideration of the ACRS are draft copies of the General Design Criteria for Nuclear Power Plant Construction Permits. This redrafted material includes a comparison of criteria contained in the Press Release dated November 22, 1965, and those contained in our latest draft dated October 20, 1966. In addition, we have included along with a revised draft of the criteria dated October 20, 1966, a comparison of the October 20 draft with the July 25 draft previously submitted and discussed with the ACRS Criteria Subcommittee.

Our October 20, 1966, draft attempts to reflect results of our last discussion with the ACRS Subcommittee, and we would like to have the scheduled November 9th meeting on criteria be based on the October 20th draft.

Sincerely yours,

J. J. DiNunno
Assistant Director for
Reactor Standards
Division of Safety Standards

Enclosures:
1. Rev. Draft dated 10/20/66 of General Design Criteria (18)
2. Comparison of Drafts dated 7/25/66 and 10/20/66 for General Design Criteria (18)
3. Comparison of Criteria in Press Release dated 11/22/65 and Those in Rev. Draft dated 10/20/66 (18)

bcc: Harold L. Price, Director of Regulation, w/encl.
~~Clifford K. Beck, Deputy Dir. of Reg., w/encl.~~
Peter A. Morris, Director, DRL, w/encl. (6)
M. M. Mann, Asst. Dir. for Nuclear Safety, REG, w/encl.

OFFICE ▶ SS/ADIR
SURNAME ▶ DiNunno:jjb

<u>REVISED DRAFT OF</u>

<u>GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANT CONSTRUCTION PERMITS</u>

October 20, 1966

The purpose of these criteria is to define or describe the basic safety objectives to be met in the design of a nuclear power plant. They are intended: (1) to serve as guidance to the applicant in preparing an application for an AEC construction permit and (2) to aid the AEC staff in reviewing that application.

The application of these criteria to a specific design involves a considerable amount of engineering judgment. There may be instances in which one or more of these criteria are unnecessary or are insufficient. It is not intended that the criteria be used as a check list of design objectives for all proposed plants, and the applicant is free to establish the safety of his design by alternative criteria. The criteria will be modified if, or as, future technological developments and experience warrant.

An applicant for a construction permit is expected to present a design approach together with data and analyses sufficient to give assurance that the design can reasonably be expected to fulfill all applicable criteria. It is recognized that the nature and detail of technical information and analysis required at the construction permit stage to provide such assurance may vary, depending on the particular criterion under consideration.

To provide guidance as to the relative emphasis expected at the construction permit stage, the criteria have been divided into two broad categories. Category A criteria involve aspects of facility design that are site-sensitive or are directly related to limiting the accidental release of radioactivity into the public domain. These aspects of facility design are also categorized by their marked influence on plans for construction and operation. From a practical viewpoint, aspects of facility design satisfying Category A criteria are relatively fixed at the construction permit stage and not amenable to change without serious disruptions of construction plans and incurrence of considerable costs. For these reasons, those aspects of facility design provided in fulfillment of Category A criteria must be dealt with in a relatively complete way at the construction permit stage.

Category B criteria are intended to reflect primarily those aspects of design that provide for safe operational control of the facility. Such features are generally less unique to a facility than those required for satisfying Category A criteria and are much less determinate of facility construction schedules. Modifications to such features that might prove necessary, for safety reasons, following issuance of a construction permit are much more likely to be accommodated without the pressures for compromise that might well accompany the more time-consuming and costly type changes. Under these circumstances, criteria principally concerned with the safe operational control of the reactor and designated as Category B may be dealt with in relatively less detail at the construction permit stage, if more detailed information is not available at that time.

All applicable safety criteria must, of course, be fulfilled as a condition for issuance of a license to operate the plant.

9.2.4.4 A capability is provided to test under conditions as close to the design as practical the full operational sequence that would bring the systems into action, including the transfer to alternate power sources.

FUEL AND WASTE STORAGE SYSTEMS

CRITERION 10 (Category B) FUEL AND WASTE STORAGE

10.0 Storage and handling systems for fuel and waste shall be designed on the basis that:

10.1 Possibilities for inadvertent criticality must be prevented by engineered systems or processes to every extent practicable. Such means as geometric safe spacing limits shall be emphasized over procedural controls.

10.2 Reliable decay heat removal means must be provided as necessary to prevent fuel or storage volume damage that could result in radioactivity release to plant operating areas or the public environs. Such means must be assured for all anticipated normal and abnormal conditions as well as those accident situations whereby normal cooling could credibly become lost.

10.3 Shielding for radiation protection shall be provided as required from considerations of 10 CFR 20.

10.4 Containment of the systems shall be provided if accidents could lead to release of undue amounts of radioactivity to the public environs.

CONTENTION TC-2: EXHIBIT 10

Letter from J. J. DiNunno, AEC, to Nunzio J. Palladino, ACRS (February 8, 1967), and attached draft of General Design Criteria (relevant excerpt)

February 8, 1967

Mr. Nunzio J. Palladino, Chairman
Advisory Committee on Reactor Safeguards
U. S. Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Palladino:

Enclosed for consideration by the Committee is a redraft of General Design Criteria. The format of the criteria has been changed. The subparts previously listed in earlier drafts have been made into separate criteria. The wording of these criteria is essentially the same as those in the October 20, 1966, draft, modified to reflect subsequent discussions held with the ACRS Subcommittee in November and recent developments of criteria for emergency core cooling systems.

An additional document showing the changes made from the last draft discussed with the ACRS is under preparation and will be forwarded by separate correspondence.

Sincerely yours,

J. J. DiNunno
Assistant Director for
Reactor Standards
Division of Safety Standards

Enclosure:
General Design Criteria for Nuclear
Power Plant Construction Permits (18)

bcc: Harold L. Price, Director of Regulation, w/encl.
Clifford K. Beck, Deputy Director of Regulation, w/encl.
M. M. Mann, Asst. Dir. for Nuclear Safety, w/encl.
C. L. Henderson, Asst. Dir. for Administration, w/encl.
Peter A. Morris, Director, DRL, w/encl. (6)
Edson G. Case, Deputy Director, DRL, w/encl.
Forrest Western, Director, DRL, w/encl.

OFFICE ▶	SS:ADIR					
SURNAME ▶	DiNunno:jjb					
	2/8/67	2/8/67				

GENERAL DESIGN CRITERIA

FOR

NUCLEAR POWER PLANT CONSTRUCTION PERMITS

February 6, 1967

Table of Contents

VIII. FUEL AND WASTE STORAGE SYSTEMS

CRITERION 61 - PREVENTION OF FUEL STORAGE CRITICALITY (Category B)

Possibilities for criticality in new and spent fuel storage shall be prevented by physical systems or processes to every extent practicable. Such means as favorable geometries shall be emphasized over procedural controls.

CRITERION 62 - FUEL AND WASTE STORAGE DECAY HEAT (Category B)

Reliable decay heat removal systems shall be designed to ensure damage to the fuel or storage facilities that could result in radioactivity release to plant operating areas or the public environs is prevented. Such means must be assured for all anticipated normal and abnormal conditions as well as those accident situations whereby normal cooling could credibly become lost.

CRITERION 63 - FUEL AND WASTE STORAGE RADIATION SHIELDING (Category A)

Shielding for radiation protection shall be provided in the design of spent fuel and waste storage facilities as required from consideration of 10 CFR 20.

CRITERION 64 - PROTECTION AGAINST RADIOACTIVITY RELEASE FROM SPENT FUEL AND WASTE STORAGE (Category B)

Containment of fuel and waste storage shall be provided if accidents could lead to release of undue amounts of radioactivity to the public environs.

IX. PLANT EFFLUENTS

CRITERION 65 - CONTROL OF RELEASES OF RADIOACTIVITY TO THE ENVIRONMENT (Category B)

The facility design shall include those means necessary to maintain control over plant radioactive effluents, whether solid, liquid, or gaseous. Appropriate

CONTENTION TC-2: EXHIBIT 11

Note by the Secretary, W.B. McCool, to AEC Commissioners re: Proposed Amendment to 10 CFR 50: General Design Criteria for Nuclear Power Plant Construction Permits (June 16, 1967) (relevant excerpts)

Parks

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AEC-R 2/57

June 16, 1967

ATOMIC ENERGY COMMISSION

PROPOSED AMENDMENT TO 10 CFR 50: GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANT CONSTRUCTION PERMITS

Note by the Secretary

1. The Director of Regulation has requested that the attached report be circulated for consideration by the Commission at an early date.

2. The Commission approved the proposed design criteria, as revised, during consideration of AEC-R 2/49 at Regulatory Meeting 223 on November 10, 1965.

W. B. McCool

Secretary

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Secretary	11	Asst. GM for Operations	1
Chairman Seaborg	4	Asst. GM for Reactors	1
Commissioner Ramey	1	General Counsel	6
Commissioner Tape	2	Compliance	6
Commissioner Nabrit	2	Congr. Relations	2
Commissioner Johnson	2	Inspection	1
General Manager	2	Materials Licensing	2
Deputy Gen. Mgr.	1	Operational Safety	2
Dir. of Regulation	3	Plans & Reports	2
Deputy Dir. of Regulation	1	Public Information	2
Asst.Dir.of Reg.for Admin.	2	Reactor Dev. & Tech.	10
Asst.Dir.of Reg.for Reactors	1	Reactor Licensing	2
Asst. Gen. Mgr.	1	Reactor Standards	2
Exec. Asst. to GM	1	State & Lic. Relations	2
Asst. GM for Admin.	1	Chairman, AS&LBP	1

LEGAL 13 Pt 50

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... staff has considered all ...

the criteria. In addition, subsequent redrafts were circulated to other divisions within the Commission. Principal comments from these divisions have been reflected in the revised criteria. Other comments from within the Commission will be considered in conjunction with public comments received after publication in the Federal Register.

6. The regulatory staff has worked closely with the Advisory Committee on Reactor Safeguards on the development of the criteria and the revision of the proposed criteria reflects ACRS review and comment. The ACRS has stated that it believes that the revised criteria are appropriate to publish for public comment.

7. It is proposed that the criteria be included as Appendix A to 10 CFR 50. The proposed amendment, which is attached as Appendix "B," provides that the General Design Criteria be used for guidance by an applicant in developing the principal design criteria for the facility. For a specific reactor case, some of the General Design Criteria may be unnecessary or inappropriate and the criteria, as a whole, may be insufficient. It is expected that additional criteria will be needed particularly for unusual sites and environmental conditions, and for new and advanced reactor types. In any case, there must be assurance that the principal design criteria proposed by an applicant encompass all those facility design features required in the interest of public safety.

8. The criteria are designated as "General Design Criteria for Nuclear Power Plant Construction Permits" to emphasize the key role they assume at this stage of the licensing process. The criteria have been categorized as Category A or Category B. Experience has shown that more definitive information is needed at the construction permit stage for the items listed in Category A than for Category B.

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9. The proposed General Design Criteria are expected to be useful as interim guidance until such time as the Commission takes further action on them.

STAFF JUDGMENTS

10. The Office of the General Counsel and the Divisions of Reactor Licensing and Compliance concur in the recommendations of this paper. The Office of Congressional Relations concurs in Appendix "C." The Division of Public Information concurs in recommendation 11.c.

RECOMMENDATION

11. The Director of Regulation recommends that the Atomic Energy Commission:

a. Approve publication of the proposed amendments to 10 CFR Part 50 contained in Appendix "B."

b. Note that the Joint Committee on Atomic Energy will be informed by letter such as Appendix "C."

c. Note that a public announcement such as Appendix "D" be issued on filing the notice of proposed rule making with the Federal Register.

LIST OF ENCLOSURES

APPENDIX "A"

LIST OF INCOMING CORRESPONDENCE ON "AEC SEEKING PUBLIC COMMENT ON PROPOSED DESIGN CRITERIA FOR NUCLEAR POWER PLANT CONSTRUCTION PERMITS" PRESS RELEASE NO. H-252 DATED NOVEMBER 22, 1965

1. J. B. McCarty, Jr., U.S. Coast Guard, 1/26/66.
2. E. P. Epler, Oak Ridge National Laboratory, 1/26/66.
3. Dr. Emerson Jones, Technical Management, Inc., 2/2/66.
4. H. C. Paxton and D. B. Hall, Los Alamos Scientific Laboratory, 2/2/66.
5. C. Starr, Atomics International, 2/4/66.
6. C. T. Chave, Stone and Webster Engineering Corporation, 2/11/66.
7. R. L. Junkins, Pacific Northwest Laboratory, 2/8/66.
8. Richard Hughes, Governor of New Jersey, 2/10/66.
9. Royce J. Rickert, Combustion Engineering, Inc., 2/11/66.
10. W. B. Cottrell, Oak Ridge National Laboratory, 2/11/66.
11. Peter A. Morris, Director, Division of Operational Safety, 2/11/66.
12. Holmes & Narver, Inc., 2/11/66.
13. CDR J. C. Ledoux, BuY&D, Dept. of Navy, 2/11/66.
14. Richard H. Peterson, Pacific Gas and Electric Company, 2/14/66.
15. Norbert L. Kopchinski, Professional Engineer, California, 2/14/66.
16. D. L. Crook, Dept. of Commerce, Maritime Adm., Wash., D.C., 2/15/66.
17. R. H. Harrison, Babcock & Wilcox, 2/22/66.
18. Theodore Stern, Westinghouse Electric Corporation, 2/25/66.
19. E. A. Wiggin, Atomic Industrial Forum, 2/28/66.
20. James G. Terrill, Jr., Dept. of Health, Education, and Welfare, Washington, D.C., 3/7/66.
21. J. P. Hogan, General Atomic, 4/30/66.
22. H. G. Rickover, Director, Division of Naval Reactors, 7/26/66.

APPENDIX "B"

[10 CFR PART 50]

LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

General Design Criteria
for Nuclear Power Plant Construction Permits 1/

The Atomic Energy Commission has under consideration an amendment to its regulation, 10 CFR Part 50, "Licensing of Production and Utilization Facilities," which would add an Appendix A, "General Design Criteria for Nuclear Power Plant Construction Permits." The purpose of the proposed amendment would be to provide guidance to applicants in developing the principal design criteria to be included in applications for Commission construction permits. These General Design Criteria would not add any new requirements, but are intended to describe more clearly present Commission requirements to assist applicants in preparing applications.

The proposed amendment would complement other proposed amendments to Part 50 which were published for public comment in the FEDERAL REGISTER on August 16, 1966 (31 F.R. 10891).

1/ Inasmuch as the Commission has under consideration other amendments to 10 CFR Part 50 (31 F.R. 10891), the amendment proposed herein would be a further revision to Part 50 previously published for comment in the FEDERAL REGISTER.

The proposed amendments to Part 50 reflect a recommendation made by a seven-member Regulatory Review Panel, appointed by the Commission to study: (1) the programs and procedures for the licensing and regulation of reactors and (2) the decision-making process in the Commission's regulatory program. The Panel's report recommended the development, particularly at the construction permit stage of a licensing proceeding, of design criteria for nuclear power plants. Work on the development of such criteria had been in process at the time of the Panel's study.

As a result, preliminary proposed criteria for the design of nuclear power plants were discussed with the Commission's Advisory Committee on Reactor Safeguards and were informally distributed for public comment in Commission Press Release H-252 dated November 22, 1965. In developing the proposed criteria set forth in the proposed amendments to Part 50, the Commission has taken into consideration comments and suggestions from divisions within the Commission, from the Advisory Committee on Reactor Safeguards, from members of industry, and from the public.

Section 50.34, paragraph (b), as published for comment in the FEDERAL REGISTER on August 16, 1966, would require that each application for a construction permit include a preliminary safety analysis report. The minimum information to be included in this preliminary safety analysis report is (1) a description and safety assessment of the site, (2) a summary description of the facility, (3) a preliminary design of the facility, (4) a preliminary safety analysis and evaluation of the facility, (5) an identification of subjects expected to be technical specifications, and (6) a preliminary plan for the organization, training, and operation. The following information is specified for inclusion as part of the preliminary design of the facility:

" (i) The principal design criteria for the facility;

(ii) The design bases and the relation of the design bases to the principal design criteria;

(iii) Information relative to materials of construction, general arrangement and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety;"

The "General Design Criteria for Nuclear Power Plant Construction Permits" proposed to be included as Appendix A to this part are intended to aid the applicant in development item (i) above, the principal design criteria. All criteria established by an applicant and accepted by the Commission would be incorporated by reference in the construction permit. In considering the issuance of an operating license under the regulations, the Commission would assure that the criteria had been met in the detailed design and construction of the facility or that changes in such criteria have been justified.

Section 50.34 as published in the FEDERAL REGISTER on August 16, 1966, would be further amended by adding to Part 50 a new Appendix A containing the General Design Criteria applicable to the construction of nuclear power plants and by a specific reference to this Appendix in §50.34, paragraph (b).

The Commission expects that the provisions of the proposed amendments relating to General Design Criteria for Nuclear Power Plant Construction

Permits will be useful as interim guidance until such time as the Commission takes further action on them.

Pursuant to the Atomic Energy Act of 1954, as amended, and the Administrative Procedure Act of 1946, as amended, notice is hereby given that adoption of the following amendments to 10 CFR Part 50 is contemplated. All interested persons who desire to submit written comments or suggestions in connection with the proposed amendments should send them to the Secretary, United States Atomic Energy Commission, Washington, D.C. 20545, within 60 days after publication of this notice in the FEDERAL REGISTER. Comments received after that period will be considered if it is practicable to do so, but assurance of consideration cannot be given except as to comments filed within the period specified. Copies of comments may be examined in the Commission's Public Document Room at 1717 H Street, N.W., Washington, D.C.

1. §50.34(b)(3)(i) of 10 CFR Part 50 is amended to read as follows:

§50.34 Contents of applications; technical information safety analysis report.2/

* * * * *

(b) Each application for a construction permit shall include a preliminary safety analysis report. The report shall cover all pertinent

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2/ Inasmuch as the Commission has under consideration other amendments to §50.34 (31 F.R. 10891), the amendment proposed herein would be a further revision of §50.34(b)(3)(i) previously published for comment in the FEDERAL REGISTER. /Additions are underscored./

subjects specified in paragraph (a) of this section as fully as available information permits. The minimum information to be included shall consist of the following:

* * * * *

(3) The preliminary design of the facility, including:

(i) The principal design criteria for the facility. Appendix A, "General Design Criteria for Nuclear Power Plant Construction Permits," provides guidance for establishing the principal design criteria for nuclear power plants.

2. A new Appendix A is added to read as follows:

(See Attachment)

(Sec. 161, 68 Stat. 948; 42 U.S.C. 2201)

Dated at ______________________________ this ______________ day of ______________ 1967.

For the Atomic Energy Commission.

____
W. B. McCool
Secretary

APPENDIX A

GENERAL DESIGN CRITERIA FOR

NUCLEAR POWER PLANT CONSTRUCTION PERMITS[3/]

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INTRODUCTION

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[illegible]

Every applicant for a construction permit is required by the provisions of §50.34 to include the principal design criteria for the proposed facility in the application. These General Design Criteria are intended to be used as guidance in establishing the principal design criteria for a nuclear power plant. The General Design Criteria reflect the predominating experience with water power reactors as designed and located to date, but their applicability is not limited to these reactors. They are considered generally applicable to all power reactors.

Under the Commission's regulations, an applicant must provide assurance that its principal design criteria encompass all those facility design features required in the interest of public health and safety. There may be some power reactor cases for which fulfillment of some of the General Design Criteria may not be necessary or appropriate. There will be other cases in which these criteria are insufficient, and additional criteria must be identified and satisfied by the design in the interest of public safety. It is expected that additional criteria will be needed particularly for unusual sites and environmental conditions, and for new and advanced types of reactors. Within this context, the General Design Criteria should be used as a reference allowing additions or deletions as an individual case may warrant. Departures from the General Design Criteria should be justified.

The criteria are designated as "General Design Criteria for Nuclear Power Plant Construction Permits" to emphasize the key role they assume at this stage of the licensing process. The criteria have been categorized as Category A or Category B. Experience has shown that more definitive information is needed at the construction permit stage for the items listed in Category A than for Category B.

I. OVERALL PLANT REQUIREMENTS

CRITERION 1 - QUALITY STANDARDS (Category A)

Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes or standards on design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety function, they shall be supplemented or modified as necessary. Quality assurance programs, test procedures, and inspection acceptance levels to be used shall be identified. A showing of sufficiency and applicability of codes, standards, quality assurance programs, test procedures, and inspection acceptance levels used is required.

CRITERION 2 - PERFORMANCE STANDARDS (Category A)

Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences shall be designed, fabricated, and erected to performance standards that will enable the facility to withstand, without loss of the capability to protect the public, the additional forces that might be imposed by natural phenomena such as earthquakes, tornadoes, flooding conditions, winds, ice, and other local site effects. The design systems into action, including the transfer to alternate power sources and the design air flow delivery capability.

VIII. FUEL AND WASTE STORAGE SYSTEMS

CRITERION 66 - PREVENTION OF FUEL STORAGE CRITICALITY (Category B)

Criticality in new and spent fuel storage shall be prevented by physical systems or processes. Such means as geometrically safe configurations shall be emphasized over procedural controls.

CRITERION 67 - FUEL AND WASTE STORAGE DECAY HEAT (Category B)

Reliable decay heat removal systems shall be designed to prevent damage to the fuel in storage facilities that could result in radioactivity release to plant operating areas or the public environs.

CRITERION 68 - FUEL AND WASTE STORAGE RADIATION SHIELDING (Category B)

Shielding for radiation protection shall be provided in the design of spent fuel and waste storage facilities as required to meet the requirements of 10 CFR 20.

CRITERION 69 - PROTECTION AGAINST RADIOACTIVITY RELEASE FROM SPENT FUEL AND WASTE STORAGE (Category B)

Containment of fuel and waste storage shall be provided if accidents could lead to release of undue amounts of radioactivity to the public environs.

CONTENTION TC-2: EXHIBIT 12

Notice of proposed rulemaking for General Design Criteria, 32 Fed. Reg. 10,213 (July 11, 1967)

52.

Published 32 Federal Register 10213, July 11, 1967

FEDERAL REGISTER will be considered before action is taken on the proposed amendment. No hearing is contemplated at this time, but arrangements for informal conferences with Federal Aviation Administration officials may be made by contacting the Chief, Air Traffic Branch. Any data, views, or arguments presented during such conferences must also be submitted in writing in accordance with this notice in order to become part of the record for consideration. The proposal contained in this notice may be changed in the light of comments received.

The Birmingham 1,200-foot transition area described in § 71.181 (32 F.R. 2148 and 3765) would be altered as follows: "* * * thence southwest along the southeast boundary of V–209 to a 19-mile radius arc centered on the Tuscaloosa, Ala., VOR; thence clockwise along this arc to longitude 87°30'00'' W.; thence north along longitude 87°30'00'' W. to point of beginning, excluding that portion that coincide with R–2101 and the Gadsden, Ala., transition area * * *." would be deleted and "* * * thence southwest along the southeast boundary of V–209 to longitude 88°00'00'' W.; thence north along longitude 88°00'00'' W., to the north boundary of V–18; thence northeast along the north boundary of V–18 to a 19-mile radius arc centered on the Tuscaloosa, Ala., VORTAC; thence clockwise along this arc to longitude 87°30'00'' W.; thence north along longitude 87°30'00'' W. to point of beginning, excluding that portion that coincides with R–2101 and the Gadsden, Ala., transition area * * *." would be substituted therefor.

The proposed additional airspace is required for the protection of IFR operations and for radar vectoring of aircraft arriving and departing the Birmingham area.

The official docket will be available for examination by interested persons at the Southern Regional Office, Federal Aviation Administration, Room 724, 3400 Whipple Street, East Point, Ga.

This amendment is proposed under section 307(a) of the Federal Aviation Act of 1958 (49 U.S.C. 1348(a)).

Issued in East Point, Ga., on June 30, 1967.

JAMES G. ROGERS,
Director, Southern Region.

[F.R. Doc. 67–7949; Filed, July 10, 1967; 8:49 a.m.]

[14 CFR Part 71]

[Airspace Docket No. 67–SO–64]

TRANSITION AREA

Proposed Designation

The Federal Aviation Administration is considering an amendment to Part 71 of the Federal Aviation Regulations that would designate the Camden, S.C., transition area.

Interested persons may submit such written data, views, or arguments as they may desire. Communications should be submitted in triplicate to the Area Manager, Atlanta Area Office, Attention: Chief, Air Traffic Branch, Federal Aviation Administration, Post Office Box 20636, Atlanta, Ga. 30320. All communications received within 30 days after publication of this notice in the FEDERAL REGISTER will be considered before action is taken on the proposed amendment. No hearing is contemplated at this time, but arrangements for informal conferences with Federal Aviation Administration officials may be made by contacting the Chief, Air Traffic Branch. Any data, views, or arguments presented during such conferences must also be submitted in writing in accordance with this notice in order to become part of the record for consideration. The proposal contained in this notice may be changed in the light of comments received.

The Camden transition area would be designated as:

That airspace extending upward from 700 feet above the surface within a 7-mile radius of Woodward Field (latitude 34°17'03'' N., longitude 80°33'53'' W.); within 2 miles each side of the 040° bearing from the Camden RBN (latitude 34°17'02'' N., longitude 80°38'42.5'' W.), extending from the 7-mile radius area to 8 miles northeast of the RBN.

The proposed transition area is required for the protection of IFR operations at Woodward Field. A prescribed instrument approach procedure to this airport utilizing the Camden (private) nondirectional radio beacon is proposed in conjunction with the designation of this transition area.

This amendment is proposed under section 307(a) of the Federal Aviation Act of 1958 (49 U.S.C. 1348(a)).

Issued in East Point, Ga., on June 21, 1967.

GORDON A. WILLIAMS, Jr.,
Acting Director, Southern Region.

[F.R. Doc. 67–7950; Filed, July 10, 1967; 8:49 a.m.]

[14 CFR Part 71]

[Airspace Docket No. 67–EA–1]

FEDERAL AIRWAYS

Supplemental Proposed Alteration

On March 1, 1967, a notice of proposed rule making was published in the FEDERAL REGISTER (32 F.R. 3402) stating that the Federal Aviation Agency was considering amendments to Part 71 of the Federal Aviation Regulations that would realign V–1 from Cape Charles, Va., via the INT of Cape Charles 013° and Salisbury, Md., 206° True radials; to Salisbury; that would designate a segment of V–139 from Norfolk, Va., via Cape Charles; to Snow Hill, Md., including a west alternate from Norfolk to Snow Hill via INT of Norfolk 360° and Snow Hill 226° True radials; and that would revoke the segment of V–194 from Norfolk to INT of Norfolk 001° and Cape Charles 313° True radials. Floors of 1,200 feet above the surface were proposed for these airway segments. These actions were proposed to simplify air traffic control procedures and flight planning in the Norfolk area.

Subsequent to publication of the notice, it was determined that the Snow Hill 226° True radial would not support a Federal airway. Accordingly, the proposals published in the notice are hereby cancelled and in lieu thereof, consideration is given to the following airway alignments that would serve the same purpose.

1. Redesignate the segment of V–194 from Norfolk via the intersection of Norfolk 001° T (008° Mag.) and Harcum, Va., 072° T (079° Mag.) radials; to the intersection of Harcum 072° and Snow Hill 211° True radials.

2. Realign V–1 from Cape Charles via the intersection of Cape Charles 009° T (016° Mag.) and Salisbury 206° T (214° Mag.) radials; to Salisbury.

Interested persons may participate in the proposed rule making by submitting such written data, views, or arguments as they may desire. Communications should identify the airspace docket number and be submitted in triplicate to the Director, Eastern Region, Attention: Chief, Air Traffic Division, Federal Aviation Administration, Federal Building, John F. Kennedy International Airport, Jamaica, N.Y. 11430. All communications received within 45 days after publication of this notice in the FEDERAL REGISTER will be considered before action is taken on the proposed amendment. The proposal contained in this notice may be changed in the light of comments received.

An official docket will be available for examination by interested persons at the Federal Aviation Administration, Office of the General Counsel, Attention: Rules Docket, 800 Independence Avenue SW., Washington, D.C. 20590. An informal docket will be available for examination at the office of the Regional Air Traffic Division Chief.

These amendments are proposed under the authority of section 307(a) of the Federal Aviation Act of 1958 (49 U.S.C. 1348).

Issued in Washington, D.C., on July 3, 1967.

T. McCORMACK,
Acting Chief, Airspace and Air Traffic Rules Division.

[F.R. Doc. 67–7951; Filed, July 10, 1967; 8:49 a.m.]

ATOMIC ENERGY COMMISSION

[10 CFR Part 50]

LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

General Design Criteria for Nuclear Power Plant Construction Permits

The Atomic Energy Comission has under consideration an amendment to its regulation, 10 CFR Part 50, "Licensing of Production and Utilization Facilities," which would add an Appendix A, "General Design Criteria for Nuclear Power

A11

Plant Construction Permits."[1] The purpose of the proposed amendment would be to provide guidance to applicants in developing the principal design criteria to be included in applications for Commission construction permits. These General Design Criteria would not add any new requirements, but are intended to describe more clearly present Commission requirements to assist applicants in preparing applications.

The proposed amendment would complement other proposed amendments to Part 50 which were published for public comment in the FEDERAL REGISTER on August 16, 1966 (31 F.R. 10891).

The proposed amendments to Part 50 reflect a recommendation made by a seven-member Regulatory Review Panel, appointed by the Commission to study: (1) The programs and procedures for the licensing and regulation of reactors and (2) the decision-making process in the Commission's regulatory program. The Panel's report recommended the development, particularly at the construction permit stage of a licensing proceeding, of design criteria for nuclear power plants. Work on the development of such criteria had been in process at the time of the Panel's study.

As a result, preliminary proposed criteria for the design of nuclear power plants were discussed with the Commission's Advisory Committee on Reactor Safeguards and were informally distributed for public comment in Commission Press Release H-252 dated November 22, 1965. In developing the proposed criteria set forth in the proposed amendments to Part 50, the Commission has taken into consideration comments and suggestions from the Advisory Committee on Reactor Safeguards, from members of industry, and from the public.

Section 50.34, paragraph (b), as published for comment in the FEDERAL REGISTER on August 16, 1966, would require that each application for a construction permit include a preliminary safety analysis report. The minimum information to be included in this preliminary safety analysis report is (1) a description and safety assessment of the site, (2) a summary description of the facility, (3) a preliminary design of the facility, (4) a preliminary safety analysis and evaluation of the facility, (5) an identification of subjects expected to be technical specifications, and (6) a preliminary plan for the organization, training, and operation. The following information is specified for inclusion as part of the preliminary design of the facility:

(i) The principal design criteria for the facility;

(ii) The design bases and the relation of the design bases to the principal design criteria;

(iii) Information relative to materials of construction, general arrangement and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety;

The "General Design Criteria for Nuclear Power Plant Construction Permits" proposed to be included as Appendix A to this part are intended to aid the applicant in development item (i) above, the principal design criteria. All criteria established by an applicant and accepted by the Commission would be incorporated by reference in the construction permit. In considering the issuance of an operating license under the regulations, the Commission would assure that the criteria had been met in the detailed design and construction of the facility or that changes in such criteria have been justified.

Section 50.34 as published in the FEDERAL REGISTER on August 16, 1966, would be further amended by adding to Part 50 a new Appendix A containing the General Design Criteria applicable to the construction of nuclear power plants and by a specific reference to this Appendix in § 50.34, paragraph (b).

The Commission expects that the provisions of the proposed amendments relating to General Design Criteria for Nuclear Power Plant Construction Permits will be useful as interim guidance until such time as the Commission takes further action on them.

Pursuant to the Atomic Energy Act of 1954, as amended, and the Administrative Procedure Act of 1946, as amended, notice is hereby given that adoption of the following amendments to 10 CFR Part 50 is contemplated. All interested persons who desire to submit written comments or suggestions in connection with the proposed amendments should send them to the Secretary, U.S. Atomic Energy Commission, Washington, D.C. 20545, within 60 days after publication of this notice in the FEDERAL REGISTER. Comments received after that period will be considered if it is practicable to do so, but assurance of consideration cannot be given except as to comments filed within the period specified. Copies of comments may be examined in the Commission's Public Document Room at 1717 H Street NW., Washington, D.C.

1. Section 50.34(b)(3)(i) of 10 CFR Part 50 is amended to read as follows:

§ 50.34 Contents of applications; technical information safety analysis report.[2]

* * * * *

(b) Each application for a construction permit shall include a preliminary safety analysis report. The report shall cover all pertinent subjects specified in paragraph (a) of this section as fully as available information permits. The minimum information to be included shall consist of the following:

* * * * *

(3) The preliminary design of the facility, including:

(i) The principal design criteria for the facility. Appendix A, "General Design Criteria for Nuclear Power Plant Construction Permits," provides guidance for establishing the principal design criteria for nuclear power plants.

* * * * *

2. A new Appendix A is added to read as follows:

APPENDIX A—GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANT CONSTRUCTION PERMITS[3]

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[1] Inasmuch as the Commission has under consideration other amendments to 10 CFR Part 50 (31 F.R. 10891), the amendment proposed herein would be a further revision to Part 50 previously published for comment in the FEDERAL REGISTER.

[2] Inasmuch as the Commission has under consideration other amendments to § 50.34 (31 F.R. 10891), the amendment proposed herein would be a further revision of § 50.34 (b)(3)(i) previously published for comment in the FEDERAL REGISTER.

* Inasmuch as the Commission has under consideration other amendments to 10 CFR Part 50 (31 F.R. 10891), the amendment proposed herein would be a further revision to Part 50 previously published for comment in the FEDERAL REGISTER.

Introduction. Every applicant for a construction permit is required by the provisions of § 50.34 to include the principal design criteria for the proposed facility in the application. These General Design Criteria are intended to be used as guidance in establishing the principal design criteria for a nuclear power plant. The General Design Criteria reflect the predominating experience with water power reactors as designed and located to date, but their applicability is not limited to these reactors. They are considered generally applicable to all power reactors.

Under the Commission's regulations, an applicant must provide assurance that its principal design criteria encompass all those facility design features required in the interest of public health and safety. There may be some power reactor cases for which fulfillment of some of the General Design Criteria may not be necessary or appropriate. There will be other cases in which these criteria are insufficient, and additional criteria must be identified and satisfied by the design in the interest of public safety. It is expected that additional criteria will be needed particularly for unusual sites and environmental conditions, and for new and advanced types of reactors. Within this context, the General Design Criteria should be used as a reference allowing additions or deletions as an individual case may warrant. Departures from the General Design Criteria should be justified.

The criteria are designated as "General Design Criteria for Nuclear Power Plant Construction Permits" to emphasize the key role they assume at this stage of the licensing process. The criteria have been categorized as Category A or Category B. Experience has shown that more definitive information is needed at the construction permit stage for the items listed in Category A than for those in Category B.

I. Overall Plant Requirements

Criterion 1—Quality Standards (Category A). Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes or standards on design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety function, they shall be supplemented or modified as necessary. Quality assurance programs, test procedures, and inspection acceptance levels to be used shall be identified. A showing of sufficiency and applicability of codes, standards, quality assurance programs, test procedures, and inspection acceptance levels used is required.

Criterion 2—Performance Standards (Category A). Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences shall be designed, fabricated, and erected to performance standards that will enable the facility to withstand, without loss of the capability to protect the public, the additional forces that might be imposed by natural phenomena such as earthquakes, tornadoes, flooding conditions, winds, ice, and other local site effects. The design bases so established shall reflect: (a) Appropriate consideration of the most severe of these natural phenomena that have been recorded for the site and the surrounding area and (b) an appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data and their suitability as a basis for design.

Criterion 3—Fire Protection (Category A). The reactor facility shall be designed (1) to minimize the probability of events such as fires and explosions and (2) to minimize the potential effects of such events to safety. Noncombustible and fire resistant materials shall be used whenever practical throughout the facility, particularly in areas containing critical portions of the facility such as containment, control room, and components of engineered safety features.

Criterion 4—Sharing of Systems (Category A). Reactor facilities shall not share systems or components unless it is shown safety is not impaired by the sharing.

Criterion 5—Records Requirements (Category A). Records of the design, fabrication, and construction of essential components of the plant shall be maintained by the reactor operator or under its control throughout the life of the reactor.

II. Protection by Multiple Fission Product Barriers

Criterion 6—Reactor Core Design (Category A). The reactor core shall be designed to function throughout its design lifetime, without exceeding acceptable fuel damage limits which have been stipulated and justified. The core design, together with reliable process and decay heat removal systems, shall provide for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and for transient situations which can be anticipated, including the effects of the loss of power to recirculation pumps, tripping out of a turbine generator set, isolation of the reactor from its primary heat sink, and loss of all offsite power.

Criterion 7—Supression of Power Oscillations (Category B). The core design, together with reliable controls, shall ensure that power oscillations which could cause damage in excess of acceptable fuel damage limits are not possible or can be readily suppressed.

Criterion 8—Overall Power Coefficient (Category B). The reactor shall be designed so that the overall power coefficient in the power operating range shall not be positive.

Criterion 9—Reactor Coolant Pressure Boundary (Category A). The reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime.

Criterion 10—Containment (Category A). Containment shall be provided. The containment structure shall be designed to sustain the initial effects of gross equipment failures, such as a large coolant boundary break, without loss of required integrity and, together with other engineered safety features as may be necessary, to retain for as long as the situation requires the functional capability to protect the public.

III. Nuclear and Radiation Controls

Criterion 11—Control Room (Category B). The facility shall be provided with a control room from which actions to maintain safe operational status of the plant can be controlled. Adequate radiation protection shall be provided to permit access, even under accident conditions, to equipment in the control room or other areas as necessary to shut down and maintain safe control of the facility without radiation exposures of personnel in excess of 10 CFR 20 limits. It shall be possible to shut the reactor down and maintain it in a safe condition if access to the control room is lost due to fire or other cause.

Criterion 12—Instrumentation and Control Systems (Category B). Instrumentation and controls shall be provided as required to monitor and maintain variables within prescribed operating ranges.

Criterion 13—Fission Process Monitors and Controls (Category B). Means shall be provided for monitoring and maintaining control over the fission process throughout core life and for all conditions that can reasonably be anticipated to cause variations in reactivity of the core, such as indication of position of control rods and concentration of soluble reactivity control poisons.

Criterion 14—Core Protection Systems (Category B). Core protection systems, together with associated equipment, shall be designed to act automatically to prevent or to suppress conditions that could result in exceeding acceptable fuel damage limits.

Criterion 15—Engineered Safety Features Protection Systems (Category B). Protection systems shall be provided for sensing accident situations and initiating the operation of necessary engineered safety features.

Criterion 16—Monitoring Reactor Coolant Pressure Boundary (Category B). Means shall be provided for monitoring the reactor coolant pressure boundary to detect leakage.

Criterion 17—Monitoring Radioactivity Releases (Category B). Means shall be provided for monitoring the containment atmosphere, the facility effluent discharge paths, and the facility environs for radioactivity that could be released from normal operations, from anticipated transients, and from accident conditions.

Criterion 18—Monitoring Fuel and Waste Storage (Category B). Monitoring and alarm instrumentation shall be provided for fuel and waste storage and handling areas for conditions that might contribute to loss of continuity in decay heat removal and to radiation exposures.

IV. Reliability and Testability of Protection Systems

Criterion 19—Protection Systems Reliability (Catergory B). Protection systems shall be designed for high functional reliability and in-service testability commensurate with the safety functions to be performed.

Criterion 20—Protection Systems Redundancy and Independence (Category B). Redundancy and independence designed into protection systems shall be sufficient to assure that no single failure or removal from service of any component or channel of a system will result in loss of the protection function. The redundancy provided shall include, as a minimum, two channels of protection for each protection function to be served. Different principles shall be used where necessary to achieve true independence of redundant instrumentation components.

Criterion 21—Single Failure Definition (Category B). Multiple failures resulting from a single event shall be treated as a single failure.

Criterion 22—Separation of Protection and Control Instrumentation Systems (Category B). Protection systems shall be separated from control instrumentation systems to the extent that failure or removal from service of any control instrumentation system component or channel, or of those common to control instrumentation and protection circuitry, leaves intact a system satisfying all requirements for the protection channels.

Criterion 23—Protection Against Multiple Disability for Protection Systems (Category B). The effects of adverse conditions to which redundant channels or protection systems might be exposed in common, either under normal conditions or those of an accident, shall not result in loss of the protection function.

Criterion 24—Emergency Power for Protection Systems (Category B). In the event of loss of all offsite power, sufficient alternate sources of power shall be provided to permit the required functioning of the protection systems.

Criterion 25—Demonstration of Functional Operability of Protection Systems (Category B). Means shall be included for testing protection systems while the reactor is in operation to demonstrate that no failure or loss of redundancy has occurred.

Criterion 26—Protection Systems Fail-Safe Design (Category B). The protection systems shall be designed to fail into a safe state or into a state established as tolerable on a defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or adverse environments (e.g., extreme heat or cold, fire, steam, or water) are experienced.

V. Reactivity Control

Criterion 27—Redundancy of Reactivity Control (Category A). At least two independent reactivity control systems, preferably of different principles, shall be provided.

Criterion 28—Reactivity Hot Shutdown Capability (Category A). At least two of the reactivity control systems provided shall independently be capable of making and holding the core subcritical from any hot standby or hot operating condition, including those resulting from power changes, sufficiently fast to prevent exceeding acceptable fuel damage limits.

Criterion 29—Reactivity Shutdown Capability (Category A). At least one of the reactivity control systems provided shall be capable of making the core subcritical under any condition (including anticipated operational transients) sufficiently fast to prevent exceeding acceptable fuel damage limits. Shutdown margins greater than the maximum worth of the most effective control rod when fully withdrawn shall be provided.

Criterion 30—Reactivity Holddown Capability (Category B). At least one of the reactivity control systems provided shall be capable of making and holding the core subcritical under any conditions with appropriate margins for contingencies.

Criterion 31—Reactivity Control Systems Malfunction (Category B). The reactivity control systems shall be capable of sustaining any single malfunction, such as, unplanned continuous withdrawal (not ejection) of a control rod, without causing a reactivity transient which could result in exceeding acceptable fuel damage limits.

Criterion 32—Maximum Reactivity Worth of Control Rods (Category A). Limits, which include considerable margin, shall be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the reactor coolant pressure boundary or (b) disrupt the core, its support structures, or other vessel internals sufficiently to impair the effectiveness of emergency core cooling.

VI. Reactor Coolant Pressure Boundary

Criterion 33—Reactor Coolant Pressure Boundary Capability (Category A). The reactor coolant pressure boundary shall be capable of accommodating without rupture, and with only limited allowance for energy absorption through plastic deformation, the static and dynamic loads imposed on any boundary component as a result of any inadvertent and sudden release of energy to the coolant. As a design reference, this sudden release shall be taken as that which would result from a sudden reactivity insertion such as rod ejection (unless prevented by positive mechanical means), rod dropout, or cold water addition.

Criterion 34—Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention (Category A). The reactor coolant pressure boundary shall be designed to minimize the probability of rapidly propagating type failures. Consideration shall be given (a) to the notch-toughness properties of materials extending to the upper shelf of the Charpy transition curve, (b) to the state of stress of materials under static and transient loadings, (c) to the quality control specified for materials and component fabrication to limit flaw sizes, and (d) to the provisions for control over service temperature and irradiation effects which may require operational restrictions.

Criterion 35—Reactor Coolant Pressure Boundary Brittle Fracture Prevention (Category A). Under conditions where reactor coolant pressure boundary system components constructed of ferritic materials may be subjected to potential loadings, such as a reactivity-induced loading, service temperatures shall be at least 120° F. above the nil ductility transition (NDT) temperature of the component material if the resulting energy release is expected to be absorbed by plastic deformation or 60° F. above the NDT temperature of the component material if the resulting energy release is expected to be absorbed within the elastic strain energy range.

Criterion 36—Reactor Coolant Pressure Boundary Surveillance (Category A). Reactor coolant pressure boundary components shall have provisions for inspection, testing, and surveillance by appropriate means to assess the structural and leaktight integrity of the boundary components during their service lifetime. For the reactor vessel, a material surveillance program conforming with ASTM-E-185-66 shall be provided.

VII. Engineered Safety Features

Criterion 37—Engineered Safety Features Basis for Design (Category A). Engineered safety features shall be provided in the facility to back up the safety provided by the core design, the reactor coolant pressure boundary, and their protection systems. As a minimum, such engineered safety features shall be designed to cope with any size reactor coolant pressure boundary break up to and including the circumferential rupture of any pipe in that boundary assuming unobstructed discharge from both ends.

Criterion 38—Reliability and Testability of Engineered Safety Features (Category A). All engineered safety features shall be designed to provide high functional reliability and ready testability. In determining the suitability of a facility for a proposed site, the degree of reliance upon and acceptance of the inherent and engineered safety afforded by the systems, including engineered safety features, will be influenced by the known and the demonstrated performance capability and reliability of the systems, and by the extent to which the operability of such systems can be tested and inspected where appropriate during the life of the plant.

Criterion 39—Emergency Power for Engineered Safety Features (Category A). Alternate power systems shall be provided and designed with adequate independency, redundancy, capacity, and testability to permit the functioning required of the engineered safety features. As a minimum, the onsite power system and the offsite power system shall each, independently, provide this capacity assuming a failure of a single active component in each power system.

Criterion 40—Missile Protection (Category A). Protection for engineered safety features shall be provided against dynamic effects and missiles that might result from plant equipment failures.

Criterion 41—Engineered Safety Features Performance Capability (Category A). Engineered safety features such as emergency core cooling and containment heat removal systems shall provide sufficient performance capability to accommodate partial loss of installed capacity and still fulfill the required safety function. As a minimum, each engineered safety feature shall provide this required safety function assuming a failure of a single active component.

Criterion 42—Engineered Safety Features Components Capability (Category A). Engineered safety features shall be designed so that the capability of each component and system to perform its required function is not impaired by the effects of a loss-of-coolant accident.

Criterion 43—Accident Aggravation Prevention (Category A). Engineered safety features shall be designed so that any action of the engineered safety features which might accentuate the adverse after-effects of the loss of normal cooling is avoided.

Criterion 44—Emergency Core Cooling Systems Capability (Category A). At least two emergency core cooling systems, preferably of different design principles, each with a capability for accomplishing abundant emergency core cooling, shall be provided. Each emergency core cooling system and the core shall be designed to prevent fuel and clad damage that would interfere with the emergency core cooling function and to limit the clad metal-water reaction to negligible amounts for all sizes of breaks in the reactor coolant pressure boundary, including the double-ended rupture of the largest pipe. The performance of each emergency core cooling system shall be evaluated conservatively in each area of uncertainty. The systems shall not share active components and shall not share other features or components unless it can be demonstrated that (a) the capability of the shared feature or component to perform its required function can be readily ascertained during reactor operation, (b) failure of the shared feature or component does not initiate a loss-of-coolant accident, and (c) capability of the shared feature or component to perform its required function is not impaired by the effects of a loss-of-coolant accident and is not lost during the entire period this function is required following the accident.

Criterion 45—Inspection of Emergency Core Cooling Systems (Category A). Design provisions shall be made to facilitate physical inspection of all critical parts of the emergency core cooling systems, including reactor vessel internals and water injection nozzles.

Criterion 46—Testing of Emergency Core Cooling Systems Components (Category A). Design provisions shall be made so that active components of the emergency core cooling systems, such as pumps and valves, can be tested periodically for operability and required functional performance.

Criterion 47—Testing of Emergency Core Cooling Systems (Category A). A capability shall be provided to test periodically the delivery capability of the emergency core cooling systems at a location as close to the core as is practical.

Criterion 48—Testing of Operational Sequence of Emergency Core Cooling Systems (Category A). A capability shall be provided to test under conditions as close to design as practical the full operational sequence that would bring the emergency core cooling systems into action, including the transfer to alternate power sources.

Criterion 49—Containment Design Basis (Category A). The containment structure, including access openings and penetrations, and any necessary containment heat removal systems shall be designed so that the containment structure can accommodate without exceeding the design leakage rate the pressures and temperatures resulting from the largest credible energy release following a loss-of-coolant accident, including a considerable margin for effects from metal-water or other chemical reactions that could occur as a consequence of failure of emergency core cooling systems.

Criterion 50—NDT Requirement for Containment Material (Category A). Principal load carrying components of ferritic materials exposed to the external environment shall be selected so that their temperatures under normal operating and testing conditions are not less than 30° F. above nil ductility transition (NDT) temperature.

Criterion 51—Reactor Coolant Pressure Boundary Outside Containment (Category A). If part of the reactor coolant pressure boundary is outside the containment, appropriate features as necessary shall be provided to protect the health and safety of the public in case of an accidental rupture in that part. Determination of the appropriateness of features such as isolation valves and additional containment shall include consideration of the environmental and population conditions surrounding the site.

Criterion 52—Containment Heat Removal Systems (Category A). Where active heat removal systems are needed under accident conditions to prevent exceeding containment design pressure, at least two systems, preferably of different principles, each with full capacity, shall be provided.

Criterion 53—Containment Isolation Valves (Category A). Penetrations that require closure for the containment function shall be protected by redundant valving and associated apparatus.

Criterion 54—Containment Leakage Rate Testing (Category A). Containment shall be designed so that an integrated leakage rate testing can be conducted at design pressure after completion and installation of all penetrations and the leakage rate measured over a sufficient period of time to verify its conformance with required performance.

Criterion 55—Containment Periodic Leakage Rate Testing (Category A). The containment shall be designed so that integrated leakage rate testing can be done periodically at design pressure during plant lifetime.

Criterion 56—Provisions for Testing of Penetrations (Category A). Provisions shall be made for testing penetrations which have resilient seals or expansion bellows to permit leak tightness to be demonstrated at design pressure at any time.

Criterion 57—Provisions for Testing of Isolation Valves (Category A). Capability shall be provided for testing functional operability of valves and associated apparatus essential to the containment function for establishing that no failure has occurred and for determining that valve leakage does not exceed acceptable limits.

Criterion 58—Inspection of Containment Pressure-Reducing Systems (Category A). Design provisions shall be made to facilitate the periodic physical inspection of all important components of the containment pressure-reducing systems, such as, pumps, valves, spray nozzles, torus, and sumps.

Criterion 59—Testing of Containment Pressure-Reducing Systems Components (Category A). The containment pressure-reducing systems shall be designed so that active components, such as pumps and valves, can be tested periodically for operability and required functional performance.

Criterion 60—Testing of Containment Spray Systems (Category A). A capability shall be provided to test periodically the delivery capability of the containment spray system at a position as close to the spray nozzles as is practical.

Criterion 61—Testing of Operational Sequence of Containment Pressure-Reducing Systems (Category A). A capability shall be provided to test under conditions as close to the design as practical the full operational sequence that would bring the containment pressure-reducing systems into action, including the transfer to alternate power sources.

Criterion 62—Inspection of Air Cleanup Systems (Category A). Design provisions shall be made to facilitate physical inspection of all critical parts of containment air cleanup systems, such as, ducts, filters, fans, and dampers.

Criterion 63—Testing of Air Cleanup Systems Components (Category A). Design provisions shall be made so that active components of the air cleanup systems, such as fans and dampers, can be tested periodically for operability and required functional performance.

Criterion 64—Testing of Air Cleanup Systems (Category A). A capability shall be provided for in situ periodic testing and surveillance of the air cleanup systems to ensure (a) filter bypass paths have not developed and (b) filter and trapping materials have not deteriorated beyond acceptable limits.

Criterion 65—Testing of Operational Sequence of Air Cleanup Systems (Category A). A capability shall be provided to test under conditions as close to design as practical the full operational sequence that would bring the air cleanup systems into action, including the transfer to alternate power sources and the design air flow delivery capability.

VIII. Fuel and Waste Storage Systems

Criterion 66—Prevention of Fuel Storage Criticality (Category B). Criticality in new and spent fuel storage shall be prevented by physical systems or processes. Such means as geometrically safe configurations shall be emphasized over procedural controls.

Criterion 67—Fuel and Waste Storage Decay Heat (Category B). Reliable decay heat removal systems shall be designed to prevent damage to the fuel in storage facilities that could result in radioactivity release to plant operating areas or the public environs.

Criterion 68—Fuel and Waste Storage Radiation Shielding (Category B). Shielding for radiation protection shall be provided in the design of spent fuel and waste storage facilities as required to meet the requirements of 10 CFR 20.

Criterion 69—Protection Against Radioactivity Release From Spent Fuel and Waste Storage (Category B). Containment of fuel and waste storage shall be provided if accidents could lead to release of undue amounts of radioactivity to the public environs.

IX. PLANT EFFLUENTS

Criterion 70—Control of Releases of Radioactivity to the Environment (Category B). The facility design shall include those means necessary to maintain control over the plant radioactive effluents, whether gaseous, liquid, or solid. Appropriate holdup capacity shall be provided for retention of gaseous, liquid, or solid effluents, particularly where unfavorable environmental conditions can be expected to require operational limitations upon the release of radioactive effluents to the environment. In all cases, the design for radioactivity control shall be justified (a) on the basis of 10 CFR 20 requirements for normal operations and for any transient situation that might reasonably be anticipated to occur and (b) on the basis of 10 CFR 100 dosage level guidelines for potential reactor accidents of exceedingly low probability of occurrence except that reduction of the recommended dosage levels may be required where high population densities or very large cities can be affected by the radioactive effluents.

(Sec. 161, 68 Stat. 948; 42 U.S.C. 2201)

Dated at Washington, D.C., this 28th day of June 1967.

For the Atomic Energy Commission.

W. B. McCOOL,
Secretary.

[F.R. Doc. 67-7901; Filed, July 10, 1967; 8:45 a.m.]

CONTENTION TC-2: EXHIBIT 13

Letter from William B. Cottrell, ORNL, to H. L. Price, AEC (September 6, 1967) and enclosed ORNL comments on proposed GDC.

DOCKET NUMBER PR 50
PROPOSED RULE

OAK RIDGE NATIONAL LABORATORY

OPERATED BY
UNION CARBIDE CORPORATION
NUCLEAR DIVISION

UNION CARBIDE

POST OFFICE BOX Y
OAK RIDGE, TENNESSEE 37830

September 6, 1967

Mr. H. L. Price
Director of Regulation
U.S. Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Price:

Subject: Review of USAEC "General Design Criteria for Nuclear Power Plant Construction Permits" Federal Register, July 11, 1967

The subject document has been reviewed by members of the staff of the Nuclear Safety Information Center. We realize and appreciate the great amount of work that your staff has done in bringing these criteria to their present form. We participated in the initial review of the criteria when they were issued in November 1965 and we are pleased to have the opportunity to review this later version. Our comments are enclosed in two parts: (1) general comments which apply to the entire set of criteria and (2) specific comments on the individual criteria and in a few cases on sections such as VII, Engineered Safety Features.

With a few exceptions, the scope of the criteria seems broad enough and generally well organized. We do have rather extensive comments on those criteria which deal with protection systems. A difficult problem is that of assessing reliability. The "single failure criterion" is an attempt to relieve this situation, but its application is subjective and it has different meanings to different individuals. Another problem area is that of the use of the same instruments for both operating the plant and providing protection. We believe that such interdependence can only degrade the reliability and performance of the protection system. Problems such as these make the task of writing criteria and standards quite difficult.

Further, the absence of clear definitions of terms, which to many are rather loosely understood, could limit the effectiveness of the criteria. We feel that there is a critical need for these definitions.

We again wish to commend you for the significant contribution represented by these criteria. If you have questions concerning our comments, we will be glad to discuss them with you.

Sincerely yours,

Wm. B. Cottrell

Wm. B. Cottrell, Director
Nuclear Safety Information Center

WBC:JRB:jt

Enclosure

cc A. J. Pressesky

General Comments

1. The ramifications of civil disobedience, riots, strikes, sabotage, and the like have not even been mentioned. With this vast potential risk in mind, should not the physical security of the plant be considered?

2. Since these criteria will be used by many groups whose terminology is not always (or even usually) in agreement, a set of definitions is badly needed. For example - what is a system, component, engineered safety feature, failure, redundancy, channel, surveillance, monitoring, malfunction, protection system, loss of coolant accident, etc.?

3. Since "single failure criteria" are to be applied to systems other than those for control (for which criterion 21 is the definition), it is extremely important that they be clearly defined for all systems.

4. Since the introduction uses the phrase "nuclear reactor plant" why is the phrase "reactor facility" used in the text of several of the criteria to mean the same thing?

Specific Comments

Title - General Design Criteria for Nuclear Power Plant Construction Permits

The title is really not grammatically correct, since it infers that we are designing a "construction permit".

Criterion 2 - Performance Standards

1. Line 7: Delete "performance" since this could be construed as applying to operating performance only.

2. In regard to earthquakes the "appropriate margin for withstanding forces greater than those recorded . . ." has not been defined here and furthermore it would be extremely difficult to do so at least with our present understanding of earthquake phenomena. Therefore, the criterion should state what constitutes an adequate margin.

Criterion 4 - Sharing of Systems

We agree with criterion 4 as it applies to the nuclear reactor plant but it should be extended to apply to systems, sub-systems, and especially engineered safety features.

Criterion 5 - Records Requirements

1. Line 2: Should read, "Records of the design, fabrication, inspection, testing and construction of . . ." to be sufficiently inclusive. The performance of engineered safety features must be determined as a datum for evaluation of subsequent tests required of the system. For example, criterion 46 states that active components be periodically tested for required performance.

2. Line 5: Change "its" to "his" to refer to the operator's control.

Criterion 8 - Overall Power Coefficient

For this entire criterion it might be better to say that "the reactor shall be designed so that either the overall power coefficient in the power operating range shall not be positive or reliable controls which will eliminate or minimize the undesirable effects of a positive power coefficient shall be provided, tested and proved effective."

Criterion 10 - Containment

We infer from subsequent criteria that the protection system is not considered an engineered safety feature even though there are reactors that depend upon the protection systems to work in order not to overstress the containment. Thus, either "engineered safety features" should be defined to include the reactor protective system, i.e., scram functions, or this and other functions should be specifically mentioned. We prefer the former alternative.

Criterion 11 - Control Room

The aims of this criterion are certainly desirable but it is difficult if not impossible to prove the criterion has been met. However, some clarification is needed, for example, if a fire in a panel renders the controls of some emergency system inoperable, the criterion can be interpreted to mean that two separate control rooms are required. Is this the intent?

Criterion 13 - Fission Process Monitors and Controls

1. Line 4: Delete "throughout core life and" since it is redundant.

2. The examples cited should either be deleted or augmented by a more comprehensive set including flux, hot spots, etc.

Criteria 14 and 15 - Core Protection Systems and Engineered Safety Features

These criteria exemplify the fact that a more detailed definition of containment and engineered safety features needs to be included. One could define the engineered safety features as including scram system, core protection system, etc., and then eliminate Criterion 14.

Suggested Criterion - Monitoring Engineered Safety Features

We suggest that this criterion be inserted at this point: Instrumentation shall be provided to monitor the performance of engineered safety features during the course of the accident and to monitor the condition of the reactor itself under these conditions.

Criterion 16 - Monitoring Reactor Coolant Pressure Boundary

This criterion defines the monitoring that is necessary to prove compliance with Criterion 9. (Similar proof is required by Criterion 36) In cases of this nature cross referencing of criteria should be made for the sake of clarity.

Criterion 17 - Monitoring Radioactivity Releases

This criterion was written to specify monitoring to meet the specifications of Criterion 70, which should be cress referenced here.

Criterion 18 - Monitoring Fuel and Waste Storage

Specification of criticality monitoring should be included in this criterion; for example, as by reference to 10 CFR, Part 70.34.

Criterion 19 - Protection Systems Reliability

There is no guide for determining whether or not the functional reliability and in-service testability is commensurate with the safety functions to be performed. Every designer could claim that his system met this criterion, and challenge a reviewer to show otherwise. Arguments about this criterion most likely will include comparisons to somewhat similar protection systems for somewhat similar nuclear power plants that have been reviewed and approved.

This criterion is of questionable value and we recommend its omission. A set of rules for designing protection systems would be more useful than a general statement of desirable results.

Criterion 20 - Protection Systems Redundancy and Independence

The criterion is not clear as to the extent of the effects of a single failure that need consideration. Apparently, considerations of effect are to be limited to a component or channel - resulting in a severe limitation in the value of this criterion. This is another example of a criterion where definitions are needed; for example, component, channel, and system need to be defined.

Criterion 21 - Single Failure Definition

A judgment of the extent of failures caused by a single event hinges on credibility. First, there is the probability of the initiating event, then the probability of progressive failures. A single event of sufficient magnitude will certainly prevent the functioning of the protection system. Detailed guidelines for describing the required independence of redundant equipment are needed. Examples are spacing between cables carrying redundant signals, methods of separating electronic equipment handling redundant signals, methods of isolating redundant logic devices which combine redundant signals, etc. Unless more detailed information is given as to what is to be considered credible, this criterion serves little purpose.

Criterion 22 - Separation of Protection and Control Instrumentation Systems

This criterion apparently recognizes the need for separating protective and control instrumentation but compromises this objective with the qualifications permitted. The net effect is to permit the intimate intermingling of the system that normally operates the plant and the system that is intended to afford protection. We strongly recommend that no exceptions be permitted to the separation of these two systems as the only effective means to insure the vital integrity of the protection system.

Both of these systems in the new and larger reactors are complex. Despite the use of buffer amplifiers in attempting to isolate the effects of failures in the two systems, the systems are not independent when the same signals are coupled into each. Additionally, the objectives of operation are not those of protection. When the two systems are intermingled, signal processing equipment is invariably designed for operating the plant rather than for protection. Inadequate control demands that corrections must be made in the equipment to allow operation, but inadequate protection equipment may be discovered only after their need during an accident. Mixing of the two systems as allowed by this criterion diverts design attention from the requirements of protection to those of operation. Such mixing also increases the probability that protection will be lost as the result of a failure in the control system that initiates the accident requiring protection.

The basic justification for independence of protection and operation systems, in our opinion, is the relative ease with which the protection function can be assured with independence, and the great difficulty of realizing such assurance with interdependence. We believe it is easier to separate the systems than to assure that their interactions are harmless. We believe it is easier to maintain independence than to insure, for the lifetime of the plant, that deliberate changes or inadvertent alteration of the operation system will not adversely affect the protection function.

The dismal list of accidents caused by design errors, and the much larger list of design errors caught before they caused accidents, lead us to believe that design errors will continue to occur. We believe further that independence of operation and protection is one of the best defenses against the possibility that a design error may cause an unprotected accident.

It may be possible that for some combinations of protection and operation instruments no conceivable failure of the operation function involved can result in a situation requiring action of the protection function involved. To the extent that this can be proved, both initially and throughout reactor lifetime, the particular interdependence could be acceptable. A hypothetical example is the instrumentation used to measure and control the pressure of a sealed containment enclosure. The operation function is used principally to provide a pressure differential between the inside of the containment and the outside, and thus to provide a means for surveillance of the leakage rate.

The protection function might be to initiate reactor shutdown, emergency cooling, and isolation of process piping if a rise in containment pressure should indicate the presence of a serious leak of potentially radioactive fluids. It might be demonstrable that no failure whatever of this instrumentation could induce a substantial leak of radioactive fluid, in which case no real interdependence of operation system and protection system would in fact exist.

The basis of the above example is the impossibility that failure of the operational function or equipment could ever, under any circumstances, lead to a situation where the protection function would be needed. Therefore, sharing of equipment (common elements) between the protection system and the operation system could not lead to interaction between the two systems. It is difficult to prove conclusively this lack of functional interaction. More difficult is the problem of ensuring that this lack of interaction can and will be maintained throughout the life of the plant. Operators are not designers; operators in charge of the plant at the end of its 40-year life are not the ones who may have discussed protection problems with the designers at the beginning. Subtle considerations are apt to be forgotten or ignored. It is easy to forget that plant protection was originally based on the impossibility that failure of certain operation instruments could result in a need for protection-system function.

Criterion 24 - Emergency Power for Protection Systems

Design requirements related to power supply include consideration of both Criteria 24 and 26. There is an anomaly here in that Criterion 24 permits the protection system to require power to provide protection, whereas Criterion 26 requires the system to fall into a safe or tolerable state on loss of power. To the extent that Criterion 26 can be met, alternate power sources become an economic or operational consideration rather than being needed for safety.

Criterion 25 - Demonstration of Functional Operability of Protection Systems

We agree with the intent of this criterion but suggest that the wording be changed to state ". . . demonstrate that no failure causing a reduction of redundancy . . ." rather than ". . . demonstrate that no failure or loss of redundancy . . .". Some systems may have extra elements whose failures do not reduce the redundancy claimed for the system.

Criterion 26 - Protection Systems Fail-Safe Design

This criterion places a requirement not only on the protection system but on the plant as well. For example, a plant design could be such that operation of the protection mechanism when not needed would be highly undesirable. (An illustration is the closure of the steam stop valves in a

BWR.) Criterion 26 requires the plant to be able to accept operation of the protection system when not needed. We believe this is a good objective and we support this criterion.

Section V - Reactivity Control

1. The title of this section should be "Reactivity Control for Reactor Shutdown".

2. This group of criteria should distinquish more clearly between functions of reactivity control; namely, the dynamic reactivity reduction process and the static holddown functions. The first function must be performed at such times as in power transients and loss-of-coolant accidents with the objective of preventing exceeding "acceptable fuel damage limits" referred to in Criteria 28 and 29. Margins expressed in terms of shutdown parameters are inappropriate and inadequate for the dynamic function.

   The reliability with which each function must be carried out depends upon the seriousness of the consequences of failure of that function.

Criterion 27 - Redundancy of Reactivity Control

This criterion is not clear. It does not state whether the two reactivity control systems (1) should both be capable of both increasing and decreasing reactivity for operation, or (2) should both be capable of fast shutdown, or (3) should one be for fast shutdown and one for holddown. We recommend that the word "shutdown" be substituted for "control" in this criterion. These systems should also meet the requirements of Criteria 28, 29, 30, 31, and 32.

Criteria 28, 29, and 30 taken together indicate that one of the shutdown systems is not required to cope with positive transients and is essentially a method of obtaining reactivity holddown capability. However, reactors that must be shut down rapidly to allow the containment system to function need two separate and fast shutdown systems. A single fast or "primary" shutdown system together with a "holddown", or slow, "secondary" shutdown system is not satisfactory in this case.

Criterion 29 - Reactivity Shutdown Capability

As stated in our comments on Criterion 27, some reactors require a shutdown to allow the containment to function. In such cases, this criterion should require that two shutdown systems be applied. Each such system should be capable of preventing an unacceptable situation.

This criterion carries a reference to shutdown margin that could well be made a separate criterion as the shutdown requirements are a function of the number of rods, reactor operating conditions and function desired (e.g., reduction of nuclear power level or holddown of the subcritical reactor). Although we have not addressed ourselves to these conditions in detail, we believe that a margin much greater than the worth of the most effective control rod is needed for reactors having many rods.

Criterion 30 - Reactivity Holddown Capability

In cases requiring the reactor to be shut down in order to achieve containment, two of these systems should be required. See comments on Criteria 27 and 29.

Criterion 31 - Reactivity Control Systems Malfunction

This criterion should be expanded to include all failures of the plant operating system that are capable of increasing reactivity. In particular this criterion should not be limited to the unplanned withdrawal of only one control rod since a failure of the control rod operating system may not be restricted to the withdrawal of only one rod. All failures that may affect the performance of the control rod operating system must be considered. Of a more general nature, all failures that can introduce reactivity increases must be considered. In addition to control rods, there are coolant temperature changes, and perhaps even void effects that need analysis.

Criterion 33 - Reactor Coolant Pressure Boundary Capability

We agree with the intent of the criterion but it is not clear what is meant by "positive mechanical means" for preventing a rod ejection. A definition is needed.

Section VII - Engineered Safety Features

With the exception of reactor shutdown systems, all other engineered safety features are discussed in this section. These are: emergency power system, emergency core cooling system, containment enclosure system, containment pressure-reducing system (including containment heat removal), and air cleaning systems.

For each of these systems, there should be criteria for design of the system and their components as well as criteria for testing and inspection.

The objective of these criteria would be clearer if each system were treated in separate subsections and the criteria for each were set up in parallel form. Thus, there would be criteria for the inspection and testing of emergency power system (now covered in only Criterion 39) as well as the inspection and testing criteria for the other engineered safety features. Criterion 52, "Containment Heat Removal Systems," would be grouped with Criteria 58-61 with which it is generally associated. Such a rearrangement raises questions on other points of apparent inconsistancy, e.g., Criterion 60 is seen to be but a special case of Criterion 61, etc.

Criterion 37 - Engineered Safety Features Basis for Design

Again a definition of engineered safety features is necessary. For example, if the scram must work in order that the containment not be overstressed, then the scram system must be considered part of an engineered safety feature.

Criterion 38 - **Reliability and Testability of Engineered Safety Features**

We agree with this criterion. However, its title and inclusion in Section VII, both of which pertain only to engineered safety features, does not reflect its more general applications which include "inherent" as well as "engineered safety features". It would more appropriately be included in Section I.

Criterion 39 - Emergency Power for Engineered Safety Features

A difficult point in the application of this criterion is that of redundancy in the offsite power system. For example, a plant failure that results in shutting off the electric generator driven by the reactor could produce the loss of all offsite power. The probability of this consequential loss of offsite power varies widely as a result of changes in the power system and of variations in power system load. As a result of this wide variation in the reliability of offsite power, we recommend that this criterion require that redundant and independent onsite power system be required such that onsite power alone be capable of supplying the needs of the engineered safety features after a failure of a single active component in the onsite power system. We do not believe that the offsite power is really independent of the power from a main generator operated from the reactor to be safeguarded.

Criterion 40 - Missile Protection

Analysis shall be made to show that fragments and components that could be ejected from highly pressurized system's rotating equipment would not impair the function of an engineered safety feature. Typical missiles requiring analyses are such items as primary system valves, flanges, instrumentation, etc. When rotating equipment is not completely contained, such as in a concrete vault, a missile map should be provided for rotating equipment (e.g., main turbines, pumps, etc.)

Criterion 41 - Engineered Safety Features Performance Capability

We agree with this criterion as far as it goes. In particular the detailed requirements for the emergency core cooling system as contained in Criterion 44 illustrate the desired amplification (but for that system only). Thus, it could be generalized and added to Criterion 41 as follows: "The performance of each engineered safety feature shall be evaluated conservatively in each area of uncertainty. The systems shall not share active components and shall not share other features or components unless it can be demonstrated that (a) the capability of the shared feature or component to perform its required function can be readily ascertained during reactor operation, (b) failure of the shared feature or component does not initiate a loss-of-coolant accident, and (c) capability of the shared feature or component to perform its required function is not impaired by the effects of a loss-of-coolant accident and is not lost during the entire period this function is required following the accident."

Criterion 42 - Engineered Safety Features Components Capability

We see no need to limit this criterion to the loss-of-coolant accident and suggest that . . . "by the effects of a loss-of-coolant accident" be changed to read "the effects of the accident for which the function is required."

Criterion 43 - Accident Aggravation Prevention

It is not obvious what purpose this criterion is intended to serve. If something specific is in mind here it should be stated, i.e., are we worried about the core becoming critical again, or inducing a thermal shock, etc. Perhaps this should not even appear here but be in the general discussion.

Criterion 44 - Emergency Core Cooling Systems Capability

As noted in the discussion on Criterion 41, we would restrict this criterion to the first two sentences (having already included the remainder of this criterion as a general requirement in Criterion 41). However, as we interpret the intent of these sentences, each of the two emergency cooling systems should cover the whole range of pipe break conditions up to the maximum. To make this point clearer, it might be better to rephrase the second sentence defining the cooling system requirements as follows: "For each size break in the reactor coolant pressure boundary, including the double-ended rupture of the largest pipe, at least two emergency core cooling systems, preferably of different design principles and each with a capability for accomplishing abundant emergency core cooling, shall be provided."

Criterion 48 - Testing of Operational Sequence of Emergency Core Cooling Systems

We agree with the intent of this criterion and suggest that in addition to "the transfer to alternate power sources" the operation of the reactivity control system (which must shutdown the reactor and then provide holddown in the cold condition after the loss-of-coolant accident) should be mentioned.

Criterion 49 - Containment Design Basis

We agree with the intent of this criterion but feel that the following need some elaboration:

Line 10: "Considerable Margin" should be defined in some manner.

Line 13: What degree of failure of the emergency core cooling system is assumed?

Criterion 50 - NDT Requirement for Containment Material

This criteria needs further clarification. The temperature of the steel members in question under normal operating and testing conditions should be defined, i.e., the temperature of the component when the ambient temperature is at its lowest recorded (or perhaps expected) value. Furthermore, the requirement of NDT + 30° F has no meaning in the eyes of the stress analyst although it has found some usage. This temperature is half way between NDT and FTE and unless there is adequate justification of which we are unaware, we recommend using NDT + 60° F which defines the transition, e.g., temperature at which cracks won't propagate at stresses less than yield.

Criterion 51 - Reactor Coolant Pressure Boundary Outside Containment

The intent of this criterion is not clear. It would appear that Criterion 53 which requires redundant valving would also cover reactor containment coolant boundaries outside containment. If, however, it is intended to require extensions of the containment, it should be specifically stated. In any event . . . delete "appropriate" and "as necessary" in lines 4 and 5 and the entire last sentence which begins, "Determination of . . .". These words do not materially contribute to the sense of the statement of the criterion and therefore should be omitted.

Criteria 54, 55, and 56 - Containment Leakage Rate Testing, Containment Periodic Leakage Rate Testing, and Provisions for Testing of Penetrations

Following the words "design pressure" it is suggested that "defined by Criterion 49" be inserted.

Criterion 56

This criterion is not sufficiently inclusive. The types of penetrations which should be tested should NOT be limited to the two that are mentioned, but for instance should also include electrical penetrations and piping penetrations that do not require expansion joints. The penetration testing is usually done at greater than design pressure.

Criterion 66 - Prevention of Fuel Storage Criticality

We do not understand the implication of "or processes" at the end of the first sentence, nor do we believe that it is practical to depend upon procedural controls to prevent accidental criticality in storage facilities of power reactors. Hence, the last sentence of this criterion should be changed to read as follows: "Such means as geometrically safe configuations shall be used to insure that criticality cannot occur."

Criterion 67 - Fuel and Waste Storage Decay Heat

To the extent that removal of decay heat is a function necessary to prevent escape of fission products, decay heat removal systems should be designed to the same requirements for redundancy, inspectability, and testability as engineered safety features on reactors. This should include facilities for supplying additional coolant fluid in the event of accidental loss.

CONTENTION TC-2: EXHIBIT 14

Letter from Edson G. Case, AEC, to Dr. Stephen H. Hanauer, ACRS (July 23, 1969), enclosing General Design Criteria for Nuclear Power Units (July 15, 1969) (relevant excerpts)

2/17/69

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

July 23, 1969

Dr. Stephen H. Hanauer, Chairman
Advisory Committee on Reactor Safeguards
U. S. Atomic Energy Commission
Washington, D. C. 20545

Dear Dr. Hanauer:

Enclosed are 18 copies of:

1. "General Design Criteria for Nuclear Power Units" revision dated July 15, 1969, which reflects the comments made by the ACRS Subcommittee at our meeting July 9, 1969, and

2. A "Comparison of Published Criteria (July 11, 1967) and Revised Criteria (July 15, 1969)."

Regarding the differences between the published and revised criteria, please note that the revised criteria:

a. Reflect comments received from industry on the published criteria and developments that have occurred since their release. In addition, they reflect comments received from the ACRS and the regulatory staff on interim drafts.

b. Establish "minimum requirements" for water-cooled reactors, whereas the published criteria were "guidance" for all reactors.

c. Are arranged in six sections, include definitions, and are not categorized (Category A or Category B).

d. Do not include the term "engineered safety features." The requirements in the published criteria for "engineered safety features" have been incorporated in the revised criteria by including the requirements in the criteria for individual systems.

e. Include criteria which do not have direct counterparts in the published criteria; these are located in the back of Enclosure 2.

ACRS review is requested as soon as possible.

Sincerely,

Edson G. Case

Edson G. Case, Director
Division of Reactor Standards

Enclosure:
As stated

GENERAL DESIGN CRITERIA FOR NUCLEAR POWER UNITS

July 15, 1969

GENERAL DESIGN CRITERIA FOR NUCLEAR POWER UNITS

Table of Contents

INTRODUCTION

Pursuant to the provisions of § 50.34, applications for construction permits must include the principal design criteria for a proposed facility. These General Design Criteria establish minimum requirements for the principal design criteria for water-cooled nuclear power units similar in design and location to units previously approved for construction by the Commission. The General Design Criteria are also considered to be generally applicable to other types of nuclear power units and are intended to be used for guidance in establishing the principal design criteria for these units.

The principal design criteria for a nuclear power unit establish necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety; that is, structures, systems, and components that prevent or mitigate the consequences or accidents which could cause undue risk to the health and safety of the public. There will be some nuclear power units for which these General Design Criteria are not sufficient for this purpose, and additional criteria must be established in the interest of public safety. It is expected that additional or different criteria will be needed to take into account unusual sites and environmental conditions, and for water-cooled nuclear power units of advanced design. Also, there may be nuclear power units for which fulfillment of some of the General Design Criteria may not be necessary or appropriate. For units such as these, departures from the General Design Criteria must be identified and justified.

DEFINITIONS

NUCLEAR POWER UNIT

A nuclear power unit means a nuclear reactor and associated equipment necessary for electrical power generation and those structures, systems, and components required to prevent or mitigate the consequences of accidents which could cause undue risk to the health and safety of the public.

REACTOR COOLANT PRESSURE BOUNDARY

The reactor coolant pressure boundary means all those pressure-containing components, such as pressure vessels, piping, pumps, and valves, within the following systems or portions of systems of pressurized and boiling water-cooled nuclear power units:

(a) The reactor coolant system. For a nuclear power unit of the boiling water type, the reactor coolant system extends to and includes the outermost containment isolation valves capable of external actuation in the main steam and feed-water lines, and the reactor safety and relief valves.

(b) Portions of associated auxiliary systems connected to the reactor coolant system. For piping of these systems which penetrates primary reactor containment, the boundary extends to and includes the first containment isolation valve outside the containment capable of external actuation. For piping of these systems which contains two valves both of which are normally closed during normal reactor operation, the boundary extends to and includes the second of these two valves (the second of which must be capable of external actuation), whether or not the system piping penetrates primary reactor containment.

(c) Portions of the emergency core cooling system connected to the reactor coolant system. For piping of this system which penetrates primary reactor containment, the boundary extends to and includes the first containment isolation valve outside containment capable of external actuation. For piping of this system which does not penetrate primary reactor containment, the boundary extends to and includes the second of two valves normally closed during normal reactor operation.

LOSS-OF-COOLANT ACCIDENTS

Loss-of-coolant accidents mean those postulated accidents that result from the loss of reactor coolant at a rate in excess of the capability of the reactor coolant makeup system from any size break in the piping, pressure vessels, pumps, and valves connected to the reactor pressure vessel and within the reactor coolant pressure boundary, up to and including a break in these components equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system.

SINGLE FAILURE

A single failure means an occurrence which results in a loss of capability of a structure, system, or component to perform its intended functions. Multiple failures resulting from a single occurrence are considered to be a single failure.

CRITERION 62 - PREVENTION OF CRITICALITY IN FUEL STORAGE AND HANDLING

Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by geometrically safe configurations.

CRITERION 63 - MONITORING FUEL AND WASTE STORAGE

Instrumentation shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of decay heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.

CRITERION 64 - MONITORING RADIOACTIVITY RELEASES

Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths and the unit environs for radioactivity that may be released from normal operations, from anticipated operational occurrences, and from postulated accidents.

CONTENTION TC-2: EXHIBIT 15

Memorandum from Edson G. Case, NRC, to Harold L. Price, et al., AEC, re: Revised General Design Criteria (October 12, 1970), and enclosed letter from Edward A. Wiggin, AIF, to Edson G. Case, NRC (October 6, 1970)

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

OCT 12 1970

Harold L. Price, Director of Regulation
Clifford K. Beck, Deputy Director of Regulation
Marvin M. Mann, Assistant Director of Regulation for Reactors
C. L. Henderson, Assistant Director of Regulation for Administration
S. H. Hanauer, Technical Advisor to the Director of Regulation
L. D. Low, Director, Division of Compliance
P. A. Morris, Director, Division of Reactor Licensing

REVISED GENERAL DESIGN CRITERIA

My memorandum of September 24, 1970, to Harold L. Price forwarded the latest revision of the General Design Criteria for your comments. Additions and changes to the June 4 version of the criteria were annotated.

Enclosed is a letter and enclosures which provide the AIF comments of the June 4 version of the criteria. Please note that the major Forum comments are discussed in the third enclosure to its October 6 letter. The revised criteria forwarded by my memorandum of September 24 appear to satisfy all of these major comments.

Please provide your comments on the revised criteria by Monday, October 19, so that review by the ACRS and final issuance of the criteria can be expedited.

Edson G. Case, Director
Division of Reactor Standards

Enclosure:
AIF Letter dated October 6, 1970,
to Edson G. Case w/encls
(except second enclosure)

cc: G. A. Arlotto, DRS

ATOMIC INDUSTRIAL FORUM INC.

475 PARK AVENUE SOUTH, NEW YORK, N.Y. 10016 · 212 725-8300

October 6, 1970

Mr. Edson G. Case, Director
Division of Reactor Standards
U.S. Atomic Energy Commission
Washington, D. C. 20545

Dear Ed:

The purpose of this letter and the enclosed material is to provide you with a commentary, developed by an ad hoc group convened under the aegis of the Forum's Committee on Reactor Safety, on the AEC-proposed "General Design Criteria for Nuclear Power Plants," as set forth in the AEC draft of June 4, 1970.

This commentary has been developed by, and represents the consensus view of, the following industry representatives, who have had an opportunity to participate either in redrafting and modifying the criteria or reviewing the same:

Robert D. Allen (Chairman) - Bechtel Corp.
Edwin A. Wiggin (Secretary) - Atomic Industrial Forum

Rennie Anderson - Combustion Engineering, Inc.
William Bley - Stone & Webster Engineering Corp.
Henry E. Bliss - Commonwealth Edison Co.
A. Philip Bray - General Electric Co.
Allan R. Collier - Westinghouse Electric Corp.
Walter D. Gilbert - General Electric Co.
Gilbert S. Keeley - Consumers Power Co.
Douglas V. Kelly - Pacific Gas & Electric Co.
William J. L. Kennedy - Stone & Webster Engineering Corp.
William Little - Babcock & Wilcox Co.
Lawrence E. Minnick - Yankee Atomic Electric Co.
James S. Moore - Westinghouse Electric Corp.
John N. Noble - Stone & Webster Engineering Corp.
Harold Oslick - Ebasco Services, Inc.
Warren H. Owen - Duke Power Co.

Rec'd Off. Dir. of Reg.
Date 10/12/70
Time

Richard F. Ranellone - General Electric Co.
William Smith - Babcock & Wilcox Co.
James E. Tribble - Yankee Atomic Electric Co.
Michael F. Valerino - Combustion Engineering, Inc.
Robert E. Wascher - Babcock & Wilcox Co.
John M. West - Combustion Engineering, Inc.
Robert A. Wiesemann - Westinghouse Electric Corp.

The enclosed material, which in its entirety comprises our commentary, includes the following five items:

1. A marked up version of the AEC draft of June 4 indicating the changes we believe should be incorporated prior to publication of the criteria.

2. A retyped version of the AEC draft of June 4 incorporating the changes referred to above.

3. A discussion of the major changes recommended. Our consensus agreement with the criteria as modified is dependent upon their acceptance.

4. An explanation of certain detailed changes which we believe to be both necessary and desirable if the criteria are to prove of maximum usefulness to the AEC and the industry. Omitted from this listing are minor changes, for the most part self-explanatory, which have been suggested in the interest of enhancing the clarity of certain criteria but which do not alter either their scope or intent.

5. An excerpt which we believe should be incorporated in the Statement of Considerations at the time the criteria are published.

We wish to emphasize the importance attached to the concerns underlying the major changes recommended. We very much hope that these concerns can be accommodated by adoption of the recommended changes or in some other equally satisfactory manner.

Submission of this consensus commentary is not intended to preclude the subsequent submission of individual comments by those named above or by other industry representatives, once the criteria have been published. Conversely, it is not expected that the group named above or the Forum Committee on Reactor Safety would wish to offer further

ATOMIC INDUSTRIAL FORUM INC.

comments if the recommendations set forth in this commentary are adopted.

Please let us know if you desire further clarification of these comments. Also, should you wish further elaboration of the comments, we would be pleased to convene a representative group of those named above to meet with you and your associates.

We appreciate the opportunity to comment on this important document.

Sincerely,

Edwin A. Wiggin

EAW:erk
Enc.

DRAF

GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANTS

June 4, 1970

APPENDIX A

GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANTS

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INTRODUCTION

Pursuant to the provisions of §50.34, an application for a construction permit must include the principal design criteria for a proposed facility. These General Design Criteria establish minimum requirements for the principal design criteria for water-cooled nuclear power units similar in design and location to units for which construction permits have been issued by the Commission. The General Design Criteria are also ~~considered to be generally applicable to other types of nuclear power units and are~~ intended to provide guidance in establishing the principal design criteria for such other/ types of nuclear power units.

The principal design criteria for a nuclear power unit establish necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety; that is, structures, systems, and components that prevent or mitigate the consequences of accidents which could cause undue risk to the health and safety of the public. Accordingly, these General Design Criteria are intended to reflect current licensing review practice. There will be some water-cooled nuclear power units for which these General Design Criteria are not sufficient for this purpose, and additional criteria must be identified and satisfied by the design in the interest of public safety. It is expected that additional or different criteria ~~will~~ may be needed to take into account unusual sites and environmental conditions, and for water-cooled nuclear power units of advanced design./ ~~Also~~ ¶ There may be water-cooled nuclear power units for which fulfillment of some of the General Design Criteria may not be necessary or appropriate. For units such as these, departures from the General Design Criteria must be identified and justified.

¶ Insert (:)-see next page

The requirements of these General Design Criteria shall be supplemented or modified as necessary to cope with the existence or consequences of a previously unidentified physical condition important to safety. The effective date for the application of industry codes and standards shall be as specified in Title 10 of the Code of Federal Regulations.

Insert (i)

The development of these General Design Criteria is not yet complete. For example, some of the definitions need further amplification. Also, certain of the specific design requirements for structures, systems, and components important to safety have not as yet been suitably defined so that they can be generalized as criteria. For these reasons it is expected that the criteria will be augmented and revised from time to time as important new or charged requirements such as these are identified and developed.

DEFINITIONS AND EXPLANATIONS

NUCLEAR POWER UNIT

A nuclear power unit means a nuclear power reactor and associated equipment necessary for electrical power generation and includes those structures, systems, and components required to prevent or mitigate the consequences of accidents which could cause undue risk to the health and safety of the public.

LOSS-OF-COOLANT ACCIDENTS

Loss-of-coolant accidents mean those postulated accidents that result from the loss of reactor coolant, at a rate in excess of the capability of the system used for normal reactor coolant makeup, ~~system~~ from ~~any size~~ breaks in the ~~piping, pressure vessels, pumps, and valves connected to the reactor pressure vessel and which are part~~ of the reactor coolant pressure boundary, up to and including a break ~~in these components~~ equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system.[1]

SINGLE FAILURE

A single failure means an occurrence which results in the loss of capability of a component to perform its intended safety functions. Multiple failures resulting from a single occurrence are considered to be a single failure. Mechanical and electrical systems are considered to be designed against an assumed single failure if neither (1) a single failure of any active component (assuming passive components function properly) nor (2) a single failure of ~~any~~ selected passive components (assuming active components function properly), results in a

[1] Further details relating to the type, size, and orientation of postulated breaks in specific components of the reactor coolant pressure boundary are under development as a general design criterion.

loss of the capability of the system to perform its safety functions.[2] ~~The failure of a passive component need not be considered in the design of mechanical systems if it can be demonstrated that the design is acceptable on some other defined basis, such as an appropriate combination of unusually high quality, high strength or low stress, inspectability, repairability, or short-term use.~~

ANTICIPATED OPERATIONAL OCCURRENCES

Anticipated operational occurrences mean those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit and include but are not limited to loss of power to the recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power.

2 Single failures of passive components in electrical systems should be assumed in designing against a single failure. The conditions under which a single failure of a passive component in a mechanical system should be considered in designing the system against a single failure are under development as a general design criterion.

CRITERIA

I. OVERALL REQUIREMENTS

CRITERION 1 - QUALITY STANDARDS AND RECORDS

Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally
, if required by unusual design or site characteristics,
recognized codes and standards are used, they shall be identified and/evaluated to determine their applicability, adequacy, and sufficiency. ~~and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function~~. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety
, as defined in 10 CFR Part 50, Appendix B,
functions. Appropriate records/of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

CRITERION 2 - DESIGN BASES FOR PROTECTION AGAINST NATURAL PHENOMENA

Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, (2) sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (3) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (4) the importance of the safety functions to be performed.

CRITERION [illegible] - FIRE PROTECTION

Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Fire fighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.

CRITERION 4 - ENVIRONMENTAL AND MISSILE DESIGN BASES

Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance and testing, and postulated accidents. These structures, systems, and components shall be

to the extent necessary
appropriately protected/~~against~~ dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures
the effects of events and conditions
and from/ ~~sources~~ outside the nuclear power unit.

~~CRITERION 5 - PROTECTION AGAINST INDUSTRIAL SABOTAGE~~

~~Structures, systems, and components important to safety shall be physically protected to minimize, consistent with other safety requirements, the probability and effects of industrial sabotage.~~

CRITERION 6 - SHARING OF STRUCTURES, SYSTEMS, AND COMPONENTS

Structures, systems, and components important to safety shall not be shared between nuclear power units unless it is shown that their ability to perform their safety functions is not significantly impaired by the sharing.

II. PROTECTION BY MULTIPLE FISSION PRODUCT BARRIERS

CRITERION 10 - REACTOR DESIGN

The reactor core and associated coolant, control and protection systems shall be designed ~~with appropriate margin~~ to assure that specified acceptable
damage
fuel ~~design~~ limits are not exceeded during all conditions of normal operation, including the effects of anticipated operational occurrences.

CRITERION 11 - REACTOR INHERENT PROTECTION

The reactor core and associated coolant system~~s~~ shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

CRITERION 12 - ~~SUPPRESSION~~ CONTROL OF REACTOR POWER OSCILLATIONS

The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding ~~of~~ specified acceptable fuel design limits are not possible or can be reliably and readily detected and ~~suppressed.~~ controlled.

CRITERION 13 - REACTOR INSTRUMENTATION AND CONTROL

Instrumentation and control shall be provided to monitor and to maintain variables within prescribed operating ranges, including those variables and systems which can affect the fission process and the integrity of the reactor core.

CRITERION 14 - REACTOR COOLANT PRESSURE BOUNDARY

The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

CRITERION 15 - REACTOR COOLANT SYSTEM DESIGN

The reactor coolant system and associated auxiliary, control, and protection systems shall be designed ~~with sufficient margin~~ to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during all conditions of normal operation, including anticipated operational occurrences.

CRITERION 16 - CONTAINMENT DESIGN

Reactor containment and associated systems shall be provided to establish an essentially leaktight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

CRITERION 17 - ELECTRICAL POWER SYSTEMS

An onsite electrical power system and an offsite electrical power system shall be provided to permit functioning of structures, systems, and components important to safety. The onsite and offsite power systems shall each ~~The safety function for each system alone shall be to~~ provide sufficient capacity and capability to assure that (1) specified acceptable fuel damage ~~design~~ limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

Element of system important to safety

The onsite electrical power ~~sources, including the batteries, and the onsite electrical distribution system~~, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.

Electrical power from the transmission network to the switchyard shall be supplie

~~Two physically independent transmission lines, each with the capability~~

by two transmission lines designed and located so as to suitably minimize the likelihood of

~~of supplying electrical power from the transmission network to the switch-~~

their simultaneous failure under operating, accident and expected environmental conditions.

~~yard, and two physically independent circuits from the switchyard to the~~

Two physically independent circuits from the switchyard to the onsite electrical distributi

~~onsite electrical distribution system shall be provided.~~ Each of these

system shall be provided.

circuits shall be designed to be available in sufficient time following a loss of ~~electrical power from all other alternating current sources, including~~

power , in the absence of a loss-of-coolant accident,

all /onsite electrical/sources/ to assure/that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are

Assuming a loss-of-coolant accident, from the switchyard to the onsite electrical

not exceeded./ One of ~~these~~ circuits ~~shall be designed to~~ be available

power distribution system shall

in sufficient time

~~immediately following a loss of coolant accident~~ to assure that core cooling,

important to safety

containment integrity, and other ~~vital safety~~ functions/are maintained.

Provisions shall be included to minimize the probability of losing

from the offsite electrical power system sources

electrical power/ ~~via any of the remaining circuits~~ as a result of, or coincident with, the loss of power generated by the nuclear power unit, ~~the loss of power from the transmission network, or the loss of power from the onsite electrical power sources.~~

CRITERION 18 - INSPECTION AND TESTING OF ELECTRICAL POWER SYSTEMS

Electrical power systems important to ~~required for~~ safety shall be designed to permit periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically (1) the operability and functional performance of the active components of the systems, such as onsite emergency power sources, relays, switches, and buses, and (2) the operability of the systems as a whole and, under conditions as close to design as practical, although not necessarily while the plant is at power, the full operational sequence that brings the systems into operation, including operation of the initiation logic ~~protection system~~, and the required transfer of power among the nuclear power unit, the offsite power system, and the onsite emergency power system.

CRITERION 19 - CONTROL ROOM

A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable emergency procedures.

III. PROTECTION AND REACTIVITY CONTROL SYSTEMS

CRITERION 20 - PROTECTION SYSTEM FUNCTIONS

The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel/~~design~~ damage limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety. The protection and reactivity control systems shall be designed to assure an extremely h gh probability of accomplishing their safety functions in the event of anticipated operational occurrences.

CRITERION 21 - PROTECTION SYSTEM RELIABILITY AND TESTABILITY

The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functional performance when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

CRITERION 22 - PROTECTION SYSTEM INDEPENDENCE

The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function. ~~in the event of systematic, nonrandom, concurrent failures of redundant elements.~~

CRITERION 23 - PROTECTION SYSTEM FAILURE MODES

The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.

CRITERION 24 - SEPARATION OF PROTECTION AND CONTROL SYSTEMS

The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired, ~~considering the possibility of systematic, nonrandom, concurrent failures of control system components or channels, or of those common to the control and protection systems~~.

CRITERION 25 - PROTECTION SYSTEM REQUIREMENTS FOR REACTIVITY CONTROL MALFUNCTIONS

The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods or unplanned dilution of soluble poison.

CRITERION 26 - REACTIVITY CONTROL SYSTEM REDUNDANCY AND CAPABILITY

Two independent reactivity control systems, ~~preferably~~ of different design principles shall be provided. ~~and preferably including a positive mechanical means for inserting control rods, shall be provided. Each system shall have the capability to control the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded~~. One of the systems shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operations, including anticipated operational occurrences, and with failure of the highest worth rod to insert, ~~appropriate margin for malfunctions such as stuck rods~~, specified acceptable fuel damage ~~design~~ limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

CRITERION 27 - COMBINED REACTIVITY CONTROL SYSTEMS CAPABILITY

The reactivity control systems shall be designed to have a combined capability in conjunction with the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions ~~and with appropriate margin for stuck rods~~ the capability to cool the core is maintained, including consideration of any rods failing to insert as a consequence of the accident.

CRITERION 28 - REACTIVITY LIMITS

The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures, or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

~~CRITERION 29 - PROTECTION AGAINST ANTICIPATED OPERATIONAL OCCURRENCES~~

~~The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences. Their design shall reflect consideration of systematic, nonrandom, concurrent failures of redundant elements.~~

IV. FLUID SYSTEMS

CRITERION 30 - QUALITY OF REACTOR COOLANT PRESSURE BOUNDARY

Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested in accordance with applicable industry codes. ~~to the highest quality standards practical.~~ Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

CRITERION 31 - FRACTURE PREVENTION OF REACTOR COOLANT PRESSURE BOUNDARY

The reactor coolant pressure boundary shall be designed ~~with sufficient margin~~ to assure that under stressed operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady-state and transient stresses, and (4) size of flaws.

CRITERION 32 - INSPECTION OF REACTOR COOLANT PRESSURE BOUNDARY COMPONENTS

Components which are part of the reactor coolant pressure boundary shall be designed to permit in accordance with applicable industry codes (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.

CRITERION 33 - REACTOR COOLANT MAKEUP

A system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for onsite and for offsite electrical power system operation the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.

CRITERION 34 - RESIDUAL HEAT REMOVAL

A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

Suitable redundancy in components and features, suitable interconnections, and leak detection and isolation capabilities shall be provide to assure that for either onsite or ~~and for~~ offsite electrical power system operation the system safety function can be accomplished assuming a single failure.

CRITERION 35 - EMERGENCY CORE COOLING

A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following a ~~any~~ loss-of-coolant accident at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts. ~~The performance of the system shall be evaluated conservatively.~~

Suitable redundancy in components and features, suitable interconnections, and leak detection, isolation, and containment capabilities shall be provided to assure that for either onsite or ~~and for~~ offsite electrical power system operation the system safety function can be accomplished assuming a single failure.

CRITERION 36 - INSPECTION AND PRESSURE TESTING OF EMERGENCY CORE COOLING SYSTEM ~~COMPONENTS~~

~~Components of~~ The emergency core cooling system shall be designed to permit periodic inspection and appropriate pressure testing of important components ~~areas and features, such as spray rings in the reactor pressure vessel, water injection nozzles, and piping,~~ to assure their structural and leaktight integrity and as a measure of the full design capability of the system.

CRITERION 37 - TESTING OF EMERGENCY CORE COOLING SYSTEM

The emergency core cooling system shall be designed to permit periodic functional testing which will provide a measure of (1) the operability and performance of the active components of the system, ~~such as pumps and valves,~~ and (2) to the extent practical, the operability of the system as a whole, ~~and, under conditions as close to design as practical,~~ the full operational sequence that brings the system into operation, including operation of the initiation logic ~~protection system,~~ the transfer between normal and emergency power sources, and operation of the associated cooling water system.

CRITERION 38 - CONTAINMENT HEAT REMOVAL

A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce, ~~rapidly,~~ consistent with the functioning of other associated systems, the containment pressure and temperature following a ~~any~~ loss-of-coolant accident and maintain them at acceptable ~~low~~ levels.

Suitable redundancy in components and features, suitable interconnections, and leak detection, isolation, and containment capabilities shall be provided to assure that either ~~for~~ onsite or ~~and for~~ offsite electrical power system operation the system safety function can be accomplished assuming a single failure.

CRITERION 39 - INSPECTION AND PRESSURE TESTING OF CONTAINMENT HEAT REMOVAL SYSTEM ~~COMPONENTS~~

~~Components of~~ The containment heat removal system shall be designed, insofar as practical, to permit periodic inspection and appropriate pressure testing of important components ~~areas and features, such as the torus, pumps, spray nozzles, and piping~~. to assure their structural and leaktight integrity as a measure of ~~and~~ the full design capability of the system.

CRITERION 40 - TESTING OF CONTAINMENT HEAT REMOVAL SYSTEM

The containment heat removal system shall be designed to permit periodic functional testing which will provide a measure of (1) the operability and performance of the active components of the system, ~~such as pumps and valves~~ and (2) to the extent practical the operability of the system as a whole, and, under conditions as close ~~to the design as practical,~~ the full operational sequence that brings the system into operation, including operation of the initiation logic ~~protection system,~~ the transfer between normal and emergency power sources, and operation of the associated cooling water system.

CRITERION 41 - CONTROL OF CONTAINMENT ATMOSPHERE ~~CLEANUP~~

Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to limit ~~reduce,~~ consistent with the functioning of other associated systems, the release ~~concentration and quantity~~ of fission products ~~released~~ to the environment following postulated accidents, such that acceptable limits are not exceeded, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

S
~~Each system shall have~~ ~~s~~uitable redundancy in components and features,
suitable shall be provided
/interconnections, and leak detection and isolation capabilities/ to assure
either or
that for/onsite/~~and for~~ offsite electrical power system operation its safety
function can be accomplished assuming a single failure.

AND PRESSURE TESTING
CRITERION 42 - INSPECTION/OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEMS ~~COMPONENTS~~

T
~~Components of~~ ~~t~~he containment atmosphere cleanup systems shall be
, insofar as practical,
designed/to permit periodic inspection and appropriate pressure testing of
components
important /~~areas and features such as filter frames, ducts, and piping~~ to
as a measure of
assure their structural and leaktight integrity /~~and~~ the full design capability
of the systems.

CRITERION 43 - TESTING OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEMS

The containment atmosphere cleanup systems shall be designed to permit
which will provide a measure
periodic functional testing/of (1) the operability and performance of the
active components of the systems ~~such as fans, filters, dampers, pumps~~
to the extent practical,
~~and valves~~ and (2)/the operability of the systems as a whole ~~and, under~~
~~conditions as close to design as practical,~~ the full operational sequence
initiation logic,
that brings the systems into operation, including operation of the/~~protection~~
~~system,~~ the transfer between normal and emergency power sources, and operation
of associated systems.

CRITERION 44 - COOLING WATER

A system to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating/or ~~and~~ accident conditions.

Suitable redundancy in components and features,/suitable interconnections, and leak detection and isolation capabilities shall be provided to assure that for/either onsite/or ~~and for~~ offsite electrical power system operation the system safety function can be accomplished assuming a single failure.

CRITERION 45 - INSPECTION/AND PRESSURE TESTING OF COOLING WATER SYSTEM ~~COMPONENTS~~

~~Components of~~ The cooling water system shall be designed/insofar as practical to permit periodic inspection and appropriate pressure testing of important/components ~~areas and features, such as heat exchangers and piping~~, to assure their structural and leaktight integrity and the full design capability of the system.

CRITERION 46 - TESTING OF COOLING WATER SYSTEM

The cooling water system shall be designed to permit periodic functional testing/which will provide a measure of (1) the operability and performance of the active components of the system, ~~such as pumps and valves,~~ and (2)/to the extent practical the operability of the system as a whole, ~~and, under conditions as close to design as practical~~ the full operational sequence that brings the system into operation for reactor shutdown and ~~for~~ loss-of-coolant accidents, including operation of the/initiation logic ~~protection system~~ and the transfer between normal and emergency power sources.

V. REACTOR CONTAINMENT

CRITERION 50 - CONTAINMENT DESIGN BASIS

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

CRITERION 51 - FRACTURE PREVENTION OF CONTAINMENT PRESSURE BOUNDARY

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

CRITERION 52 - CAPABILITY FOR CONTAINMENT LEAKAGE RATE TESTING

The reactor containment and other equipment which may ~~necessarily~~ be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at pressures up to and, if necessary, including the ~~containment~~ design pressure.

CRITERION 53 - PROVISIONS FOR CONTAINMENT TESTING AND INSPECTION

The reactor containment shall be designed to permit insofar as practical (1) visual inspection of all ~~important areas, such as~~ penetrations, (2) an appropriate materials surveillance program, and (3) periodic testing at containment design pressure of the leaktightness of penetrations which have resilient seals and expansion bellows ~~at containment design pressure~~.

INSERT (2) - see next page

CRITERION 54 - PIPING SYSTEMS PENETRATING CONTAINMENT

Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

INSERT (2)

CRITERION 54 - PROVISIONS FOR CONTAINMENT ISOLATION

Piping which penetrates the containment must be provided with two isolation barriers; one barrier must be located outside the containment and one must be inside the containment, unless it can be demonstrated that the design is acceptable on some other defined basis.

The definition of an isolation barrier is either a suitably designed closed system trip valve, check valve or a manually closed valve under administrative control.

Using this definition four general classifications are derived:

1. Two closed systems - one inside, one outside, no isolation valves required.
2. No closed systems - one valve inside and one valve outside required.
3. Closed system inside - no valve inside, valve required outside.
4. Closed system outside - no valve outside, valve required inside.

NOTE 1: The same criteria apply to lines which are used after an accident except that manual isolation is acceptable and in the case of instrument lines, a check valve or manual valve inside or outside containment is acceptable.

NOTE 2: An isolation valve outside containment shall be located as close to to the containment as practical and upon loss of actuating power the automatic isolation valves shall be designed to take the position that provides greater safety.

CRITERION 55 - REACTOR COOLANT PRESSURE BOUNDARY PENETRATING CONTAINMENT

Each line which is part of the reactor coolant pressure boundary and which penetrates primary reactor containment shall be provided with one automatic isolation valve inside and one automatic isolation valve, other than a simple check valve, outside of containment, unless it can be demonstrated that the design is acceptable on some other defined basis. The valve outside of containment shall be located as close to containment as practical and upon loss of actuating power the automatic isolation valves shall be designed to take the position that provides greater safety.

Other appropriate requirements to minimize the probability o consequences of an accidental rupture of these lines or of lines connected to them shall be provided as necessary to assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication, and testing, additional provision for inservice inspection, protection against more severe natural phenomena, and additional isolation valves and containment, shall include consideration of the population density, use characteristics, and physical characteristics of the site environs.

CRITERION 56 - CONTAINMENT PRESSURE BOUNDARY ISOLATION VALVES

Each line which connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with one automatic isolation valve inside and one automatic isolation valve, other than a simple check valve, outside of containment, unless it can be demonstrated that the design is acceptable on some other defined basis. The valve outside of containment shall be located as close to containment as practical and upon loss of actuating power the automatic isolation valves shall be designed to take the position that provides greater safety.

CRITERION 57 - CLOSED SYSTEMS ISOLATION VALVES

Each line which penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one isolation valve, other than a simple check valve. This valve shall be outside of containment and shall be located as close to containment as practical.

VI. FUEL AND RADIOACTIVITY CONTROL

CRITERION 60 - CONTROL OF RELEASES OF RADIOACTIVE MATERIALS TO THE ENVIRONMENT

The nuclear power unit design shall include means to ~~maintain suitable~~ control ~~over~~ the handling and release of radioactive materials in gaseous and liquid effluents and in solid wastes produced during normal reactor operation, including anticipated operational occurrences, within acceptable limits. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon their release to the environment.

CRITERION 61 - FUEL STORAGE AND HANDLING AND ~~RADIOACTIVITY CONTROL~~ RADIOACTIVE WASTE SYSTEMS

The fuel storage and handling and radioactive waste systems ~~and other systems which may contain radioactivity~~ shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be ~~designed~~ provided with (1) with a capability to permit inspection and testing of ~~important areas and features of the~~ components important to safety ~~of these systems~~, (2) with suitable shielding for radiation protection, (3) ~~with~~ appropriate ~~containment~~, confinement, and filtering systems, and (4) ~~with~~ a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and designed ~~(5)~~ to prevent significant reduction in fuel storage coolant inventory under accident conditions.

CRITERION 62 - PREVENTION OF CRITICALITY IN FUEL STORAGE AND HANDLING

Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

CRITERION 63 - MONITORING FUEL AND WASTE STORAGE

Instrumentation
~~Appropriate systems~~ shall be provided in fuel storage and radioactive waste systems and associated handling areas ~~(1)~~ to detect/conditions and alarm any that may result in loss of residual heat removal capability and excessive radiation levels. ~~and (2) to initiate appropriate safety actions.~~

CRITERION 64 - MONITORING RADIOACTIVITY RELEASES

Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.

(Sec. 161, 68 Stat. 948; 42 U.S.C. 2201)

Dated at ______________________________ this ______________

day of ______________________ 1970.

For the Atomic Energy Commission

____

W. B. McCool
Secretary

A Discussion of Major Changes Recommended

There are a number of criteria which as drafted cannot be accepted by the industry for one or more of the following reasons: (1) it represents an unnecessary and unjustified escalation of licensing requirements, (2) there is no clear or common understanding on the part of the AEC and the licensee as to what it would take to meet the requirement, and (3) it is premature to attempt to incorporate the requirement into general design criteria inasmuch as the technical rationale for the requirement has not been fully developed.

Loss-of-Coolant Accident

The definition of the loss-of-coolant accident as set forth in the AEC draft of June 4 clearly represents an escalation of licensing requirements inasmuch as it refers to "any size break" in the "pressure vessels, pumps, and valves connected to the reactor pressure vessel" as well as to a break in the piping. These additional breaks should not be postulated by license reviewers and certainly should not be incorporated into general design criteria in the absence of a realistic technical rationale, the basis for which can be developed only through further study. That study is now being undertaken by an ACRS subcommittee and by an ad hoc Forum group.

Single Failure

As the definition of "single failure" appears in the AEC draft of June 4, it postulates the failure of passive components in both mechanical and electrical systems. Although current licensing review practice assumes the failure of passive components in electrical systems, the extension of the general concept to mechanical systems represents an escalation of licensing requirements for which no technical rationale has been developed. Further, the definition leaves open ended the number and type of mechanical systems to which it could be applied. Indeed, an undisciplined application of the definition would presumably lead to postulating such failures as to make it impossible to design operable systems. Clearly, a single failure concept which would permit the indiscriminate application of postulated failures of passive components in mechanical systems should not be incorporated into general design criteria.

Industrial Sabotage

The AEC draft of June 4 includes as Criterion 5 "Protection Against Industrial Sabotage" which reads" "Structures, systems, and components important to safety shall be physically protected to minimize, consistent with other safety requirements, the probability and effects of industrial sabotage."

Policy considerations involved in the proposed requirement are of such significance that a direct discussion of top utility management personnel with members of the Commission would appear to be prerequisite to resolution of the issues that would be raised in implementing the proposed criterion.

Transmission of Offsite Electrical Power

Criterion 17, "Electrical Power Systems," as it appears in the June 4 draft, includes the requirement: "Two physically independent transmission lines, each with the capability of supplying electrical power from the transmission network to the switchyard, and two physically independent circuits from the switchyard to the onsite electrical distribution system shall be provided."

A literal interpretation of this requirement would call for two transmission lines mounted on different sets of towers located on different rights-of-way. Not only is this an unwarranted escalation of licensing requirements, but for many sites the requirement would neither be desirable nor possible to meet. Further, such a requirement would be contradictory in many instances with requirements being imposed on licensees by environmental considerations.

License applicants should be permitted the option of satisfying the integrity of emergency offsite electrical power service by means other than would be permitted by the criterion as now drafted.

Systematic, Nonrandom, Concurrent Failures of Redundant Elements

Criteria 22, 24 and 29, as set forth in the AEC draft of June 4, all deal with protection and reactivity control systems and all postulate "systematic, nonrandom, concurrent failures of redundant elements." This postulated failure mode is not acceptable to the industry for the following reasons: (1) there is no indication of what requirements are involved, (2) it would provide a "hunting license" for an undisciplined imposition of requirements, (3) there is no logical basis for limiting the concept to protection and reactivity control systems, and (4) the reactor systems suppliers are only now in the early stages of studies which the AEC regulatory staff has asked them to undertake in this area.

Until such time as the requirements which would be imposed by this postulated failure mode can be clearly defined and supported by sound technical rationale, they should not be incorporated into general design criteria.

Containment Isolation

Criterion 54 through Criterion 57, as set forth in the AEC draft of June 4, provide a number of requirements dealing with containment isolation. As drafted, some of these requirements are difficult to interpret and appear to represent an escalation of current licensing practice. Informal discussions with the AEC regulatory staff have not proved successful in developing a mutually satisfactory format for these criteria.

CONTENTION TC-2: EXHIBIT 16

Final Rule, General Design Criteria for Nuclear Power Plants, 36 Fed. Reg. 3,255 (February 20, 1971)

Act of February 2, 1903, as amended the Act of March 3, 1905, as amended, the Act of September 6, 1961, and the Act of July 2, 1962 (21 U.S.C. 111-113, 114g, 115, 117, 120, 121, 123-126, 134b, 134f), Part 76, Title 9, Code of Federal Regulations, restricting the interstate movement of swine and certain products because of hog cholera and other communicable swine diseases, is hereby amended in the following respects:

In § 76.2, the reference to the State of Ohio in the introductory portion of paragraph (e) and paragraph (e)(9) relating to the State of Ohio are deleted.

(Secs. 4-7, 23 Stat. 32, as amended, secs. 1, 2, 32 Stat. 791-792, as amended, secs. 1-4, 33 Stat. 1264, 1265, as amended, sec. 1, 75 Stat. 481, secs. 3 and 11, 76 Stat. 130, 132; 21 U.S.C. 111, 112, 113, 114g, 115, 117, 120, 121, 123-126, 134b, 134f; 29 F.R. 16210, as amended.)

Effective date. The foregoing amendment shall become effective upon issuance.

The amendment excludes a portion of Clinton County, Ohio, from the areas quarantined because of hog cholera. Therefore, the restrictions pertaining to the interstate movement of swine and swine products from or through quarantined areas as contained in 9 CFR Part 76, as amended, will not apply to the excluded area, but will continue to apply to the quarantined areas described in § 76.2(e). Further, the restrictions pertaining to the interstate movement of swine and swine products from nonquarantined areas contained in said Part 76 will apply to the excluded area. No areas in Ohio remain under the quarantine.

The amendment relieves certain restrictions presently imposed but no longer deemed necessary to prevent the spread of hog cholera and must be made effective immediately to be of maximum benefit to affected persons. It does not appear that public participation in this rule making proceeding would make additional information available to this Department. Accordingly, under the administrative procedure provisions in 5 U.S.C. 553, it is found upon good cause that notice and other public procedure with respect to the amendment are impracticable and unnecessary, and good cause is found for making it effective less than 30 days after publication in the Federal Register.

Done at Washington, D.C., this 16th day of February 1971.

F. J. Mulhern,
Acting Administrator,
Agricultural Research Service.

[FR Doc.71-2380 Filed 2-19-71;8:49 am]

[Docket No. 71-520]

PART 76—HOG CHOLERA AND OTHER COMMUNICABLE SWINE DISEASES

Areas Quarantined

Pursuant to provisions of the Act of May 29, 1884, as amended, the Act of February 2, 1903, as amended, the Act of March 3, 1905, as amended, the Act of September 6, 1961, and the Act of July 2, 1962 (21 U.S.C. 111-113, 114g, 115, 117, 120, 121, 123-126, 134b, 134f), Part 76, Title 9, Code of Federal Regulations, restricting the interstate movement of swine and certain products because of hog cholera and other communicable swine diseases, is hereby amended in the following respects:

In § 76.2, in paragraph (e)(13) relating to the State of Texas, subdivision (xvi) relating to Smith County is deleted, and new subdivisions (xxii) and (xxiii) relating to Bexar County are added to read:

(13) *Texas.* * * *

(xxii) That portion of Bexar County bounded by a line beginning at the junction of Interstate Highway 410 and Farm-to-Market Road 78; thence, following Farm-to-Market Road 78 in a northeasterly direction to Farm-to-Market Road 1518; thence, following Farm-to-Market Road 1518 in a southeasterly and then southwesterly direction to U.S. Highway 87; thence, following U.S. Highway 87 in a northwesterly direction to Interstate Highway 410; thence, following Interstate Highway 410 in a northwesterly direction to its junction with Farm-to-Market Road 78.

(xxiii) That portion of Bexar County bounded by a line beginning at the junction of the Bexar-Medina County line and State Highway 16; thence, following State Highway 16 in a southeasterly direction to Farm-to-Market Road 471; thence, following Farm-to-Market Road 471 in a southwesterly and then northwesterly direction to Farm-to-Market Road 1957; thence, following Farm-to-Market Road 1957 in a southeasterly and then southwesterly direction to the Bexar-Medina County line; thence, following the Bexar-Medina County line in a northerly direction to its junction with State Highway 16.

(Secs. 4-7, 23 Stat. 32, as amended, secs. 1, 2, 32 Stat. 791-792, as amended, secs. 1-4, 33 Stat. 1264, 1265, as amended, sec. 1, 75 Stat. 481, secs. 3 and 11, 76 Stat. 130, 132; 21 U.S.C. 111, 112, 113, 114g, 115, 117, 120, 121, 123-126, 134b, 134f; 29 F.R. 16210, as amended)

Effective date. The foregoing amendments shall become effective upon issuance.

The amendments quarantine portions of Bexar County, Tex., because of the existence of hog cholera. This action is deemed necessary to prevent further spread of the disease. The restrictions pertaining to the interstate movement of swine and swine products from or through quarantined areas as contained in 9 CFR Part 76, as amended, will apply to the quarantined portions of such county.

The amendments also exclude a portion of Smith County, Tex., from the areas quarantined because of hog cholera. No areas in Smith County, Tex., remain under the quarantine. Therefore, the restrictions pertaining to the interstate movement of swine and swine products from or through quarantined areas as contained in 9 CFR Part 76, as amended, will not comply to the excluded area, but will continue to apply to the quarantined areas described in § 76.2(e). Further, the restrictions pertaining to the interstate movement of swine and swine products from nonquarantined areas contained in said Part 76 will apply to the area excluded from quarantine.

Insofar as the amendments impose certain further restrictions necessary to prevent the interstate spread of hog cholera, they must be made effective immediately to accomplish their purpose in the public interest. Insofar as they relieve restrictions, they should be made effective promptly in order to be of maximum benefit to affected persons.

Accordingly, under the administrative procedure provisions in 5 U.S.C. 553, it is found upon good cause that notice and other public procedure with respect to the amendments are impracticable, unnecessary, and contrary to the public interest, and good cause is found for making them effective less than 30 days after publication in the Federal Register.

Done at Washington, D.C., this 16th day of February 1971.

F. J. Mulhern,
Acting Administrator,
Agricultural Research Service.

[FR Doc.71-2339 Filed 2-19-71;8:46 am]

Title 10—ATOMIC ENERGY

Chapter I—Atomic Energy Commission

PART 50—LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

General Design Criteria for Nuclear Power Plants

The Atomic Energy Commission has adopted an amendment to its regulations, 10 CFR Part 50, "Licensing of Production and Utilization Facilities," which adds an Appendix A, "General Design Criteria for Nuclear Power Plants."

Section 50.34(a) of Part 50 requires that each application for a construction permit include the preliminary design of the facility. The following information is specified for inclusion as part of the preliminary design of the facility:

(i) The principal design criteria for the facility

(ii) The design bases and the relation of the design bases to the principal design criteria

(iii) Information relative to materials of construction, general arrangement, and the approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety.

The "General Design Criteria for Nuclear Power Plants" added as Appendix A to Part 50 establish the minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission. They also provide guidance in establishing the principal design criteria for other types of nuclear power plants. Principal design criteria established by an applicant and accepted by the Commission will be incorporated by reference in the construction permit. In considering the issuance of an operating license under Part 50, the Commission will require assurance that these criteria have been satisfied in the detailed design and construction of the facility and that any changes in such criteria are justified.

A proposed Appendix A, "General Design Criteria for Nuclear Power Plant Construction Permits" to 10 CFR Part 50 was published in the FEDERAL REGISTER (32 F.R. 10213) on July 11, 1967. The comments and suggestions received in response to the notice of proposed rule making and subsequent developments in the technology and in the licensing process have been considered in developing the revised criteria which follow.

The revised criteria establish minimum requirements for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission, whereas the previously proposed criteria would have provided guidance for applicants for construction permits for all types of nuclear power plants. The revised criteria have been reduced to 55 in number, include definitions of important terms, and have been rearranged to increase their usefulness in the licensing process. Additional criteria describing specific requirements on matters covered in more general terms in the previously proposed criteria have been added to the criteria. The Categories A and B used to characterize the amount of information needed in Safety Analysis Reports concerning each criterion have been deleted since additional guidance on the amount and detail of information required to be submitted by applicants for facility licenses at the construction permit stage is now included in § 50.34 of Part 50. The term "engineered safety features" has been eliminated from the revised criteria and the requirements for "engineered safety features" incorporated in the criteria for individual systems.

Further revisions of these General Design Criteria are to be expected. In the course of the development of the revised criteria, important safety considerations were identified, but specific requirements related to some of these considerations have not as yet been sufficiently developed and uniformly applied in the licensing process to warrant their inclusion in the criteria at this time. Their omission does not relieve any applicant from considering these matters in the design of a specific facility and satisfying the necessary safety requirements. These matters include:

(i) Consideration of the need to design against single failures of passive components in fluid systems important to safety.

(ii) Consideration of redundancy and diversity requirements for fluid systems important to safety. A "system" could consist of a number of subsystems each of which is separately capable of performing the specified system safety function. The minimum acceptable redundancy and diversity of subsystems and components within a subsystem and the required interconnection and independence of the subsystems have not yet been developed or defined.

(iii) Consideration of the type, size, and orientation of possible breaks in the components of the reactor coolant pressure boundary in determining design requirements to suitably protect against postulated loss of coolant accidents.

(iv) Consideration of the possibility of systematic, nonrandom, concurrent failures of redundant elements in the design of the protection systems and reactivity control systems.

In addition, the Commission is giving consideration to the need for development of criteria relating to protection against industrial sabotage and protection against common mode failures in systems, other than the protection and reactivity control systems, that are important to safety and have extremely high reliability requirements.

It is expected that these criteria will be augmented or changed when specific requirements related to these and other considerations are suitably identified and developed.

Pursuant to the Atomic Energy Act of 1954, as amended, and sections 552 and 553 of title 5 of the United States Code, the following amendment to 10 CFR Part 50 is published as a document subject to codification to be effective 90 days after publication in the FEDERAL REGISTER. The Commission invites all interested persons who desire to submit written comments or suggestions in connection with the amendment to send them to the Secretary, U.S. Atomic Energy Commission, Washington, D.C. 20545, Attention: Chief, Public Proceedings Branch, within 45 days after publication of this notice in the FEDERAL REGISTER. Such submissions will be given consideration with the view to possible further amendments. Copies of comments may be examined in the Commission's Public Document Room at 1717 H Street NW., Washington, DC.

1. Section 50.34(a)(3)(i) is amended to read as follows:

§ 50.34 Contents of applications; technical information.

(a) *Preliminary safety analysis report.* Each application for a construction permit shall include a preliminary safety analysis report. The minimum information to be included shall consist of the following:

* * * * *

(3) The preliminary design of the facility including:

(i) The principal design criteria for the facility.[1] Appendix A, General Design Criteria for Nuclear Power Plants, establishes minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have previously been issued by the Commission and provides guidance to applicants for construction permits in establishing principal design criteria for other types of nuclear power units;

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[1] General design criteria for chemical processing facilities are being developed.

2. A new Appendix A is added to read as follows:

APPENDIX A—GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANTS

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INTRODUCTION

DEFINITIONS

INTRODUCTION

Pursuant to the provisions of § 50.34, an application for a construction permit must include the principal design criteria for a proposed facility. The principal design criteria establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety; that is, structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public.

These General Design Criteria establish minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission. The General Design Criteria are also considered to be generally applicable to other types of nuclear power units and are intended to provide guidance in establishing the principal design criteria for such other units.

The development of these General Design Criteria is not yet complete. For example, some of the definitions need further amplification. Also, some of the specific design requirements for structures, systems, and components important to safety have not as yet been suitably defined. Their omission does not relieve any applicant from considering these matters in the design of a specific facility and satisfying the necessary safety requirements. These matters include:

(1) Consideration of the need to design against single failures of passive components in fluid systems important to safety. (See Definition of Single Failure.)

(2) Consideration of redundancy and diversity requirements for fluid systems important to safety. A "system" could consist of a number of subsystems each of which is separately capable of performing the specified system safety function. The minimum acceptable redundancy and diversity of subsystems and components within a subsystem, and the required interconnection and independence of the subsystems have not yet been developed or defined. (See Criteria 34, 35, 38, 41, and 44.)

(3) Consideration of the type, size, and orientation of possible breaks in components of the reactor coolant pressure boundary in determining design requirements to suitably protect against postulated loss-of-coolant accidents. (See Definition of Loss of Coolant Accidents.)

(4) Consideration of the possibility of systematic, nonrandom, concurrent failures of redundant elements in the design of protection systems and reactivity control systems. (See Criteria 22, 24, 26, and 29.)

It is expected that the criteria will be augmented and changed from time to time as important new requirements for these and other features are developed.

There will be some water-cooled nuclear power plants for which the General Design Criteria are not sufficient and for which additional criteria must be identified and satisfied in the interest of public safety. In particular, it is expected that additional or different criteria will be needed to take into account unusual sites and environmental conditions, and for water-cooled nuclear power units of advanced design. Also, there may be water-cooled nuclear power units for which fulfillment of some of the General Design Criteria may not be necessary or appropriate. For plants such as these, departures from the General Design Criteria must be identified and justified.

DEFINITIONS AND EXPLANATIONS

Nuclear power unit. A nuclear power unit means a nuclear power reactor and associated equipment necessary for electrical power generation and includes those structures, systems, and components required to provide reasonable assurance the facility can be operated without undue risk to the health and safety of the public.

Loss of coolant accidents. Loss of coolant accidents mean those postulated accidents that result from the loss of reactor coolant at a rate in excess of the capability of the reactor coolant makeup system from breaks in the reactor coolant pressure boundary, up to and including a break equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system.[1]

Single failure. A single failure means an occurrence which results in the loss of capability of a component to perform its intended safety functions. Multiple failures resulting from a single occurrence are considered to be a single failure. Fluid and electrical systems are considered to be designed against an assumed single failure if neither (1) a single failure of any active component (assuming passive components function properly) nor (2), a single failure of a passive component (assuming active components function properly), results in a loss of the capability of the system to perform its safety functions.[2]

Anticipated operational occurrences. Anticipated operational occurrences mean those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power.

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[1] Further details relating to the type, size, and orientation of postulated breaks in specific components of the reactor coolant pressure boundary are under development.

[2] Single failures of passive components in electrical systems should be assumed in designing against a single failure. The conditions under which a single failure of a passive component in a fluid system should be considered in designing the system against a single failure are under development.

CRITERIA

I. Overall Requirements

Criterion 1—Quality standards and records. Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

Criterion 2—Design bases for protection against natural phenomena. Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed.

Criterion 3—Fire protection. Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Firefighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.

Criterion 4—Environmental and missile design bases. Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.

Criterion 5—Sharing of structures, systems, and components. Structures, systems, and components important to safety shall not be shared between nuclear power units unless it is shown that their ability to perform their safety functions is not significantly impaired by the sharing.

II. Protection by Multiple Fission Product Barriers

Criterion 10—Reactor design. The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

Criterion 11—Reactor inherent protection. The reactor core and associated coolant systems shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

Criterion 12—Suppression of reactor power oscillations. The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

Criterion 13—Instrumentation and control. Instrumentation and control shall be provided to monitor variables and systems over their anticipated range for normal operation and accident conditions, and to maintain them within prescribed operating ranges, including those variables and systems which can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems.

Criterion 14—Reactor coolant pressure boundary. The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

Criterion 15—Reactor coolant system design. The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

Criterion 16—Containment design. Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

Criterion 17—Electrical power systems. An onsite electrical power system and an offsite electrical power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

The onsite electrical power sources, including the batteries, and the onsite electrical distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.

Electrical power from the transmission network to the switchyard shall be supplied by two physically independent transmission lines (not necessarily on separate rights of way) designed and located so as to suitably minimize the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. Two physically independent circuits from the switchyard to the onsite electrical distribution system shall be provided. Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power sources and the other offsite electrical power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a loss-of-coolant accident to assure that core cooling, containment integrity, and other vital safety functions are maintained.

Provisions shall be included to minimize the probability of losing electrical power from any of the remaining sources as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electrical power sources.

Criterion 18—Inspection and testing of electrical power systems. Electrical power systems important to safety shall be designed to permit periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically (1) the operability and functional performance of the components of the systems, such as onsite power sources, relays, switches, and buses, and (2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operation sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system.

Criterion 19—Control room. A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

III. Protection and Reactivity Control Systems

Criterion 20—Protection system functions. The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

Criterion 21—Protection system reliability and testability. The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

Criterion 22—Protection system independence. The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.

Criterion 23—Protection system failure modes. The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.

Criterion 24—Separation of protection and control systems. The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired.

Criterion 25—Protection system requirements for reactivity control malfunctions. The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods or unplanned dilution of soluble poison.

Criterion 26—Reactivity control system redundancy and capability. Two independent reactivity control systems of different design principles and preferably including a positive mechanical means for inserting control rods, shall be provided. Each system shall have the capability to control the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operations, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

Criterion 27—Combined reactivity control systems capability. The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

Criterion 28—Reactivity limits. The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

Criterion 29—Protection against anticipated operational occurrences. The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

IV. Fluid Systems

Criterion 30—Quality of reactor coolant pressure boundary. Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

Criterion 31—Fracture prevention of reactor coolant pressure boundary. The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady-state and transient stresses, and (4) size of flaws.

Criterion 32—Inspection of reactor coolant pressure boundary. Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.

Criterion 33—Reactor coolant makeup. A system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.

Criterion 34—Residual heat removal. A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Criterion 35—Emergency core cooling. A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of coolant accident at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Criterion 36—Inspection of emergency core cooling system. The emergency core cooling system shall be designed to permit periodic inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles, and piping, to assure the integrity and capability of the system.

Criterion 37—Testing of emergency core cooling system. The emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

Criterion 38—Containment heat removal. A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Criterion 39—Inspection of containment heat removal system. The containment heat removal system shall be designed to permit periodic inspection of important components, such as the torus, sumps, spray nozzles, and piping to assure the integrity and capability of the system.

Criterion 40—Testing of containment heat removal system. The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole, and, under conditions as close to the design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

Criterion 41—Containment atmosphere cleanup. Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

Each system shall have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) its safety function can be accomplished, assuming a single failure.

Criterion 42—Inspection of containment atmosphere cleanup systems. The containment atmosphere cleanup systems shall be designed to permit periodic inspection of important components, such as filter frames, ducts, and piping to assure the integrity and capability of the systems.

Criterion 43—Testing of containment atmosphere cleanup systems. The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems.

Criterion 44—Cooling water. A system to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Criterion 45—Inspection of cooling water system. The cooling water system shall be designed to permit periodic inspection of important components, such as heat exchangers and piping, to assure the integrity and capability of the system.

Criterion 46—Testing of cooling water system. The cooling water system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and the performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation for reactor shutdown and for loss-of-coolant accidents, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.

V. Reactor Containment

Criterion 50—Containment design basis. The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and, with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and energy from metal-water and other chemical reactions that may result from degraded emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.

Criterion 51—Fracture prevention of containment pressure boundary. The reactor containment boundary shall be designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining (1) material properties, (2) residual, steady-state, and transient stresses, and (3) size of flaws.

Criterion 52—Capability for containment leakage rate testing. The reactor containment and other equipment which may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.

Criterion 53—Provisions for containment testing and inspection. The reactor containment shall be designed to permit (1) inspection of all important areas, such as penetrations, (2) an appropriate surveillance program, and (3) periodic testing at containment design pressure of the leaktightness of penetrations which have resilient seals and expansion bellows.

Criterion 54—Piping systems penetrating containment. Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

Criterion 55—Reactor coolant pressure boundary penetrating containment. Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

(1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or

(2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or

(3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them shall be provided as necessary to assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication, and testing, additional provisions for inservice inspection, protection against more severe natural phenomena, and additional isolation valves and containment, shall include consideration of the population density, use characteristics, and physical characteristics of the site environs.

Criterion 56—Primary containment isolation. Each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

(1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or

(2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or

(3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to the containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Criterion 57—Closed system isolation valves. Each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be either automatic, or locked closed, or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.

VI. Fuel and Radioactivity Control

Criterion 60—Control of releases of radioactive materials to the environment. The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.

Criterion 61—Fuel storage and handling and radioactivity control. The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.

Criterion 62—Prevention of criticality in fuel storage and handling. Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

Criterion 63—Monitoring fuel and waste storage. Appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.

Criterion 64—Monitoring radioactivity releases. Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.

(Secs. 161, 182, 68 Stat. 948, 953; 42 U.S.C. 2201, 2232)

Dated at Washington, D.C., this 10th day of February 1971.

For the Atomic Energy Commission.

W. B. McCOOL,
Secretary of the Commission.

[FR Doc.71-2370 Filed 2-19-71;8:48 am]

Title 14—AERONAUTICS AND SPACE

Chapter I—Federal Aviation Administration, Department of Transportation

[Docket No. 71-EA-13; Amdt. 39-1155]

PART 39—AIRWORTHINESS DIRECTIVES

American Aviation Corp.

The Federal Aviation Administration is amending § 39.13 of Part 39 of the Federal Aviation Regulations so as to issue an airworthiness directive applicable to

CONTENTION TC-2: EXHIBIT 17

Letter from Donna B. Alexander, CP&L, to U.S. NRC (October 15, 199), enclosing letter from Scott H. Pellet, Holtec International, to Steven Edwards, CP&L (October 11, 1999)

CP&L

OCT 15 1999

Carolina Power & Light Company
Harris Nuclear Plant
PO Box 165
New Hill NC 27562

SERIAL: HNP-99-156

United States Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, DC 20555

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
SUPPLEMENTAL INFORMATION REGARDING THE LICENSE AMENDMENT REQUEST TO PLACE HNP SPENT FUEL POOLS 'C' AND 'D' IN SERVICE

Dear Sir or Madam:

Enclosure 8 of the HNP license amendment request (ref. SERIAL: HNP-98-188, dated December 23, 1998) provided a detailed Alternative Plan for demonstrating compliance with ASME Boiler & Pressure Vessel Code requirements for spent fuel pool cooling and cleanup system piping in accordance with 10 CFR 50.55a(a)(3)(i). By letter dated March 24, 1999, the NRC issued a request for additional information (RAI) related to the Harris Nuclear Plant (HNP) license amendment request to place spent fuel pools C and D in service. The March 24, 1999 RAI included a request to identify each of the embedded field welds within the scope of the Alternative Plan. The HNP response (ref. SERIAL: HNP-99-069, dated April 30, 1999) provided a field weld matrix which identified the field welds to be inspected by using a high resolution remote video camera. The sample size was selected based on a feasibility walkdown, with the camera vendor. CP&L has continued, however, to investigate alternative inspection methods with other vendors. Through these efforts with another vendor, CP&L has successfully performed a remote camera inspection of all 15 embedded field welds included within the scope of the Alternative Plan. In the course of the inspection, two field welds (2-SF-1-FW-3 and 2-SF-1-FW-6) which were not embedded in concrete, but within the scope of the Alternative Plan, were cut out to facilitate removal of piping to provide access for the camera inspections. An updated field weld matrix will be provided to reflect the removal of these two welds and the inspection of all 15 embedded field welds.

In addition, by letter dated April 29, 1999, the NRC issued an RAI related to the criticality control provisions in the HNP license amendment request. Item 1 of this RAI requested information regarding a postulated fresh fuel assembly misloading event. As a supplement to our June 14, 1999 response (ref. SERIAL: HNP-99-094) to requested item 1 of the RAI, we had our vendor, Holtec International, perform additional fuel assembly misloading analyses. The results of these analyses are included as an Enclosure to this letter. These analyses demonstrate that criticality will not occur as a result of the postulated misloading of a fresh fuel assembly in the spent fuel storage racks for HNP pools C and D.

This information is provided as a supplement to our December 23, 1998 license amendment request and does not change our initial determination that the proposed license amendment represents a no significant hazards consideration.

Please refer any questions regarding the enclosed information to Mr. Steven Edwards at (919) 362-2498.

Sincerely,

J.H. Eads for DBA

Donna B. Alexander
Manager, Regulatory Affairs
Harris Nuclear Plant

KWS/kws

Enclosure:

c: (all w/ Enclosure)

Mr. J. B. Brady, NRC Senior Resident Inspector
Mr. Mel Fry, N.C. DEHNR
Mr. R. J. Laufer, NRC Project Manager
Mr. L. A. Reyes, NRC Regional Administrator - Region II

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bc: (all w/ Enclosure)

Mr. K. B. Altman
Mr. G. E. Attarian
Mr. R. H. Bazemore
Mr. C. L. Burton
Mr. S. R. Carr
Mr. J. R. Caves
Mr. H. K. Chernoff (RNP)
Mr. B. H. Clark
Mr. W. F. Conway
Mr. G. W. Davis
Mr. M. J. Devoe
Mr. W. J. Dorman (BNP)
Mr. R. S. Edwards
Mr. R. J. Field
Mr. K. N. Harris

Ms. L. N. Hartz
Mr. W J. Hindman
Mr. C. S. Hinnant
Mr. W. D. Johnson
Mr. G. J. Kline
Mr. B. A. Kruse
Ms. T. A. Head (PE&RAS File)
Mr. R. D. Martin
Mr. T. C. Morton
Mr. J. H. O'Neill, Jr.
Mr. J. S. Scarola
Mr. J. M. Taylor
Nuclear Records
Harris Licensing File
Files: H-X-0511
H-X-0642

HOLTEC INTERNATIONAL

Holtec Center, 555 Lincoln Drive West, Marlton, NJ 08053
Telephone (609) 797-0900
Fax (609) 797-0909

October 11, 1999

Mr. Steven Edwards
Manager of Projects
Carolina Power & Light Company
Harris Nuclear Plant
P.O. Box 165
New Hill, NC 27562

References: Holtec Project 70324
CP&L Contract XTA7000024

Subject: Additional Criticality Analysis Results

Dear Mr. Edwards,

Per your request, and in support of the recent NRC RAIs pertaining to the criticality evaluations performed for fuel storage in pools C and D, we have performed additional analyses.

RAI #1 from the NRC stated that an evaluation of a fuel assembly misloading event should be analyzed. Holtec's previous response drew upon earlier spent fuel rack evaluations and stated that the k_{eff} would remain below 0.95 with a minimum of 400 ppm soluble boron in the pool.

As a supplement to this response, Holtec International has performed additional analyses for the Harris Spent Fuel Pools C and D to determine the amount of soluble boron required to maintain k_{inf} below 0.95 with a misloaded fresh PWR fuel assembly. The results of this analysis are summarized here.

The inadvertent misloading of a fresh PWR fuel assembly into Harris Pools C and D was analyzed using MCNP-4A and CASMO-3. A delta-k_{inf} for the misloading event was calculated using MCNP and this delta-k_{inf} was applied to the maximum k_{inf} in the licensing amendment report (LAR) to determine the maximum k_{inf} under the misloading scenario. This accident scenario consisted of a single 5 wt.% ^{235}U PWR fresh fuel assembly misloaded into the PWR racks surrounded by fuel of maximum reactivity as determined by the burnup and enrichment curve in the LAR. The k_{inf} for the PWR racks with the misloaded fresh assembly, without taking credit for soluble boron, was determined to be 0.9916 with a 95%/95% confidence level.

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Mr. Steven Edwards
Carolina Power & Light Company
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A second scenario was also analyzed in which the fresh 5 wt.% ^{235}U PWR fuel assembly was placed in a PWR storage cell adjacent to the BWR storage racks. The PWR and BWR racks were filled with fuel of maximum permissible reactivity. The k_{inf} for this scenario with the misloaded fresh 5 wt.% ^{235}U PWR fuel assembly, without taking credit for soluble boron, was 0.9932 with a 95%/95% confidence level.

These results clearly demonstrate that the spent fuel pool will remain subcritical even with a fresh 5 wt.% ^{235}U PWR fuel assembly misloaded in the PWR racks.

The April 1978 NRC letter to All Power Reactor Licensees states that "The double contingency principle of ANSI N-16.1-1975 shall be applied. It shall require two unlikely, independent, concurrent events to produce a criticality accident." Consistent with this approach, credit for soluble boron, which is normally in the spent fuel pool, was taken when the misloaded fresh 5 wt.% ^{235}U PWR fuel was analyzed. It was determined that the maximum k_{inf} for the misloading accident is 0.9352 with 400 ppm soluble boron in the spent fuel pool water. Therefore, the minimum amount of soluble boron required to maintain k_{inf} less than the regulatory limit of 0.95 under all postulated abnormal and accident conditions is 400 ppm.

Additional calculations were also performed to determine the k_{inf} for the misloading accident with 1000 and 2000 ppm soluble boron in the spent fuel pool water. The maximum k_{inf} was calculated to be 0.8671 and 0.7783 for the 1000 and 2000 ppm respectively. These results demonstrate that there is considerable un-credited margin in the criticality analysis of Harris Spent Fuel Pools C and D.

If you have any questions please feel free to contact me.

Sincerely,

Scott H. Pellet

Scott H. Pellet
Project Manager

cc: Holtec Engineering File 80964
Holtec Contracts file

Document ID: 80964SP1

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555-0001

June 25, 1999

NRC INFORMATION NOTICE 99-21: RECENT PLANT EVENTS CAUSED BY HUMAN PERFORMANCE ERRORS

Addressees

All holders of licenses for nuclear power, test, and research reactors.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice to ale addressees to a recently apparent increase in human performance weaknesses that have resulted in plant transients. It is expected that recipients will review the informat applicability to their facilities and consider actions, as appropriate, to prevent a s occurrence. However, suggestions contained in this information notice are not NRC requirements; therefore, no specific action or written response to this notice is requ

Description of Circumstances

Salem Unit 1

At 1:38 a.m. on February 28, 1999, the Salem Unit 1 reactor automatically shut down be of a low bearing oil pressure turbine trip. The unit was operating at 60-percent powe the shutdown and was being maintained at this power to allow troubleshooting to be per on a main feedwater pump. Preparations were also being made to allow maintenance to r a leaking main turbine lube oil cooler. One of the two oil coolers had developed a le previous shift, and the operators were adjusting the position of a cooler isolation va attempt to more tightly close the valve.

While adjusting the isolation valve, the operators inadvertently positioned the valve seat, allowing oil from the in-service cooler to enter the partially drained out-of-se This diverted flow caused a momentary drop in the turbine bearing oil pressure and res the automatic main turbine trip and subsequent reactor trip.

The cause of the transient has been attributed to misoperation of the cooler isolation The valve used to swap the main turbine lube oil coolers is a Schutte & Koerting six-w isolation valve. This type of valve is only used for the main turbine lube oil cooler Units 1 and 2, and the valve is operated very infrequently. The operators did not kno attempts to more tightly close the valve would result in moving the valve off its clos

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The operators responded to the automatic shutdown as directed by the plant's emergency operating procedures and the unit was stabilized in a shutdown condition.

Diablo Canyon Unit 1

At 5:06 p.m. on February 25, 1999, the Unit 1 annunciator alarmed in the control room "Spent Fuel Pool Level/Temperature." Operators verified the alarm by checking the pla computer, which indicated an elevated temperature of 125 degrees F in the spent fuel p

The shift foreman dispatched a nuclear operator to the spent fuel pool area. The nucl operator noted that the local spent fuel pool temperature gauge indicated 126 degrees nuclear operator subsequently found that spent fuel pool pump 1-2 was not operating as expected and restarted the pump at the direction of the shift foreman.

The licensee's investigation into the event revealed that operator logs prepared earli February 25, 1999, had verified that the spent fuel pool pump 1-2 was operating as req and that spent fuel pool temperature was 100 degrees F. Further investigation reveale during the day, relay CIAX-H was replaced. This relay was associated with the contain Phase A isolation signal. The control circuit associated with the CIAX relay trips th pool cooling pumps during an accident scenario to prevent overloading of the emergency generators. The relay had been replaced at approximately 1 p.m., and as a result, spe pool cooling had been lost for approximately 4 hours before the high level/temperature was received in the control room. Licensee engineers determined that the spent fuel p heatup rate was approximately 8 degrees F per hour and would have resulted in spent fu boiling after approximately 16 hours.

A review of the work order associated with the relay replacement revealed that the cle associated with the procedure did not contain any precautions or limitations to notify operators of the trip of the spent fuel pool cooling pump as a result of removal of th The pre-job briefing apparently did not identify the condition, nor were the operators electricians who performed the relay replacement aware of the resultant condition of t spent fuel pool cooling pumps.

A second factor that appears to have contributed to the duration of the event was a la controls or indications in the control room of the status of the spent fuel pool cooli temperature of the spent fuel pool, or the level of the spent fuel pool, other than t aforementioned level/temperature alarm. These indications and controls were available in the spent fuel pool area but, as directed by plant procedures, were required to be and logged only once every 12 hours during operator rounds.

Vogtle Unit 2

At 2:07 a.m. on March 2, 1999, operators manually shut down the Unit 2 reactor from 100-percent power because of an observed low water level in the #3 steam generator and

concurrent alarm of the "Steam Flow/Feed Flow Mismatch" annunciator. The cause of the decrease was due to the unexpected closing of the Unit 2 loop 3 main feedwater isolati The loop 3 main feedwater isolation valve closed because plant equipment operators mistakenly pulled the control power fuses to the Unit 2 isolation valve while hanging clearance tag on the Unit 1 isolation valve.

Following the event, the licensee initiated a root cause analysis to determine the cau operator performance errors and determined that multiple factors contributed to the ev as failure to implement self-checking using the dual concurrent verification (i.e., bo were present, performed the function, and verified the correctness of the actions); la feedback between the operators regarding the complete component identification tag num including unit designation; and work schedule factors (one of the operators was workin sixth 12-hour shift of nine scheduled consecutive twelve hour shifts).

San Onofre Unit 2

At 9:59 a.m. on February 1, 1999, a loss of shutdown cooling occurred at San Onofre Un
The unit was in mode 6 and refueling was in progress. Before the event occurred, the
4.16-kV vital bus 2A04 was being fed from the offsite transmission system by the unit
transformer. Train A bus 2A04 was the protected supply to the operating shutdown cool
pump and to the containment spray pump which was providing spent fuel pool cooling.

At the time of the event, the licensee was implementing a clearance order to facilitat
maintenance on the reserve auxiliary transformer, which was an alternate power supply
Train A 4.1-kV bus 2A04. The clearance called for racking out the already opened Trai
4.16-kV breaker to the reserve auxiliary transformer. In preparation for the activiti
reserve auxiliary transformer was disconnected from the switchyard and all three groun
disconnect switches on the primary side of the transformer (220-kV side) were closed.
Subsequently, while attempting to rack out the breaker, electricians performing the wo
that the breaker was stuck and would not disengage.

Licensee personnel involved with the evolution discussed the matter and incorrectly co
that discharging the closing springs would prevent the breaker from inadvertently clos
attempting to again rack out the breaker. The operators and electricians involved in
believed that pushing a lever that discharges the closing springs would not cause the
close. They based this belief on previous experience with using this button while the
was in the racked-out position and not having the breaker close as a result. However,
electricians performed the action on the racked-in breaker, the breaker did close. Th
in the grounded high side of the reserve auxiliary transformer becoming a near-infinit
the low side, which was being supplied by Bus 2A04 through the now closed breaker. Th
created an undervoltage condition on Bus 2A04. All of the supply breakers for the aff
tripped open, except for the breaker to the reserve transformer which was in an off no
configuration due to the actions of the electricians.

The standby emergency diesel generator automatically started but did not close onto th
affected bus because of a protective relay lockout that prevented more than one feed t
at any one time. The standby emergency diesel generator was not designed to be capabl
maintaining bus voltage under these circumstances. As a result, the affected bus deen
thereby causing a loss of the shutdown cooling and spent fuel pool cooling functions
approximately 26 minutes.

Following the event, the licensee initiated an investigation and determined that the p
directing the grounding of the high side of the reserve auxiliary breaker before racki
4.16-kV breakers was inadequate in that the order of the activities should have been r
Additionally, it was determined that although the plant personnel and management invol
recognized the potential for serious consequences if the breaker inadvertently closed,
planning and control of the evolution did not adequately reflect the increase in risk
with these activities.

Discussion

The NRC has noticed an apparent increase in human performance related events that have
resulted in plant transients. The four examples described above represent a sample of
recent events in which human performance played a key role, and each highlights the no
challenges that human performance weaknesses may present to plant operation. The
importance of human error in determining risk from nuclear power plants is well known
discussed in NUREG/CR-5319, "Risk Sensitivity to Human Error", April 1989. NUREG/CR-5

"Risk Sensitivity to Human Error in the LaSalle PRA", March 1990, presents detailed ri sensitivity studies involving human performance that had previously shown notable sens risk to changes in human error probabilities. In light of these findings, there appea large risk incentive to ensuring that human performance does not degrade below the performance level assumed in the plant-specific probabilistic risk assessments and rem consistent with licensee management expectations.

IN 99-21
June 25, 1999
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This information notice requires no specific action or written response. However, rec reminded that they are required by 10 CFR 50.65 to take industry-wide operating experi (including information presented in NRC information notices) into consideration, when when setting goals and performing periodic evaluations. If you have any questions abo information in this notice, please contact one of the technical contacts listed below appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

/s/'d by J..E. Lyons
for Ledyard B. Marsh, Chief
Events Assessment, Generic
and Non-Power Reactors Branch
Division of Regulatory Improvement Programs
Office of Nuclear Reactor

Technical contacts: Greg S. Galletti, NRR
301-415-1831
E-mail: gsg@nrc.gov

Nick Fields, NRR
301-415-1173
E-mail: enf@nrc.gov

Attachments: List of Recently Issued NRC Information Notices

Date: 3/17/00 1:07 PM
Sender: DaveL
To: nancyburtonesq@hotmail.com
irss@igc.apc.org
Priority: Normal
Subject: Millstone Documents

Hello Nancy and Gordon:

I spent several hours the past three days reviewing our files and collecting documents for use in the Millstone 3 spent fuel pool case. The attached listing covers the documents I may use to support Contentions 4, 5, and 6. I am mailing copies of every document on this listing to both of you.

We should list these documents in our discovery response.

I'll try to come up with a list of waht we should ask for in our discovery request.

Thanks,

Dave Lochbaum
Nuclear Safety Engineer
Union of Concerned Scientists
1616 P Street NW Suite 310
Washington, DC 20036
(202) 332-0900
(202) 332-0905 fax
website: www.ucsusa.org

Millstone 3 Discovery Documents.doc

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555-0001

September 3, 1997

NRC INFORMATION NOTICE 97-68: LOSS OF CONTROL OF DIVER IN A SPENT FUEL STORAGE POOL

Addressees

Holders of a facility license or construction permit issued for a power reactor pursuant to 10 CFR Part 50.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice to inform addressees of inadequacies in licensee control of work in a spent fuel storage pool at a power reactor facility which resulted in a diver getting close to very high radiation fields emanating from recently discharged spent fuel. It is expected that recipients will review the information in this notice for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, suggestions contained in this information notice are not NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances

On April 3, 1997, the Calvert Cliffs Unit 2 facility, owned and operated by Baltimore Gas and Electric Company (the licensee), was in Mode 6 with reactor defueling on hold because of a malfunction of the Unit 2 fuel transfer system. The fourth in a series of diving activities to effect repairs to the fuel transfer system was conducted in the spent fuel pool. Previous dives had been made in the refueling cavity to repair the system. The diver entered the spent fuel pool at about 9:00 a.m. to commence work on an upender limit switch at the south end of the fuel transfer area, the only surveyed and authorized work area. The fuel transfer area runs the length of the west side of the Unit 2 spent fuel pool. No wall or shield (other than the pool water) separates the area from the fuel storage racks on the east side.

As with the previous dives, normal diving controls were specified by a licensee-approved procedure and a job-specific radiation work permit. Multiple thermoluminescence dosimeters (TLDs) were attached to the diver's wrists, head, chest, back, and thighs and feet. Monitoring of the diverþs dose was provided in real time with teledosimetry devices attached to his wrists, thighs (above knee), chest, and back. The diver was also provided with two radiation detector probes attached to a shaft approximately 76 cm (30 in) long for the purpose of surveying his immediate vicinity. Each teledosimetry device was set to alarm at the surface monitoring station on detecting an integrated dose of 1.0 mSv (100 mrem) or a dose rate of 8.95 mSv/hr (895 mrem/hr). Radiation protection (RP) technicians continuously

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monitored the instrument readouts, and relayed the information through an intercom to the diver who had no local indication of monitor readings or alarms. Unlike previous dives into the refueling cavity which employed underwater closed-circuit television (video) to visually monitor the diver, a technician at the pool surface was assigned to observe the diver through a floating window box during the fourth dive.

Following the repair of the limit switch, the diver asked for some materials that he needed to complete the work. While he was waiting for the materials to be lowered, the diver told the support team that he wanted to inspect a kink in the upender cable. He was referring to a cable kink at the north end of the pool. However, the RP technicians assumed that the diver was referring to the cable in the authorized work area. Accordingly, the health physicist

(HP) technician approved the request, and the diver headed toward the north end of the pool. At the north edge of the authorized work zone, the diver inflated his diving suit, ascended and hovered above the pool floor to inspect cabling. He then vented his suit, descended to the pool floor and continued toward the north leaving the authorized area. The observer at the surface did not detect the diverþs unauthorized entry into the north end of the fuel transfer area because the vented air bubbles ascending through the pool water obscured his view of the diver. The observer subsequently was distracted with other duties and never regained visual contact with diver. Therefore, the dive tender continued to provide cable and breathing air line to the diver unchallenged.

Near the north end of the transfer system, the diver stopped to survey a pipe on the west wall of the pool that he did not recognize. During the survey, the monitors on the diver's right and left wrists alarmed and increased to 90 mGy/hr (9 rad/hr) and 23 mGy/hr (2.3 rad/hr) respectively. The RP technicians instructed the diver to retreat to a lower dose area. The RP technician was not aware that the diver had actually encountered the radiation field from recently off-loaded spent fuel located in the racks on the east side of the transfer area. Still believing that the diver was at the south end of the pool, the RP technician instructed the diver to survey the area to locate the source of the unexpected radiation. When the survey meter readout increased to 30 mSv/hr (3 rem/hr), the dive was suspended. Only after the diver surfaced, did the RP personnel realize that the diver had actually been in the north end of the pool. The subsequent assessment of the event revealed that the diver crossed about 4.6 meters (15 feet) of unsurveyed fuel transfer area floor and came within a few feet of radiation dose rates ranging from 120 to 200 Gy/hr (12,000 to 20,000 rad/hr).

The diver's TLDs were subsequently processed, but not before he was allowed to re-enter the radiation control area (RCA) to support another diving operation as a standby safety diver. The licensee allowed the re-entry to the RCA prior to dosimetry processing based on a preliminary assessment that the teledosimetry readings indicated that the diver received no significant radiation exposure.

Following TLD processing, the licensee calculated a maximum dose to the extremities (right knuckles) of 8.85 mSv (885 mrem) based on a wrist TLD badge shallow dose equivalent result of 4.24 mSv (424 mrem). The licensee also calculated a dose of 2.7 mSv (270 mrem) to the highest exposed portion of the whole body (arm above the elbow) as compared to a

maximum TLD reading on the head of 1.37 mSv (137 mrem). The maximum dose to the lower extremity (ankle) was 0.021 mSv (21 mrem) shallow dose equivalent.

Discussion

The NRC has noted several deficiencies in the preplanning and controls implemented to support the April 3, 1997, diving operations at Calvert Cliffs. These include:

1. The scope of work was not clearly understood by all parties involved.

   During the formal pre-job briefing, on the morning of the dive, it was noted that the scope of work included an inspection of the cable kink at the north end of the transfer mechanism þif radio-logical conditions permitted.þ The RP personnel in attendance were not sure if the radiation survey made to support the dive covered the north end of the pool. After the briefing, the RP supervisor determined that the survey was limited to the south end of the fuel transfer area and informed the dive engineering support personnel that work in the north of the pool was not authorized. No one gave this information to the diver or the dive tender.

2. The diver was given inadequate instructions about the location and magnitude of the radiation sources accessible to him.

   A second communications failure took place at the dive site when the RP technician briefed the diver on the radiation levels in the work area. A map indicating the results of the radiation survey was shown to the diver. However, this map was an enlarged view of the south end of the transfer area. Due to a lack of perspective, the diver believed he was being shown the radiation levels in the entire pool. This mis-communication reinforced the diver's incorrect understanding of the scope of the authorized work.

3. Positive control over the diver in the pool was inadequate.

   Guidance on effective access control over divers in the spent fuel pool is given in Regulatory Guide (RG) 8.38, "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants," and in such industry standards as the Electric Power Research Institute's (EPRI's) "Underwater Maintenance Guide" (EPRI NP-7088-R2). Appendix A to RG 8.38 discusses six areas of concern where control needs to be exercised over diving operations. This list is a compilation of the result of lessons learned from previous diving events at nuclear power plants. The licensee failed to implement effective controls in five of these six areas.

   At the time visual contact with the diver was lost, the licensee had, in effect, lost control of the dive. As stated in Section 1.5.4 of the EPRI guide, visual contact should be maintained throughout the entire dive to be sure of the

   diver's location and proximity to all known underwater radiation sources. The licensee failed to recognize the significance of maintaining visual contact with the diver. The inattentiveness and lack of a questioning attitude by the dive support personnel contributed significantly to the loss of control.

   The licensee's investigation of this event determined that several of the people involved did not clearly understand the scope of their responsibilities for the planning and conduct of the diving operation. One of the root causes identified was the practice at Calvert Cliffs of providing management expectations on how a task is to be performed in documents other than the formal job procedure. For example, the licensee found that the individual assigned to observe the diver did not understand that he was to continuously observe the diver and thought that this task was optional since it was not stated in the procedure. The requirement to maintain visual contact with the diver was in an RP Job Coverage Standard instead of in a formal procedure. The licensee is revising plant procedures so that they will contain all critical steps needed to exercise adequate controls over on-site work.

4. Licensee failed to adequately evaluate the diverþs exposure status before authorizing additional work in the RCA.

Given the complexity of the diving environment, the licensee's assessment of the diverþs dose based on the teledosimetry readings was not sufficiently comprehensive. Teledosimetry located on the diverþs thigh is not adequate to determine whether an overexposure to the diver's extremity occurred during this event. The diver could have received a high dose to his feet while walking across the unsurveyed section of the pool floor without exceeding the alarm setpoint on the thigh monitors because of the shielding provided by the pool water. In addition, the estimate of the whole body dose did not consider the possibility of exposure to neutron radiation since the detectors were not sensitive to neutrons. Subcritical spent fuel is a significant neutron source due to alpha-n reactions and spontaneous fission of curium in the fuel. In response to the NRC inspectorþs questions, the licensee subsequently determined that the diver would have to be within 0.6 meters (2 feet) of the fuel for neutrons to be a factor. The TLD readings verified that the diver received no measurable neutron dose.

Although it appears that the radiation doses received by the diver did not exceed the dose limits given in 10 CFR Part 20, the breakdowns noted above resulted in the diver being able to gain access to a very high radiation area contrary to the requirements of 10 CFR 20.1602. During normal operations spent fuel pools are neither high nor very high radiation areas since the radioactive sources in them are usually covered by at least 3.3 meters (10 feet) of water and are thus considered inaccessible to personnel (see Regulatory Position 4.2 in RG 8.38). However, consistent with Regulatory Position 1.5 in RG 8.38, once an inaccessible area is made accessible, in this case by conducting diving operations, the applicable controls for a high or very high radiation area must be provided. This includes the access control

requirements of 10 CFR 20.1601 and 20.1602 as well as appropriate posting at the entrance to the area consistent with the requirements of 10 CFR 20.1902.

The combination of an extremely intense radiation source and the very steep dose gradients that can be encountered as a diver moves through his shielding (water), make diving in areas where irradiated fuel can be accessed a uniquely hazardous operation. Had the circumstances of this event been only slightly altered, the diver could have been exposed to much higher dose rates. Even with continuous teledosimetry monitoring, it is possible for a diver to inadvertently enter a radiation field and receive a serious radiation dose, in a matter of seconds. Establishing and maintaining proper effective controls is critical to worker safety.

Related NRC Communications and Correspondence

The following related communications and correspondence are noted:

- NRC Information Notice 82-31, "Overexposure of Diver During Work in Fuel Storage Pool," July 28, 1982

- NRC Information Notice 84-61, "Overexposure of Diver in Pressurized Water Reactor (PWR) Refueling Cavity," August 8, 1984

- NRC Regulatory Guide 8.38, "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants," June 1993, Appendix A, "Procedure for Diving Operations in High and Very High Radiation Areas"

- NRC Inspection Reports 50-317/97-02 and 50-318/97-02, May 29, 1997

- NRC Enforcement Action EA97-192 dated August 11, 1997

This information notice does not require any specific action or written response. If you have any questions about the information in this notice, please contact one of the technical contacts listed below.

signed by

Jack W. Roe, Acting Director
Division of Reactor Program Management
Office of Nuclear Reactor Regulation

Technical contacts: Ronald L. Nimitz, RI
(610) 337-5267
E-mail: rln@nrc.gov

John R. White, RI
(610) 337-5114
Email: jrw@brc.gov

Roger L. Pedersen, NRR
301-415-3162
E-mail: rlp@nrc.gov

Attachment: List of Recently Issued NRC Information Notices

United States Nuclear Regulatory Commission
Office of Public Affairs
Washington, DC 20555
Phone 301-415-8200 Fax 301-415-2234
Internet:opa@nrc.gov

No. 96-74

FOR IMMEDIATE RELEASE
(Wednesday, May 22, 1996)

NRC STAFF COMPLETES SURVEY OF REFUELING PRACTICES AT NATION'S NUCLEAR POWER PLANTS

A Nuclear Regulatory Commission staff survey has found that most of the 110 nucle power plants in the nation are conducting refueling practices consistent with their li requirements.

The survey further found, however, that 15 nuclear power plants at nine sites may have acted consistent with license commitments when they moved all the fuel from the r the spent fuel storage pool during refueling, rather than a partial offload. Since th completed, utilities which operate these plants have either updated their license docu reflect the refueling practices or have firm commitments to NRC that such action will completed before the next refueling outage.

Unloading the full reactor core during refueling is practiced by the majority of power plant licensees. NRC considers the practice one which can be beneficial to safe particularly in reducing the hazard to workers who perform maintenance in or around th vessel during outages. The agency's concern is that, before it is used, the practice analyzed and documented.

NRC staff conducted this survey as part of a wider project to measure the extent problems encountered at the Millstone Unit 1 plant in Connecticut exist at other facil

A copy of the survey report is attached.

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Attachment:
As stated

May 21, 1996

MEMORANDUM TO: Chairman Jackson
Commissioner Rogers
Commissioner Dicus

FROM: James M. Taylor
Executive Director for Operations

SUBJECT: REPORT ON SURVEY OF REFUELING PRACTICES

In my memorandum to Chairman Jackson dated December 28, 1995, I committed the staff to activities that would measure the extent to which problems encountered at Millstone Un compliance with the Final Safety Analysis Report (FSAR) existed at other facilities. which is the focus of the attached report, compared current refueling practices agains (drawn from the FSAR, Technical Specifications, license amendments and other docketed decay heat removal from spent fuel pools for all operating reactors. The second activ licensee compliance with other aspects of the facility description contained in the FS inspection guidance. The results of this latter activity will be presented separately

The staff's goal was to complete the spent fuel pool related survey for those plants w before refueling and no later than May 1996 for all other facilities. Although there that had started refueling before the staff's review, the review results from these pl with the comments and findings noted herein for other plants.

As described in the attached report, the staff has completed its review of core offloa operating reactor. Based on the survey, the staff has concluded that plants have spen systems and backup cooling capability that the NRC staff had reviewed and approved. S and licensee operating practices were found to be adequate in assuring protection for safety. It is noted that margins of safety, although adequate, and the clarity of req plant to plant. In addition, the staff concludes that, based on the information colle the specific licensee actions taken and commitments made during the course of this rev practices are currently consistent with the spent fuel pool decay heat removal licensi plants or will be prior to the next refueling outage. However, during the course of t determined that nine sites (fifteen units) needed to modify their licensing bases or p pursuant to 10 CFR 50.59 or 10 CFR 50.90, to ensure that their reload practices were i their licensing basis. This is an indication that, similar to Millstone, Unit 1, a nu appear to have previously performed full core offloads inconsistent with their licensi

To gain additional perspective on these nine sites, the staff is examining the results scope FSAR compliance regional inspection activities to see if these sites show eviden programmatic FSAR non-compliance problems. The broad base FSAR compliance review at a comparison of existing FSAR compliance data for the nine sites documented in the attac report is ongoing. The results of these activities will be presented separately.

Due to previously identified concerns regarding FSAR compliance at the Millstone and H staff has issued letters pursuant to 10 CFR 50.54(f) to Northeast Utilities regarding FSAR for those plants. The problems identified to date at the shutdown Millstone Unit broader in scope and more serious in nature than the core offload compliance discrepan attached report. The staff has not identified any concerns at Haddam Neck regarding c core offload and spent fuel pool decay heat removal licensing basis as a result of its survey. However, through the 10 CFR 50.54(f) request mentioned above, the staff is se which is broader in scope than the spent fuel pool survey and the resulting informatio alter our current findings on Haddam Neck.

In addition to the nine plants mentioned above, the staff noted that the FSARs for ten units) did not reflect the most recent licensing basis information as required by 10 C since the affected information was already captured elsewhere in the licensing basis (submittals and staff safety evaluations), the staff considers the non-timely updating indication of administrative program failures to maintain plant documentation. Such a failures could have safety significance if they were widespread and resulted in violat requirements such as 10 CFR 50.59.

The staff is taking steps to ensure that the details of the staff findings for these p in inspection reports. It is expected that the characterization of the report finding plants may be revised as the staff completes the detailed documentation activity. Th documentation of spent fuel pool survey discrepancies in plant-specific inspection rep Concurrently, the staff is developing enforcement guidance to address the instances of the FSAR identified in the spent fuel pool survey and the broader FSAR compliance revi any enforcement guidance to the findings of potential non-compliance identified throug would follow accordingly.

In addition to the compliance issues identified, the staff will review the data collec survey and will identify specific plant design features and operating practices which safety enhancements using the backfit process (10 CFR 50.109). An example of potentia design enhancement is spent fuel pool instrumentation. The staff will develop its pla backfit activities or generic improvements in the regulation of spent fuel pool decay

After addressing the compliance issues and potential safety enhancements, the staff wi improve the clarity and consistency of our spent fuel pool requirements. One approach consideration is to include spent fuel pool design and operational issues in the shutd activity which is already well underway.

This will be made publicly available in five working days from the date of this memo.

Attachment: Refueling Practice Survey: Final Report

cc: SECY
OGC
OCA

REFUELING PRACTICE SURVEY: FINAL REPORT

1.0 INTRODUCTION

In July 1995, Northeast Utilities submitted a proposed license amendment for Millstone that the staff approve a full core offload as a normal refueling practice, approve cer analyzing decay heat removal, and approve certain new technical specification (TS) req 1995, a petition was filed, pursuant to 10 CFR 2.206 which sought, among other things, Millstone 1 license amendment request. The staff conducted an extensive review of the and the issues raised in the 2.206 petition. To address the amendment request and the examined a number of design issues, operating and administrative procedural issues, an issues.

One of the fundamental issues in both proceedings concerned the operational limits on as documented in the Millstone 1 licensing basis. The Millstone 1 spent fuel pool coo designed to remove heat from the spent fuel pool, and the system design capabilities a various licensing documents. During the review, the staff became concerned about whet practice of conducting full core offloads was consistent with the licensing basis of t decay heat removal systems. The staff was concerned that routine refueling offload pr been consistent with licensing basis assumptions regarding routine or normal operation

Through the fall of 1995, the staff became aware of several discrepancies between the pool licensing basis and the refueling practices at other nuclear plants. The staff i Notice 95-54, "Decay Heat Management Practices During Refueling Outages," dated Decemb the industry of the discrepancies observed at Millstone 1 and the other facilities. B ongoing "Task Action Plan for Spent Fuel Storage Pool Safety," which is intended to ad concerns, and the continued confidence of the staff in the substantial, though variabl provided by the design of existing spent fuel pools and their associated support syste consider immediate regulatory actions appropriate to address the observed licensing ba

On December 28, 1995, the staff forwarded a memorandum to the NRC Chairman on the less Millstone 1 review. In this memorandum, the staff committed to review the refueling p operating reactors against the current spent fuel pool decay heat removal licensing ba the Final Safety Analysis Report (FSAR) and other documents) for those facilities. Th review each operating plant before the next scheduled refueling outage, but no later t plants. Although there were several plants that had started refueling before the staf results from these plants were consistent with the comments and findings noted herein

To meet the spent fuel pool survey commitments described in the December 28, 1995, mem developed a program to evaluate each operating reactor's refueling practices relative current spent fuel pool decay heat removal licensing basis. Table 1 lists the plants the survey as well as the routine offload practice at that plant.

2.1 Licensing Basis Review

To determine the licensing basis, the staff reviewed relevant licensing documents that capability to remove decay heat from the spent fuel pool including:

- relevant Final Safety Analysis Report (FSAR) sections
- relevant Technical Specifications
- documentation associated with license amendments related to the spent fuel pool (this are amendments that increase the licensed storage capacity of the spent fuel amendments), amendments that increase a reactor's licensed thermal power ("power amendments), and amendments that increase the licensed enrichment or energy produ ("burnup") of fuel.)
- other docketed correspondence addressing spent fuel pool decay heat removal capab further defined the systems or practices

The staff focussed on certain specific areas in determining the licensing requirements

- descriptive phrases that imply the frequency of certain offload sequences (e.g., emergency)
- configuration of spent fuel pool cooling systems assumed to be operating in the d (e.g., single failure considerations, backup systems)
- assumptions that affected spent fuel pool heat load (e.g., delay time and operati
- plant specific spent fuel pool temperature limits and the bases for those limits.

The staff observed that the licensing basis for spent fuel pool decay heat removal var This variation in licensing bases stems from differences in spent fuel decay heat remo accepted by the staff, from differences in the level of detail provided in the licensi evolving NRC review criteria. As a result, the margin of safety with respect to syste reliability differs from plant to plant.

2.2 Operating Practice Review

To determine the operating practices of each operating reactor during refueling, the s site or the utility corporate office and reviewed plant specific operating procedures, controls and engineering analyses. The staff compared the operating practices with th developed for each operating reactor to identify discrepancies with the design basis a spent fuel pool cooling system.

As part of the survey, the staff also gathered detailed design information on the spen facilities at all operating reactors. This information will be used in developing pla the staff's "Task Action Plan for Spent Fuel Storage Pool Safety."

3.0 COMMENTS AND FINDINGS

Operating Practice Reviews

Except as noted in the attached Tables 2 and 3, the staff found that for most plants, were consistent with the licensing basis and the FSAR reflected most recent changes to Also, applicable procedures and administrative controls were found to be adequate. Th however, that many plants were considering or processing changes to the FSAR to correc inconsistencies and were enhancing applicable procedures to better control outage oper

However refueling practices at several plants were potentially in conflict with that p decay heat removal licensing basis. Plants that the staff concluded were in this cond Table 2. In some cases, the licensee had independently recognized the discrepancy and the plant's spent fuel pool decay heat removal design basis or the plant's refueling p cases, the staff promptly informed the licensee about its understanding of both the pl and the plant's operating practices and identified the discrepancies between the two. affected licensees took action or committed to take action to reconcile refueling offl licensing requirements before the next core offload activity took place.

Because these plants may have operated outside their licensing basis during previous r staff will document the detailed findings for these facilities in appropriate inspecti take appropriate regulatory action.

Periodic FSAR Updates

Some plant-specific FSARs did not reflect information associated with spent fuel pool from applicable license amendments. An NRC regulation, specifically 10 CFR 50.71(e), be periodically updated to reflect such information. Plants where this discrepancy wa in Table 3 although such plants may be within the update periodicity provided in 10 CF may not be in violation of 10 CFR 50.71(e). The staff will document the detailed find facilities in appropriate inspection reports and will take appropriate regulatory acti

Control of Design Basis Assumptions

At a number of plants, the staff identified weaknesses in the procedural control of de assumptions. An example of this includes weak procedure control of in-vessel decay ti transfer. In some cases, licensees upgraded procedures to directly implement the desi in other cases the licensee revised existing procedures and performed engineering anal pursuant to 10 CFR 50.59, as necessary, to ensure the planned activities would not exc assumptions. In addition, some plants perform outage-specific analyses as a matter of plants performed (or plan to perform) such analyses in response to the staff's survey.

Other Observations

A number of facilities were found to remove spent fuel pool cooling systems and/or sup service for maintenance during refueling outages. While the staff did not identify a of potential non-compliance in this regard, the staff will consider the appropriate le controls during refueling outages as part of the Task Action Plan on Spent Fuel Storag an issue may be appropriately addressed within the context of shutdown risk rulemaking

Also, a number of facilities had installed significant additional spent fuel pool deca capability pursuant to 10 CFR 50.59 (Table 4). For example, one plant installed a two decay heat removal system such that either train could reject the heat associated with offload. The system was installed to facilitate performance of full core offloads whi simultaneous maintenance of RHR systems which might otherwise be needed for spent fuel

The staff reviewed the 10 CFR 50.59 evaluations for several of these modifications and In general, it is expected that this kind of modification could be performed under 50. approval. However, plant specific circumstances may require a license amendment.

4. CONCLUSIONS

During the course of the survey, the staff evaluated the compliance of refueling opera operating reactor with respect to that plant's spent fuel decay heat removal licensing concludes that, based on the information collected and reviewed and the specific licen commitments made during the course of this review, refueling operating practices are c with the spent fuel pool decay heat removal licensing basis for all plants or will be refueling outage. However, the survey determined that nine sites (fifteen units) nee licensing bases or plant practices, pursuant to 10 CFR 50.59 or 10 CFR 50.90, during t review to ensure that their reload practices were in compliance with their licensing b indication that some of these plants may have previously performed core offloads incon licensing basis. In addition, the staff noted that the Final Safety Analysis Reports (eighteen units) did not reflect the most recent licensing basis information.

The staff is taking steps to ensure that the details of the staff findings for these p in inspection reports. It is expected that the characterization of the report finding plants may be revised as the staff completes the detailed documentation activity. Th documentation of spent fuel pool survey discrepancies in plant-specific inspection rep Concurrently, the staff is developing enforcement guidance to address the instances of the FSAR. Application and implementation of any enforcement activities to the finding

The staff did not identify any safety issues regarding spent fuel pool decay heat remo practices that have not been captured for resolution as part of the staff's "Task Acti Storage Pool Safety." As part of the action plan, the staff concluded that the existi operation of spent fuel pool systems do not pose an undue risk to public health and sa variation in spent fuel pool cooling system design bases accepted by the staff in the concluded that compliance with design limits does not reflect a consistent margin of s this conclusion and other technical concerns, the staff is examining spent fuel pool d identify safety enhancements through the implementation of the staff's action plan for pool safety.

During the course of the survey, the staff also collected detailed design information systems for all operating reactors. This information will be used in developing plans staff's "Task Action Plan for Spent Fuel Storage Pool Safety." Plans for resolving ac well as separate plans to address the license compliance issues described in this repo by June 28, 1996.

Table 1: Routine Offload Practices

PLANT	OFFLOAD METHOD
ANO-1	Full Core
ANO-2	Full Core
Big Rock Point	Full Core
Beaver Valley 1,2	Full Core
Braidwood 1,2	Full Core
Browns Ferry 1,2,3	Full Core
Brunswick 1,2	Full Core
Byron 1,2	Full Core
Callaway	Full Core
Calvert Cliffs 1,2	Partial Core
Catawba 1,2	Full Core
Clinton	Partial Core
Comanche Peak 1,2	Full Core
D.C. Cook 1,2	Full Core
Cooper	Full Core
Crystal River	Full Core
Davis Besse	Full Core
Diablo Canyon 1,2	Full Core
Dresden 2,3	Full Core
Duane Arnold	Full Core
Farley 1,2	Full Core
Fermi 2	Full Core
Fitzpatrick	Full Core
Fort Calhoun	Full Core
Ginna	Full Core
Grand Gulf	Partial Core
Haddam Neck	Full Core
Harris	Full Core
Hatch 1,2	Full Core
Hope Creek	Partial Core
Indian Point 2	Full Core
Indian Point 3	Full Core
Kewaunee	Partial Core

LaSalle 1,2	Full Core
Limerick 1,2	Partial Core
Maine Yankee	Full Core
McGuire 1,2	Full Core
Millstone 1	Full Core
Millstone 2	Partial Core
Millstone 3	Full Core
Monticello	Partial Core
Nine Mile Point 1,2	Partial Core
North Anna 1,2	Full Core
Oconee 1,2,3	Full Core
Oyster Creek	Partial Core
Palisades	Partial Core
Palo Verde 1,2,3	Full Core

PLANT	OFFLOAD METHOD
Peach Bottom 2,3	Partial Core
Perry	Partial Core
Pilgrim	Partial Core
Point Beach 1,2	Full Core
Prairie Island 1,2	Partial Core
Quad Cities 1,2	Full Core
River Bend	Partial Core
Robinson	Full Core
Salem 1,2	Full Core
San Onofre 2,3	Full Core
Seabrook	Full Core
Sequoyah 1,2	Full Core
South Texas 1,2	Full Core
St. Lucie 1,2	Partial Core
Summer	Full Core
Surry 1,2	Full Core
Susquehanna 1,2	Partial Core
TMI-1	Partial Core
Turkey Point 3,4	Full Core
Vermont Yankee	Partial Core
Vogtle 1,2	Full Core
Waterford	Partial Core
Watts Bar	Full Core
WNP-2	Partial Core
Wolf Creek	Full Core
Zion 1,2	Full Core

Table 2: Past Offloads in Potential Non-Compliance with Current Licensing Basis

Cooper
McGuire 1,2
Millstone 1
North Anna 1,2
Oconee 1,2,3
South Texas 1,2
Summer
Turkey Point 3,4
Vogtle 1

Table 3: FSAR Update Needed to Achievew Consistence Within Licensing Basis

Browns Ferry 1,2,3
Crystal River
Fermi 2
Kewaunee
LaSalle 1,2
Millstone 1,2,3
Salem 1,2
Sequoyah 1,2
Vermont Yankee
Zion 1,2

Table 4: Significant Plant Modifications Under 50.59

Brunswick 1,2
FitzPatrick
Hatch 1,2
Seabrook
Susquehanna 1,2

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Send Questions or Comments to opa@nrc.gov

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555

April 22, 1992

NRC INFORMATION NOTICE 92-21, SUPPLEMENT 1: SPENT FUEL POOL REACTIVITY CALCULATIONS

Addressees

All holders of operating licenses or construction permits for nuclear power reactors.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this supplemental information notice to update information initially supplied by ABB Combustion Engineering (CE) and incorporated in Information Notice (IN) 92-21. It is expected that recipients will review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, suggestions contained in this information notice are not NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances

IN 92-21 described errors which were discovered in reactivity calculations for spent fuel pools. Initially, CE attributed one of the errors in their calculations for the Millstone, Unit 2, spent fuel pool to using the transport cross section as an approximation for the total cross section in a region containing a strong neutron absorber (Boraflex). CE has recently informed the NRC that, upon further investigation, they believe that this approximation had little effect on calculated reactivity. CE now attributes the error to incorrect treatment of the self-shielding effect in Boraflex for the epithermal energy group. Revised calculations have shown that the absorption cross section in Boraflex for the epithermal energy group is significantly self-shielded; however, this was not accounted for in the original calculations. This oversight resulted in overestimating neutron absorption and a corresponding lower calculated keff in that region. The remainder of the discrepancy is still attributed to the inaccurate geometric buckling term used, as discussed in IN 92-21.

Discussion

Criticality calculational methodologies are benchmarked against criticality experiments which include neutron absorbers; however, the negative reactivity worth of the neutron absorber is usually low compared to that utilized in an

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actual spent fuel pool storage rack (approximately 2 percent versus 20 percent). Modest errors in calculated absorber worth in these criticality experiments would result in relatively small errors in total reactivity. However, similar errors when applied to the much higher absorber worth in actual spent fuel pool storage racks may result in significantly larger errors in total reactivity.

This information notice requires no specific action or written response. If you have any questions about the information in this notice, please contact one of the technical contacts listed below or the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

Charles E. Rossi, Director
Division of Operational Events Assessment
Office of Nuclear Reactor Regulation

Technical contacts: Jack Ramsey, NRR
(301) 504-1167

Larry Kopp, NRR
(301) 504-2879

Attachment: List of Recently Issued NRC Information Notices

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555-0001

December 1, 1995

NRC INFORMATION NOTICE 95-54: DECAY HEAT MANAGEMENT PRACTICES DURING REFUELING OUTAGES

Addressees

All holders of operating licenses or construction permits for nuclear power reactors.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice to alert addressees to recent NRC assessments of licensee control of refueling operations and the methods for removing decay heat produced from the irradiated fuel stored in the spent fuel pool. It is expected that recipients will review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, suggestions contained in this information notice are not NRC requirements; therefore, no specific action or written response is required.

Background

The staff recently reviewed a design change and associated procedural controls regarding spent fuel pool decay heat removal systems at Millstone Nuclear Power Station, Unit 1, and full-core offloading controls at Cooper Nuclear Station. The staff evaluated overall controls on irradiated fuel movement and the control of irradiated fuel decay heat removal during refueling operations, including adequate adherence to final safety analysis report commitments, implementation of procedures, procedural adequacy, and effectiveness of training.

Description of Circumstances

Millstone Unit 1

On October 18, 1993, the licensee for Millstone Unit 1 submitted Licensee Event Report (LER) 93-11, in which it reported that it had determined through engineering analysis that conditions may have existed during which the spent fuel pool cooling system may have been incapable of maintaining spent fuel pool temperature below the licensee's criteria of 66øC [150øF] design limit for continued operation. Specifically, the LER stated that (1) the licensee had made inappropriate assumptions in the analysis performed in support of a 1988 spent fuel pool re-rack project, (2) the "normal" refueling sequence described in the Millstone Unit 1 Updated Final Safety Analysis Report assumed offload of only one third of a core, (3) a full-core offload considered in the safety analysis report as an "emergency" (or abnormal discharge) offload was normally performed at Millstone Unit 1, and (4) under certain circumstances

Millstone Unit 1 may have operated outside its design basis for the spent fuel pool.

The licensee recently implemented a modification to the shutdown cooling system to provide additional spent fuel pool decay heat removal capability. In a July 28, 1995 submittal, the licensee stated that the modification would enable Millstone Unit 1 to perform a full core discharge as a normal offload practice. Coincident with the development of the modification, the licensee proposed a license amendment to impose technical specification controls on shutdown cooling system operability, spent fuel pool temperature and decay time prior to beginning offload activities. In response to staff questions, the licensee stated it had concluded during a review pursuant to 10 CFR 50.59, that the proposed modification did not represent an unreviewed safety question and as such did not require prior NRC approval, and that the license amendment was not required but was being submitted to remove ambiguity regarding the full core offload refueling practice.

During its review of the procedural controls for the shutdown cooling-spent fuel pool cooling cross-connect, the staff found that the administrative procedures for the cross-connect, including controls for the cross-connect valves and the spent fuel pool-reactor vessel weir gate were not sufficiently explicit. The licensee addressed these concerns. Because the requested specifications did not meet the criteria of 10 CFR 50.36 for inclusion as limiting conditions of operation in the technical specifications, the staff issued a license condition. The license condition specifies that refueling operations that include full core offload be conducted in accordance with the revised controls proposed by the licensee.

Cooper Nuclear Station

On October 20, 1995, the operators of the Cooper nuclear station halted movement of fuel from the reactor vessel to the spent fuel pool to perform a review of the design and licensing basis and administrative controls associated with the removal of decay heat from the spent fuel pool. The licensee concluded that no licensing restrictions regarding the practice of conducting a full-core offload existed with regard to decay heat removal. The licensee further concluded that the installed spent fuel pool cooling system and backup fuel pool cooling inter-tie from the residual heat removal system had sufficient capacity to remove the decay heat from the irradiated spent fuel and reactor cavity for postulated heat loads up to and including those associated with a full-core offload.

However, the licensee acknowledged that the description of the spent fuel pool cooling system in the Cooper Updated Safety Analysis Report was confusing and ambiguous. Consequently, the licensee proposed revisions to that document to clarify ambiguous language and performed a 10 CFR 50.59 analysis, which documented the evaluation of the plant's licensing basis for the design and operation of the spent fuel pool cooling system. Upon approval of the changes by the Station Operations Review Committee, the licensee updated its refueling procedures to be consistent with the revised safety analysis report, and proceeded with the full-core offload.

Discussion

The functional capability to protect irradiated fuel from damage due to inadequate decay heat removal is an important safety attribute. Maintaining spent fuel pool water temperature below boiling temperature provides adequate cooling for stored irradiated fuel. However, prolonged operation at elevated spent fuel pool temperatures may impair the capability of the purification system to remove contaminants from the spent fuel pool coolant and increase the rate of heat addition to the fuel storage area atmosphere to a value above that assumed in the ventilation system design. In addition, high spent fuel pool temperatures may exceed the temperature used for thermal stress computation in the structural analysis of the spent fuel pool liner and the spent fuel pool structure itself.

Shutdown cooling systems, which are aligned to directly cool the reactor vessel or reactor coolant system, are designed to remove the residual and decay heat associated with the irradiated fuel in the reactor vessel in order to bring the reactor coolant system to the cold shutdown condition. Licensees typically have procedures to maintain the removal of decay heat from the vessel at all times while irradiated fuel is in the reactor vessel.

Similarly, systems are also installed in nuclear power plants to remove from the spent fuel pool, decay heat generated by the stored irradiated fuel. However, these spent fuel pool cooling systems are designed with a lower heat removal capacity relative to the shutdown cooling system based on the decrease in decay heat generation within the irradiated fuel as the time after reactor shutdown increases. At some facilities, including most boiling water reactors and some pressurized water reactors, the spent fuel pool cooling system is not designed to remove the decay heat associated with a full core immediately after shutdown and still maintain a bulk spent fuel pool temperature below design-basis limits. However, these facilities are designed with backup spent fuel pool cooling systems, which are generally alternative operating modes of the residual heat removal or shutdown cooling systems, that supplement the spent fuel pool cooling system during periods shortly after reactor shutdown when the decay heat load of a full core may exceed the heat removal capacity of the normal spent fuel pool cooling system.

The capability of spent fuel pool cooling systems and backup spent fuel decay heat removal systems is described in the Final Safety Analysis Report, as updated for nuclear power plants. The decay heat load scenarios used to evaluate the adequacy of system heat rejection capability may be based on a series of core offloads and an associated decay time. These scenarios may be described as "normal" or "abnormal" maximum heat loads for this purpose, which is consistent with the NRC staff guidance for the review of spent fuel pool cooling system design contained in Section 9.1.3 of the Standard Review Plan (NUREG-0800).

Recent licensee reviews of refueling outage practices at Millstone Unit 1 and Cooper found that the system design bases specified in Final Safety Analysis Reports, as related to core offload practices, were ambiguous. Administrative controls on refueling outage plans and practices were inconsistent in regard

December 1, 1995
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to ensuring that temperature commitments for the spent fuel pool were maintained through all phases of refueling operation. Both licensees, after clarifying and improving the design bases and administrative controls related to refueling outages, determined that a routine practice of performing full core offloads was acceptable.

The NRC has issued two information notices to alert licensees to potential risks associated with a loss of spent fuel pool cooling. NRC Information Notice 93-83, "Potential Loss of Spent Fuel Pool Cooling After a Loss-of-Coolant Accident," was issued October 7, 1993, and described concerns found at

Susquehanna Steam Electric Station. NRC Information Notice 93-83, Supplement 1, was issued August 8, 1995, to inform licensees of the results of the NRC review of the concerns at Susquehanna.

The events described in this and previous information notices, and the plant reviews discussed above, illustrate the importance of:

þ assuring that planned core offload evolutions, including refueling practices and irradiated decay heat removal, are consistent with the licensing basis, including the Final Safety Analysis Report, technical specifications, and license conditions;

þ assuring that changes are evaluated through the application of the provisions of 10 CFR Part 50.59, as appropriate; and

þ assuring that all relevant procedures associated with core offloads have been appropriately reviewed.

The staff is continuing to review this matter with respect to the need to issue additional generic communications.

This information notice requires no specific action or written response. If you have any questions about the information in this notice, please contact one of the technical contacts listed below or the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

/s/'d by DMCrutchfield

Dennis M. Crutchfield, Director
Division of Reactor Program Management
Office of Nuclear Reactor Regulation

Technical contacts: Joseph W. Shea, NRR (301) 415-1428

Steven R. Jones, NRR (301) 415-2833

David L. Skeen, NRR (301) 415-1174

September 30, 1996

EA 96-202

Guy R. Horn, Vice President - Nuclear
Nebraska Public Power District
1414 15th Street
Columbus, Nebraska 68601

SUBJECT: NOTICE OF VIOLATION
(NRC Inspection Report 50-298/95-18 and Investigation Case No. 4-96-002)

Dear Mr. Horn:

This refers to the matters discussed at the predecisional enforcement conference conducted on August 5, 1996, at the NRC's office in Arlington, Texas. As indicated in our letter dated June 27, 1996, the conference was conducted to discuss apparent violations related to a control rod mispositioning event that occurred on January 7, 1996, at the Cooper Nuclear Station (CNS). A summary of the predecisional enforcement conference, including the information presented by the Nebraska Public Power District (NPPD) at the conference, was issued on August 7, 1996. Subsequent to the conference with NPPD, the NRC also conducted individual conferences with two former CNS licensed operators who were involved in the rod mispositioning event.

Based on the information developed during the inspection and investigation, a review of NPPD's investigation of this matter, and the information obtained from the conferences, the NRC has determined that violations of NRC requirements occurred. These violations are cited in the enclosed Notice of Violation (Notice). Each involves a failure by licensed operating personnel to follow procedural requirements, including: 1) a failure to insert control rods in the proper sequence following a loss of a reactor recirculation pump; 2) a failure to notify shift supervision of an unexpected situation, i.e., a mispositioned control rod, for approximately 20 minutes; and 3) a failure to obtain the concurrence of the shift supervisor and reactor engineer in developing a recovery plan for a mispositioned control rod.

This event began when the involved operators, after being directed to insert control rods in reverse sequence following a reactor recirculation pump trip, mistakenly inserted control rods on the wrong page of the control rod sequence book. The operators recognized their mistake but continued inserting control rods without notifying shift supervisory personnel of their error and without seeking concurrence in a recovery plan. This event was investigated by NPPD and resulted in NPPD terminating the involved licensed operators.

The NRC agrees with NPPD's expectation that the operators should have promptly informed shift supervisory personnel of their mistake and the abnormal conditions that developed. The information available to the NRC, however, does not support a conclusion that they intentionally violated any CNS procedural requirements. Although their actions violated CNS procedures and NPPD management expectations, the operators appear to have been focused on inserting control rods to avoid exceeding plant administrative limits and an automatic plant trip. And, while they should have been mindful of the procedural requirements, they were not. The facts that they maintained accurate logs and informed the reactor engineer of the mistake when he approached the panel do not suggest a deliberate intent to cover up their mistake or violate procedures.

The NRC recognizes that the actions taken by the involved operators did not place the plant in an unsafe condition. Nonetheless, there is regulatory significance to licensed operators not recognizing their obligation to obtain shift supervisor and reactor engineer concurrence before proceeding to insert control rods in this situation. The NRC also attaches regulatory significance to the fact that the control room supervisor, despite being aware that the operators were inserting rods on the Emergency Control Rod Movement sheet, an unusual situation, did not take action to determine what was occurring and to understand the situation. As noted in NPPD's investigation of this matter, the control room supervisor's attention appears to have been focused heavily on balance-of-plant activities. While the NRC does not conclude that the control room supervisor's actions violated the Conduct of Operations procedures, an apparent violation discussed at the conference, this remains a concern. Finally, the NRC notes that NPPD's investigation team found inconsistent crew members' knowledge of the requirements of CNS procedure 10.13, "Control Rod Sequence and Movement Control," which calls into question the adequacy of CNS's training on the specific requirements of this procedure.

Based on the regulatory significance of these violations, they have been categorized in the aggregate in accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions" (Enforcement Policy), NUREG-1600, as a Severity Level III problem. In accordance with the Enforcement Policy, a civil penalty, with a base value of $50,000, is considered for a Severity Level III problem. Because your facility has been the subject of escalated enforcement actions within the 2 years preceding the identification of this problem,1 the NRC considered whether credit was warranted for *Identification* and *Corrective Action* in accordance with the civil penalty assessment process in Section VI.B.2 of the Enforcement Policy.

These violations were identified as a result of the involved operators informing NPPD managers of their mistake, and NPPD's follow-up investigation into this matter. Thus, credit for identification is warranted. The NRC also has determined that NPPD is deserving of credit for its corrective actions, which consisted of: immediate actions to assure the safety of the facility and assure that thermal limits had not been exceeded; meetings with all operating crews to discuss issues arising from this event; initiation of an independent review team investigation; disciplinary action against the involved operators; clarification and revisions to procedures and Ops Instructions; and assessment of the environment for reporting errors.

Therefore, to encourage prompt identification and comprehensive correction of violations, I have been authorized, after consultation with the Director, Office of Enforcement, not to propose a civil penalty in this case. However, significant violations in the future could result in a civil penalty.

You are required to respond to this letter and should follow the instructions specified in the enclosed Notice when preparing your response. The NRC will use your response, in part, to determine whether further enforcement action is necessary to ensure compliance with regulatory requirements.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response will be placed in the NRC Public Document Room (PDR).

Sincerely,

ORIGINAL SIGNED BY

L.J. Callan
Regional Administrator

Docket No. 50-298
License No.: DPR-46

Enclosure: Notice of Violation

cc w/Enclosure:
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Nebraska Public Power District
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Kansas Radiation Control Program Director

NOTICE OF VIOLATION

Nebraska Public Power District
Cooper Nuclear Station

Docket No. 50-298
License No. DPR-46
EA 96-202

During an NRC investigation concluded on May 8, 1996, violations of NRC requirements were identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," NUREG-1600, the violations are listed below:

A. 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that, "Activities affecting quality shall be . . . accomplished in accordance with these instructions, procedures, and drawings."

Step 8.2.6 5 of Cooper Nuclear Station Operations Manual, "Conduct of Operations Procedure 2.0.3," Revision 20, dated August 21, 1995, states, "Operators should notify the control room supervisor and shift supervisor of any unexpected situations encountered in monitoring the main control boards."

Contrary to the above, on January 7, 1996, operators did not notify the control room supervisor and shift supervisor of a mispositioned control rod, an unexpected situation encountered in monitoring the main control boards, until approximately 20 minutes after discovery. (01013)

B. 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that, "Activities affecting quality shall be . . . accomplished in accordance with these instructions, procedures, and drawings."

Step 8.1.5 of Cooper Nuclear Station Operations Manual, Nuclear Performance Procedure 10.13, "Control Rod Sequence and Movement Control," Revision 26, dated December 24, 1995, requires that operators, ". . . not deviate from the sequence unless approved by a reactor engineer (or shift supervisor in an emergency) or per a SORC approved procedure."

Contrary to the above, on January 7, 1996, operators deviated from the approved sequence when operators inserted control rods starting with the incorrect page of the control rod sequence book without the express permission of a reactor engineer or the shift supervisor, or a SORC approved procedure. (01023)

C. 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that, "Activities affecting quality shall be . . . accomplished in accordance with these instructions, procedures, and drawings."

Step 8.4.4 of Cooper Nuclear Station Operations Manual, Nuclear Performance Procedure

10.13, "Control Rod Sequence and Movement Control," Revision 26, dated December 24, 1995, requires that operators, "With concurrence of the shift supervisor and reactor engineer, implement a recovery plan . . . ." when recovering from mispositioned control rods.

Contrary to the above, on January 7, 1996, operators failed to properly implement this procedure when the control room operators took actions to recover from mispositioned control rods using their own judgement rather than a recovery plan which had been concurred in by the shift supervisor and the reactor engineer. (01033)

These violations represent a Severity Level III problem (Supplement I).

Pursuant to the provisions of 10 CFR 2.201, Nebraska Public Power District is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555 with a copy to the Regional Administrator, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011, and a copy to the NRC Resident Inspector at the facility that is the subject of this Notice, within 30 days of the date of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation" and should include for each violation: (1) the reason for the violation, or, if contested, the basis for disputing the violation, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further violations, and (4) the date when full compliance will be achieved. Your response may reference or include previous docketed correspondence, if the correspondence adequately addresses the required response. If an adequate reply is not received within the time specified in this Notice, an order or a Demand for Information may be issued as to why the license should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

Under the authority of Section 182 of the Act, 42 U.S.C. 2232, this response shall be submitted under oath or affirmation.

Because your response will be placed in the NRC Public Document Room (PDR), to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be placed in the PDR without redaction. If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.790(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

Dated at Arlington, Texas
this 30th day of September 1996

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555

May 5, 1988

NRC INFORMATION NOTICE NO. 88-20: UNAUTHORIZED INDIVIDUALS MANIPULATING CONTROLS AND PERFORMING CONTROL ROOM ACTIVITIES

Addressees:

All holders of operating licenses or construction permits for nuclear power, test and research reactors, and all licensed operators.

Purpose:

This information notice is being provided to alert addressees to potential problems resulting from unauthorized persons manipulating controls and performing control room activities. It is expected that recipients will review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, no specific action or written response is required.

Description of Circumstances:

On September 13, 1987, with Turkey Point Unit 3 at 30% power, a non-licensed individual (who was not enrolled in an operator training program) under the direct supervision of a licensed operator turned the reactor control make-up switch to "start" on two occasions in close succession. At the time of the event, a 30% power flux map was in progress. Negative reactivity was being added to the reactor because of xenon buildup. To minimize flux distortion, control rod motion was being minimized, and the negative reactivity insertion from the xenon buildup was being countered by lowering the boron concentration in the RCS (dilution). The dilution was being performed by periodically adding preset quantities of non-borated water to the RCS. The licensed operator preset the non-borated water quantity to 30 gallons, then permitted the non-licensed individual to turn the reactor control make-up switch to the "start" position. The system response to the 30-gallon dilution was not sufficient, and the above evolution was repeated, with the non-borated water quantity preset by the licensed operator at 20 gallons. The licensed operator directly supervised both dilutions. This event is discussed further in Licensee Event Report (LER) 250/87-024 (Voluntary Report).

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The NRC resident inspectors at Braidwood were informed of a similar event by that facility's licensee. An individual licensed at another facility (but not enrolled in an operator training program) was temporarily contracted to Braidwood as an equipment attendant. On September 20, 1987, this individual was allowed to manipulate the controls on the Unit 1 reactor for a dilution of the RCS and for a control rod movement.

No serious consequences occurred from either of these events. However, these incidents are contrary to the requirements of 10 CFR 50.54(i) and 10 CFR 55.3 because these actions directly affected the reactivity and power of the reactor, and the individuals performing the actions did not meet the exemption requirements of 10 CFR 55.13. (10 CFR 55.13 allows an individual, under the direction and in the presence of a licensed operator, to manipulate the controls of a facility as a part of the individual's training in a facility licensee's training program.) Because neither individual was enrolled in an approved operator training program, the exemption requirements of 10 CFR 55.13 were not satisfied.

Discussion:

The events discussed above are contrary to 10 CFR Parts 50 and 55. Addressees may wish to ensure that they are familiar with these requirements. In addition, the requirements of 10 CFR Part 55 were revised in 1987 and made effective on May 26, 1987. These revisions, as well as other relevant information, are discussed in NUREG-1262, "Answers to Questions at Public Meetings Regarding Implementation of Title 10, Code of Federal Regulations, Part 55 on Operators' Licenses," published in November 1987.* A copy of this document was provided to all power and non-power reactor licensees and applicants for licenses by Generic Letter 87-16. Addressees who have not already done so may wish to obtain and review a copy of NUREG-1262 to ensure that they are familiar with the revisions made to 10 CFR Part 55.

The NRC has required individuals licensed at near-term operating license (NTOL) facilities to obtain operational experience at like facilities through cooperative arrangements. Nothing in this information notice is intended to preclude such arrangements.

*A copy of NUREG-1262 can be ordered by writing to the Superintendent of Documents, U.S. Government Printing Office, Post Office Box 37082, Washington, D.C. 20013-7082, or by calling (202) 275-2060.

No specific action or written response is required by this information notice. If you have any questions about this matter, please contact the technical contact listed below or the Regional Administrator of the appropriate regional office.

Charles E. Rossi, Director
Division of Operational Events Assessment
Office of Nuclear Reactor Regulation

Technical Contact: John N. Hannon, NRR
(301) 492-1031

Attachment: List of Recently Issued NRC Information Notices

Attachment
IN 88-20
May 5, 1988
Page 1 of 1

LIST OF RECENTLY ISSUED
NRC INFORMATION NOTICES

Information Notice No.	Subject	Date of Issuance	Issued to
88-19	Questionable Certification of Class 1E Components	4/26/88	All holders of OLs or CPs for nuclear power reactors.
88-18	Malfunction of Lockbox on Radiography Device	4/25/88	All NRC licensees authorized to manufacture, distribute, and/or operate radio-graphic exposure devices.
88-17	Summary of Responses to NRC Bulletin 87-01, "Thinning of Pipe Walls in Nuclear Power Plants"	4/22/88	All holders of OLs or CPs for nuclear power reactors.
88-16	Identifying Waste Generators in Shipments of Low-Level Waste to Land Disposal Facilities	4/22/88	Radioactive waste collection and service company licensees handling prepackaged waste, and licensees operating low-level waste disposal facilities.
88-15	Availability of U.S. Food and Drug Administration (FDA)-Approved Potassium Iodide for Use in Emergencies Involving Radioactive Iodine	4/18/88	Medical, Academic, and Commercial licensees who possess radioactive iodine.
88-14	Potential Problems with Electrical Relays	4/18/88	All holders of OLs or CPs for nuclear power reactors.
88-13	Water Hammer and Possible Piping Damage Caused by Misapplication of Kerotest Packless Metal Diaphragm Globe Valves	4/18/88	All holders of OLs or CPs for nuclear power reactors.

OL = Operating License
CP = Construction Permit

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555

November 4, 1992

NRC INFORMATION NOTICE 92-73: REMOVAL OF A FUEL ELEMENT FROM A RESEARCH REACTOR CORE WHILE CRITICAL

Addressees

All holders of operating licenses or construction permits for nuclear power reactors.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice to alert licensees to a recent event in which licensed operators at a research reactor inadvertently removed a fuel element from a reactor core that was critical. It is expected that recipients will review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, suggestions contained in this information notice are not NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances

On June 8, 1992, at the University of Michigan's (the licensee's) Ford research nuclear reactor facility the Assistant Reactor Manager for Operations (ARM) and two other senior reactor operators (SROs) were conducting tests to measure changes in core reactivity. In each test, the operators would perform the following: move the fuel, bring the reactor to low power, collect data, and shut down the reactor. After collecting data following the third fuel movement, and with the reactor still critical at low power (8 kW), the ARM directed the two SROs to move the fuel a fourth time. The SRO acting as the control room operator then informed the ARM that the 2-hour control room log readings were due. The ARM then gave the SROs instructions on what to do while he obtained the log readings. The ARM subsequently told the NRC that he instructed the SROs to prepare for the fourth move; however, the SROs believed that they had clear direction to move fuel. The SROs then began moving the fuel. While one SRO monitored the test, the other latched a fuel element with the fuel handling tool, and then removed the fuel element. The research reactor immediately went subcritical and the control rod's servo-mechanism switched out of automatic control. At that time, another SRO not directly involved in the fuel movement, but recognizing what had happened, entered the control room and manually inserted the shim rods and control rod. The equipment performed as designed and the reactor remained in a safe condition.

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IN 92-73
November 4, 1992
Page 2 of 3

Discussion

As discussed in NRC's Augmented Inspection Team (AIT) Report No. 50-002/92001, dated July 9, 1992, and associated correspondence, several factors led to this event.

1. There was frequent informal turnover of control room responsibilities between the ARM and the control room operator during the fuel movements. Who was to have overall control of the reactor between the third and fourth fuel movements was not clearly established.

2. The ARM and the other two SROs moving the fuel did not communicate well. For example, both SROs believed that they had clear instructions to move the fuel, while the ARM believed that he only instructed them to prepare to move fuel. An intercom system between the control room and the fuel handling bridge was not used until the fuel element was being moved.

3. Relying on their experience and the routine nature of the fuel moves, the SROs did not use or review the procedures that applied to moving fuel either before or during the actions to move the fuel.

4. An excessive work load may have contributed to the event. For example, the NRC inspectors found that, after correcting a previous problem with the control and shim rod magnets, the SRO's had only four hours during their normal shift schedule to complete the planned fuel moves. This was said to create a rushed atmosphere for the test activities.

In moving the fuel element while the reactor was critical, the two SROs handling the fuel indicated that they did not clearly know the condition of the reactor when they removed the fuel element. The ARM did not maintain adequate control over the entire test activity. The distraction of completing the control room log contributed to poor communications with the other SROs.

The licensee has modified its procedures and will install illuminated indicators on the rod drive housing located on the bridge. The indicators will be illuminated only when the rods are fully inserted. These changes to the procedures and equipment will give more positive communication, enable operators to better control fuel changes, and visually indicate the status of the control rods.

This event is an example where licensed operators at a research reactor became so involved in tasks that they failed to maintain adequate control of the reactor. The operators did not maintain current knowledge of the condition of the reactor and therefore were not cognizant of the effect that their actions would have on that condition.

This information notice requires no specific action or written response. If you have any questions about this matter, please contact the technical contact listed below or the appropriate Nuclear Reactor Regulation (NRR) project manager.

ORIGINAL SIGNED BY

Brian K. Grimes, Director
Division of Operating Reactor Support
Office of Nuclear Reactor Regulation

Technical contact: Charles Cox, RIII
(708) 790-5298

Project Manager: Theodore S. Michaels, NRR
(301) 504-1102

Attachment: List of Recently Issued NRC Information Notices

SSINS No.: 6835
IN 85-12

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

February 11, 1985

IE INFORMATION NOTICE NO. 85-12: RECENT FUEL HANDLING EVENTS

Addressees:

All nuclear power reactor facilities holding an operating license (OL) or construction permit (CP).

Purpose:

This information notice is provided as a notification of potentially significant problems pertaining to recent fuel handling events. This notice supplements Information Notice 80-01, which discussed similar events. It is expected that recipients will review the information for applicability to their facilities and consider actions, if appropriate, to preclude similar problems from occurring at their facilities. However, suggestions contained in this information notice do not constitute NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances:

Two events have occurred recently at nuclear power plants in which fuel was dropped because of failures or deficiencies in hoist equipment. More details are provided in Attachment 1.

(1) At Hatch 1 on October 6, 1984, a spent fuel bundle was dropped into its storage cell because of a possible inadvertent actuation of the fuel grapple hook position switch. The switch cover was missing.

(2) At Millstone 2 on November 8, 1984, a fuel pin dropped in the spent fuel pool during fuel assembly reconstitution because the gripping collet fingers slipped.

Several additional events have occurred that are noteworthy because they involve deficiencies or maloperation of fuel handling equipment or procedures. These are briefly summarized below; more detailed information is given in Attachment 1.

(1) At Monticello on November 29, 1984, a spent fuel bundle handle was deformed during transportation because of inadequate cask loading procedures.

(2) At Palisades on April 4, 1984, a new fuel bundle was stuck in the refueling machine because of inadequate spreader bar air supply pressure.

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(3) At Turkey Point 4 on April 5, 1983, a spent fuel assembly dropped back into its storage cell when the hoist limit switches failed to prevent upward movement of the assembly. This event also involved a procedural inadequacy concerning these limit switches.

(4) A second event at Turkey Point 4 on April 17, 1983, resulted in an improperly loaded (leaning) fuel assembly.

(5) At Cook 1 on June 19, 1981, a fuel assembly was damaged in a collision with a shield wall because an entangled air hose had tripped a limit switch.

(6) Also at Cook 1 on August 4, 1982, a fuel assembly was cocked and lodged in the manipulator bridge mast because the fuel handling procedures were not properly followed.

Discussion:

This information notice briefly describes several events involving failures or deficiencies in fuel handling equipment or procedures. In addition, Information Notice 80-01 discusses two similar events at Pilgrim. In one, a spent fuel assembly was inadvertently raised high enough in the fuel pool to activate area radiation alarms because the lifting hook was caught between the lifting bail and the assembly channel. In the other, a new fuel assembly dropped onto the top of the storage fuel racks when the auxiliary hook latching device failed to hold the lifting bail when the assembly struck the top edge of the racks. The radiological consequences of these events were minimal. Nevertheless, the events are considered significant, in that they could have compromised plant safety and could have been prevented. Licensees may wish to review their procedures in view of these events.

No specific action or written response is required by this information notice. If you have any questions about this matter, please contact the Regional Administrator of the appropriate NRC regional office or this office.

Edward L. Jordan, Director
Division of Emergency Preparedness
and Engineering Response
Office of Inspection and Enforcement

Technical Contact: C. V. Hodge, IE
(301) 492-7275

Attachments:
1. Description of Recent Fuel Handling Events
2. List of Recently Issued IE Information Notices

Attachment 1
IN 85-12
February 12, 1985
Page 1 of 3

Description of Recent Fuel Handling Events

Hatch 1

This event involved a possible inadvertent actuation of the fuel grapple hook position switch. On October 6, 1984, with core unloading in progress, a spent fuel bundle was inadvertently dropped into its storage rack cell (a distance of about 12 feet), slightly deforming and scratching the bundle and rack. Before the event, no trouble had been experienced in grappling bundles. When the bridge operator lowered the affected bundle and detected contact of the bundle with the rack, he stopped to align the bundle with its storage cell; then the bundle dropped. The licensee declared an unusual event and terminated it on confirming that no fission gases had been released.

Grapple tests and operator interviews indicated that the operator actions required to position or rotate the fuel bundle could have resulted in inadvertently operating the fuel grapple hook position actuation switch. General Electric Service Information Letter (SIL) No. 298, dated August 1979, describes the potential for inadvertent switch operation in conjunction with a slack grapple hoist cable before the operator has verified that the fuel bundle is properly seated. General Electric recommends that the owners of BWRs 1 through 4 install a commercially available snap cover over the switch. The licensee had installed the switch coVers on the refueling platforms of Units 1 and 2; however, between 1979 and the present, the covers had been removed. The licensee originally used an epoxy-type adhesive to secure the covers, but now has bolted them into place.

Millstone 2

This event involved mechanical slipping of the fuel holding mechanism. On November 8, 1984, during fuel assembly reconstitution in the spent fuel pool, a single spent fuel pin was dropped during eddy current testing for cladding defects. The pin was gripped by collet fingers inside a long cylindrical probe. Evidently these fingers slipped, possibly because of a weld repair at the top of the pin. The fingers were adjusted to provide a more positive grip. Although this pin was retrieved, inspected, and showed no defects, it was replaced in its position in the fuel assembly by a stainless steel spacer. The licensee instituted an additional check for proper gripping of each fuel pin and completed the fuel assembly reconstitution.

Monticello

This event illustrates the need for an explicit checkpoint in the fuel cask loading procedure. On November 29, 1984, the handle on a spent fuel bundle was found deformed when it was off-loaded from a transportation cask to a storage rack at the GE Morris spent fuel storage facility. The bundle had not been seated properly in the cask because horizontal tabs at the top of the bundle had not been aligned properly with the cask, preventing the bundle from being fully inserted. No radiological effects were caused, but the event is significant because the fuel loading procedures were not carefully followed.

Surveillance was conducted for this loading of the cask, but there was not an explicit check for proper seating of the bundles before the cask cover was bolted in place. The licensee's corrective action is to institute such an explicit check in the fuel loading procedures.

Palisades

This event involved inoperability of the fuel hoist mechanism. On April 4, 1984, while reloading the core, a new fuel bundle stuck in the refueling machine. A combination of low spreader bar air supply pressure (40 psi vs normal 50 psi) and air leakage from the spreader bar retraction hose fitting resulted in the spreader bar extending downward one inch below the hoist bottom. An interlock for the extended spreader bar prevented movement of the bridge trolley. After evaluating the situation, the licensee increased the air supply pressure and inserted the bundle into the core. The licensee then completed core reload without further problem.

Turkey Point 4

This event involved a malfunction of the limit switches on the spent fuel pit hoist and disclosed a procedural inadequacy. On April 5, 1983, during refueling after a six month outage for steam generator repair, partially burned fuel assembly X-13 was being lifted from its storage rack. The limit switches failed to stop the upward movement of X-13, resulting in parting of the hoisting cable and causing the assembly to drop back into its rack.

The crane design provides two different limit switches to restrict upper motion: a power circuit limit switch and a geared limit switch. About three weeks before actual fuel movement, testing indicated the switches would work, but the investigation after the event revealed that a linkage in the power limit switch was unhooked, which disabled the trip feature, and the geared limit switch was out of adjustment. Had the licensee tested the upper limit switch under no load at the beginning of each shift,as required by OSHA regu-lations [29 CFR 1910.179(n)(4)] or recommended by industry guidance (ANSI B30.2-1976, "Overhead and Gantry Cranes"), this event could have been prevented.

The procedural inadequacy was the incorrect designation of the limit switches. The spent fuel pit crane test procedure indicated that the power circuit switch backed up the geared switch; the operating procedure for that crane incorrectly indicated the opposite. The operating procedure also contradicted the prohibition stated in both procedures against using the two switches as normal stopping devices.

A second event occurred shortly afterward in which improper placement of a fuel assembly into the core was not readily detected. Because of the X-13 drop, it was necessary to reconfigure the core loading sequence. Because only the central area was to be reconfigured, the approved fuel loading sequence started with assemblies on the core perimeter and spiraled inward. This sequence only provided one or two adjacent surfaces (fuel or baffle plate), instead of the usual four, to guide an assembly being inserted.

On April 17, 1983, a small maladjustment of the fuel handling bridge position (less than an inch deviation) coupled with a slight bow in

A.1 Reactivity Management

Zion Technical Specification 6.2, Procedures and Programs, requires, in part, that "[w]ritten procedures including applicable checkoff lists covering items listed below shall be prepared, implemented, and maintained: a.) The applicable procedures recommended in Appendix A, of Regulatory Guide 1.33, Revision 2, February 1978."

Appendix A of Regulatory Guide 1.33, Revision 2 dated February 1978, recommends that written procedures should cover activities associated with "Authorities and Responsibilities for Safe Operation," (paragraph 1.b) and "General Plant Operating Procedures," (paragraph 2) activities associated with all modes of operation.

a. General Operating Procedure (GOP) 4, "Plant Shutdown and Cooldown," Revision 13, partially implements paragraph 2 of Appendix A to Regulatory Guide 1.33. In Mode 2 with the reactor critical, step 5.21.f, states, Hold "#363, ROD MOTION CONTROL" switch IN to minimize dumping steam and establish power at or less than the point of adding heat." (2.5 x 10E-2% intermediate range (IR) or 0.025 percent power).

Contrary to the above, on February 21 1997, with the reactor critical in Mode 2, the primary Nuclear Station Operator (NSO) manipulated the control rods, but did not accomplish this activity in accordance with GOP-4, step 5.21.f. Specifically, the primary NSO did not establish power at or less than the point of adding heat in accordance with GOP-4, step 5.21.f, but made the reactor sub-critical, i.e., entered Mode 3 (01013)

b. Zion Administrative Procedure (ZAP) 300-01B, "Reactivity Management Guidelines," Revision 1, partially implements paragraph 1.b of Appendix A to Regulatory Guide 1.33. Section G.2.c.1, states that strict reactivity controls are required to minimize the potential for core damage, and that all plant personnel, particularly operators, must stop and question unexpected situations involving reactivity, criticality, power level, or core anomalies.

Contrary to the above, on February 21, 1997, the primary Nuclear Station Operator failed to implement strict reactivity controls when he did not stop and question unexpected changes in reactivity, criticality, and power level. The primary Nuclear Station Operator made the reactor substantially sub-critical (non-power) and, instead of stopping and evaluating the unexpected change in reactivity and criticality, he attempted to return the reactor to the point of adding heat (0.025 percent power) by continuously withdrawing control rods. (01023)

c. ZAP 300-01B, "Reactivity Management Guidelines," Revision 1, partially implements paragraph 1.b of Appendix A to Regulatory Guide 1.33. Section G.1.1, states the Qualified Nuclear Engineer's (QNE) responsibility to implement the reactivity management policy by providing technical advice on reactivity related matters.

Contrary to the above, on February 21, 1997, the QNE failed to provide technical advice on reactivity related matters to either the primary Nuclear Station Operator or the Shift Supervisor concerning the excessive control rod manipulations and resultant reactivity changes. Specifically, the QNE observed the Nuclear Station

Operator (NSO) take the reactor subcritical and then observed the NSO inappropriately add positive reactivity by withdrawing control rods in an unauthorized manner in an attempt to reachieve criticality. Instead of promptly addressing the inappropriateness of the NSO's actions, the QNE, by his inaction, allowed the unauthorized control rod withdrawal to continue and left the control room to discuss the matter with the Lead Nuclear Engineer. (01033)

A.2 Command, Control, and Communication

Zion Technical Specification 6.2, Procedures and Programs, requires that "[w]ritten procedures including applicable checkoff lists covering items listed below shall be prepared, implemented, and maintained: a.) The applicable procedures recommended in Appendix A, of Regulatory Guide 1.33, Revision 2, February 1978."

Appendix A of Regulatory Guide 1.33, Revision 2 dated February 1978, states in paragraph 1.b, that written procedures should cover activities associated with "Authorities and Responsibilities for Safe Operation."

a. ZAP 300-01, "Conduct of Operations," Revision 3, partially implements paragraph 1.b of Appendix A to Regulatory Guide 1.33. Section VI.A, states, in part, that all operations personnel from the Operations Manager to the Nuclear Station Operator share the responsibility for the reactor core. The Nuclear Station Operator, Unit Supervisor and station Reactor Engineer have a greater responsibility, and share this ownership on a continuous basis. Section VI.A further states, in part, that: (1) Operations personnel SHALL be attentive to the condition of the plant at all times and (4) the Unit Supervisor and Shift Engineer should NOT become so involved in any single operation that distracts them from the overall perspective of plant operations.

Contrary to the above, on February 21, 1997, while the licensee was performing a Unit 1 shutdown, the Shift Engineer and the Unit Supervisor were so focused on containment spray pump restoration activities that they failed to be attentive to the fact that the Nuclear Station Operator had made the reactor substantially subcritical and was withdrawing control rods, in an attempt to return the reactor to criticality. (01043)

b. ZAP 300-01, "Conduct of Operations," Revision 3 partially implements paragraph 1.b of Appendix A to Regulatory Guide 1.33. Attachment A, Section VI.A.3, requires that a briefing be conducted prior to the conduct of an infrequently performed evolution.

ZAP 300-01, "Conduct of Operations," Revision 3, Section IV.G.2 defines infrequently performed evolutions as evolutions whereby the performance frequency is greater than annually AND the evolution requires the coordination of two or more departments, OR three or more individuals, AND has the potential to adversely affect reactivity control, OR core cooling.

Contrary to the above, on February 21, 1997, the licensee failed to perform a briefing prior to conducting an activity to maintain the reactor at the point of adding heat (2.5 x 10E-2% IR), which was an infrequently performed evolution in that it had not been performed in more than a year, required the coordination of three individuals, and had the potential to adversely affect reactivity control.

(01053)

c. ZAP 300-01, "Conduct of Operations," Revision 3 partially implements paragraph 1.b of Appendix A to Regulatory Guide 1.33. Section IX.E.3, requires the individual who is to perform the activity to be responsible for adequately reviewing the procedure and fully understanding assigned responsibilities, and cognizant of all the limitations, precautions, and requirements.

Contrary to the above, on February 21, 1997, the primary Nuclear Station Operator and Unit Supervisor, failed to adequately review GOP-4, "Plant Shutdown and Cooldown," prior to performing the Unit 1 shutdown and failed to understand their assigned responsibilities as evidenced by the actions of the NSO in driving in control rods until the point of adding heat as measured on the intermediate range monitors. In doing so, the NSO failed to account for the time delay associated with the addition of substantial negative reactivity and as a result inadvertently made the reactor subcritical. The inadequate review and failure to understand assigned responsibilities was further demonstrated by the Nuclear Station Operator's actions in subsequently withdrawing control rods once he recognized that he had made the reactor subcritical. (01063)

d. ZAP 300-01A, "Control Room Access and Conduct," Revision 4 partially implements paragraph 1.b of Appendix A to Regulatory Guide 1.33. Section VIII.A.2, requires that Control Room business SHALL be conducted at a location and in such a manner that neither on-shift licensed personnel attentiveness nor the professional atmosphere is compromised.

Contrary to the above, on February 21, 1997, the presence of an excessive number of individuals in the control room -- 39 people were in the control room envelope, with 15 people in the immediate vicinity of the primary Nuclear Station Operator and Unit Supervisor -- caused a loud and disruptive environment. The high ambient noise level due to the large number of individuals present made communications between operators and operations supervision difficult. This created a control room environment that was not conducive to conducting a controlled and orderly shutdown. As a result, licensed personnel attentiveness and the professional atmosphere of the control room were compromised. (01073)

e. ZAP 300-09, "Station Operational Communications," Revision 3 partially implements paragraph 1.b of Appendix A to Regulatory Guide 1.33. Section VII.A.3., requires, in part, that if the receiver does not understand an operational communication, then the receiver shall promptly inform the sender and ask the sender to repeat or rephrase the message.

Contrary to the above, on February 21, 1997, the primary Nuclear Station Operator (the receiver) failed to promptly inform the Unit Supervisor (the sender) that he did not understand the communication concerning establishing power at the point of adding heat and ask the Unit Supervisor (the sender) to repeat or rephrase his message. The Nuclear Station Operator did ask the Unit supervisor if he wanted him (the Nuclear Station Operator) to drive rods in, indicating he did not understand the instruction, and instead of explaining or rephrasing, the Unit supervisor simply reread the step aloud. (01083)

These violations represent a Severity Level III problem (Supplement I) - $110,000

B. Corrective Actions -- Reactivity Management Event

10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Actions," requires, in part, that measures be established to assure that conditions adverse to quality are promptly corrected, and in the case of significant conditions adverse to quality, that measures be established to assure that the cause of the condition is determined and corrective actions taken to preclude recurrence.

1. Contrary to the above, as of February 21, 1997, following the identification by the licensee of an adverse trend in reactivity management activities, which was documented in an internal Zion Station memorandum, "ZNG:96-002" dated February 12, 1996, and which constituted a significant condition adverse to quality, corrective actions were not prompt and commensurate with the safety significance of the issue and failed to preclude recurrence of a reactivity management event on February 21, 1997. (02013)

2. Contrary to the above, as of February 21, 1997, following issuance of a Notice of Violation (50-304/96005-03) on April 8, 1996, that identified an inadvertent mode change -- a significant condition adverse to quality that was caused by poor communications, weak command and control, and poor reactivity management -- corrective actions taken to preclude recurrence were not adequate as demonstrated by the February 21, 1997, actions of the Unit 1 operating crew, where failures in communications, command and control, and reactivity management directly caused an unauthorized and uncontrolled positive reactivity addition.(02023)

3. Contrary to the above, as of February 21, 1997, following a command and control, communication and reactivity management problem during the Unit 1 startup on September 16, 1996, which was a significant condition adverse to quality, corrective actions taken to preclude recurrence were not effective as demonstrated by the February 21, 1997, reactivity management event. The problem that occurred on September 16, 1996 was the subject of a Notice of Violation (50-295/96014-02) issued on January 28, 1997. (02033)

These violations represent a Severity Level III problem (Supplement I) - $110,000

C. Corrective Actions -- Reactor Voiding Event

10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Actions," requires, in part, that measures be established to assure that conditions adverse to quality are promptly identified and corrected, and in the case of significant conditions adverse to quality, that measures be established to assure that the cause of the condition is determined and corrective actions taken to preclude recurrence.

Contrary to the above, from September 2, 1996, through March 8, 1997 the licensee failed to take prompt corrective actions commensurate with the undetected accumulation of gas in the reactor vessel head which displaced reactor coolant and threatened the ability to maintain decay heat removal. Engineering and Operations personnel had determined the corrective actions necessary to preclude recurrence but failed to revise appropriate operations procedures prior to the recurrence of the event on March 8, 1997, in both Units 1 and 2. (03013)

This is a Severity Level III Violation (Supplement I) - $110,000

II. Violations Not Assessed a Civil Penalty

A. Failure to Comply With a Limiting Condition for Operation (LCO)

Technical Specification (TS) 3.1. "Reactor Protection Instrumentation and Logic," requires, that with the minimum number of operable channels below the limits specified by Table 3.1-1, "Reactor Protection Systems - Limiting Operation Conditions and Setpoints," plant operation shall be as specified in Column 5 of Table 3.1-1. Table 3.1-1 specifies that if there are less than a minimum of two operable channels per loop of Low Primary Coolant Flow, maintain Hot Shutdown and, if the minimum conditions are not met within 24 hours, the unit shall be in the Cold Shutdown condition within an additional 24 hours.

Contrary to the above, on February 24, 1997, the licensee failed to comply with the Limiting Condition for Operation of TS 3.1 when Unit 1 was not placed in cold shutdown conditions within 48 hours of rendering all three-reactor coolant system loop "A" flow instrumentation channels inoperable. (04014)

This is a Severity Level IV Violation (Supplement I)

B. Undetected Displacement of Reactor Coolant

10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that activities affecting quality be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and be accomplished in accordance with these instructions, procedures, or drawings.

Contrary to the above, as of March 8, 1997, procedures for activities affecting quality, extended operation in cold shutdown, were not appropriate to the circumstances. Specifically, no operating procedures were prescribed which included measures to diagnose or prevent the displacement of reactor coolant from the reactor vessel caused by the undetected accumulation of nitrogen gas in the reactor coolant system. (05014)

This is a Severity Level IV Violation (Supplement I)

C. Failure to Report the Accumulation of Gas in the Reactor Coolant System

10 CFR Part 50.72(b)(2)(iii)(B) requires that the licensee shall notify the NRC as soon as practical, and in all cases within four hours, of any event or condition that alone could have prevented the fulfillment of the safety function of structures or systems that are needed to remove residual heat.

10 CFR Part 50.73(a)(2)(v)(B) requires that the licensee shall submit a Licensee Event Report within 30 days after the discovery of the event, for any event or condition that alone could have prevented the fulfillment of the safety function of structures or systems that are needed to remove residual heat.

Contrary to the above, the licensee did not notify the NRC within four-hours and did not submit to the NRC a Licensee Event Report within 30 days after discovery that nitrogen gas had accumulated in the reactor vessel head on both Unit 1 and Unit 2 on March 8, 1997. This condition (nitrogen gas accumulation in the reactor coolant system) alone could have caused the loss of both trains of shutdown cooling prior to the nitrogen gas bubble reaching the size where pressurizer level would have provided direct indication of reactor vessel water level and, therefore, could have prevented the fulfillment of the safety function of a system needed to remove residual heat. Additionally, the nitrogen gas in the reactor coolant system could have accumulated in the steam generators which would have resulted in the obstruction of natural circulation cooling. This condition also could have prevented the fulfillment of the safety function of a system needed to remove residual heat. (06014)

This is a Severity Level IV Violation (Supplement I)

Pursuant to the provisions of 10 CFR 2.201, Commonwealth Edison Company (the Licensee) is hereby required to submit a written statement or explanation to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, within 30 days of the date of this Notice of Violation and Proposed Imposition of Civil Penalties (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation" and should include for each alleged violation: (1) admission or denial of the alleged violation, (2) the reasons for the violation if admitted, and if denied, the reasons why, (3) the corrective steps that have been taken and the results achieved, (4) the corrective steps that will be taken to avoid further violations, and (5) the date when full compliance will be achieved. If an adequate reply is not received within the time specified in this Notice, an order or a Demand for Information may be issued as to why the license should not be modified, suspended, or revoked or why such other action as may be proper should not be taken. Consideration may be given to extending the response time for good cause shown. Under the authority of Section 182 of the Act, 42 U.S.C. 2232, this response shall be submitted under oath or affirmation.

Within the same time as provided for the response required above under 10 CFR 2.201, the Licensee may pay the civil penalties by letter addressed to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, with a check, draft, money order, or electronic transfer payable to the Treasurer of the United States in the amount of the civil penalties proposed above, or the cumulative amount of the civil penalties if more than one civil penalty is proposed, or may protest imposition of the civil penalties in whole or in part, by a written answer addressed to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission. Should the Licensee fail to answer within the time specified, an order imposing the civil penalties will be issued. Should the Licensee elect to file an answer in accordance with 10 CFR 2.205 protesting the civil penalties, in whole or in part, such answer should be clearly marked as an "Answer to a Notice of Violation" and may: (1) deny the violation(s) listed in this Notice, in whole or in part, (2) demonstrate extenuating circumstances, (3) show error in this Notice, or (4) show other reasons why the penalties should not be imposed. In addition to protesting the civil penalties, in whole or in part, such answer may request remission or mitigation of the penalties.

In requesting mitigation of the proposed penalties, the factors addressed in Section VI.B.2 of the Enforcement Policy should be addressed. Any written answer in accordance with 10 CFR 2.205 should be set forth separately from the statement or explanation in reply pursuant to 10 CFR 2.201, but may incorporate parts of the 10 CFR 2.201 reply by specific reference (e.g., citing page and paragraph numbers) to avoid repetition. The attention of the Licensee is directed to the other provisions of 10 CFR 2.205, regarding the procedure for imposing a civil penalties.

Upon failure to pay any civil penalties due which subsequently has been determined in accordance with the applicable provisions of 10 CFR 2.205, this matter may be referred to the Attorney General, and the penalties unless compromised, remitted, or mitigated, may be collected by civil action pursuant to Section 234c of the Act, 42 U.S.C. 2282c.

The response noted above (Reply to Notice of Violation, letter with payment of civil penalties, and Answer to a Notice of Violation) should be addressed to: Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555 with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region III and a copy to the NRC Resident Inspector at the Zion facility.

Because your response will be placed in the NRC Public Document Room (PDR), to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be placed in the PDR without redaction. If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.790(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

Dated at Lisle, Illinois
this 2nd day of September 1997

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555

February 22, 1994

NRC INFORMATION NOTICE 94-13: UNANTICIPATED AND UNINTENDED MOVEMENT OF FUEL ASSEMBLIES AND OTHER COMPONENTS DUE TO IMPROPER OPERATION OF REFUELING EQUIPMENT

Addressees

All holders of operating licenses or construction permits for nuclear power reactors.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice to alert addressees to potential problems resulting from inadequate oversight of refueling operations and inadequate performance on the part of refueling personnel. It is expected that recipients will review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, suggestions contained in this information notice are not NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances

Vermont Yankee Events

The Vermont Yankee facility was in a refueling outage with fuel movement in progress on September 3, 1993, when an irradiated fuel assembly became detached from the grapple after being lifted out of its position in the reactor core. The assembly fell approximately 2.4 m [8 ft] back into its original location in the reactor core. The licensee suspended fuel handling and investigated the event. The licensee determined that the grapple had not properly engaged the lifting bail on the fuel assembly and that the personnel performing the fuel handling activities had failed to verify proper grapple engagement. After completing the investigation and taking corrective actions, the licensee resumed fuel handling activities on September 7, 1993.

On September 9, 1993, a fuel assembly that was being moved to a fuel sipping can was inadvertently lowered, instead of raised, striking another core component. The potentially damaged fuel assembly was then moved to the fuel sipping can and the licensee again suspended fuel handling activities. The NRC dispatched an augmented inspection team (AIT) on September 9, 1993, to investigate the fuel handling incidents.

The AIT documented its findings in NRC Inspection Report 50-271/93-81, issued October 21, 1993. The AIT concluded that mistakes made by refueling personnel

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were the immediate causes of both events. In addition, weaknesses in the human factors aspects of the controls for the fuel handling equipment contributed to the event in which a fuel assembly was lowered rather than raised. The controls for the fuel handling equipment had been modified shortly before this event occurred. The team concluded that the root cause of the events was a significant weakness in management oversight of fuel handling activities. Weak management oversight had allowed many of the measures intended to prevent a fuel handling accident to be neglected. The AIT found that (1) design changes were not transmitted to allow timely and accurate training on modifications to the refueling bridge, (2) procedures were not always used and, when they were used, they were not always adhered to, and (3) supervisors did not ensure that procedures were followed. In addition, the AIT found that training was not effective in that operators were not aware of certain key procedure steps in most instances. Specifically, the personnel monitoring the fuel handling activities were not aware of the requirement to visually verify grapple closure when engaging and lifting fuel assemblies. The AIT found that management did not communicate expectations and provide proper oversight of fuel handling activities.

Peach Bottom Events

With Unit 3 shut down for refueling on September 23, 1993, a fuel assembly could not be fully inserted into its spent fuel rack cell. It was thought that the fuel assembly had swelled due to irradiation in the core, and the fuel assembly was successfully placed in a different cell. It was further postulated that there might be some debris in the cell, and that the cell should be checked at some future date. On September 24, 1993, another fuel assembly became stuck in its spent fuel rack cell. The licensee evaluated the material condition of the fuel assembly, calculated an allowable lifting force, and conferred with the fuel vendor. The licensee increased the load limit of the refueling hoist and the fuel assembly was freed from the rack with no damage to the fuel assembly. Subsequent examinations revealed that sections of local power range monitor instrument strings that had previously been cut up were in the bottoms of three cells in the rack, including the two cells with which difficulties were experienced. The licensee believes that the debris may have fallen into the cells during a fuel pool cleanup effort conducted during the previous summer.

The licensee is currently investigating why the debris was in the spent fuel pool and why the refueling personnel did not ensure that the spent fuel rack cells did not contain any debris prior to inserting the fuel assemblies.

Susquehanna Events

The Susquehanna Steam Electric Station Unit 1 was shut down with defueling in progress on October 6, 1993, when the personnel performing the fuel handling activities removed an incorrect fuel assembly from a peripheral location in the core. The personnel involved realized they had removed the wrong assembly and they inappropriately decided to return the assembly to its prior position in the core. The appropriate action, per licensee procedures, would have been to place the bundle in the spent fuel pool and secure fuel handling activities until the cause of the error was determined and corrected.

On October 26, 1993, while lowering a fuel assembly into the core during refueling, an unexpected drop of 25 to 38 cm [10 to 15 in] of one of the sections of the fuel handling mast occurred. The fuel assembly was not released and it did not strike the vessel internals. Subsequent testing reproduced mechanical binding of the mast, and a bend in the mast was observed. The binding temporarily restrained one section of the mast while a lower section extended. Eventually weight or motion caused the bound section to release and drop down a limited distance. The licensee subsequently determined that the mast had been bent by an impact with the flange protector for the reactor vessel while traversing through the "cattle chute" between the core and fuel pool, because the mast was not raised high enough. The Unit 2 refueling bridge was transferred to Unit 1 and, after satisfactory completion of surveillance testing, refueling was resumed.

On October 27, 1993, while transferring a double blade guide to the spent fuel pool, the blade guide hit the side of the reactor vessel because it was not raised high enough to clear the vessel. The licensee suspended refueling activities, revised the associated procedure, and inspected the mast. The core reload was resumed after surveillances on the fuel handling equipment were successfully conducted. On October 28, 1993, while attempting to grapple a new fuel assembly in the fuel pool, the personnel performing the fuel handling activities heard two loud bangs and observed bubbles in the pool for 5 to 10 seconds. Subsequent inspection revealed that one section of the mast from Unit 2 was bent. The licensee believes that the mast was weakened by the impact with the reactor vessel that occurred during the October 27 event.

On October 29, 1993, the NRC dispatched an AIT to the site to review the events. The AIT documented its findings in Inspection Report 50-387/93-80, issued on December 21, 1993. The AIT concluded that facility management did not maintain proper oversight of refuel floor activities and that inadequate corrective actions were implemented in the past for problems with the fuel handling equipment. The AIT also concluded that the licensee fuel handling procedures were adequate for the proper completion of the fuel handling activities, although certain improvements could be made to increase the awareness of the operators concerning potential problems.

Nine Mile Point Event

Nine Mile Point Unit 2 was shut down with refueling in progress on November 1, 1993, when a blade guide was moved from the core into the spent fuel pool. The contractor refueling operator disengaged the grapple and observed the correct light indication on the bridge. There was no procedural requirement to visually verify disengagement or for the Senior Reactor Operator Limited to Fuel Handling (LSRO) or the spotter to verify disengagement. The refueling operator noticed increased drag after the refueling bridge crane had been moved approximately 23 cm [9 in] toward the next location. At that time, licensee personnel determined that the blade guide was still engaged on the grapple. The bridge was returned to its previous position, the blade guide was lowered and disengaged (positive verification was obtained this time), and the operator proceeded to move the next component, which was a fuel assembly. While lowering that fuel assembly

into the core, the refueling operator noticed that the mast was binding. At this point, the LSRO became involved and directed that the fuel assembly be returned to the fuel pool. While lowering the fuel assembly into the rack in the fuel pool, the inner section of the mast dropped between 61 and 76 cm [24 and 30 in]. However, the fuel assembly was not released. After the fuel assembly was lowered, the grapple was disengaged and the LSRO halted further fuel movement. The licensee subsequently determined that the mast was bent and that the blade guide was not damaged. After the licensee reviewed the event, modified the procedure, and repaired the fuel handling equipment, fuel movements were resumed on November 4, 1993.

The licensee determined that there were several personnel performance issues that needed to be addressed. The refueling operator had been trained to verify disengagement after releasing each component, although the procedure only required verification of ungrappling when handling fuel assemblies. Disengagement was to be verified by raising and rotating the mast. The refueling operator did not verify disengagement after releasing the blade guide. In addition, the refueling operator did not notify the LSRO of the unanticipated equipment response (remaining connected to the blade guide while traversing the bridge). Also contributing to the event was the fact that the LSRO was observing a refueling bridge trolley bearing about which he was concerned, rather than the handling of the blade guide. Licensee review determined that management expectations regarding the supervision of refueling activities had not been clearly expressed to the LSROs.

Discussion

Refueling activities are safety-significant operations that are not conducted on a routine basis. In addition, fuel handling activities are often performed by contractor personnel under the supervision of licensee personnel. As a result, fuel handling personnel may not be familiar with the fuel handling equipment or may feel that their experience in fuel handling operations permits them to ignore some requirements for procedural use and adherence. Either of these situations could require increased management attention and oversight by the licensee to ensure proper and safe performance of fuel handling activities.

Appendix B to Part 50 of Title 10 of the Code of Federal Regulations (10 CFR 50) requires licensees to have appropriate procedures to control activities affecting quality (such as the actions to be taken during operation of refueling equipment), and that the procedures are used and followed. In addition, 10 CFR 50.120 requires licensees to implement a training program for various categories of nuclear power plant personnel to ensure that those personnel have the necessary knowledge, skills, and abilities to perform their assigned jobs competently. This rule applies to the personnel (including contractors) who operate or supervise the operation of the refueling equipment. The cases discussed in this notice include situations in which the licensees failed to conduct appropriate training in the use of their refueling equipment, particularly with respect to design modifications made to the controls for the fuel mast. These events also demonstrated that the fuel

handling personnel involved in certain instances were variously not aware that management expected them to identify deviations from expected results, cease operations when an unexpected or abnormal condition is encountered, and notify operations and/or plant management of unexpected or abnormal conditions.

This information notice requires no specific action or written response. If you have any questions about the information in this notice, please contact one of the technical contacts listed below, or the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

ORIGINAL SIGNED BY

Brian K. Grimes, Director
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Attachment:
List of Recently Issued NRC Information Notices

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555-0001

October 14, 1994

NRC INFORMATION NOTICE 94-75: MINIMUM TEMPERATURE FOR CRITICALITY

Addressees

All holders of operating licenses or construction permits for pressurized-water reactors (PWRs).

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice to alert addressees to potentially non-conservative initial conditions that were used in the analysis of some design-basis transients. As a result, some plant technical specifications for minimum temperature for criticality may not be adequately conservative. It is expected that recipients will review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, suggestions contained in this information notice are not NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances

Commonwealth Edison Company (Zion Nuclear Station Units 1 and 2) and New York Power Authority (Indian Point Nuclear Power Plant Unit 3) have informed the NRC that their technical specifications for minimum temperature for criticality were not supported by the safety analyses for their plants. The licensing analysis performed by Westinghouse for Zion assumed a nominal hot-zero-power (no-load) operating temperature of 286 oC [547 oF], but Zion's technical specifications allow criticality if the average reactor coolant system temperature is greater then 260 oC [500 oF]. The safety analysis for Indian Point 3 was also performed at 286 oC [547 oF], but its technical specifications allow criticality at 232 oC [450 oF], a limit which was set by reactor vessel material considerations. A review of Indian Point records indicated that the reactor was brought critical below 286 oC [547 oF] several times in the early life of plant operations (before 1988). The lowest temperature during these instances was 272 oC [521 oF].

After discovering a potential to operate the plant in a region outside that analyzed, both licensees instituted administrative controls to ensure that the minimum temperatures for criticality are bound by the safety analyses performed for their plants. In addition, both licensees submitted license amendments to NRC to revise the minimum temperature for criticality.

Further details concerning these events are in a 10 CFR Part 21 report to NRC prepared for the Zion plant, dated March 18, 1993, and in Indian Point 3 Licensee Event Report 93-046-00, dated December 1, 1993.

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Discussion

For transient analysis performed for hot-zero-power cases, small changes (e.g., 3 oC [6 oF]) in initial conditions such as allowed by standard technical specifications would have a negligible impact on analysis results. However, if PWRs are allowed to achieve criticality significantly below the temperature that was previously analyzed at hot-zero-power, the following safety concerns would be raised:

(1) The transient analyses, such as "rod withdrawal from subcritical," "rod ejection," "zero power feedwater malfunction," and "boron dilution event" documented in the Final Safety Analysis Report, might not have been analyzed at temperatures below hot-zero-power and could be non-conservative. This could cause the analyses results with small margins to violate specified fuel design limits (i.e., centerline fuel melt or departure from nucleate boiling) for one or more of these postulated transients.

(2) The response of the power range ex-core nuclear instrumentation may be adversely effected by the increased density of the reactor coolant at lower temperatures. This could result in a higher power being reached before a power range reactor trip occurs which might violate specified fuel design limits for transients that rely on this trip.

(3) The moderator temperature coefficient will become more positive, perhaps causing a violation of existing technical specifications. Higher values of moderator temperature coefficient could exceed those used in some of the transient safety analyses. A more positive moderator temperature coefficient at power would result in reactivity insertion that could increase the consequences of an anticipated-transient-without-scram event.

New analyses may justify criticality at somewhat lower temperatures. For example, Commonwealth Edison Company performed the necessary safety analyses for the Zion station and the staff approved a technical specification amendment to change the minimum temperature for criticality to 277 oC [530 oF].

This information notice requires no specific action or written response. If you have any questions about the information in this notice, please contact the technical contact listed below or the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

original signed by

Brian K. Grimes, Director
Division of Project Support
Office of Nuclear Reactor Regulation

Technical contact: George A. Schwenk, NRR
(301) 504-2814

Attachment: List of Recently Issued NRC Information Notices

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555

May 9, 1988

NRC INFORMATION NOTICE NO. 88-21: INADVERTENT CRITICALITY EVENTS AT OSKARSHAMN AND AT U.S. NUCLEAR POWER PLANTS

Addressees:

All holders of operating licenses or construction permits for nuclear power reactors.

Purpose:

This information notice is being provided to alert addressees to undesirable procedural practices that could lead to inadvertent criticality events in nuclear power plants. It is expected that recipients will review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, suggestions contained in this information notice do not constitute NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances:

On July 30, 1987, an unplanned criticality event occurred at Oskarshamn Unit 3, a boiling water reactor (BWR) in Sweden, during routine control rod shutdown margin testing. A night shift team, consisting of a shift supervisor, a physicist, and an operator, had decided to proceed with shutdown margin testing, even though they knew that the fast-acting hydraulic scram system was inoperable. A slower acting electric rod insertion system and the boron injection system remained operational.

Upon partial withdrawal of the first control rod, the core unexpectedly went critical. Although the flux rise was indicated on the instrument panels, the team was not immediately aware that the reactor was critical. However, the control logic for the electric system was initiated by the high flux signal, blocking further withdrawal and reinserting the control rod.

The team then reset the electric control system and continued the test on a second control rod without further incident. The night shift was in the process of testing a third control rod when they were relieved by the day shift. The night shift apparently failed to inform the day shift of the

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inoperable fast-acting hydraulic scram system. The day shift had completed testing the third rod and had started testing the fourth rod when they discovered that the hydraulic scram was inoperable. They then stopped further testing and reported the situation to their supervisors.

Several unplanned criticality events have occurred at nuclear power plants in the United States, although none have been reported in which the crew deliberately withdrew rods with an inoperable scram system.

On November 7, 1973, an inadvertent criticality event occurred at the Vermont Yankee nuclear power plant when an operator withdrew a control rod with the adjacent rod already in the fully withdrawn position. At the time of the event, the reactor was shut down with the reactor vessel and primary containment heads removed, and the refueling cavity above the reactor vessel flooded. Control rod friction tests and core verification procedures were in progress simultaneously. To allow traversing of the television camera mounted on the fuel grapple while rods were being withdrawn for the friction tests, the operators used jumpers to defeat the refueling interlock of the manual control system for the control rods. Contrary to the normal refueling condition, this action permitted the withdrawal of more than one rod at a time.

As the rod testing progressed, the rod in position 30-23 was inadvertently left in the fully withdrawn position. Meanwhile, the core verification procedure was completed, but the interlock jumpers were not immediately removed. The reactor operator conducting the rod testing failed to observe that rod 30-23 was still withdrawn and withdrew an adjacent rod in position 26-23. At about rod notch position 16, the reactor went critical. Somewhere between notch positions 20 and 26, the operator saw the power rising on the nuclear instruments and attempted to insert the rod. However, a full scram was initiated by the high-high flux signals on the intermediate-range monitors.

Because the scram system remained operational and terminated the power rise, the event did not cause any serious consequences. The dosimeter readings of the personnel working on the refueling floor were normal. Five fuel assemblies in the affected area were removed for inspection and testing, and no damage was found.

On November 12, 1976, an inadvertent criticality event occurred at Millstone Unit 1 when an operator withdrew the wrong control rod during a shutdown margin test on a partially loaded core. The operator was supposed to withdraw two diagonally adjacent control rods, in positions 46-23 and 42-19, as part of a shutdown margin test during a core-loading procedure. The operator had positioned rod 46-23 at notch 10, but then erroneously selected the rod at position 46-19, which was directly adjacent to the first rod, and withdrew it to notch 10 also. He then continued to withdraw rod 46-23 in steps. When rod 46-23 was

withdrawn from notch 14 to notch 16, the reactor went critical and scrammed a few seconds later. A few minutes later, the operator made the same error, withdrawing both rods 46-23 and 46-19 to notch 10 and then attempted to withdraw rod 46-23 further. This time the operator saw that the startup-range monitor was increasing and inserted the rods before another scram occurred.

Once again, the presence of an operational scram system prevented any serious consequences from this event. Personnel exposures were normal. The four fuel bundles surrounding rod 46-23 were removed, partially disassembled, and examined. No damage was found.

Discussion:

These events highlight the importance of maintaining an operable fast-acting scram capability whenever any coupled control rods are withdrawn from a reactor core. Licensees are encouraged to review their procedures and training programs to ensure there is no ambiguity on this point.

These events also highlight the importance, during control rod manipulations, of following procedures and staying alert to the relevant instrumentation, even when the reactor is not expected to become critical. In each of the three cases described, procedures were violated. At Oskarshamn, withdrawing control rods with the fast-acting scram inoperable was a violation of the plant procedures. At Vermont Yankee, the "Lifted Lead Log" procedure that was required to be used for jumper installation was not adhered to. The jumper installation was not recorded in the general plant log and consequently operating personnel were not adequately informed of the jumpered interlock status. In addition, the jumpers were not removed immediately after core verification. At Millstone, the operator who was performing the shutdown margin test reselected the incorrect control rod and tried to withdraw it a second time without determining the cause of the initial reactor scram.

In each of the three cases, the operators failed to observe indications on the instruments that could have prevented or mitigated the event. At Oskarshamn, when the core unexpectedly went critical, the flux rise was indicated on the instrument panels. However, the operators were not immediately aware that the reactor was critical. At Vermont Yankee, the operator failed to observe that rod 30-23 was mistakenly left in the withdrawn position, though it was later proved that the rod's digital position display was functioning properly. At Millstone, the operator observed that the startup-range monitor was increasing during the second erroneous rod withdrawal and managed to prevent a second reactor trip. If the operator had observed that the startup-range monitor was increasing the first time, both the initial criticality and the subsequent repetition of the error might have been prevented.

No specific action or written response is required by this information notice. If you have any questions about this matter, please contact the technical contacts listed below or the Regional Administrator of the appropriate regional office.

Charles E. Rossi, Director
Division of Operational Events Assessment
Office of Nuclear Reactor Regulation

Technical Contacts: Robert J. Giardina, NRR
(301) 492-1188

Donald C. Kirkpatrick, NRR
(301) 492-1152

Attachment: List of Recently Issued NRC Information Notices

Attachment
IN 88-21
May 9, 1988
Page 1 of 1

LIST OF RECENTLY ISSUED
NRC INFORMATION NOTICES

Information Notice No.	Subject	Date of Issuance	Issued to
88-20	Unauthorized Individuals Manipulating Controls and Performing Control Room Activities	5/5/88	All holders of OLs or CPs for nuclear power, test and research reactors, and all licensed operators.
88-19	Questionable Certification of Class 1E Components	4/26/88	All holders of OLs or CPs for nuclear power reactors.
88-18	Malfunction of Lockbox on Radiography Device	4/25/88	All NRC licensees authorized to manufacture, distribute, and/or operate radio-graphic exposure devices.
88-17	Summary of Responses to NRC Bulletin 87-01, "Thinning of Pipe Walls in Nuclear Power Plants"	4/22/88	All holders of OLs or CPs for nuclear power reactors.
88-16	Identifying Waste Generators in Shipments of Low-Level Waste to Land Disposal Facilities	4/22/88	Radioactive waste collection and service company licensees handling prepackaged waste, and licensees operating low-level waste disposal facilities.
88-15	Availability of U.S. Food and Drug Administration (FDA)-Approved Potassium Iodide for Use in Emergencies Involving Radioactive Iodine	4/18/88	Medical, Academic, and Commercial licensees who possess radioactive iodine.
88-14	Potential Problems with Electrical Relays	4/18/88	All holders of OLs or CPs for nuclear power reactors.

OL = Operating License
CP = Construction Permit

MEMORANDUM FOR: Jack E. Rosenthal, Chief
Reactor Operations Analysis Branch
Division of Safety Programs
Office for Analysis and Evaluation
of Operational Data

FROM: Sanford L. Israel
Reactor Systems Section W and B&W
Reactor Operations Analysis Branch
Division of Safety Programs
Office for Analysis and Evaluation
of Operational Data

SUBJECT: REVIEW OF MISPOSITIONED EQUIPMENT

Enclosed for your information and use is a technical review report on mispositioned equipment. The safety importance of this issue is the unavailability of safety-related equipment when needed to mitigate an accident. This study examined over 190 mispositioned equipment events for the period 1990 to 1993. Most of the events concerned mispositioned valves and about 15 percent of them involved multiple components. The personnel errors associated with these situations varied widely from improvisation in the absence of adequate procedures to apparent false sign-off on check lists. About half of the events were cited in NRC inspection reports and about one-third were identified as violations. Some of the violations had fines from $25,000 to $150,000.

Regulatory Guide 1.47 addresses automatic status indication for safety systems and TMI Action Plan Item I.C.6 addresses independent verification of alignment when returning a system from maintenance or testing. This guidance appears adequate to address the issue, but its implementation is deficient from time to time. A rough analysis of the human error probabilities and the potential system unavailabilities associated with the data indicates that the safety impact is below what was previously analyzed in probabilistic risk assessments. In addition, the NRC inspectors monitor configuration control at the plants using inspection modules and are accustomed to citing violations. Consequently, no new initiatives appear to be warranted at this time.

Original signed by

Sanford L. Israel
Reactor Systems Section W and B&W
Reactor Operations Analysis Branch
Division of Safety Programs
Office for Analysis and Evaluation
of Operational Data

Enclosure: As stated
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TECHNICAL REVIEW REPORT

REVIEW OF MISPOSITIONED EQUIPMENT EVENTS

MAY 1994

Prepared by:
Sanford Israel

Reactor Operations Analysis Branch
Division of Safety Programs
Office for Analysis and Evaluation
of Operational Data

cc w/enclosure: J. Wiggins, RI
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AEOD/T94-02

TECHNICAL REVIEW REPORT

REVIEW OF MISPOSITIONED EQUIPMENT EVENTS

MAY 1994

Prepared by:
Sanford Israel

Reactor Operations Analysis Branch
Division of Safety Programs
Office for Analysis and Evaluation
of Operational Data

SUMMARY

Over 190 mispositioning events were reviewed for the period 1990 to 1993. Most of the events involved mispositioned valves and about 15 percent involved mispositioning multiple components. The personnel errors associated with these events cover a wide range of lapses. The independent verification process meant to catch mispositioned equipment is not always successful. The licensees generally discipline or counsel the personnel involved in the error rather than make tangible plant modifications such as status alarms and position markers. The overall safety impact of these deficiencies appears to be small.

INTRODUCTION

An Enforcement Action (EA) ($100,000 fine) for mispositioned root valves at Catawba in 1990 prompted this review of mispositioned equipment caused by personnel errors. Restoration errors occur following maintenance, surveillance, and refueling outages. The Catawba event happened during a reactor vessel refill evolution that involved isolated pressure sensors that simultaneously defeated overpressure protection actuation and reactor vessel pressure readout in the control room (CR) and resulted in an unnoticed plant pressurization. Similarly, closed valves in the emergency feedwater system contributed to the initiation of the Three Mile Island (TMI) accident. Subsequent to that accident, the NRC issued Bulletin 79-06 which in part required verification of the operability of all safety-related systems when they are returned to service following maintenance or testing. TMI Action Plan, Item I.C.6 (NUREG-0737), required verification of system configuration when returning from maintenance and testing. Information Notice (IN) 84-51 provided additional amplification on independent verification and summarized several mispositioned equipment events observed after the TMI accident.

The NRC has a long standing concern about mispositioned equipment going back to Criterion XIV, "Inspection, Test, and Operating Status," of 10 CFR 50, Appendix B and Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems." This guidance calls for automatic indication in the CR of inoperable trains of safety systems. In addition to the above verification requirements, the NRC required post-maintenance testing in Generic Letter (GL) 83-28, "Required Actions Based on Generic Implications of Salem ATWS Event" which was issued in response to the Salem ATWS event. Post-maintenance testing may capture some of the mispositioned equipment situations, but it may also be the source of mispositioning equipment (inadequate system restoration following the test). The NRC Inspection Manual has modules that include monitoring independent verification of system status and operability testing of equipment being returned to service.

The industry has produced more than 10 studies on mispositioned equipment; the most recent was issued in 1992. These reports provide suggestions for remedial actions based on licensee corrective actions. In 1986, AEOD issued a technical review report (AEOD/T612) on this subject.

A program will be developed that will clearly identify instruments within the Control Room that are either out of service or known to be out of calibration.

River Bend - IR 458/93-20 - The inspectors noted several situations related to system alignment. The following are excerpts from the report:

> The inspectors observed portions of a high pressure core spray (HPCS) valve and pump test. Several steps during the restoration of the HPCS system required an independent verification of the proper valve alignment. During one such verification, the operator performing the procedure handed the verifier the procedure, pointed to the valve switches to be verified, and requested that he perform an independent verification of these valves. The first performer did not appear to realize that he could have defeated the independence of the verifier by pointing out the specific valve switches to be verified.
>
> The licensee's administrative procedure ADM-0022 states, "Independent verification is intended to mean a second check of the position or status of a component or system. The independent verification will be performed separately without visual or audible contact with the first performer."
>
> In another instance, the licensee noted that a test fixture on a source range monitor remained installed for four months since the previous test. A review of the previous test document indicated that the removal of the test fixture had been signed off by an independent verifier.
>
> The corrective actions included:
>
>> Modifications of procedures by re .oving unnecessary verifications, in-process verifications were clearly identified as requiring completion before proceeding, and restoration verifications were in a separate section at the end of the procedure. A human performance engineer was designated to set up a consistent and effective independent verification program. Operations and maintenance departments would be provided instructions on independent verification that would be unique to their respective disciplines. Plant management would hold individual verifiers personally accountable for their actions.

Indian Point 3 - IR 286/91-14 - During a walkdown, an NRC resident inspector noted a boron injection valve fully open (according to the local stem indicator) while tagged in a shut position under an operating order providing reactor coolant system (RCS) protection during mid-loop operations. The valve was presumably set at a throttled position one month earlier. The licensee never determined how the valve became backseated in the full open position.

It was surmised that the reactor operator who was supposed to close the valve for mid-loop operation never moved the valve off of its backseat. Based on a similar incident, the licensee concluded that the reactor operator turned the handwheel only enough turns the inspectors did not identify why the valves were open. The documentation associated with the testing indicated that the testing was complete and the independent verification step of the restoration process, which required the valves to be shut, had been completed even though previous steps of the restoration section had not been performed.

A review of the watch station turnover sheets in the CR contained a note that the valves should be closed following recharging of the nitrogen cylinders. The inspector noted "Although the operations department administrative procedures allow procedure steps to be performed out of sequence with the unit/shift supervisor approval, and the entry on the unit supervisor's turnover sheet satisfies the intent of the administrative procedures regarding control of components manipulated outside of prescriptive procedures, the lack of a unit log entry indicating the manipulation of major components and the lack of awareness by the reactor operator of the valves' positions and purpose was identified by the inspectors to the licensee as a poor operating practice."

After soliciting suggestions from the operating staff, the licensee instituted a valve manipulation log sheet to record the manipulations of any valve performed without specific procedural control.

DISCUSSION

Background

The requirements for configuration control arise from the regulations in 10 CFR 50, Appendix B and 10 CFR 50.55a which embraces IEEE Standard 279-1971. Regulatory Guide 1.47 expands on IEEE Standard 279 by defining an acceptable method for implementing this requirement with respect to indicating the bypass or inoperable status of portions of the protection system, systems actuated or controlled by the protection system, and auxiliary or supporting systems:

1. Administrative procedures should be supplemented by a system that automatically indicates at the system level the bypass or the deliberately induced inoperability of the protection system and the systems actuated or controlled by the protection system.

2. The indicating system of one above should also be activated automatically by the bypassing or deliberately induced inoperability of any auxiliary or supporting system that effectively renders inoperable the protection system and the systems actuated or controlled by the protection system.

3. Automatic indication in accordance with 1 and 2 above should be provided in the CR for each bypass or deliberately induced inoperable status.

This guidance was to be implemented, where practical, recognizing all the possible means by which safety related systems could be completely or partially rendered inoperable.

SUMMARY

Over 190 mispositioning events were reviewed for the period 1990 to 1993. Most of the events involved mispositioned valves and about 15 percent involved mispositioning multiple components. The personnel errors associated with these events cover a wide range of lapses. The independent verification process meant to catch mispositioned equipment is not always successful. The licensees generally discipline or counsel the personnel involved in the error rather than make tangible plant modifications such as status alarms and position markers. The overall safety impact of these deficiencies appears to be small.

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The NRC has a long standing concern about mispositioned equipment going back to Criterion XIV, "Inspection, Test, and Operating Status," of 10 CFR 50, Appendix B and Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems." This guidance calls for automatic indication in the CR of inoperable trains of safety systems. In addition to the above verification requirements, the NRC required post-maintenance testing in Generic Letter (GL) 83-28, "Required Actions Based on Generic Implications of Salem ATWS Event" which was issued in response to the Salem ATWS event. Post-maintenance testing may capture some of the mispositioned equipment situations, but it may also be the source of mispositioning equipment (inadequate system restoration following the test). The NRC Inspection Manual has modules that include monitoring independent verification of system status and operability testing of equipment being returned to service.

The industry has produced more than 10 studies on mispositioned equipment; the most recent was issued in 1992. These reports provide suggestions for remedial actions based on licensee corrective actions. In 1986, AEOD issued a technical review report (AEOD/T612) on this subject.

DESCRIPTION OF EVENTS

Over 190 licensee event reports (LERs) and inspection reports (IRs) involving mispositioned equipment were collected for the period 1990 through 1993. Summaries of these situations are presented in the Appendix. The reports are about evenly divided between LERs and IRs. Violations were reported in about one-third of the references and eight licensees were fined $25,000 to $150,000 for infractions related to mispositioned equipment.

The number of events reviewed in this study does not reflect the frequency of this problem. Note that only half of the events were reported in LERs, while the others were uncovered by the NRC inspectors. One licensee indicated that he recorded 10 times more mispositioned equipment events than were reported in LERs. Since these additional events did not meet the reporting threshold, they may have less safety significance. However, the high incidence may be indicative of the general issue of configuration control.

Several of the more illuminating events are described below:

Catawba - Special Report (DPC, 4/26/90) - On March 20, 1990, the plant was in Operating Mode 5 performing a pressure vessel fill and vent evolution. The operators at the controls were unaware of the primary system pressure increase because the root valves for system pressure transmitters were valved out by instrumentation and electrical (I&E) personnel on February 7, 1990, for previously scheduled maintenance on compression fittings. This maintenance work was completed on February 21, 1990. These pressure transmitters not only provided indication in the CR, they also actuated the power operated relief valves (PORVs) as part of the low temperature overprotection system (LTOP). The pressure increase occurred when the head vents were isolated while the charging system continued to add fluid to the primary system. Relief valves in the residual heat removal (RHR) suction lines relieved the charging flow so the repressurization stayed within acceptable limits.

As noted in a human factors review of the Catawba event in Ref. 1, "No permanent record or tag-out of the inoperability of these three pressure instruments is made in the CR, (i.e., no out-of-service tag is hung on the indicators). The I&E group is considered to have operational responsibility for instruments..." The licensee's corrective actions to avoid further violations (Ref. 2) were:

> The program to assure equipment operability during mode and condition changes was more fully described in station procedures. Procedure sign-offs for other groups are being incorporated into the controlling procedures for identified condition changes.
>
> The outage schedule will provide logic for scheduling of Technical Specification plant conditions as identified.

A program will be developed that will clearly identify instruments within the Control Room that are either out of service or known to be out of calibration.

River Bend - IR 458/93-20 - The inspectors noted several situations related to system alignment. The following are excerpts from the report:

The inspectors observed portions of a high pressure core spray (HPCS) valve and pump test. Several steps during the restoration of the HPCS system required an independent verification of the proper valve alignment. During one such verification, the operator performing the procedure handed the verifier the procedure, pointed to the valve switches to be verified, and requested that he perform an independent verification of these valves. The first performer did not appear to realize that he could have defeated the independence of the verifier by pointing out the specific valve switches to be verified.

The licensee's administrative procedure ADM-0022 states, "Independent verification is intended to mean a second check of the position or status of a component or system. The independent verification will be performed separately without visual or audible contact with the first performer."

In another instance, the licensee noted that a test fixture on a source range monitor remained installed for four months since the previous test. A review of the previous test document indicated that the removal of the test fixture had been signed off by an independent verifier.

The corrective actions included:

Modifications of procedures by removing unnecessary verifications, in-process verifications were clearly identified as requiring completion before proceeding, and restoration verifications were in a separate section at the end of the procedure. A human performance engineer was designated to set up a consistent and effective independent verification program. Operations and maintenance departments would be provided instructions on independent verification that would be unique to their respective disciplines. Plant management would hold individual verifiers personally accountable for their actions.

Indian Point 3 - IR 286/91-14 - During a walkdown, an NRC resident inspector noted a boron injection valve fully open (according to the local stem indicator) while tagged in a shut position under an operating order providing reactor coolant system (RCS) protection during mid-loop operations. The valve was presumably set at a throttled position one month earlier. The licensee never determined how the valve became backseated in the full open position.

It was surmised that the reactor operator who was supposed to close the valve for mid-loop operation never moved the valve off of its backseat. Based on a similar incident, the licensee concluded that the reactor operator turned the handwheel only enough turns

(1-2/3) in order for the clutch keys on the clutch sleeve to mate with the lugs on the bottom of the handwheel thus allowing the resistance of the backseat to be felt. The operations personnel did not trust the local valve stem indication because a temporary procedure change had deleted reference to the stem indicator because it was considered unreliable.

The corrective actions included:

> The training department would stress the importance of evaluating a situation when plant indication contradicts the expected plant conditions. The valves were modified with permanent reliable position indicators. The long disconnected motor operators on the valves were removed and replaced with manual operators.

Summer - IR 395/90-18 - An auxiliary operator discovered the motor of a component cooling water pump to be hotter than normal. Investigation revealed that a chiller water outlet valve was closed instead of open and a crossconnect valve was open instead of closed. The valves were apparently mispositioned about a week earlier during a train swap-over evolution. The sign off sheets on the swap-over of the outlet valve were signed off by two auxiliary operators. The crossconnect valve should not have been disturbed by the evolution. The valves were separated by 15 feet, easily identifiable, and located above each motor.

Turkey Point - IR 250/93-22 - Prior to a maintenance activity, the NRC inspector reviewed the clearance and determined that the tags were clearly printed and positioned. During a follow-up inspection after the system was restored, the inspector identified a mispositioned valve that was locked open instead of closed. The clearance documentation indicated that the valve was locked closed by one operator and verified closed by another operator 17 minutes later. The two individuals involved were an experienced non-licensed auxiliary nuclear plant operator and an experienced senior reactor operator (SRO) who was nuclear watch engineer. Both individuals were disciplined.

Braidwood - IR 456/91-24 - An operator attempted to change out a seal filter on Unit 2, but instead the filter was partially ejected from the housing and contaminated water spilled. The investigation revealed that, earlier, the technician and independent verifier had entered the valve room for Unit 1 to isolate the filter. The inspector noted that the out-of-service restoration required several sets of anti-contamination clothing to complete the task. Subsequently, the licensee determined that the independent verifier had not taken an adequate number of sets and concluded that the verifier attempted to verify the valve positions associated with the event from beyond the radiological barrier.

The licensee installed more visible valve tags, disciplined the individuals involved, and remarked the valve rooms to clearly indicate the contaminated valve's respective unit.

Comanche Peak 2 - IR 445/93-26 - While performing a control board walkdown following testing of feedwater isolation valves, the inspectors observed that all four valves were opened. A review of the unit log and questioning the operator at the controls by the inspectors did not identify why the valves were open. The documentation associated with the testing indicated that the testing was complete and the independent verification step of the restoration process, which required the valves to be shut, had been completed even though previous steps of the restoration section had not been performed.

A review of the watch station turnover sheets in the CR contained a note that the valves should be closed following recharging of the nitrogen cylinders. The inspector noted "Although the operations department administrative procedures allow procedure steps to be performed out of sequence with the unit/shift supervisor approval, and the entry on the unit supervisor's turnover sheet satisfies the intent of the administrative procedures regarding control of components manipulated outside of prescriptive procedures, the lack of a unit log entry indicating the manipulation of major components and the lack of awareness by the reactor operator of the valves' positions and purpose was identified by the inspectors to the licensee as a poor operating practice."

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2. The indicating system of one above should also be activated automatically by the bypassing or deliberately induced inoperability of any auxiliary or supporting system that effectively renders inoperable the protection system and the systems actuated or controlled by the protection system.

3. Automatic indication in accordance with 1 and 2 above should be provided in the CR for each bypass or deliberately induced inoperable status.

This guidance was to be implemented, where practical, recognizing all the possible means by which safety related systems could be completely or partially rendered inoperable.

The scope and depth of the implementation of this guidance varies among the plants. At some plants, all the components in the support systems are included in the safety system status indication; at others, status indication includes only limited frontline components.

The Three Mile Island accident was initiated because discharge valves on the auxiliary feedwater system were incorrectly closed. Immediately following the accident, the NRC issued Bulletin 79-06 which required, in part, that procedures be reviewed to assure that valves remain positioned in a manner to ensure the proper operation of engineered safety features and that they are returned to their correct positions following necessary manipulations. Further, procedures were to be reviewed and modified to ensure verification of the operability of all safety related systems when they are returned to service following maintenance or testing.

About 1 year later, this bulletin was followed by TMI Action Plan Item, I.C.6, which required in part that procedures be reviewed and revised to assure that for the return-to-service of equipment important to safety, a second qualified operator should verify proper system alignment unless functional testing can be performed without compromising plant safety, and can prove that all equipment, valves, and switches involved in the activity are correctly aligned.

In a related circumstance, the Salem ATWS event in 1983 precipitated GL 83-28 which required, in part, that licensees review procedures to assure that post-maintenance operability testing of all safety related equipment is required to be conducted and that the testing demonstrates that the equipment is capable of performing its safety functions before being returned to service.

In 1984, the NRC issued IN 84-51, "Independent Verification," because of continuing mispositioned equipment events. Three observations were made in this IN:

> "Functional tests used in lieu of independent verification, should be examined to ensure they test the entire portion of the system affected by the previous actions. For example, performing a normal surveillance by running a pump on recirculation may not verify correct alignment of all valves in the system.
>
> Independent verification should be independent with respect to personnel, i.e., two appropriately qualified individuals, operating independently, should verify that equipment has been properly returned to service. Both verifications are to be implemented by procedure and documented by the initials or signature of the two individuals performing the alignment and verification.
>
> In certain instances it may be possible to accomplish one verification from observing CR instruments, annunciators, valve position indicators, etc. This is acceptable as long as the CR indication is a positive one and is directly observed and documented."

Thus, on numerous occasions the NRC has clearly enunciated a concern about restoration of system function following equipment manipulations. This concern is further reenforced by the use of inspection modules to monitor the licensee's tag-out process, independent verification of equipment status, and operability testing when returning equipment to service.

Based on discussions with several licensees, there appears to be a general form. : for administrative controls of equipment status. Standard check lists and independen: verification are used at each plant to assure that safety systems are operable ..n changing operating modes during a return to power. These check lists and verification processes vary from plant to plant and new items are added when omissions are discovered. The verifier may be someone who accompanies the restorer or someone truly independent who walks down the system with a check list after it is restored.

Administrative controls for taking systems out of service and restoring them following maintenance or testing are more variable depending on the equipment in question. The most stringent controls appear to be applied to work orders for mechanical components such as pumps, valves, and piping. There is a detailed tagging process that includes detailed procedures identifying specific boundary components that will be manipulated, maintenance or test procedures that identify specific components that will be manipulated within the service boundary, a dedicated reactor operator in the CR who processes these work orders to assure the specified component lists are complete, approval by the shift supervisor of the work package, and independent verification that the specified components have been restored.

Generally, there is no tagging process for work orders on electronic components, although there may be status indication in the CR. The instrumentation and control (I&C) personnel do not have detailed procedures with check-off lists identifying manipulated components. There is no CR review similar to that imposed on maintenance activities and the "independent verifier" accompanies the technician during his activities. The argument for this approach is that the I&C perform troubleshooting which can not be easily prescribed beforehand so they are given latitude in their activities. The second person (verifier) is supposed to assure that all the various reconnections and root valve manipulations are performed based on his continuous observation of the technician's activities.

Chemistry personnel have similar latitude in their actions and may not have a second verifier in tow. The procedures may be more specific regarding opening and closing valves in sample lines. In one instance, a technician forgot to close a redundant sampling valve in an evolution that he had previously performed over 50 times.

Evaluation of Operational Data

Examination of the events in the appendix indicates that most involved mispositioned valves with mispositioned switches a distant second. Mispositioned drain plugs, circuit breakers, fire barriers, dampers, and sensors accounted for less than 10 percent of the events. Only 15% of the events involved multiple components. The occurrence of these events does not appear to be changing significantly over the past 4 years.

The reasons for the mispositioned equipment cover a wide range of human foibles. A sampling of the errors noted in the events includes:

The procedures were incomplete so the technician improvised.

The independent verifier would have to change in and out of anti-contamination gear several times to perform check-off function. A suit count indicated that he didn't bother and checked off valve positions anyway.

Shift supervisor permitted impromptu change in evolution that he noted in his blackbook, but he didn't inform the rest of the CR staff who were following formal equipment control documents.

Test procedure was aborted in the middle and the equipment was never properly restored.

Independent verifier accompanied the technician who indicated the steps and equipment he was to sign-off on.

Equipment left in wrong position even though there was a sign-off by an independent verifier.

No procedure used in evolution, so the technician incorrectly improvised based on his knowledge.

Technician rotated manual valve 1-2/3 turns and felt resistance that he incorrectly interpreted as a closed valve.

Deleted valve position markers from procedures.

Technician aware that another test was to be performed with some of the equipment so he abandoned the current test procedure before the equipment was restored.

Tag-out sheet used incorrect valve names but correct valve numbers. The technician focused on the valve name, not the number.

Many valves had missing name tags, and the licensee delayed replacing them.

Duct tape was not removed from the exhaust ports of several air operated valves even though action was specified in procedures. When installed, the exhaust port was not readily visible and the tape was the same color as the valve.

All four AFW flow control valves were in wrong position for four days, 11 shift turnovers. It was surmised that everybody thought the positions were correct because they were all the same.

Freeplay in handwheel on a butterfly valve negated the use of the number of turns as a basis for maintaining correct valve throttle position.

Deficient independent verification is a concern as noted in five of the examples discussed above. At River Bend, independent verification was defeated when the technician who aligned the system prompted the "independent" verifier to sign-off a specific action. It appears that the "independent" verifiers falsely signed-off at Summer, Turkey Point, Braidwood, and Comanche Peak. At Turkey Point, the "independent" verification was performed presumably 17 minutes later by an experienced SRO after the valve was restored. The root causes of these types of human failures are usually not determined, but probably range from deliberate (we are in a hurry to get the plant started) to boredom (I have done this many times before) to lazy (I don't want to suit up to check the valve position) because the equipment are normally found in their correct positions.

Lack of control of I&C activities is another problem area as noted above. At Catawba, the I&C personnel left the root valves closed on redundant pressure instruments. Other examples can be found in the Appendix. One licensee indicated that 50 percent of the mispositioned equipment events were caused by I&C personnel, though this large fraction was not indicated by the events in the Appendix.

Events at Summer and Braidwood above involved wrong component or wrong unit. In both cases the verifier signed-off the check list. At Comanche Peak, informal approvals by the shift supervisor left the operator at the controls out of the communications loop which contributed to a mispositioned valve. At Indian Point, the operators ignored a valve stem indication and incorrectly judged resistance to manual movement to mean a closed valve. Technicians improvised in the absence of specific directions, sometimes they improvise in spite of having directions available.

The licensee corrective actions varied from soft:

> Operations senior management formally established a policy for restoration of equipment to operable status; the licensee held shift briefings to stress the importance of self-checking and independent verification; disciplinary actions against personnel directly involved; more control of contractor personnel; verbatim compliance with procedures was reemphasized; management suspended all work and held meetings with all personnel stressing their expectation regarding procedural compliance. If the procedure can not be performed as written, personnel were to stop and have the procedure changed. Steps are to be performed in sequence (a change from previous policy);

to semi-soft:

> The surveillance test procedure was modified to require a second verification that a component out of service is being cleared; independent verification procedure was modified to include safety related manual valves; requirement not to sign clearance tags until component is actually observed in proper position and the tags were modified to include a space for the independent verifier's signature;

to hard:

> Marking rings have been installed on all emergency core cooling system (ECCS) throttle valves; developed a device to fix dampers in position and indicate if their position has changed from previous check; Enhancements have been made to local valve position indication; valves were painted purple to identify them as potential release paths; put covers over switches; added a redundant alarm independent of switch position; alarm annunciation installed.

This same range of corrective actions was identified in industry reports on mispositioned equipment.

A frequent licensee correction is to reiterate self verification or STAR - Stop, Think, Act, Review. This is an important consideration, but does not comply with regulatory expectations of "independent" verification or automatic status indication. Discussions with licensees indicate that the shift supervisor determines how verification is to be accomplished. If time is not critical, then a second individual may be sent out after completion of the equipment alignment to verify its status using check lists. If time is money, then the verifier will accompany the equipment restorer. As noted above, I&C actions do not appear to have separate "independent" verifiers as a rule.

In several instances, the licensee implemented corrective actions relevant to regulatory expectations. They installed marking rings so that correct throttle positions could be ascertained easily and they installed alarms in the CR for easy operator recognition that safety equipment is not available. A palpable action to correct mispositioned equipment deficiencies instead of a management/procedure modification was not a major resolution for most of these events.

Safety Importance

The importance of this issue is that mispositioned equipment may leave a safety system unavailable to mitigate an accident for which it was designed. This is especially a concern if both trains are affected.

About 200 mispositioned equipment events were collected for a four year period in this study. Discussions with a licensee indicate that the actual number could be ten times larger (equivalent to 2000) because most events involve only a single train and therefore do not exceed the reporting threshold. A rough estimate of the number of opportunities to misposition equipment at all plants (100) in four years is 4x10(6) based on an assumed 10,000 opportunities per plant-year. Thus, an estimated probability of mispositioning a single component is 5x10(-4) to 5x(10-5). This estimate, even considering uncertainties, is below the estimate of human error probabilities of 10(-3) to 0.5 calculated for mispositoned equipment in Ref. 3.

Similarly, about 30 events involved multiple components. Using this number as a first approximation of common-mode failure of a system caused by mispositioned equipment, an estimate of the probability of system loss by this mechanism is 8x10(-6). Two train system unavailabilities estimated in plant PRAs range from 10(-2) to 10(-3) and 10(-3) to (10-5) for three train systems (Ref. 4). Thus, the estimated contribution of mispositioned equipment to system unavailability is not a major contributor on an industry wide basis.

FINDINGS

1. Mispositioned equipment continues to .cur despite NRC and industry actions. Regulatory guide 1.47 and TMI Action Plan Item I.C.6 impose specific expectations regarding means to minimize these occurrences. The industry has issued over ten reports on the topic. The NRC inspectors have cited numerous violations for mispositioned equipment.

2. The personnel errors leading to mispositioned equipment vary widely. There appears to be a breakdown in the independent verification process which is supposed to provide regulatory assurance that the safety systems are properly aligned.

3. The licensee corrective actions generally do not include tangible modifications such as status alarms and position markers, but rather, they lean toward employee discipline and counseling.

4. Mispositioned equipment appears to be a small contributor to system unavailability on an industry wide basis.

CONCLUSIONS

No new initiatives are warranted at this time. The safety impact of mispositioned equipment is small and existing regulatory guidance addresses the issue adequately. In addition, the NRC inspectors monitor configuration control at the plants through inspection modules and are accustomed to writing citations for observed infractions.

REFERENCES

1. J. Harbour (EGG), Trip Report of Onsite Analysis of Human Factors of Event at Catawba Unit 1 on 900320 (Overpressurization of RHR System), May 31, 1990.

2. Letter from H. Tucker (DPC) to NRC, June 15, 1990.

3. W. Vesely to A. Thadani, Swain's Human Error Probabilities for Leaving Valves in Misconfigured Positions, NRC Memorandum, March 4, 1981.

4. A. El-Bassioni et al, PRA Review Manual, NUREG/CR-3485, August 1985.

APPENDIX

SUMMARIES OF MISPOSITIONED EQUIPMENT EVENTS

Catawba - IR 413/93-34 - Mode change made with manual valves closed on turbine driven auxiliary feedwater pump (TDAFWP). Informal implementation of the removal and restoration process and a misinterpretation of the technical specifications (TS) contributed to this deficiency. Licensee cited for violation.

Salem - IR 311/93-23 - Two instances of improper valve restorations were noted in the IR. These incidents resulted in unexpected fluid discharges. $50,000 fine.

Turkey Point - IR 250/93-26 - Five chemical and volume control (CVCS) valves were found closed after they had been independently verified to be open. The two operators involved stated that they had performed the valve alignment together rather than separately as required by the licensee procedures and training. According to the IR, the lack of independent verification did not violate NRC requirements.

North Anna - IR 338/93-27 - Violation for incorrectly opened diesel generator (DG) breaker after a test by putting switch in pull to lock position.

Browns Ferry - IR 260/93-12 - Violation for five instances with hold order tags not in place with clearances still active and two hold order tags did not correctly specify component position on the sheet.

Millstone - IR 336/93-28; 93-03 - Violation for incorrectly throttled high pressure safety injection (HPSI) valves.

Zion - LER 295/93-08 - A motor operated valve was incorrectly logged back in-service following surveillance activity. There were two previous LERs - 92-23 and 89-06 - which concerned switches that were mispositioned and not identified during control board walkdowns.

Dresden - IR 237/93-27 - A non-cited violation concerned inadequate restoration after a surveillance.

Quad Cities - LER 254/93-17 - A systems engineer initiated draining exhaust pots in high pressure coolant injection lines prior to testing and failed to have valves restored to operable positions following the test. There were two previous LERs - 92-01 and 92-24 - that were concerned with valve misposition.

Diablo Canyon - LER 323/93-02 - Maintenance personnel disabled a second damper in a ventilation system while performing preventive maintenance. There was one previous LER - 92-11.

San Onofre - LER 361/93-05 - A management walkdown discovered bolts missing or broken on tornado blowout panels on 7/15/92. Panels were restored on 9/30/92.

Sequoyah - LER 278/93-02 - Routine containment integrity surveillance identified five, 1/2 inch, drain valves unsecured and two open one-turn.

Pilgrim - LER 293/93-20 - Two ATWS pressure transmitters valved out for 3 hours. The valves were closed during a backfilling procedure which was unclear about which of the two valves in series to close. As a result, the I&C personnel left the valve closest to the instrument rack closed.

Grand Gulf - IR 416/93-11 - Licensee cited for operator failure to follow procedure which resulted in an individual control rod scram from the wrong position. This was the third rod mispositioning in four months.

River Bend - IR 458/93-20 - The licensee was cited for two examples of possible flaws in their independent verification program. There was an inappropriate communication between the performer of system restoration and the independent verifier.

Haddam Neck - LER 213/93-12 - I&C personnel discovered all four steam line flow transmitters isolated and an equalizer valve open while in Mode 3.

Robinson - LER 261/93-06 - The licensee discovered an air return damper inappropriately blocked open with a wooden wedge.

ANO 1 - IR 313/93-06 - The licensee was cited because of a mispositioned locked throttle valve in the AFW bearing cooling return line. The licensee identified several other cases of mispositioned valves.

Prairie Island - IR 282/93-10 - A non-cited violation was noted for failure to perform independent verification of equipment control tags used for configuration control during maintenance activities.

Dresden - LER 249/93-09 - The licensee discovered an isolation valve for a pressure switch closed during a calibration test. Two previous root valve mispositionings were noted in LERs - 93-90 and 92-28.

Quad Cities - LER 254/93-07 - During planned bus manipulations, power was removed from the sample pump for the toxic gas analyzer and wasn't discovered for 7 hours after completion of the bus manipulations.

Waterford - IR 382/93-19 - During the inspection, one violation was noted regarding the failure to adequately implement a plant status control requirement for a locked valve.

Palisades - IR 255/93-12 - Non-cited violation was noted pertaining to the restoration of a hydrogen recombiner following maintenance. There was a failure to execute a restoration switching and tagging order.

Wolf Creek - IR 482/93-14 - A mispositioned valve that rendered a hydrogen analyzer inoperable resulted in a non-cited violation involving an inadequate procedure.

Three Mile Island - IR 289/93-13 - Valve mispositioning event discovered.

Grand Gulf - IR 416/93-07 - Non-cited violation involved mispositioned valves in the RHR system.

ANO 1 - LER 368/93-01 - Original installation of reactor vessel level system probes had miswired sensors whose polarity was reversed. Correction was made at instrument panel. Subsequent sensor replacement with correct polarity did not correct polarity adjustment at instrument panel. Error undetected for 6 months.

Limerick - IR 352/93-09 - Mispositioned valves found during essential service water lineup verification.

Vogtle - LER 425/93-02 - Discovered that interlock for containment building personnel airlock door was defeated.

Cooper - LER 298/93-06 - Two fire barrier doors in reactor building (RB) found open and obstructed with no fire watch assigned.

Braidwood - LER 457/93-01/06 - Head vent inappropriately isolated during RCS draindown - resulted in holding up the water level and providing incorrect level indication. One previous related LER - 92-42.

Peach Bottom - LER 278/93-03 - The head vent valves closed because the instrument air supply valves were closed. The problem was attributed to the operator not fully moving instrument air switch to automatic.

Brunswick - LER 324/93-04 - RHR system isolated when an incorrect fuse was removed from back panel. Caused by incorrect labeling.

Vogtle - LER 424/93-01 - Valving error caused the opposite train to be removed from service.

Millstone - LER 336/93-03 - Licensee discovered mispositioned HPSI valve. Previous LER - 92-04 - had problem with the same system.

Braidwood - IR 456/92-25 - Violation for not implementing corrective action from LER 456/90-14 concerning deferred restoration of equipment.

Summer - IR 395/93-03 - A non-cited violation identified a mispositioned switch on the local control panel for the containment hydrogen analyzer.

Crystal River - IR 302/92-30 - A mispositioned valve was noted in the spent fuel cooling system.

Oconee - IR 269/93-03 - Violations: Unit 1 not maintained in accordance with refueling procedure and in Unit 3, valves not placed in "auto" after restarting main feedwater system.

LaSalle - LER 373/93-02 - Safety relief valve (SRV) stuck open because of duct tape over actuators air valve manifold exhaust port. Other SRVs also had tape on their exhaust ports.

Callaway - IR 483/92-15 - Improper tagging of valve.

Sequoyah - IR 327/92-36 - Violation (EN-93-020) involved inadequate procedures and failing to follow procedures which resulted in mispositioning throttle valves. $50,000 fine.

Peach Bottom - LER 277/92-26 - Outside Appendix R because an emergency service water sluice gate power feed was in the "on" position.

Zion - LER 295/92-23 - Operator discovered that defeat switches were not returned to normal following an abnormal operating procedure action. Previous LER - 89-06 - involved the same switches.

Wolf Creek - IR 482/92-30 - $50,000 fine for mispositioned locked throttle valve in essential service water system (SWS).

Three Mile Island - IR 289/92-20 - Atmospheric monitor not returned to service following surveillance and diesel inoperable for 1 month because of a mispositioned cooling water valve.

Crystal River - IR 302/92-27 - Violation for not following procedures which resulted in misalignment of a valve.

Calvert Cliffs - IR 317/92-27 - Violation for the isolation of the common miniflow line for all ECCS.

Perry - LER 440/92-23 - Discovered mispositioned instrument isolation valve for pressure transmitter.

Turkey Point - LER 250/92-12 - Discovered airlock vent valve open. Caused by incorrect indication.

San Onofre - LER 361/92-09 - Discovered emergency seal water isolation valve closed for salt water cooling pump.

Zion - LER 295/92-20 - AFW discharge valve locked closed. Previous event noted in a DVR in 1990.

Perry - LER 440/92-19 - Valve positioning error disabled both SLCS trains.

Quad Cities - LER 254/92-24 - Drywell vent valve closed because air supply valve closed during scaffold construction. Occurred in spite of extensive prejob briefing of contractor personnel about air valves in the vicinity of the work area.

Davis Besse - LER 346/92-08 - Equalizing valve for pressure switch found open and inoperable.

Brunswick - LER 325/92-25 - Discovered that effluent sampling system not in service when reactor building ventilation started.

Oconee - LER 269/92-13 - Containment isolation valve found open.

Diablo Canyon - IR 275/92-22 - Identified three instances of mispositioned equipment.

Catawba - IR 413/92-22 - Violation for valve misalignments in CVCS, ECCS, and steam generator (SG) blowdown line. One deficiency was the operators incorrectly assumed that alignment was returned by fill and vent procedure. In another instance, the operators failed to close valves within block tag-out. These errors resulted in fluid discharge. The cause of the misalignment of the SG blowdown valves was not determined.

Hatch - LER 386/92-14 - Personnel error resulted in mispositioned valve.

Millstone - IR 423/92-16 - Increase in the number of mispositioned safety-related valves because of procedural inadequacies and personnel errors.

St. Lucie - IR 335/92-11 - Violation noted because of maintenance personnel not restoring peripheral services following equipment modification.

Brunswick - LER 325/92-22 - Main steam line drain valve open while clearance tag indicates it is closed.

South Texas - LER 498/92-06 - All four AFW control valves closed after recovering from reactor trip.

Hatch - LER 321/92-11 - Control switch found in open position rendering excessive flow check valve inoperable.

Millstone - LER 423/92-08 - Plant personnel discovered that eight valves not included in service water system TS valve lineup.

Comanche Peak - LER 445/91-10 - AFW recirculation test line had isolation valve 1/4 turn open even though independently verified after test 14 days earlier.

Millstone - LER 423/92-04 - CR pressurization bottles were found isolated by two manual valves.

Millstone - LER 336/92-04 - HPSI train header valve discovered closed while in Mode 3.

Oconee - LER 287/91-09 - Containment integrity valve found mispositioned during forced outage. Could have been open for 8 months.

Comanche Peak - LER 445/91-30 - Entered Mode 3 with two mispositioned ECCS valves.

Perry - LER 440/91-24 - Discovered keepfill pressure below limit because of mispositioned valve.

Comanche Peak - LER 445/91-29 - Handswitch positions for steam supply valves left in pull to lock after entering Mode 3 thus defeating TDAFWP.

WNP2 - LER 397/91-34 - RHR system differential pressure switch found isolated.

Catawba - LER 413/91-20 - Discovered breaker open for one train of the CR ventilation and chilled water system.

McGuire - LER 369/91-14 - Air handling unit outlet control found in the closed position.

Millstone - IR 336/91-28 - Weakness in the tag-out restoration process was noted in the IR.

Palo Verde - LER 530/91-11 - Equalizing valve on AFW flow transmitter found open. The licensee acknowledged other mispositioned valve events.

Limerick - LER 353/91-12 - Two floor drain plugs needed for RB integrity were removed by maintenance personnel.

Nine Mile Point - LER 410/91-16 - Mispositioned valves identified.

North Anna - IR 338/91-16 - Non-cited violation because a technician failed to close a valve after taking a sample of the demineralizer. The independent verification did not occur.

Millstone - LER 423/91-21 - Containment isolation valve found mispositioned.

Browns Ferry - IR 259/91-24 - Adjacent and different sized fuses were reinstalled in the wrong locations during an equipment restoration evolution. Procedures were not followed.

Indian Point - IR 286/91-14 - Violation cited because personnel failed to close a valve during a maintenance evolution and, despite its position indicator showing the valve full open, proceeded to tag it as shut.

Sequoyah - LER 327/91-17 - Containment radiation monitor (RM) inoperable because inlet valve closed.

Sequoyah - LER 328/91-03 - Breaker for operator for cold leg accumulator incorrectly locked in closed position.

Callaway - IR 483/91-13 and LER 483/91-03 - Violation for inadequate surveillance of position of throttle valve in the SI system.

Vogtle - LER 425/91-08 - SI pump tagged out for maintenance. Caused by procedure inadequacy.

Peach Bottom - LER 277/91-20 - Two diesels discovered inoperable because of a mispositioned fuel oil valve.

Prairie Island - LER 282/91-06 - RM switch in the reset position instead of the operate position.

McGuire - LER 370/91-02 - TDAFWP inoperable because of a mispositioned sliding link on a pressure switch. I&E error.

Salem - IR 272/91-09 - Violation for not releasing tag for ECCS pump and not repositioning suction valve per tagging release work sheet and work order for tagging not signed-off.

Catawba - Special Report 4/22/91 - Diesel didn't reach speed because of mispositioned fuel oil strainer.

Catawba - IR 413/91-11 - Violation because personnel failed to complete assignment of sequence numbers for restoration of generator tag-out equipment.

Seabrook - LER 443/91-03 - Unlocked instrument root isolation valves eventually mispositioned because of inadequate procedure.

Surry - LER 280/91-04 - Fuel oil transfer pump erroneously tagged-out and secured making one of the diesels inoperable.

Catawba - LER 413/91-02 - One train of low pressure safety injection inoperable during power escalation because of closed suction valve.

Surry - LER 280/90-19 - All six main feedwater flow transmitters found isolated, equalized and drained.

Millstone - LER 336/90-22 - Service water, cross-tie header valve found open. $37,500 fine.

Perry - LER 440/90-39 - Both loops of containment spray mode of the RHR system inoperable because of a mispositioned valve. Procedure problem.

Perry - LER 440/90-38 - CR RM isolated for more than 7 days.

San Onofre - EA 90-115 - $150,000 fine for TDAFW inoperable for 55 days.

Perry - LER 440/90-34 - Mispositioned equalizing valve on RV water level instrumentation.

San Onofre - IR 361/90-37 - Violation for leaving sump valve open 4 days.

Catawba - IR 413/90-29 - Violation for not following a procedure that resulted in a mispositioned valve and the spray-down of a pump room.

Fermi - IR 341/90-13 - Violation for HPSI suction valve mispositioned for 19 hours after surveillance test.

Prairie Island - LER 282/90-13 - Inadvertent mispositioning of 11 heater controls.

Hatch - IR 321/90-15 - Violation for mispositioned valves in the core spray system.

Harris - IR 400/90-14 - Violation because essential chiller was inoperable due to a mispositioned valve.

Robinson - LER 261/90-11 - Fire damper found in the open position instead of closed. This is the only damper that must be closed to be operable.

Maine Yankee - LER 309/90-05 - "Summer Control Switch" in the wrong position which impacted calorimetric calculations.

Summer - IR 395/90-18 - Violation for two chiller system valves mispositioned and two operators failed to verify correct positions which resulted in overheating of component cooling water pump motor.

Zion - LER 295/90-13 - Discovered both primary and emergency water makeup lines isolated seal water tank. Procedural deficiency.

Palo Verde - IR 528/90-20 - Violations for not following procedures for maintaining a locked open valve for an atmospheric dump valve and incorrectly opening a valve which overpressurized the postaccident sampling system.

Turkey Point - IR 250/90-14 - Violation for changing modes with no reactor vessel level instrumentation system operable and one ECCS flow path unavailable. Also noted a containment isolation valve pinned open instead of closed.

St. Lucie - IR 389/90-09 - Violation concerning the control of plant work order tags.

Millstone - LER 423/90-17 - Accumulator isolated unknowingly for 4 hours because operator failed to reopen a valve following a fill operation.

Salem - LER 311/90-24 - Radwaste effluent line monitor left isolated by chemistry personnel.

Peach Bottom - LER 277/90-12 - Valves left closed after removal of blocking permit.

Calvert Cliffs - LER 317/89-19 - HPSI discharge header valves not locked shut per LTOP requirements.

Harris - LER 400/90-13 - Misaligned valve caused unplanned release from waste gas system.

South Texas - LER 498/90-07 - All three trains of containment ventilation isolation in test mode and incapable of actuation for 35 minutes while fuel movement occurring.

Hatch - LER 321/90-08 - Two RV head vent valves found closed.

Seabrook - LER 443/90-12 - Numerous instrumentation valves found mispositioned.

Palisades - LER 255/90-05 - AFW inoperable because backup nitrogen bottles isolated.

Sequoyah - LER 327/90-04 - Handswitch controlling steam supply to AFW pump in manual.

Trojan - IR 344/90-02 - Temporary modification tags still in place 5 months after closeout.

Trojan - LER 344/90-29 - Control switches for HPSI found in pull-to-lock position.

Salem - IR 272/92-01 - RCP seal return RV had an unauthorized gagging device installed.

Catawba - LER 414/90-09 - "Audible rate multiplier" switch found in "off" position during refueling.

Indian Point - IR 247/92-07 - Violation for numerous errors found during a walk-down of a diesel using licensee's check-off list. Both missing valves and mispositioned valves.

Indian Point - LER 286/93-17 - Three-way valve on gas sampling monitor out of position for 1 month.

Indian Point - LER 286/93-42 - SWS in configuration not controlled by plant procedures.

Oconee - LER 270/93-06 - Containment isolation valve mispositioned.

Harris - LER 400/92-06 - Excess flow check valves were mispositioned for 5 years.

Perry - LER 440/92-08 - Discovered that outboard containment isolation valve on RHR system was open and deenergized for 5 hours in Mode 5. Opened as part of a tag restoration evolution.

Catawba - IR 314/90-09 - Violation for leaving block valves closed 3 days on SG PORVs and leaving containment valve seal water system isolated.

Catawba - IR 413/90-10 - $100,000 fine for leaving pressure instrumentation isolated (root valves) when refilling plant.

Clinton - LER 461/90-11 - SW was isolated on both diesels. Operators relied on counting turns on the manual valve, but freeplay in handwheel defeated action.

Zion - ENS 17756 - Mechanic incorrectly turns off dc power switch on diesel during walkdown in preparation for maintenance.

Turkey Point - IR 250/93-22 - Non-cited violation of a mispositioned fire water system valve that was not restored properly by tag-out routine even though there was independent verification.

Nine Mile Point - LER 220/92-05 - Gate to screenhouse forebay was inappropriately closed resulting in net positive suction head problems for SW pumps.

Dresden - LER 249/92-22 - Drained condensate line on isolation condenser degraded performance of system because it allowed the condenser to be bypassed.

Quad Cities - LER 254/93-04 - Drain plugs not installed during removal of floor drain isolation valves.

Point Beach - LER 266/91-07 - Fire barriers had holes without compensating fire watch.

Limerick - LER 352/91-16 - Changed modes with reactor core isolation cooling (RCIC) inoperable.

Limerick - LER 352/91-17 - Fuse was not replaced after performing maintenance on a safeguard transformer and was not discovered for 2 years.

LaSalle - LER 374/91-01 - Open penetration in TS related fire wall without compensating fire watch.

Salem - LER 311/93-01 - Underfrequency protection inoperable because of mispositioned test switch.

Sequoyah - IR 327/92-17 - Violation for entering Mode 4 with inoperable containment spray system.

South Texas - IR 498/92-08 - Violation for four circuit breakers not tagged.

McGuire - IR 369/92-10 - Non-cited violation regarding a containment pressure transmitter valved out because of failure to follow procedures.

North Anna - IR 338/92-03 - Violation for not having emergency diesel generator bypass valve opened and locked during operating procedure.

Perry - IR 440/92-02 - Operators failed to implement written instruction resulting in valve lineup error which caused loss of instrument air to main steam isolation valves (MSIVs).

Oconee - IR 269/91-35 - Violation for misconfigured valves affecting containment isolation and an inadvertent boron dilution of a storage tank over several days.

Catawba - IR 413/91-27 - Violation for three configuration control problems.

Comanche Peak - IR 445/91-62 - Two violations were noted for improper system alignments entering Mode 3.

Braidwood - IR 456/91-24 - Violation for the failure of an independent verifier to note that a seal injection filter was not properly isolated.

Haddam Neck - IR 213/91-25 - Violation for fuel movement without sufficient containment closure.

ANO - IR 313/91-30 - Violation for inadvertently disabling HPSI train.

Byron - IR 454/91-27 - Violation for entering Mode 4 with both trains of containment spray inoperable.

Palisades - IR 255/91-18 - Violation for having pressure switch inoperable for 2 weeks.

Farley - IR 348/91-19 - Violation for both air start headers inoperable on one diesel.

Millstone - IR 245/91-16 - Violation for charging header isolation valve to CR hydraulic unit being mispositioned.

Wolf Creek - IR 482/91-30 - Violation for inoperable RM in containment blowdown path.

Millstone - IR 423/91-16 - Violation for working outside of workscope and rendering PORV inoperable.

Seabrook - IR 443/91-29 - Violation for leaving demineralized water line unisolated following restoration of system.

Farley - IR 348/91-17 - $25,000 fine for leaving recirculation bypass valve open on AFW train.

Palisades - IR 255/91-17 - Violation for failure to return containment spray pumps to service prior to criticality.

Zion - IR 295/91-15 - Violation for entering Mode 3 with an AFW pump inoperable for 2 days.

South Texas - IR 489/91-11 - Violation for finding a number of plant valves with handwheels locked.

Oconee - IR 287/91-09 - Violation for leaving certain valves open when start up initiated.

River Bend - LER 458/92-27 - RCIC not placed in standby prior to changing modes.

Hatch - IR 321/92-12 - Excess flow check valve inoperable and bypassed for 18 hours.

Limerick - LER 92-07 - Reactor enclosure isolation valves reset switches not returned to auto position.

Turkey Point - IR 250/92-10 - A turbine operator replaced back-up nitrogen bottles for MSIVs and failed to realign the valves properly to two MSIVs.

Indian Point - IR 286/91-26 - Violation for automatic voltage control being out of position on DG.

Millstone - LER 423/91-25 - Failed to deenergize solid state protection input relays for cold overpressure protection.

Sequoyah - IR 327/93-09 - Seven safety-related valves mispositioned.

St. Lucie - IR 389/93-05 - Safety injection tank isolation valve left open following test in Mode 5.

Nine Mile Point - LER 220/93-04 - Selector switch for two monitoring systems placed in a position that interrupted auxiliary systems.

Indian Point - LER 286/93-12 - A penetration supply line left disconnected following an integrated leak rate test.

Wolf Creek - LER 482/93-10 - Entered Mode 4 with switches for motor drive auxiliary feedwater pumps in pull to lock. $50,000 fine.

Diablo Canyon - LER 275/92-30 - Valves not sealed open due to personnel error.

Summer - IR 395/90-21 - Violation for not taking adequate corrective action for mispositioned valve events.

South Texas - EA 90-138 - Violation for mispositioned AFW recirculation valve.

Peach Bottom - LER 277/93 07 - Purge valve mispositioned thus defeating RM in dry well.

River Bend - LER 458/92-18 - System pressurized with automatic depressurization system train isolated because of closed root valve.

St. Lucie - LER 389/91-03 - Mispositioned component cooling water valve disabled heat exchanger.

Oconee - IR 269/92-24 - Mispositioned valve in low pressure service water system on Unit 3.

Comanche Peak - IR 445/93-26 - All four feedwater isolation valves found open after a surveillance test and restoration signed off that they were closed.

Comanche Peak - IR 446/92-201 - Violation for not correcting mispositioned valve events.

Salem - LER 272/92-18 - Containment spray system valves found closed during plant start-up.

Turkey Point - IR 250/92-34 - Letdown heat exchanger vent valve open and uncapped.

OE 2473 I HECKMAN 16-MAR-88 16:37 PENNSYLVANIA POWER (PPL)
Subject: Drained Water From Spent Fuel Pools

SUSQUEHANNA UNIT 1
DATE 9/12/87
NSSS/AE GE BWR/BECHTEL
RATING 1085 MWE
DATE OF COMMERCIAL OPERATION UNIT 1, 6/8/83

EVENT DESCRIPTION:

Approximately 11,000 gallons of water were inadvertently drained from the Unit 1 and 2 spent fuel pools through a mispositioned valve. The Unit 1 third refueling outage had just begun. Prior to the outage, the gates between the Unit 1 and Unit 2 spent fuel pools and the spent fuel cask storage pit had been removed. The cask storage pit and each fuel pool are separate pools but are cross-connected when the fuel pool gates are removed. Before flooding the Unit 1 reactor cavity to support fuel movements, a leak test was conducted on the reactor cavity seals. Following completion of the leak test, maintenance personnel restored the valve lineup in accordance with the test procedure. The test procedures restoration valve lineup directed the maintenance procedure assumed the gates between the fuel pools and cask storage pit were installed and the storage pit drained. The valve was opened. The allowed operators were alerted to the problem when low surge tank levels were received on the fuel pool cooling system. The mispositioned valve was closed and the loss of water was stopped.

COMMENTS:

The spent fuel pools were designed to prevent inadvertant draining. However, when the fuel pool(s) are cross-connected to the cask storage pit, the potential for draining the fuel pool(s) exist. To minimize this potential, plant policy and procedures have been revised to minimize the length of time the fuel pool gates are removed. Additional controls have been instituted in the operating procedures to either lock or tag cask storage pit valves to more effectively control their status. The maintenance procedure has been revised to perform valve lineups depending on the status of the gates.

Information Contact: Pat Taylor (717) 542-3188

NORTHEAST UTILITIES

THE CONNECTICUT LIGHT AND POWER COMPANY
WESTERN MASSACHUSETTS ELECTRIC COMPANY
HOLYOKE WATER POWER COMPANY
NORTHEAST UTILITIES SERVICE COMPANY
NORTHEAST NUCLEAR ENERGY COMPANY

September 4, 1987

NE-87-L-915

MEMO

ref: LER 87-015-00

9/19/87

TO: D. B. Miller

Robert L. McGuinness

FROM: R. L. McGuinness
(Ext. 3297)

SUBJECT: Reportability of Spent Fuel Pool Cooling

The Regulatory Compliance Section of Generation Facilities Licensing has investigated(1) the potential reportability of the Haddam Neck Spent Fuel Pool Cooling event of August 14, 1987. Based on this investigation I have concluded that the event is not required to be reported as prompt report nor a 30-day LER, but should never-the-less be reported as a voluntary 30-day LER, designated as "Other" in the LER Reporting form and specifically identified as a voluntary report in the text.

---

(1) The investigation included a review of:

- 10CFR50, Appendix A, GDC-34 and GDC-61
- 10CFR50.72 and 50.73
- The actual event where the non-vital power supply was temporarily lost
- The cooling system design (P & ID).
- The operator and management actions as discussed with your Staff Assistant, August 25, 1987.
- NUREG 1022 and Supplements 1 and 2, "License Event Report System"
- Standard Review Plan (NUREG-0800), Section 9.1.3, "Spent Fuel Pool Cooling....".
- Haddam Neck FSAR, Section 9.1, "Fuel Storage and Handling".
- Haddam Neck Technical Specifications
- NU letter to NRC, providing the Safety Assessment Report for SEP Topic IX-1, "Fuel Storage", dated August 31, 1981.
- NRC letter to NU, providing the final evaluation of SEP Topic IX-1, "Fuel Storage", dated March 1, 1982.
- Discussions held with Andra Asars and Tom Shedlosky (of NRC, Region 1, August 31, 1987).
- Discussions held with Wayne D. Lanning, Branch Chief of the Events Assessment Branch of NRR, August 31, 1987.

SEP 9

Several of the important factors supporting this conclusion derive from my discussion with Wayne Lanning, Chief of the NRC Events Assessment Branch on August 31, 1987. Wayne was very familiar with the event and felt it was reportable before I talked with him. During our discussion I pointed out that NRC guidance provided in NUREG-1022, Supplement 1, page 10, allows that "reasonable operator actions to correct minor problems may be considered." The fact that the Haddam Neck operator could have taken reasonable action to restore the non-vital power supply in a matter of minutes, but instead chose an even more reasonable approach of involving management, since he had so much time available, is a direct application of the NRC guidance. Wayne agreed with this point.

Wayne also agreed that the classic definition of "operability" does not apply to the safety function provided by the spent fuel pool cooling system since the "operation" of it is not required by a Technical Specification. Thus, engineering judgement is reasonable in concluding the operator actions mitigated the significance of this event.

Reporting this as a voluntary LER would result in all of the same information submitted to the NRC with the only difference being the designation of the LER category. A voluntary LER is credited by the NRC in SALP performance according to Wayne Lanning. He felt strongly that a voluntary LER is needed versus an "information letter", and in conclusion, that a voluntary LER is appropriate versus a "required LER".

cc: P. M. Kacich
P. A. Blasioli
G. P. van Noordennen
J. M. Powers
T. J. Dente
J. G. Resetar
B. M. Pinkowitz
M. Bigiarelli
J. L. Majewski
GFL Memo File

Threshold = 2

CONNECTICUT YANKEE ATOMIC POWER COMPANY

HADDAM NECK PLANT

RR#1 • BOX 127E • EAST HAMPTON, CONN. 06424

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D. C. 20555

September 11, 1987
Re: 10CFR50.73

Reference: Facility Operating License No. DPR-61
Docket No. 50-213
Reportable Occurrence LER 50-213/87-015-00

Gentlemen:

This letter forwards Licensee Event Report 87-015-00, required to be submitted pursuant to the requirements of Connecticut Yankee Technical Specifications.

Due to problems experienced on the initial classification of this event, no prompt report was made pursuant to the requirements of 10CFR50.72. Subsequent discussions with NRC staff have resulted in classification of this event as reportable per 10CFR50.73(a)(2)(v)(B), even though the licensee considers the premise that this event alone could have resulted in the complete loss of decay heat removal capability to be inconsistent with the design basis of the system. Technical details are included in the report. The licensee feels that this event warrants reporting even though the exact classification is unclear and will continue to report similar events in the future.

Very truly yours,

Donald B. Miller, Jr.
Station Superintendent

Total No. of Commitments = 1
#01 2-C ELC 460-600 V AC Electrical System
SFC Spent Fuel Pool Cooling System

DBM:JJL/dfv

Attachment: LER 87-015-00

cc: W. T. Russell, Regional Administrator, Region I
J. T. Shedlosky, Senior Resident Inspector, Haddam Neck

NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Haddam Neck	0 5 0 0 0 2 1 3	1 OF 0 8

TITLE (4)
Personnel Error Causes Temporary Loss of Spent Fuel Pit Cooling Pumps Power

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
0 8	1 4	8 7	8 7	— 0 1 5	— 0 0	0 9	1 1	8 7		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9)	6
POWER LEVEL (10)	0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
20.405(a)(1)(i)	50.36(c)(1)	X 50.73(a)(2)(v)	73.71(c)
20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	OTHER (Specify in Abstract below and in Text, NRC Form 366A)
20.405(a)(1)(iii)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
20.405(a)(1)(iv)	50.73(a)(2)(ii)	50.73(a)(2)(viii)(B)	
20.405(a)(1)(v)	50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (AREA CODE)
D. J. Ray, Engineering Supervisor	2 0 3 2 6 7 - 2 5 5 6

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
X	E C	2 7	G 0 8 0	Y					

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
X YES (If yes, complete EXPECTED SUBMISSION DATE)	NO		0 1	2 0	8 8

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

ABSTRACT

[redacted] at 1415 hours with the plant shutdown in mode 6, [redacted]

Operators recognized the loss of spent fuel pit cooling, and the fact that adequate time was available to restore power to the system using a properly reviewed and approved jumper control sheet. The jumper control sheet was approved by the Plant Operations Review Committee and spent fuel pit cooling was restored at 1545 hours. The loss of cooling existed for 80 minutes and resulted in an increase in spent fuel pit temperature of 6 degrees fahrenheit. No temperature limits were exceeded.

The cause of the event was personnel error. Corrective measures include better training, and increased supervisory involvement. Problems experienced with the General Electric HEA relays involved in this event are currently under evaluation and the results will be forwarded in a supplemental report.

This event has been classified as reportable under 10CFR50.73(a)(2)(v)(B) with the assistance of NRC staff.

NRC Form 366 (9-83)

BACKGROUND

I) Spent Fuel Pit Cooling Design

The original design basis for the spent fuel pit cooling loop was to provide the ability to remove the residual heat produced by 40 percent of a 1473 MWth reactor core (1/3 core offload), 150 hours after reactor shutdown, while maintaining the pit water temperature at or below 116 degrees fahrenheit with 80 degree fahrenheit river water. The design Westinghouse used to meet this basis included one 40 horsepower 610 gpm single stage centrifugal pump (P-21-1A) supplying flow to a shell and tube type heat exchanger (E-10-1A) with a 1,620 square foot surface area sized to transfer 5.43 E6 BTU per hour under design conditions. The design addressed loss of spent fuel pit cooling pump by stating that a portable sump pump could be used temporarily if necessary.

In 1976 the second pump (P-21-1B) and heat exchanger (E-10-1B) were added to increase the heat removal capacity of the system under Plant Design Change Request 212 "Connecticut Yankee Spent Fuel Pool Cooling System Modification". The second pump and heat exchanger were sized to provide cooling (2.0 E7 BTU per hour) adequate to maintain spent fuel pit temperature at or below 140 degrees fahrenheit with 85 degree fahrenheit river water after a full core offload at 150 hours after shutdown. The design addressed a loss of power to both spent fuel pit cooling pumps by stating that either pump could manually be powered from either emergency diesel generator (EG2A or EG2B).

II) Haddam Neck Electrical Distribution

The Haddam Neck electrical distribution is illustrated on Figure 1. This figure shows that the spent fuel pit cooling pumps, P-21-1A and P-21-1B, are powered through Motor Control Center 2 (MCC-2) from 480 volt buses 4 and 5 respectively. It is important to note that both P-21-1A and P-21-1B normally receive emergency power from the same source, EG2A.

The undervoltage protection scheme for emergency 4160 volt bus 8 is as follows. An undervoltage condition on bus 8 causes breakers 8T2, 2T8, and 4841 to trip and lockout, thus isolating the normal 4160 volt power source (bus 1-2) from the emergency source (bus 8). Breaker 4T5 closes to connect bus 4 to the emergency power supply. EG2A then starts and when it is ready, its output breaker closes, supplying power to buses 4, 5, and 8.

Prior to this event, EG2A and bus 8 had been removed from service for outage related work. Bus 4 was receiving 480 volt power from its normal source, bus 1-2, and it was supplying power to bus 5 through the 4T5 breaker.

At the start of this event, ultrasonic inspection of spent fuel assemblies was in progress. The full core had been offloaded to the spent fuel pit in support of this inspection. For a fuel assembly to undergo this inspection, it must be suspended from the spent fuel building crane. This crane is powered from MCC-2. At the time of initiation of this event, 1415 hrs on August 14, 1987, an assembly was suspended from the crane. In this position, the activated region of the suspended fuel assembly was below the lowest penetration (spent fuel pit pump suction) in the spent fuel pit, and covered by approximately 8 feet of water.

EVENT DESCRIPTION

On August 14, 1987, the plant was shutdown in mode 6. Problems had been experienced with mechanical binding of the bus 8 undervoltage relay 27Y1-8 (EIIS System Code: EC; Component Code: 27). A work order was generated to trouble shoot and replace the 27Y1-8 relay. This work order was reviewed and approved in accordance with station procedures.

In conjunction with removing bus 8 from service, its undervoltage protection scheme (discussed in BACKGROUND) had been disabled by opening the 125 volt direct current (DC) control power breaker in the EG2A auxiliary board. It was, however, still possible to initiate the undervoltage scheme by manually tripping the 27Y1-8 relay. The maintenance electrician working on the 27Y1-8 relay was not aware of this fact, and at 1425 hours pushed the manual trip latch mechanism. This action initiated the bus 8 undervoltage scheme and resulted in complete loss of buses 4 and 5 since bus 8 was out of service.

The electrician immediately notified the Supervising Control Operator (SCO) that he had manually tripped the 27Y1-8 relay. The SCO, in turn, recognized that a partial loss of 480 volt power had occurred. He attempted to restore power by: resetting the 27Y1-8 relay, resetting the bus 4 and 5 main undervoltage relays (27Y-4 and 27Y-5), and attempting to reset the bus 4 and 5 auxiliary undervoltage relays (27Y-4B and 27Y-5B). The 27Y-4B and 27Y-5B relays would not reset. Resetting 27Y-4 and 27Y-5 with buses 4 and 5 still de-energized would not have been possible if the trip coils were still good. It appears that the coils partially shorted when the relays were reset with a trip signal still present. Although the SCO should not have reset the undervoltage relays prior to re-energizing bus 4 and 5, this improper operation action did not adversely contribute to this event. The SCO, upon observing failure of the bus 4 and 5 auxiliary relays to reset, had the 4841 and 4T5 breakers closed to restore power to these buses.

At about the same time that the 4841 and 4T5 breakers were shut by the control operator, the maintenance electrician noticed that smoke was issuing from the auxiliary panel that houses the bus 4 and 5 undervoltage relays. This smoke was coming from the energized coils of the 27Y-4 and 27Y-5 relays. A production test electrician, who was present in the control room recognized what was happening. He entered the panel and manually tripped the 27Y-4 and 27Y-5 relays. This action stopped the smoldering of the relay coils that was causing the smoke. The time was 1417, approximately 2 minutes had elapsed since initiation of the event.

The operators recognized immediately that spent-fuel pit cooling had been lost. They also learned that an assembly was suspended from the south spent fuel building crane for inspection at the time MCC-2 was lost. Since the bus 4 and 5 undervoltage relays were all tripped, and the 27Y-4 and 27Y-5 relays were damaged, it was not possible to simply restore MCC-2 by shutting the supply breakers from buses 4 and 5. The operators recognized that the undervoltage relays would have to be defeated in order to close the bus 4 and 5 feed breakers to MCC-2. The operators also realized that adequate time was available to perform this action using the appropriate review and approval process, before the loss of spent fuel pit cooling became a concern.

Plant management was informed of the situation and they immediately responded to the control room. After being briefed on the situation, management concurred on the use of an approved jumper control sheet to remove the fuses for DC control power to the bus 4 undervoltage relays. Removal of the DC control power would deenergize the trip coils and allow the undervoltage relays to be reset. This, in turn, would allow the bus 4 feed breaker to MCC-2 to be shut, restoring power to the spent fuel pit cooling pumps and the south spent fuel building crane.

The jumper control sheet was prepared and reviewed by the Plant Operations Review Committee. The committee approved the jumper control sheet for use. The DC control power fuses were removed and at 1545 hours MCC-2 was reenergized and spent fuel pit cooling was restored.

Spent fuel pit cooling was lost for approximately 80 minutes. During this time spent fuel pit temperature increased 6 degrees fahrenheit, resulting in a heatup rate of 4.5 degrees fahrenheit per hour. At no time were any temperature limits for the pit water exceeded.

CAUSE OF THIS EVENT

There were two independent problems involved in this event. The first problem was the fact that the maintenance electrician who initiated this event did not adequately understand the operation of the relay he was working on. In addition, the work order did not mitigate the consequences of this knowledge deficiency by providing adequate cautions on the potential consequences of the work. Finally, supervisory guidance on this job was not sufficient to compensate for the electrician's inexperience.

The second cause of this event involved problems with the General Electric HEA relays (27Y1-8, 27Y-4 and 27Y-5) that were involved. The root cause of the relay failure has not yet been determined. Mechanical binding of the 27Y1-8 relay was the reason for performing the maintenance that initiated this event.

SAFETY ASSESSMENT

This event caused a partial loss of non-vital 480 volt power which resulted in a total loss of normal power to the spent fuel pit cooling system. An additional complication was introduced by the fact that a spent fuel assembly was left suspended above the spent fuel racks when power to the south spent fuel building crane was lost (powered from MCC-2).

The complete loss of spent fuel pit cooling is an analyzed event. In the original (pre-1976) system configuration loss of spent fuel pit cooling pump P-21-1A constituted a complete loss of cooling since it was the only pump. In the design basis documentation, it was stated that a portable sump pump could be used for pumping water through the system in the event of a complete loss of P-21-1A. The Facility Description and Safety Analysis (FDSA) amplified this guidance to indicate that blind flanges are provided on the pump suction and discharge to facilitate connection of a temporary pump.

Loss of both spent fuel pit cooling pumps in the present (post 1976) configuration due to a loss of power was specifically analyzed as part of Plant Design Change Request No. 212. This analysis concluded that loss of non-vital 480 volt power was acceptable because either pump could be manually connected to either emergency diesel generator within the 8 hours available before boiling occurred in the pit due to decay heat of the spent fuel assemblies under worst case conditions.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Haddam Neck	0 5 0 0 0 2 1 3	8 7	— 0 1 5	— 0 0	0 6 OF 0 8

TEXT (If more space is required, use additional NRC Form 366A's) (17)

The fact that a spent fuel assembly was left hanging from the south spent fuel building crane during this event does not change the results of the decay heat removal analyses discussed above. It did, however, result in a condition where a fuel assembly was suspended above the spent fuel pit racks, and there was no immediate means available to lower it back into the rack. Should a seismic event have occurred while the spent fuel assembly was suspended, the assembly could have dropped and/or a portion of the spent fuel pit cooling system piping could have been damaged, causing a partial draindown of the spent fuel pit.

Dropping of the spent fuel assembly is an analyzed event, described in the Updated Final Safety Analysis Report. This accident is demonstrated to have acceptable consequences and bounds the current postulated dropped spent fuel assembly accident.

A partial draindown of the spent fuel pit with the spent fuel assembly suspended from the crane would, in the worst case, result in a final water level approximately six inches above the activated region of the fuel assembly. Keeping the assembly covered with water would ensure adequate cooling of the assembly and thus this postulated condition would not increase the safety consequences of an earthquake damaging the spent fuel pit cooling system.

Based on the discussion above of the worst case accident scenarios that could have resulted from this event, the event is judged to have limited safety significance. There was no threat to the public health and safety posed by either the actual or postulated worst case events.

This event has been classified as reportable under 10CFR50.73(a)(2)(v)(B) with the assistance of the NRC staff.

CORRECTIVE ACTION

In order to address the personnel and related problems associated with this event, the following corrective actions are planned or in progress. Supervisory involvement in critical activities will be increased. This increased involvement will include increased supervisor presence at the work site as well as improved written guidance on work orders. All electricians have been verbally reminded that they must understand the operation of a circuit prior to starting work on it.

#01

Type 2C

The problems identified with the General Electric HEA relays are currently under evaluation. The results of this evaluation will be presented in a supplemental report.

See B.S. memo of 7-1-96

NRC Form 366A
(9-83)

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150–0104
EXPIRES: 8/31/88

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Haddam Neck	0 5 0 0 0 2 1 3	8 7	— 0 1 5	— 0 0	0 7 OF 0 8

TEXT (If more space is required, use additional NRC Form 366A's) (17)

ADDITIONAL INFORMATION

The relays involved in this event are type HEA 61 relays with 24 volt DC coils and 20 ampere continuously rated contacts manufactured by General Electric.

PREVIOUS SIMILAR EVENTS

None

*U.S.GPO:1986-0-624-538/455

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
...ddam Neck	0 5 0 0 0 2 1 3	8 7	— 0 1 5 —	0 0	0 8 OF 0

...XT (If more space is required, use additional NRC Form 366A's) (17)

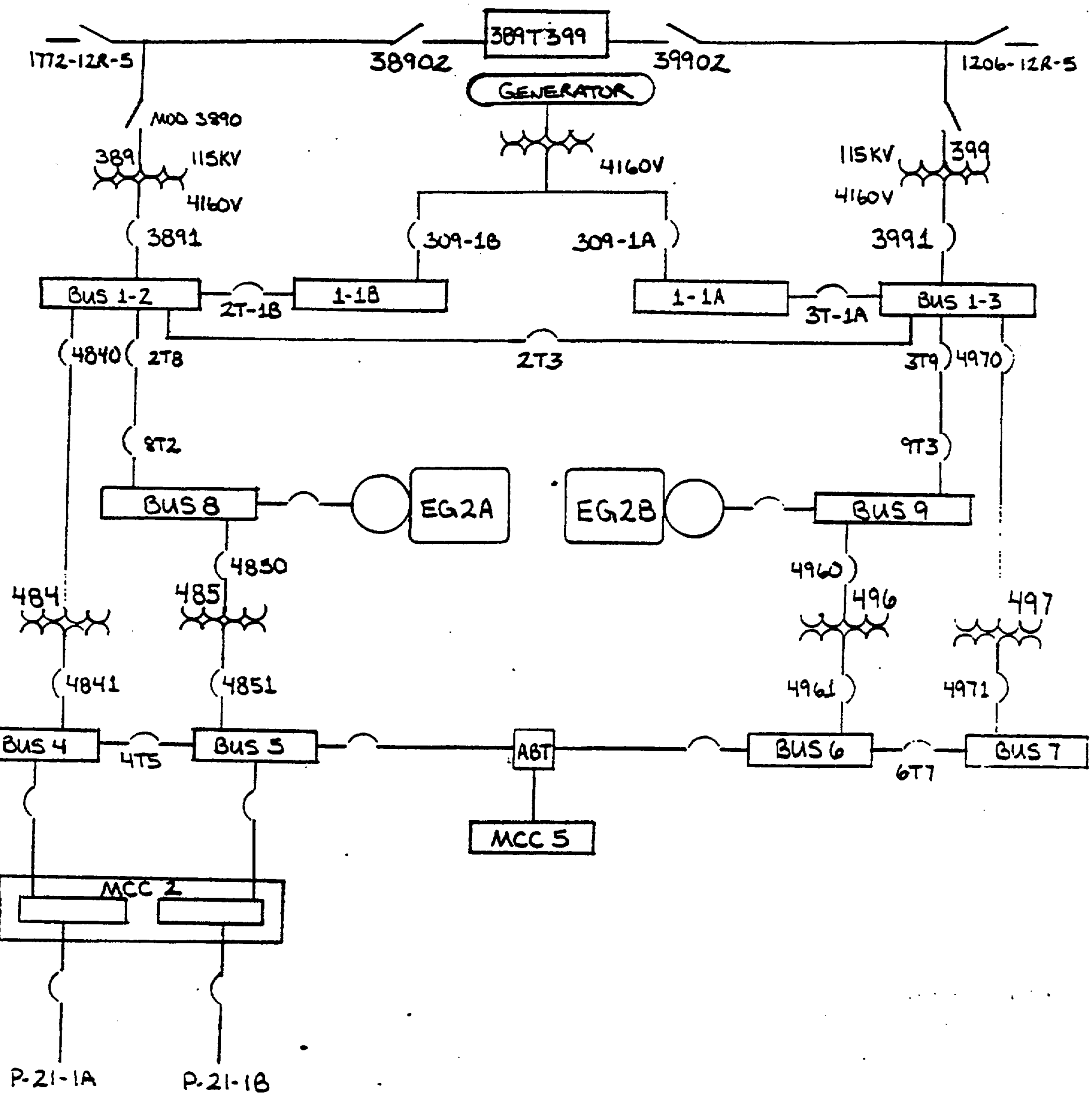

FIGURE 1
HADDAM NECK
ELECTRICAL DISTRIBUTION

OE 7218 I HAMILTON 11-APR-95 08:23 EST
INDIANA MICHIGAN POWER (IME)
Subject: Incorrect Fuel Assembly Moved During Examination

Cook Nuclear Plant offers the following information:

UNIT:...................	Cook Nuclear Plant, Units 1 & 2
DOC NO................	50-315, 50-316
EVENT DATE............	2/07/95
NSSS/AE...............	Westinghouse/AEPSC
RATING................	3250 Mwt/3411 Mwt

Siemens fuel rod hi-mag/eddy current examinations were in progress at Cook Nuclear Plant. Fuel Assembly AM43 was the next assembly to be moved and examined in the spent fuel pool. The Reactor Engineer (RE) wrote AM43 on the white board used to communicate cell location to the fuel handler. The fuel handler positioned the crane on row 43 and the RE verified the location. The fuel handler positioned the crane to alpha location BM and, without any position verification being performed, latched onto and moved assembly BM43 to the test rig. Testing of the assembly was completed. While returning the assembly to the AM43 location the fuel handler found an assembly in AM43. Using a camera, the crew determined the assembly moved was from location BM43. The assembly was returned to location BM43. The individuals involved did not immediately report the event to supervision. Fuel movement was stopped on the following shift. An investigation of the event was performed. Corrective actions were developed and provided to plant management before work resumed.

Causes of this event are attributed to the fuel handler not receiving position verification before latching and moving the assembly and the failure to establish direct communications between the RE and fuel handler.

Planned Preventive Actions:

1. Plant Engineering (PLE) and Operations (OPS) will devise additional Spent Fuel Pool Indexing for use in dual concurrent verification of fuel handling equipment positions.

2. PLE and OPS will revise procedures for spent fuel pool evolutions to require concurrent dual verification of:

   spent fuel pool location written on the white board.

   crane X-Y index (north/south - east/west) location prior to lowering the refueling tool.

   fuel handling tool location prior to latching or lowering a fuel assembly.

   The use of audio communications equipment and repeat backs will be required to confirm instructions are understood. A positive response from the verifier, for position confirmation, and permission will be required prior to performing the next step.

3. PLE and OPS will develop a method to ensure fuel movement precursor events occurring during all fuel movement activities, including refueling, are communicated to personnel associated with movement activities.

4. PLE will revise their job briefing form to:

Ensure applicable industry and plant operating experience events are addressed during job briefings.

Discuss managements expectations for conservative decision making when unanticipated events occur. Emphasize on placing equipment in a safe condition and immediately notifying management/supervision of the problem.

5. OPS, PLE, and Quality Control will determine minimum staffing requirements and define the expectations, responsibilities, and requirements for each role regarding dual concurrent verification during all fuel movement activities.

6. OPS will revise the Error Free Refueling Plan to include management expectations for conservative decision making when unanticipated events occur.

7. OPS will revise the Refueling Brief Guidelines to include management expectations for conservative decision making when unanticipated events occur.

Information Contact: Randy Keppeler (616) 465-5901 ext. 1339

Information Contact: Randy Keppeler (616) 465-5901 ext. 1339

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INDIANA MICHIGAN POWER (IME)
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Events
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Subject: Significant Event Notification (SEN) 127, Recurring Events April June 1995

The following events have been screened as significant during the second quarter of 1995 and are recurrent; i.e., similar in many aspects to events for which the "lessons learned" are adequately described in a previous SEE-IN product or in a Significant By Others Report reference document. This list is published to help maintain awareness that similar events continue to occur.

1. Recurring Event: Reactor Recirculation Pump Restarted Without Meeting Differential Temperature Requirements

Description:

On January 10, 1995, with Dresden Unit 2 operating at 31 percent power, operators restarted an idle reactor recirculation pump without meeting the required temperature limits. Reactor recirculation pump 2B had inadvertently tripped approximately two hours earlier when instrument technicians began work on the incorrect reactor recirculation motor-generator set. When plant operators prepared to start the idle pump, they were unable to comply with the requirement that the differential temperature between the bottom head drain line and reactor steam space be less than 145 degrees F before starting an idle reactor recirculation pump. This differential temperature is measured using the bottom head drain line temperature; however, the bottom head drain line has been blocked for several years, and the bottom head drain line temperature indication was unreliable. Additionally, the temperature difference between the reactor steam space and the reactor bottom head metal temperature (an alternate temperature measurement) exceeded the station procedural limit of 145 degrees F for starting an idle recirculation pump.

When the procedural limit could not be met without a plant shutdown to reduce the temperature differential and allow restarting the idle pump, control room supervisors decided that the intent of the temperature limit requirement could be met by using the active recirculation loop discharge temperature to indicate temperature in the bottom head region. This method is not consistent with vendor guidance for appropriate differential temperature measurement locations, but the vendor information was not included in the procedure. Control room supervision made this decision without involving station or operations management and without benefit of an engineering analysis. Prior to starting the idle recirculation pump, an Independent Safety Engineering Group engineer questioned their decision, but the crew continued. Subsequent analysis by the vendor indicated that the conditions necessary for thermal stratification were not present; therefore, the pump start had minimal adverse thermal impact on reactor vessel bottom head penetrations.

Consequences/Comments:

Significant aspects of this event include a nonconservative decision made by control room supervisors that had the potential to subject reactor pressure vessel bottom head penetrations to unnecessary thermal stresses. Several of the lessons learned in INPO SOER 94-01, Revision 1, "Nonconservative Decisions and Equipment Performance Problems Result in a Reactor Scram, Two Safety Injections, and Water-Solid Conditions," are similar to some of the causal factors in this event. Control room supervisors made a decision to operate outside of procedural guidance and in a manner that did not comply with station technical specifications in order to restart the reactor recirculation pump. This decision was made without consulting station management and without benefit of an engineering evaluation. The clogged bottom head drain is a plant condition that has existed for several years. Plans are in place to correct this long-standing condition. Additionally, the procedure written to restart an idle reactor recirculation pump without valid indication of bottom head drain temperature did not provide the operators with the information necessary to obtain accurate differential temperatures.

Plant/NSSS:	Dresden Nuclear Power Station Unit 2/General Electric (BWR 2)
Event Date:	January 10, 1995

2. Recurring Event: Automatic Actuation of Low Temperature Overpressure Protection System

Description:

On January 4, 1995, McGuire Unit 2 experienced a reactor coolant system (RCS) pressure transient while in a water-solid condition. RCS pressure increased from approximately 250 psig to 371 psig over a nine minute period. The transient was terminated when a pressurizer power-operated relief valve (PORV) automatically opened in response to a signal from the low temperature overpressure protection system.

Background - McGuire Unit 2 was starting up after a refueling outage. The reactor was in cold shutdown, and operations personnel had just completed venting the RCS. Because of the increased work load associated with unit startup, the control room crew had been augmented with an additional reactor operator. However, control room activities eventually reached the point where all three control room operators were involved in multiple startup tasks that diverted their attention from monitoring the RCS.

RCS Pressure Transient - The reactor operator assigned to monitor RCS conditions noticed that RCS pressure had decreased to 250 psig and adjusted letdown flow to raise system pressure to approximately 275 psig. (During water-solid operation, RCS pressure is raised and lowered by adjusting the mass flow out of the system while charging flow is held constant.) He also adjusted component cooling water (CCW) flow to the residual heat removal system heat exchangers to raise

RCS temperature. Immediately after adjusting letdown and CCW flow, the operator was called away to assist maintenance personnel who were testing a feedwater system pump. The other reactor operators were also busy with other startup activities and were not monitoring RCS parameters. Nine minutes later, a control room annunciator actuated indicating that a pressurizer PORV was open. Noting that RCS pressure had risen to 371 psig, the operator readjusted letdown flow to lower RCS pressure, reseat the PORV, and stabilize RCS pressure.

Consequences/Comments:

Continuous monitoring of critical plant parameters could have prevented actuation of the LTOP system. Operators assigned to monitor the RCS during water-solid operation were distracted from their primary function of maintaining and controlling RCS pressure and temperature. Control room supervisory personnel need to maintain an awareness of the level of activity in the control room and curtail nonessential tasks when operators are no longer able to devote complete attention to the reactor and its support systems.

Plant/NSSS: McGuire Unit 2/Westinghouse
Event Date: January 4, 1995

3. Recurring Event: Undetected Loss of Spent Fuel Cooling Due to Improper Maintenance Activities

Description:

On August 10, 1994, a Seabrook station mechanic inadvertently closed the outlet throttle valve on the operating train A spent fuel pool cooling pump, while adjusting the valve packing. Spent fuel pool temperature increased from 95 degrees F to 120 degrees F over a 24-hour period before operators discovered the condition. The mechanic was not aware that the valve was a throttle valve, and for unknown reasons, thought the valve was initially closed (the valve was actually 25 to 40 percent open). Procedure guidance allows a valve manipulation of 1/2 turn during packing adjustments. Immediately after closing the valve, a spent fuel pool cooling pump low discharge flow alarm annunciated in the control room. However, due to other unrelated higher priority alarms, the senior reactor operator (SRO) did not pursue the low discharge flow alarm. This condition went unnoticed for the next 24 hours, including two shift changes, until the original SRO assumed the watch the following day and noticed the same low flow alarms were illuminated. No damage was done to the pump, and no system operational setpoints or limitations were exceeded.

Significant Event Report, SER 18-93, "Undetected Loss of Spent Fuel Cooling," and Significant Event Notification (SEN) 119, "Undetected Loss of Spent Fuel Cooling," address similar industry events. Both of these industry events emphasize the importance of operator control board awareness, monitoring spent fuel pool conditions, and thorough shift turnovers.

Consequences/Comments:

The undetected loss of spent fuel pool cooling event resulted from several human performance problems. Those human performance problems are summarized as follows:

- An incomplete pre-job briefing between the work control coordinator and the station mechanic did not establish the as-found and as-left valve position.

- The pre-job briefing failed to recognize that the valve to be worked was the in-service throttle valve for the operating spent fuel pool cooling loop. Covering the status of the valve in a pre-job briefing would have alerted the mechanic to the consequences of reconfiguring the valve.

- The work control supervisor did not communicate to control room personnel that work was about to begin on the in-service spent fuel pool cooling throttle valve. Had this communication occurred, it could have alerted control room personnel to subsequent spent fuel pool cooling system alarms.

- Control room personnel did not respond aggressively to the spent fuel pool cooling pump low discharge flow alarm. The senior control room operator (SCRO) did not inform other control room team members of the existence of the alarm. In addition, the SCRO did not contact the primary nonlicensed operator to check system conditions upon receipt of the alarm.

- Insufficiently effective work practices permitted a station mechanic to reconfigure the throttle valve closed when the valve was actually 25-40 percent open. As a result, upon work completion, the mechanic returned the valve to the incorrectly recorded as-found position (closed).

- Incomplete shift turnovers also contributed to this event. Two shift turnovers were performed with the spent fuel pump low discharge flow alarm annunciated, and neither document reviews nor control board walkdowns prompted questions concerning the alarm.

- The maintenance procedure data sheet lacked sufficient detail to ensure the valve was left in the proper position. The procedure only required documentation of the as-left position and did not specifically address throttle valves.

Plant/NSSS: Seabrook/Westinghouse
Event Date: August 10, 1994

4. Recurring Event: Reactivity Excursion While Placing An Unsaturated Mixed-Bed Demineralizer In Service

Description:

On February 5, 1995, McGuire Unit 2 experienced an inadvertent dilution of reactor coolant system (RCS) boron concentration and a subsequent reactivity excursion when an unsaturated mixed-bed demineralizer was placed in service. Prompt response by the operating crew limited reactor power to 100.5 percent and average RCS temperature within one degree F of its programmed value. Inappropriate actions by chemistry department personnel because of inadequate written and verbal communication and failure to follow approved plant procedures were determined to be the root causes of this event.

Background - On January 18, 1995, chemistry personnel loaded a standby mixed-bed demineralizer (A) with fresh resin and coordinated with operations personnel to flush it for 30 minutes. (Flushing a freshly loaded demineralizer ensures the resin is saturated with boron and that it will not remove boron from the RCS when placed in service.) After flushing for 30 minutes, the chemical and volume control system (CVCS) was realigned to its normal letdown lineup, and the in-service demineralizer (B) was returned to service. When the sample results for demineralizer A were reviewed, the values for lithium and boron were not as expected for saturated resin. The chemistry technician determined the sample was invalid and the demineralizer needed to be flushed and sampled again. The technician then recorded the analysis results in the CVCS demineralizer status sheets but did not annotate the sheets to indicate the sample results were questionable. The technician then left the site for the day. Based on previous conversations with other chemistry department personnel regarding the need to reflush demineralizer A, the technician assumed the on-coming shift would rerun the flushing procedure. The off-going technician did not discuss the evolution with his relief, ensure pertinent information was recorded in the department turnover sheets, or verify the status of demineralizer A when he returned to work the following day.

RCS Dilution - Eighteen days later, another chemistry technician identified a need to place demineralizer A in service to remove lithium from the RCS. Seeing no indications of a problem with demineralizer A in the logbook or demineralizer status sheets, the technician contacted operations and requested demineralizer A be placed in service for 45 minutes. Operations personnel realigned the CVCS as requested. Shortly after aligning demineralizer A for service, the reactor operator added water to the volume control tank through the boric acid blender. Because of the water addition, the operator was closely monitoring RCS temperature and noticed RCS temperature increasing. The operator referred to the abnormal operating procedure for boron dilution events. The boron dilution procedure directed stopping the primary water makeup pumps, initiating emergency boration, and implementing the abnormal operating procedure for emergency boration. When directed by the emergency boration procedure, the operator bypassed demineralizer A. The control room operator then requested a sample of the demineralizer. The sample results indicated demineralizer A had not been fully boron saturated when placed in service.

Consequences/Comments:

The reactivity excursion that resulted from this event required the reactor operator to take compensatory actions to mitigate the excursion by adding 45 gallons of boric acid, reducing main turbine load, and inserting control rods. The prompt response by the operator prevented a main turbine runback and possible reactor scram.

Similar RCS dilution events were addressed in INPO SEN-106, "Recurring Event: Unexpected Reactivity Additions While Placing Primary Demineralizers In Service," and SOER 94-2, "Boron Dilution Events In Pressurized Water Reactors." Unexpected RCS dilutions occurred at McGuire in December 1993 and August 1994. Internalization of both industry and in-house events has been ineffective for several reasons:

- McGuire problem reports involving primary demineralizers were sometimes not coded as "reactivity management" events.
- The station analyses of individual events sometimes did not identify all the contributing causes for boron dilution.
- Communication of reactivity management problems was not fully effective in developing an awareness of, appreciation for, and sensitivity to plant evolutions that can result in a dilution event.

Incomplete Flushing and Nonrepresentative Sampling - The CVCS purification demineralizers are maintained by the Chemistry Department personnel at McGuire. As such, chemistry personnel are responsible for ensuring freshly loaded resin is flushed and sampled. However, the flushing process requires coordination with control room personnel to ensure valves are properly aligned. For this event, flushing and sampling were conducted as independent tasks. There was no pre-job briefing, and communication between operations and chemistry personnel was ineffective.

Improper Documentation and Follow Up - When demineralizer A sample results were recorded in the CVCS demineralizer status sheets, the responsible technician did not include any notations or other indications that the entries were abnormal. When the sample results were checked prior to placing the standby demineralizer in service, there was nothing in the records to indicate the resin was not completely saturated.

After determining the boron and lithium concentrations at the influent and effluent sample points for demineralizer A, the chemistry technician concluded the resin would have to be reflushed and resampled. However, he did not inform control room personnel or conduct a formal turnover on the status of the demineralizer with his relief. The chemistry technician assumed the task would be completed during the evening, but he did not verify completion when he returned to work the following day. If the chemistry technician had informed control room personnel demineralizer A was not flushed, the reactivity excursion may have been prevented.

Plant/NSSS: McGuire Unit 2/Westinghouse

Event Date: February 5, 1995

5. Recurring Event: Control Switches For Both Unit 3 High Head Safety Injection Pumps Placed In Pull-to-Lock

Description:

On November 3, 1994, Turkey Point Unit 3 was operating at 100 percent power, and Unit 4 was in mode 5 conducting engineered safety features (ESF) testing when control switches for both Unit 3 high head safety injection pumps were incorrectly placed in the pull-to-lock position. Placing control switches for both pumps in pull-to-lock prevented the pumps from automatically starting on a safety injection signal. With one unit at power, station technical specifications require that three of the four shared high head safety injection pumps be operable. (Unit 3 and Unit 4 both have two high head safety injection pumps, and all four pumps receive a start signal in response to a safety injection actuation signal in either unit.) A licensed operator, coordinating the test, improperly directed that control switches for both pumps be placed in pull-to-lock. Operations management identified the improper switch position for the pumps approximately 2 3/4 hours later during a control board walkdown after testing was completed. Both Unit 4 pumps were available and would have started on demand.

Consequences/Comments:

INPO SER 22-93, "Inappropriate Disabling of Automatic Actuation Functions for Safety-related Components," describes several similar events. In this event, the licensed test director inappropriately directed placing the control switches for the Unit 3 pumps in the pull-to-lock position even though the procedure for ESF testing directed they be placed in the stop position. The operator who placed the pumps' switches in the pull-to-lock position did not question the appropriateness of that action. Additionally, operators on Unit 3 insufficiently responded when alarms were received indicating the automatic start capability of both Unit 3 high head safety injection pumps had been disabled. One of the lessons learned in SER 22-93 discusses the need for operators to instinctively question the appropriateness of disabling safety systems before performing the activity.

Plant/NSSS: Turkey Point Unit 3/Westinghouse
Event Date: November 3, 1994

6. Recurring Event: Potential Pressure-Locking of Containment Sump Isolation Valves

Description:

On January 25, 1995, Millstone Unit 2 reported that the redundant motor-operated isolation valves (MOVs) between the containment sump and the containment spray/emergency core cooling pump suction piping could be susceptible to thermally induced bonnet pressurization under postulated design-basis accident conditions. The containment sump isolation valves are

24-inch, double-disc gate valves that are required to open during the recirculation phase of a loss of coolant accident (LOCA). This flow path alignment ensures a source of water to the containment spray pumps and high pressure safety injection pumps after the refueling water storage tank (RWST) inventory has been depleted. If the containment sump isolation valves fail to open, long-term core cooling and reduction of containment temperature and pressure could be jeopardized.

Consequences/Comments:

The containment sump isolation valves were evaluated in 1990 in response to INPO SOER 84-7, "Pressure Locking and Thermal Binding of Gate Valves." The evaluation concluded that the valves were not susceptible to pressure-locking or thermal-binding (PL/TB). In 1994, Millstone personnel identified some inconsistencies in the 1990 PL/TB evaluation of Millstone Unit 1 and initiated a reanalysis of the Unit 2 MOVs. The new analysis identified a condition, overlooked during the 1990 evaluation, where the sump isolation MOVs could be rendered inoperable by pressure-locking in the later stages of a LOCA.

Basis for Concern - During normal operation, the sump MOVs are maintained closed and function as non-Appendix J containment isolation valves. The upstream (containment) piping and valve disks should be dry because they connect to the recirculation sump, which does not normally contain water. The downstream valve disks are exposed to the static head (approximately 35 psig) of the RWST due to backleakage through the check valves located in the downstream piping. The plant safety analysis assumes that the upstream piping is dry and the downstream piping is full of water. In reality, the upstream piping contains water. Because the valve design does not preclude water from entering the valve bonnet, it is assumed that periodic cycling during surveillance testing could cause water from the RWST to become trapped in the valve bonnet. Under postulated design-basis accident conditions, the valves would remain closed for approximately 45 minutes while a combination of reactor coolant, safety injection, and containment spray water collected in the recirculation sump. The temperature of the liquid in the containment sump would be approximately 200 degrees F.

Identification of Potential Inoperability - The 1994 analysis hypothesized that the high temperature fluid in the containment sump could cause heating of trapped fluid in the valve bonnets and lead to pressurization of the area between the valve disks while the valves are closed. When Millstone personnel learned of the potential for pressure-locking, the valve manufacturer was contacted in order to determine the maximum bonnet pressure that could effect valve operability. (All pressure locking evaluations were conducted under the assumption that there is water in the valve bonnet because there is no way of ensuring that water is excluded from the bonnet area.) The valve manufacturer determined that an internal pressure of 150 psig in the bonnet could impose forces against the valve disks that could not be overcome by the motor operator. This pressure equated to a temperature rise of only 5 degrees F, assuming the bonnet area is water-solid. Plant management declared both containment sump isolation valves inoperable and developed an action plan to correct the problem.

The containment sump isolation valves were not identified as being susceptible to pressure-locking during the 1990 evaluation because the analysis assumed that valves with operating temperatures less than or equal to 200 degrees F and operating pressures less than or equal to 150 psig were not susceptible to PL/TB. The original evaluators did not realize that the containment environment could communicate with the valves through the recirculation piping. They believed the upstream piping was filled with water.

Utility Actions - In April 1995, Millstone completed a detailed analysis of the containment sump isolation valves, including in-plant radiographic examination of the isolation valves to determine if water was present in the valve bonnets. Plant personnel also contracted an independent laboratory to conduct bonnet pressurization experiments using a mockup of the sump piping and valve. A 6-inch double-disc gate valve was exposed to conditions similar to the post-LOCA containment environment. The results of the inspection, analysis, and experiments indicated that the containment sump isolation valves were operable and were not susceptible to thermally induced bonnet pressurization. A key factor was the presence of a small amount of air trapped in the bonnet. (The Millstone engineering staff cautions that these results are for valves exposed to low pressure, low temperature, and short duration heating.) In addition to analyzing and examining the valves, plant personnel also drilled a small hole in each upstream valve disc to ensure that pressure-locking will not occur.

Plant/NSSS: Millstone Unit 2/Combustion Engineering
Event Date: January 25, 1995

7. Recurring Event: Potential Pressure-Locking of Safety-Related Gate Valves

Description:

On March 9, 1995, Connecticut Yankee reported that a new engineering analysis of safety-related, motor-operated valves (MOVs) indicated that multiple valves in the emergency core cooling system (ECCS) may have been technically inoperable because of the potential for pressure-locking. A follow-up report issued on March 15, 1995, indicated that the on-going MOV review had identified additional ECCS and shutdown cooling valves that might also be susceptible to pressure-locking under design-basis accident conditions. The potential for pressure-locking identified at Connecticut Yankee is significant because of the large number of valves involved. Pressure-locking because of rapid system depressurization, excessive differential pressure, or thermally induced bonnet pressurization could have potentially affected valves supporting the following ECCS functions:

- high pressure safety injection
- low pressure safety injection
- containment sump recirculation
- two-path safety injection
- residual heat removal (shutdown cooling)

The valves that were identified as being susceptible to pressure-locking during this latest evaluation were:

- four redundant high pressure safety injection isolation valves
- two high pressure safety injection pump suction isolation valves
- two redundant low pressure safety injection isolation valves
- one common low pressure safety injection isolation valve
- two redundant charging injection isolation valves
- two redundant high pressure safety injection cross-connect valves from the recirculation header
- four series isolation valves between the reactor coolant system and the residual heat removal system

When Millstone Unit 2 personnel discovered that the containment sump isolation valves may have been rendered inoperable because of the potential for thermally-induced bonnet pressurization (January 1995), the Connecticut Yankee staff increased the urgency and depth of their on-going evaluation. (The previous analyses at Connecticut Yankee and Millstone were performed by the same contractor organization.) During the expedited review, several cases were found where information, not available at the time of the original review, resulted in increasing the analyzed loadings on valve actuators. For the high pressure and low pressure safety injection valves, use of actual valve factors measured during the 1995 refueling outage and an improved methodology, resulted in larger pull-out loads than assumed. Also, the use of guidance and criteria available in NUREG 1275 (published in 1993) resulted in more pressure-locking concerns due to thermal effects. Connecticut Yankee discovered the following instances where pressure-locking had not been fully addressed:

- the potential for thermally induced pressure-locking caused by water entrapment in the valve bonnets
- the susceptibility to pressure-locking of valves that must be repositioned during the later stages of a design-basis accident. If the lines are stagnant and subjected to high external environmental (temperature)

conditions, bonnet temperatures and pressures could increase, potentially pressure-locking the valve.

Consequences/Comments:

Information and recommendations on pressure-locking were provided in 1984 in INPO SOER 84-7, "Pressure Locking and Thermal Binding of Gate Valves." The response to SOER 84-7, and the follow-on SEE-IN documents related to pressure-locking (SER 8-88, SEN-R 89-02, and SEN-R 91-03) has not been sufficiently effective in identifying, evaluating, and correcting conditions that might lead to pressure-locking of safety-related gate valves. Industry response to pressure-locking may have been affected by several factors:

First, while much information had been issued, there has been no industry agreed-upon method or criteria to assure comprehensive screening and evaluation. A comprehensive evaluation for pressure-locking requires knowledge of detailed valve performance characteristics and an understanding of valve environment under normal, accident, and test conditions.

Second, many facilities had no prior history of pressure-locking and may have lacked an understanding of the conditions under which pressure-locking could occur or an appreciation of the safety significance of the phenomena.

Finally, pressure-locking was sometimes viewed as an engineering issue rather than an operational safety concern. As an engineering problem, bounding conditions were established based on a fixed operational environment. Many valves were excluded from further evaluation because some analyses did not include evaluations of changes in operating conditions.

Other criteria that may have been improperly used to exclude valves from a detailed analysis were:

- the presence of check valves in the upstream or downstream piping
- valve orientation
- assumed leakage from the valve bonnet assembly
- valves equipped with air or hydraulic operators

As a result of limited or no experience with pressure-locking, many managers may have concluded that pressure-locking was not a credible failure mode. In some cases, corrective actions were not implemented when a potential pressure-locking problem was identified. Documentation and review of evaluations may not have been detailed enough to detect nonconservatism in the evaluation process. A tendency to rely solely on engineering judgment could preclude an in-depth review that might have identified system and plant operating regimes where precursors for pressure-locking conditions could exist. For example, valves that are normally closed, but are cycled during surveillance testing, or during draining and venting evolutions, have a higher probability for liquid entrapment in their bonnet assemblies. Very few, if any, analyses addressed the potential for pressure-locking that might occur when valves were realigned for testing purposes.

Plant/NSSS: Connecticut Yankee/Westinghouse
Event Date: March 9, 1995

8. Recurring Event: Loss of Shutdown Cooling During Installation of a Plant Modification

Description:

On February 13, 1995, Catawba Unit 1 was in the third day of a refueling outage when a suction valve for the operating train B of residual heat removal (RHR) inadvertently closed. Core cooling was interrupted for approximately 20 minutes, and reactor coolant system temperature increased from 110 degrees F to 135 degrees F (75 degrees F/hour heatup rate). A modification to remove the auto-close interlock circuitry for RHR train A suction valves was in progress with RHR train B in service. As a result of a change to the modification procedure for a similar modification previously performed on Unit 2, technicians were directed to remove power from a protective relay for personnel safety concerns while modifying the relay contacts. When the relay was deenergized, one of the two RHR train B suction valves closed (one suction valve in each train receives operator power from the opposite train). Operators quickly diagnosed the problem and took action in accordance with the abnormal operating procedure to restore RHR flow. The causes of this event were an inadequate engineering and operations modification review and incomplete updating of the modification package from the previous outage. In particular, the modification package was not updated to include the effect of deenergizing the RHR suction valve relays relating to required equipment isolations. Additionally, the modification package did not take into account that the previous modification was performed while the Unit 2 RHR suction valves were deenergized (plant was in mode 5), as opposed to the condition at the time of this event where the RHR suction valves were energized (plant was in mode 4). The modifications engineer, was aware that the suction valves could close but believed operations personnel had taken action to prevent the operating B train suction valve, powered from the A train, from closing.

A pre-outage safety review determined that with one train of RHR (A train) out of service for modification work (mode 5), at least two steam generators should be available as heat sinks. During the loss of RHR flow, all four steam generators were available as a secondary heat sink, and the A train of RHR could have been returned to service after realigning the system.

Consequences/Comments:

Events relating to loss of shutdown cooling continue to occur in the industry. In this case, the loss of shutdown cooling occurred as a result of an incomplete modification package that did not consider the modification would be performed under different plant conditions than had existed when a similar modification was performed on the other unit. Similar events are discussed in INPO SOER 92-01, "Reducing the Occurrence of Plant Events Through Improved Human Performance," and SOER 85-04, "Loss or Degradation of Residual Heat Removal Capability In PWRs." Additionally, the station has experienced other problems relating to the preparation and review of modification packages.

Plant/NSSS: Catawba/Westinghouse
Event Date: February 13, 1995

9. Recurring Event: Safety-Related Equipment Damage and Potential For Personnel Injury During Concrete Drilling

Description:

On March 6, 1995, D.C. Cook workers were assigned to install a security card reader, conduit, and support anchor bolts for design change work in the Unit 1 AB diesel generator room. A pre-job walkdown was performed by a supervisor and craft foreman. No problems were identified during the walkdown. A worker encountered a void in the concrete while drilling the third hole. The worker suspected that the void was formed by insufficient concrete fill during initial plant construction. The worker moved the drilling location 1 inch and drilled a 3/8-inch diameter hole successfully. During the attempt to drill a fourth hole, another void was encountered. Two additional attempts were made in different locations, and voids were encountered each time. The worker informed the foreman of the void problems. The foreman subsequently identified the drilling location as a vertical electrical pilaster. Site design engineering personnel reviewed plant drawings and confirmed the location as an electrical pilaster. The pilaster contained the 4 kV output cables from Unit 1 AB diesel generator to the T11A bus. The cables are encased in plastic conduit. The cables were not energized at the time of the incident.

During the pre-job walkdown, neither individual recognized the installation location as a electrical pilaster because it had a nondistinct physical appearance and other equipment was mounted on the pilaster. The pilaster was excavated to inspect for damage to the electrical cables, and the Unit 1 AB diesel generator was declared inoperable. Two conduits had been penetrated but no cable damage occurred. The station has a drilling permit program. The program requires a drilling permit when drilling 1/2-inch holes or larger and also requires a drilling permit for all drilling on concrete beams, columns, and structural pilasters regardless of hole size. Plant policy, in effect at the time of this incident, was not to use electrical pilasters as a support point.

This event is similar to an event described in SER 86-84, "Industrial Accident Involving Electrical Shock Caused Two Fatalities." This SER recommended that structures containing embedded circuits be prominently marked or posted with warning signs. This recommendation was not implemented by the station. Posting and marking was considered difficult to accomplish, since embedded circuits were common throughout the plant. In this event, the designer thought the work location was a structural pilaster but did not adequately depict it on the drawing. Consequently, the existing drilling program, designed to prevent events of this type, was not implemented.

Consequences/Comments:

Inadequate work package preparation, unmarked pilasters, lack of identification of the structure type by designers, and weaknesses in the station concrete drilling program resulted in inappropriate drilling, that under different circumstances could have resulted in personnel injury. Contributing factors to this event included:

- o electrical pilasters in the plant were not marked or labeled for ease of identification
- o site design department had not clearly identified electrical or structural pilasters on design drawings
- o relocating and redrilling after encountering a void was allowed by station procedures

The station "three-step method" of drilling, which could have prevented this event, was not implemented because the work location was not identified as a pilaster. (The three step method requires utilizing a masonry drill bit that prevents penetrating metallic objects. Drilling is done in short increments and a hammer is utilized to chip away concrete until the desired depth is obtained.)

Plant/NSSS: D.C. Cook/Westinghouse
Event Date: March 6, 1995

10. Recurring Event: Excessive Unplanned Personnel Radiation Exposure As a Result of Inadequate Survey and Improper Response to Alarming Dosimetry Alarming Dosimetry

Description:

At Maine Yankee, on March 24, 1995, two workers received unplanned radiation exposures in excess of administrative limits as a result of an inadequate pre-job survey and improper personnel action upon receipt of dosimetry alarms. Two contract workers were assigned to prepare areas of a highly radioactive reactor coolant pump impeller shaft for nondestructive testing. A survey had been performed the previous shift by a contract health physics technician. However, the survey was rapidly performed in an effort to reduce radiation exposure and lacked sufficient detail of the specific work area. Consequently, the dose rate at the work area was incorrectly documented as approximately 1 rem per hour, when in fact it was about 20 rem per hour. A contract health physics technician was assigned to directly monitor the job. This technician did not completely review the radiological work permit package for the job and did not brief the workers before they started the job. The health physics technician erroneously assumed the ALARA group had briefed the workers. Additionally, station procedures required that a verification survey be performed prior to the start of the job, and this was not done. The workers and the health physics technician entered the work area, and the workers' dosimetry immediately alarmed. The health physics technician observed that the dosimetry was alarming on dose rate (set to alarm at 4 rem per hour) and not on accumulated dose (set to alarm at 400 mrem) and allowed the work to continue. Rather than stop work and resurvey the area, the health physics technician started an air filtration unit and an air sampler. The health physics technician then surveyed the area and observed higher than expected dose rates. The accumulated dose on the workers' alarming dosimeters was then checked, and it had exceeded the alarm setpoint. The health physics technician then stopped the job, and upon exiting the area, it was noted that the workers' self-reading dosimeters were off-scale high. One worker received a radiation dose of 3.3 rem, and the other received 3.1 rem.

Station radiological protection supervisory involvement in the job was limited, although the potential for high radiation exposure from the highly radioactive impeller shaft was recognized. A high emergent workload had limited station supervisory involvement in this and other radiological protection jobs. INPO SOER 85-3, "Excessive Personnel Radiation Exposure," recommends direct supervisory involvement in jobs where high radiation exposures can be received in a short time interval. Specific steps were not included in the radiation work permit or the applicable maintenance procedure to prevent excessive radiation exposure, another recommendation in SOER 85-3. Additionally, the radiological training program for nonstation personnel was not effective in fostering a high level of awareness and sense of individual responsibility regarding control of personnel radiation exposure. This is also an SOER 85-3 recommendation.

Consequences/Comments:

An inadequate pre-job survey and noncompliance with station procedures regarding survey verification and response to alarming dosimetry were direct causes of two personnel receiving unplanned radiation exposures greater than 1 rem. Insufficiently effective implementation of some SOER 85-3 recommendations also contributed to the event.

Plant/NSSS: Maine Yankee/Combustion Engineering
Event Date: March 24, 1995

11. Recurring Event: Increased Potential for Excessive Unplanned Personnel Radiation Exposure

Description:

On March 9, 1995, with Arkansas Nuclear One Unit 1 in a refueling outage, unanticipated high dose rates were experienced while returning the reactor core support assembly (CSA) to the reactor vessel. Performance of a 10-year in-service inspection of the reactor vessel internal welds required removal of the CSA. The station provided extensive details of this event in NUCLEAR NETWORK Operating Experience entry OE 7304, "Higher Than Expected Dose Rates Experienced During Installation of the Reactor Core Support Assembly," and OE 7304 should be referred to for the event description. Among the inappropriate actions documented by the OE entry were: 1) a failure to treat the CSA movement as an infrequently performed test or evolution, and 2) some personnel did not immediately exit the area when electronic dosimetry alarmed. In this event, individual exposures received were below station administrative limits, with the highest individual dose being 608 millirem.

Consequences/Comments:

INPO SOER 91-1, "Conduct of Infrequently Performed Tests or Evolutions," contains several recommendations regarding performance of infrequently performed evolutions, such as the CSA movement. Although the activity had the potential for high radiation exposure and is normally only performed every 10 years, it was not identified as an infrequently performed test or evolution. Enhanced briefings and additional supervisory oversight as recommended in SOER 91-1 could have resulted in more effective mitigation of the event. Additionally, INPO SOER 85-3, "Excessive Personnel Radiation Exposure," discusses several instances where personnel took nonconservative actions when unusual radiological conditions were encountered.

Plant/NSSS: Arkansas Nuclear One Unit 1/B&W
Event Date: March 9, 1995

12. Recurring Event: Excessive Unplanned Personnel Radiation Exposure from an Undetected Activated Stellite Particle

Description:

On February 13, 1995, while Maine Yankee was in a refueling outage, a pre-job radiation survey was performed in the fuel transfer canal, and a small activated stellite particle was discovered inside the tip of a hydrolazing wand suspended in the canal. Dose rates on contact with the tip of the wand read up to 400 rem/hour. Work had been performed on the fuel assembly upender near the suspended hydrolazing wand two days previously. In light of the February 13 survey results, radiological protection personnel reviewed a videotape of the previous two days work and observed that workers were in close proximity to the hydrolazing wand, with one worker contacting the hydrolazer wand. The geometry of the particle caused the radiation field to be highly directional, with a steep gradient to the dose rate. As a result, the workers' dosimetry may not have accurately measured the whole body exposure actually received. Using the videotape and extensive worker interviews, the workers' whole body exposure was calculated. The calculated two-day dose to the worker who contacted the wand was 3.103 rem. The station's administrative limit is 1 rem. The calculated exposure for the other workers was below 1 rem.

The general area radiation dose rates were relatively high around the fuel assembly upender, and the pre-job survey for the upender work was not detailed enough to identify the activated stellite particle on the wand. Prior to the work in the fuel transfer canal, the area was decontaminated, and the stellite particle became attached to the hydrolazing wand used for this activity. The hydrolazing wand was left suspended in the transfer canal. Dose rates at the fuel assembly upender were dependent on the water level in the transfer canal, and at the time of the pre-job survey, were 300-400 millirem per hour. Water level could not be increased to reduce the dose rates because the water level was limited to support other scheduled work. The high general area dose rates affected the ability to differentiate the particle's radiation field and specifically identify it. When the pre-job survey for additional work was performed on February 13, the water level in the transfer canal was higher, and general area dose rates were less, making it easier to identify the radiation field from the stellite particle. Finally, the hydrolazing tool that had been used for decontamination of a highly contaminated area had been allowed to remain at a work site where it would not be used.

Consequences/Comments:

Surveys were performed with insufficient detail to accurately determine the radiological conditions at the job site. Additionally, leaving an unrelated tool, used in a highly contaminated area at the job site, was an improper radiological work practice. Removal of the tool to an area with lower dose rates could have facilitated earlier discovery of the activated stellite particle. This event is similar to events described in INPO SOER 85-3, "Excessive Personnel Radiation Exposures."

Plant/NSSS: Maine Yankee/Combustion Engineering
Event Date: February 12, 1995

INTERNATIONAL EVENTS

1. Recurring Event: Power Oscillations at a Boiling Water Reactor Result in a Manual Scram

Note: The following information is based on NUCLEAR NETWORK Operating Experience entry OE 7075 and on a World Association of Nuclear Operators (WANO) Event Analysis Report dated August 10, 1995.

Background:

Laguna Verde is a 654-MWe GE BWR/5 that began commercial operation in 1990.

Description:

On January 24, 1995, Laguna Verde 1 was in power ascension at approximately 37 percent power (approximately 66 percent rod-line and 38 percent core flow), and operators were preparing to shift the reactor recirculation pumps to fast speed. As the recirculation flow control valves were throttled to the minimum flow position to enable shifting the recirculation pumps to fast speed, the average power range monitors (APRMs) indicated the onset of power oscillations (maximum magnitude of the oscillations was approximately 11 percent peak-to-peak). There were no high- or low-scale APRM alarms present. Reactor operators reopened the flow control valves to increase core flow in an attempt restore core flow to the conditions prior to the onset of the power oscillations. However, the power oscillations continued. Approximately six minutes after the onset of the power oscillations, operators manually scrammed the reactor to terminate the power oscillations.

Consequences/Comments:

The power-to-flow map in the plant technical specifications defines expected regions of instability. When the power oscillations occurred, the power-to-flow conditions were not within the regions where instability was expected to occur. However, a similar instability event, described in SER 19-92, "Power Oscillations at Boiling Water Reactors," had identified the possibility of instability occurring when operating near the analytically-determined regions of instability. As a result of the 1992 event, the GE BWR Owners Group revised the interim corrective actions for BWR thermal instability to emphasize the need for caution when operating near the region of instability. The interim corrective actions also stressed the need to ensure operators are trained to recognize that the analytically-determined region boundaries are not an absolute indicator of the potential for instability under all conditions. Because Laguna Verde had not received the revised interim corrective actions document, operators and reactor engineering personnel were not aware of the potential for power oscillations while operating under these conditions. A later review of the event determined that the Laguna Verde core, at the time of the instability, was bottom-peaked axially and had a low boiling boundary. The potential effects of these factors were specifically discussed in the revised interim corrective actions; however, this information was not available to Laguna Verde.

Plant/NSSS: Laguna Verde 1/BWR (GE)
Event Date: January 24, 1995

2. Recurring Event: Water Hammer in Main Steam Lines Causes Main Steam Safety Valve Safety Valve Damage

Note: The following information is based on a World Association of Nuclear Operators (WANO) Event Analysis Report dated December 12, 1994.

Background:

Cattenom 1 is a 1,300-MWe Framatome/EdF pressurized water reactor that began commercial operation in 1987.

Description:

On June 12, 1994, during Cattenom Unit 1 start-up activities following an outage, delays in opening the main steam line drain valves resulted in several water hammers and damage to the main steam safety valves. At the time of the event, reactor pressure was 2,248 psia, and temperature was 563 degrees F, with a secondary pressure of 1,131 psia. During plant heatup and main steam drain system warm-up, operators implement three procedures affecting the alignment of main steam line drains. When coordination problems delayed full implementation of these procedures, the unit heatup was continued without assuring that the steam lines were properly drained. While in this undrained condition, the main steam atmospheric dump valves were opened to purge noncondensable gases from the steam generators. When the dump valves were opened with residual water in the lines, water hammers were experienced, resulting in safety valve damage.

Consequences/Comments:

This event occurred because of insufficient coordination of plant heatup activities. The control room staff became focused on start-up coordination problems and did not adequately control draining of the main steam lines and atmospheric dump operation. The outage was extended by 40 days to repair safety valve damage and to examine main steam lines and supports. Similar events are discussed in SER 23-87, "Water Hammer in Main Steam Lines," and SER 69-84, "Damage in Main Steam and Feedwater Systems Caused By Water Hammer and Rapid Thermal Transients."

Plant/NSSS: Cattenom Unit 1/Framatome, EdF
Event Date: June 12, 1994

3. Recurring Event: Maintenance On the Wrong Train Causes Fire Suppression Systems Actuation

Note: The following information is based on a World Association of Nuclear Operators (WANO) Event Analysis Report dated May 4, 1995.

Background:

Balakovo Unit 4 is a 1,000-MWe, four-loop Ministry of Heavy Engineering, VVER-1,000 design pressurized water reactor that began commercial operation in 1993.

Description:

On September 21, 1994, during fire suppression system maintenance at Balakovo Unit 4, the fire suppression system in cable vault 1 actuated unexpectedly. The train 2 fire suppression system had been taken out of service for authorized maintenance; however, the workers began work on safety system train 1. The train 1 fire suppression system actuated when fire detectors were removed.

Consequences/Comments:

Although the consequences were minimal during this event, personnel safety hazards and the potential for damage to safety-related equipment existed. This event occurred because, contrary to station expectations, the work plan lacked specific directions regarding the location where the work was to be performed, and supervisory oversight was insufficiently effective in assuring that the work was performed on the intended train. In addition, similarities in the labeling ("Cable Room 1 I-SS" versus "Cable Room II-SS") of the rooms that contained the equipment contributed to this event.

Events involving operation or maintenance on the wrong unit or wrong train are discussed in the following significant event reports:

1. SER 23-88, "Work On Wrong Train of Radwaste Evaporator Results in Personnel Injuries"
2. SER 25-84, "Trips of Both Reactors At Site Following a Spurious Transformer Fault and Jarring of Relays"
3. SER 24-83, "Opening of Both Power Operated Relief Valves"
4. SER 22-83, "Operator Performs Valve Alignment on Wrong Unit"

SOER 92-1, "Reducing the Occurrence of Plant Events Through Improved Human Performance," discusses the need for clear direction and supervisory oversight of plant activities.

Plant/NSSS:	Balakovo Unit 4/Ministry of Heavy Engineering, VVER-1,000
Event Date:	September 21, 1994

LIMITED DISTRIBUTION

Document Copy Contact: INPO Records Center, (404) 644-8697

Information Contact: Fred Rehrig, INPO, (404) 644-8631

ed
IS 1241 I NEWTON 22-JUN-95 13:19 EST
INPO (INP)
Subject: SER 15-95, Spent Fuel Pool Liner Punctured by Dropped Equipment

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SUMMARY:

At two plants, equipment dropped into the spent fuel pool punctured the pool stainless steel liner plates. In one event, inadequate training of contract workers resulted in improper fabrication of a lifting sling. The sling failed while supporting a core shroud head bolt, and the bolt punctured the pool floor liner plate. In the second event, an unauthorized modification to a tool made it difficult to attach a safety lanyard. As a result, the lanyard was not installed, and the tool was inadvertently dropped and punctured the pool floor liner plate. During these events, plant personnel terminated spent fuel pool leakage prior to significant decrease in water level.

These events are significant because dropped loads, which were not considered in the spent fuel pool safety analyses, punctured pool liner plates and resulted in pool leakage.

LESSONS LEARNED FROM THESE EVENTS:

1. Plant safety analyses for bounding the effects of a load dropped over the spent fuel pool may consider that a dropped fuel bundle is the worst case event. In these events, a reactor component and a tool, both much lighter than a fuel assembly, caused damage to the spent fuel pool liner when dropped from near the surface of the spent fuel pool.

2. The weight, elevation, path of travel, and geometry of a suspended load need to be considered when evaluating the potential consequences of a dropped load.

3. Training contract workers on plant expectations and procedures applicable to assigned work continues to be an important method for ensuring high standards of performance are met. In one event, deficient contractor knowledge and understanding of utility requirements for lifting slings resulted in improperly constructed slings that failed under load.

UNIT: E. I. Hatch Nuclear Plant Unit 1 (Georgia Power Company)

YEAR COMMERCIAL: 1975

OE 7443 I LLOYD 25-AUG-95 15:44 EST
PENNSYLVANIA POWER (PPL)
Subject: New Fuel Bundle Dropped in Spent Fuel Pool

SUSQUEHANNA UNIT 2
AUGUST 25, 1995
NSSS/GE BWR-4
RATING 1135 MWe
DATE OF COMMERCIAL OPERATION UNIT 1, 6/8/83 UNIT 2, 2/12/85

EVENT DESCRIPTION

On 8/22/95 while performing a fuel shuffle from the Unit 2 New Fuel Vault to the Fuel Prep machine, a new unirradiated fuel bundle fell approximately 15-20 feet into the Fuel Prep Machine in the Spent Fuel Pool when the air operated general purpose grapple separated from the jib crane. The fuel bundle went through the top guide of the Prep Machine and impacted the lower carriage support plate.

The bundle came to rest at a slight angle, leaning on the east roller guide and the west rail of the lower carriage. The west rail of the lower carriage support plate was found to have substantial damage.

The bundle is completely submerged. No personnel injury occurred, and no obvious visual damage to the fuel bundle was noted during initial inspection. There was no increase in radiation in the immediate vicinity.

The cause of the event is under investigation. It is not known at this time why the general purpose grapple became detached from the jib crane cable swivel connector. Concurrent with the cause investigation, a recovery plan is being implemented to remove the fuel bundle from the Prep Machine.

Information Contact: John Finnegan (717) 542-3242

REACTOR TYPE (SIZE): Boiling Water (850 MWe)
REACTOR MANUFACTURER: General Electric
TURBINE MANUFACTURER: General Electric
PLANT DESIGNER: Southern Company Services & Bechtel
EVENT DATE: December 28, 1994

UNIT: Tricastin Unit 1 (Electricite de France)
YEAR COMMERCIAL: 1980
REACTOR TYPE (SIZE): Pressurized Water (915 MWe)
REACTOR MANUFACTURER: Framatome/Creusot-Loire
TURBINE MANUFACTURER: Alsthom
PLANT DESIGNER: Electricite de France
EVENT DATE: January 31, 1994

REFERENCES:

1. INPO Significant Event Notification (SEN) 123, "Spent Fuel Pool Liner Plate Punctured When Shroud Head Bolt Dropped Due to Rigging Failure," January 13, 1995, (IS 1212)

2. WANO-Paris Center, MER PAR 94-009, "Spent Fuel Pool Leak," January 10, 1995, (ME 480)

E. I. HATCH NUCLEAR PLANT UNIT 1

Description

Seven defective core shroud head bolts were identified during the October 1994 refueling outage. The defective bolts were removed from the shroud head and moved to the spent fuel pool. The route for bolt movement was selected to minimize the potential for passing over fuel assemblies. Once in the spent fuel pool, the shroud head bolts were stored by suspending them from the side of the pool using site-fabricated stainless steel wire rope lifting slings. Shroud head bolts are about 3 inches in diameter and 17 feet long and weigh 365 pounds.

On December 28, 1994, the seven defective core shroud head bolts were being removed for shipment and burial. To move the bolts to the shipping container, the stainless steel wire rope slings were attached to the refueling floor crane auxiliary hoist with nylon slings. The wire rope sling used to lift the first bolt failed when the bottom of the bolt was about 1 foot above the spent fuel pool water surface. The shroud head bolt dropped into the pool, glanced off the side wall of the spent fuel pool, and punctured the 3/16-inch thick floor liner plate. The contract crew was unaware that the liner had been punctured and continued to remove a second bolt without notifying the control room of the dropped bolt. The contract crew successfully lifted the second shroud head bolt and placed it in the shipping container.

When the spent fuel pool liner plate was punctured, approximately 2,000 gallons of spent fuel pool water drained to the radioactive waste system via a normally open leak detection drain line. Spent fuel pool water level decreased about 2 inches in 23 minutes, actuating the low spent fuel pool cooling surge tank level alarm in the control room and tripping the operating spent fuel pool cooling pump on low suction pressure. An equipment operator sent to the refueling floor to investigate the cause of the pump trip noted the spent fuel pool water level decrease and discovered that a core shroud head bolt had punctured the spent fuel pool liner plate. The operator notified the control room and halted shroud head bolt lifting activities. Operations personnel subsequently isolated the leak detection drain line to the radioactive waste system, restored spent fuel pool water level to normal, and returned the spent fuel pool cooling system to service.

The remaining five core shroud head bolts were removed from the spent fuel pool using a double-rigging arrangement that included load-tested nylon slings. On December 31, 1994, the core shroud head bolt that had punctured the bottom of the spent fuel liner was removed from the spent fuel pool, and a temporary rubber patch was installed over the hole in the liner plate. On January 2, 1995, divers installed a permanent welded patch. The repair effort resulted in about 2 man-rem of radiation exposure to the divers.

Analysis

Safety Analysis

The potential for damage to the spent fuel pool liner from a dropped core shroud head bolt was not considered during work planning or in the facility safety analysis. The safety analysis determined that, as a bounding case, a dropped fuel assembly would not puncture the liner plate, but it did not address the forces applied by a smaller diameter object. Therefore, perforation of the Hatch fuel pool liner by the dropped core shroud head bolt was not expected. As a result, no special precautions, such as using certified lifting slings, were required for lifting the core shroud head bolts.

Barrier Analysis:

Barrier -- Contractor Training

Contract maintenance workers fabricated a lifting sling for each bolt using about 7 feet of 3/16-inch stainless steel wire rope with an eye at each end. The eyes were formed by looping the wire and joining the cable to itself by crimping the wire with two stainless steel compression sleeves. The contract maintenance personnel who made the lifting slings were not trained on the proper procedure for fabrication of wire rope slings. The workers did not know that the vendor instruction manual contained the only available information for proper installation of the compression sleeves. As a result, the crimping tool used was for soft metal sleeves, such as copper or aluminum, and inadequate sleeve compression was obtained. The sling failed during use when the cable dead end at the top eye pulled out of the improperly crimped compression sleeves, even though the slings had been tested with a 456-pound load.

Similarly, the contract workers removing the bolts from the spent fuel pool and the health physics technician providing job oversight had not been trained or briefed to notify the control room if an abnormal condition was encountered during work. Additional utility oversight was not considered necessary for this job.

Corrective Actions:

1. The leak in the spent fuel pool liner was permanently repaired with a welded stainless steel plate.

2. The station has discontinued site fabrication of lifting slings. Only certified prefabricated slings are used.

3. Procedural guidance for control of rigging activities in or over the spent fuel pool will be improved.

TRICASTIN UNIT 1 (FRANCE)

Description

On January 31, 1994, contract maintenance workers were removing a control rod cluster control guide tube from an irradiated fuel assembly in the spent fuel pool fuel elevator. After replacing the fuel element top nozzle assembly, a 44-pound, 15-foot long screwdriver was being returned to its storage bracket. As the screwdriver was manually removed from the spent fuel pool building crane hook, it dropped into the spent fuel pool. The contract workers immediately notified the plant work supervisor that the screwdriver had been dropped. Plant personnel quickly assessed the event and concluded that the screwdriver could have pierced the spent fuel pool stainless steel liner.

The shift supervisor was notified of the potential for a spent fuel pool liner leak, and water was observed leaking out of the refueling building. The water level in the spent fuel pool decreased about 4 inches during the event. Plant personnel determined that without operator action, the spent fuel pool cooling pumps could have operated for 16 days before tripping on low suction head. A suction cup plug was used to temporarily repair the pool liner. Permanent repair was made by welding a stainless steel plate over the hole in the liner.

Analysis

Barrier Analysis

Barrier -- Modification Control

The screwdriver at Tricastin had been modified by station personnel without approval or knowledge of the engineering department responsible for tool design. The screwdriver had a U-shaped lifting ring with a crossbar that rested on the storage bracket. This crossbar made it difficult to attach the ratchet wrench used to turn the screwdriver. Workers, who were not aware of its purpose, removed the crossbar. This modification caused the tool to rest on the storage bracket lifting ring, using the same location where a lanyard or lifting sling would be attached. As a result, a safety lanyard was not installed when the tool was being placed on the storage bracket because it would have interfered with hanging the screwdriver on the storage bracket. A properly installed lanyard could have prevented the screwdriver from reaching and piercing the spent fuel pool liner.

Corrective Actions:

1. The leak in the spent fuel pool liner was permanently repaired by welding a stainless steel plate to the

liner.

2. The fuel handling tools were restored to their original design.

3. A suction cup plug is stored in the fuel building for emergency use.

4. The spent fuel pool leakage system was modified to contain leakage in the fuel building.

5. Station work plans now require a risk analysis for work performed in the fuel building.

Corrective actions 2 through 5 were implemented at all Electricite de France nuclear plants.

EVENT CRITERIA:	Fuel handling/storage event
CAUSE CATEGORIES:	Construction (fabrication of lifting slings) Design (unauthorized tool modification) Managerial methods (insufficient training and oversight)
MALFUNCTIONING SYSTEMS:	NPRDS System Codes: None NPRDS Function Codes: None
ATTACHMENTS:	None

This document is based on technical information provided by, Georgia Power Company (E. I. Hatch Nuclear Plant) and Electricite de France (Tricastin).

Utilities and participants are requested to provide feedback on similar occurrences and solutions at their plants or on their equipment to the information contact listed below.

LIMITED DISTRIBUTION

KEYWORDS: contractor, heavy load, human performance, lifting device, sleeve, spent fuel pool, training

Information Contact: Paul Hoffmeier, INPO, (404) 644-8474

GE Nuclear Energy

GENE 523-A085-0895

Independent Assessment of Spent Fuel Pool Cooling at Millstone 1 Nuclear Power Station

August 31, 1995

Prepared By

Dinesh Saxena
George Stramback

Approved By: George Stramback
Licensing Services

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1.0 Introduction

Spent Fuel Pool Cooling and Cleanup (SFPCC) systems are utilized on all BWRs and PWRs in the US to provide cooling of the storage pool which contains used radioactive fuel removed from the reactor. The spent (used) fuel is stored under water and cooled in the pool until it can be shipped to an off-site location. Because of institutional and governmental delays in operation of approved off-site locations, Northeast Utilities Service Company's (NUSCo) has redesigned its storage facilities (storage racks) located at the bottom of the storage pools to accommodate an increased number of fuel assemblies which achieves a longer term storage of spent fuel. During each refueling outage some additional spent fuel assemblies are added to the pool. Because utilities typically plan a combination of maintenance, modification and testing activities to coincide with the refueling activities, there are technical and safety benefits derived from the removal from the reactor vessel of part or all of the reactor fuel during the relatively short time of the refueling outage.

A recent letter to the NRC (Reference 1) has expressed a concern for the NUSCo refueling practice of off-loading the entire core at Millstone 1. In support of providing a NUSCo response to that letter, GE Nuclear Energy was requested to perform an independent review and assessment of the Millstone 1 Nuclear Power Station SFPCC design. The assessment was to include review 1) of the bases and processes used to demonstrate compliance to existing requirements using the system currently in place and 2) the planned and in-process modifications that are the basis for a Technical Specification change request recently submitted to the NRC.

2.0 Conclusion

The NRC has reviewed and approved the modifications to the SFPCC, their application and supporting analyses over the years, as evidenced by multiple NRC Safety Evaluation Reports. Early identification of the intent to utilize full core off-loading is contained in the *Introduction* statements of the NUSCo 1976/77 Amendment 39 report (Reference 2) where it states:

> *It is prudent to reserve storage space in the spent fuel pool to receive an entire reactor core, i.e., a full core off-load, should unloading of the core be necessary or desirable because of operational considerations. This together with the fact that spent fuel reprocessing facilities cannot assuredly be available to process Millstone 1 discharge fuel prior to the mid 1980's at the earliest (and, therefore, no additional spent fuel will be shipped offsite) lead to the conclusion that an increase in spent fuel storage capacity is necessary to accommodate both subsequent spent fuel discharge and to maintain the full core off-load capability.*

With consideration to full core off-loads at Millstone 1, spent fuel cooling and refueling operations have been conducted consistent with industry practice and at all times the health and safety of the public has been adequately protected. The existing documentation of the system designs and the analyses performed over the years demonstrates that adequate cooling, radiation protection and coolant makeup capability has existed and such capability is being further enhanced.

Formal processes are in place to evaluate and resolve issues which may arise with respect to the Spent Fuel Pool (SFP) system. These evaluation processes are similar to existing industry practice, which includes reporting to the NRC, if necessary. The NRC also is informed of NUSCo identified SFP issues through the Licensee Event Report (LER) process, (i.e., 10CFR50.73) which includes an evaluation of the basis for continued plant operation, if necessary, and the identification of any corrective or mitigating actions that are taken, underway or planned.

3.0 SFPCC Design Evolution

3.1 Industry Practice and Status

The industry practice for the BWR fleet has been to provide sufficient Spent Fuel Pool Cooling capability in conjunction with either the Shut Down Cooling system or the Residual Heat Removal system for the purpose of permitting complete core off-load into the SFP when necessary during a refueling outage or for repair and maintenance of the reactor internals. The storage of the complete core off-load in the SFP normally requires a period of about 2-4 weeks to perform work inside the Reactor Pressure Vessel (RPV)or during the refueling operation to load new replacement fuel bundles.

During a refueling operation either the entire core or only that portion of the core that is to be replaced with the new fuel could be off-loaded. This operation is dependent upon operational and maintenance requirements specified by the outage planning process. In accordance with NRC recommended limits, the design of the SFP and the systems which are connected to it provides sufficient heat removal for the already stored spent fuel and for a complete core off-load. Then the heat removal capability is sufficient to maintain the SFP cooling water below the acceptance criteria of 150°F.

3.2 Current Design and Previous Modifications

Initial Plant Design

The initial design of the Spent Fuel Pool (SFP) included cooling water covering the storage space for 880 fuel assemblies (Reference 3) containing spent (used) fuel. In 1976

(Reference 3) additional storage racks of the same design as the original storage racks were installed to increase the spent fuel storage to 1100 fuel assemblies. Connected to the SFP is the Spent Fuel Pool Cooling and Cleanup (SFPCC) system which was designed to have a decay heat removal capacity of 7.84 Million Btu/Hr (Reference 3). This cooling capacity is provided by two sets of identical equipment (trains of equipment), each having a heat removal capacity of 50% of the total capacity (Reference 3). Each train of equipment included a pump, a heat exchanger, piping, valves, instrumentation and controls. In addition to these equipment trains there is a Spent Fuel Pool Filter and a Spent Fuel Pool Demineralizer to maintain the water quality in the storage pool.

The SFPCC system is designed and analyzed to maintain the SFP bulk temperature below 125°F (Reference 3). To demonstrate that this temperature can be met, the analysis conditions include the decay heat from one quarter of the core off-loaded during the refueling operation and the spent fuel already in the pool storage racks.

The SFPCC system also is designed to work in conjunction with the Shutdown Cooling (SDC) system during refueling outages. This facilitates performance of reactor maintenance with the reactor fuel removed. During such an outage, the entire reactor core (100% – equaling 580 fuel assemblies) may be removed from the reactor pressure vessel and stored in the SFP. Depending upon the heat removal needs for the outage, the combined cooling capacity of SFPCC system trains and the one train of the SDC system could be used to provide 29.86 Million Btu/Hr or, if less equipment capacity is needed, 25.94 Million Btu/Hr (References 4 & 5) could be provided by only including one train each of the SFPCC and SDC systems.

Spent Fuel Pool Re-racking (Millstone Amendment 39) ('76/'77)

A modification to the Millstone 1 SFP storage capacity was filed by NUSCo in July 1976 to replace the existing fuel storage racks with a new rack design which would increase the storage capacity from 1100 fuel assemblies to 2184 fuel assemblies. This included storage of 1512 spent fuel assemblies until the 1984 refueling operation and reserve storage for one full core off-load of 580 fuel assemblies. The NRC approved Amendment 39 (Reference 2) to the Millstone 1 operating license.

This Amendment identified the NUSCo intent to utilize full core off-loading, as operational considerations warrant, and states this intention followed up by supporting analysis. The NUSCo report *Introduction* states:

> *It is prudent to reserve storage space in the spent fuel pool to receive an entire reactor core, i.e., a full core off-load, should unloading of the core be necessary or desirable because of operational considerations. This together with the fact that spent fuel reprocessing facilities cannot assuredly be available to process Millstone*

1 discharge fuel prior to the mid 1980's at the earliest (and, therefore, no additional spent fuel will be shipped offsite) lead to the conclusion that an increase in spent fuel storage capacity is necessary to accommodate both subsequent spent fuel discharge and to maintain the full core off-load capability.

With this modification in place, the SFPCC system heat exchanger design capacity (both trains) of 7.84 Million Btu/Hr is capable of handling the decay heat from the 1512 spent fuel assemblies while maintaining the fuel pool bulk temperature at or below 125°F. The supporting analyses also indicated that the SFPCC system, in conjunction with SDC train A, was capable of maintaining the fuel pool bulk temperature at or below 140°F while removing the decay heat from 1512 spent fuel pool assemblies and 580 fuel assemblies, comprising the full core off-load from the RPV. The combined decay heat from the spent fuel and full core off-load (after 250 hours) was estimated by NUSCo to be 17.9 Million Btu/Hr (Reference 3). This is significantly less than the combined cooling capacity of one SFPCC train and one SDC train which is 25.92 Million Btu/Hr; only one of each train is necessary and one train of SFPCC is in reserve. Even though this condition of full core off-load is not a single failure analysis basis, this reserve capacity greatly extends the time to potential boiling of the SFP. This extension allows additional time for operator action to recover SFP cooling or provide one or more of the alternate makeup sources.

The NRC Standard Review Plan (SRP) "Technical Position APCSB 9-2" concluded that the NUSCo analysis was adequately conservative. Part of the NUSCo analysis concluded that, after a full core off-load, if there was to be a complete loss of all spent fuel pool cooling, it would take more than eleven hours to heat the fuel pool water volume of 40,000 cubic feet from 125°F to 212°F. Eleven hours would be sufficient time either to make repairs or to add water from one of the alternate sources of coolant make up to the SFP (Reference 2).

The NRC review found that the analyses, design, fabrication and installation of the proposed storage racks to be in accordance with accepted criteria, and in conformance with the rules of subsection NF of Section III of the ASME B&PV Code. The storage racks are designed as Seismic Category 1 structures.

The NRC concluded that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

Spent Fuel Pool Re-racking (Millstone Amendment 40) ('88/'89)

In July 1988, NUSCo filed another proposed modification to Millstone 1 Spent Fuel Storage Pool, again to increase the storage capacity. This modification increased the SFP storage capacity from the existing 2184 fuel assemblies by 1045 to 3229 fuel assemblies and 20 defective fuel containers. Again the new SFP storage capacity included full core (580 fuel assemblies) reserve off-load capability. Analyses to support this amendment were provided, including the same type of full core off-load and spent fuel heat loads evaluation (Reference 6).

The NRC issued Amendment 40 to Millstone 1 Facility Operating License No. DPR-21 in November 1989. The amendment included the addition of a new upper limit on the number of fuel assemblies, supporting expansion of the acceptable capacity of the SFP to 3229 fuel assemblies (Reference 7).

The analysis results indicated that the maximum SFP water bulk temperature, with full core off-load and the existing spent fuel, was 140°F (Reference 8) which is below the acceptance criteria limit of 150°F (References 7 & 9). One train of the SDC system acting in conjunction with one train of the SFPCC system, with one train of SFPCC in reserve was assumed in the supporting analysis.

The staff reviewed the NUSCo analysis regarding the adequacy of the spent fuel pool cooling system in light of the increased fuel storage capacity and found that the cooling system met the criteria in the SRP and was, therefore, acceptable.

3.3 Spent Fuel Pool Mod (Proposed Modification) ('95)

In July 1995, NUSCo submitted to the NRC additional Technical Specification constrains on operation of Millstone 1 to accompany an enhancement modification to the SDC system interface with the SFPCC system. This modification included a request to change the operating license condition to permit Millstone 1 to perform a full core discharge as a normal end-of-cycle event. NUSCo requested NRC to review and approve this modification pursuant to 10CFR50.90 and 10CFR50.92.

The enhancement modification includes the installation of a SDC system train B cross-tie connection to the SFPCC system. This cross-tie will provide cooling capability similar to that of SDC system train A and satisfies the need to accommodate a single failure while still providing adequate cooling. In the event of a single active failure, the new SDC train B cross-tie will allow for a minimum of one SFPCC system pump and one SDC system pump to provide the necessary cooling. This configuration maintains the SFP bulk temperature below the 140°F technical specification limit.

Additionally, NUSCo performed a cycle specific analysis for the 1994 refueling outage which showed that the spent fuel pool bulk temperature would not exceed the 150°F limit for a full core off-load (after 150 hours decay time) with a single active failure (References 9 & 10).

4.0 Issue Identification & Evaluation

The Reference 1 letter makes several technical statements with respect to the NUSCo practice at Millstone 1 of complete core off-load during the refueling outages. The following review addresses various concerns raised in the above petition:

Concern:

The Spent Fuel Pool does not have sufficient cooling capability to maintain the pool at or below the required temperature limit of 140°F with the complete core off-loaded during the refueling outage.

Review Comments:

A review of the NRC approved Amendments 39 and 40 to the Millstone 1 Facility Operating License DPR-21 indicates that appropriate design analyses were performed by NUSCo to support the acceptable performance capability of the Spent Fuel Storage Pool. This included the improved design of the new spent fuel storage racks now in place in the SFP and sufficient SFP cooling capability to handle the decay heat from the already stored spent fuel and the entire core off-load consisting of 580 fuel assemblies from the RPV during the refueling and/or maintenance operation. This process is a common practice in the rest of the BWR fleet.

The maximum cooling capability, assuming the single failure of one SFPCC pump, is available from the operation of one SFPCC system train and one SDC train. The total combined cooling capacity of these two trains is 25.92 Million Btu/hr which is far greater than the total decay heat load of 17.9 Million Btu/hr in the SFP. This evaluation includes the heat load from the already stored spent fuel assemblies and full core off-load consisting of 580 fuel assemblies. Under these heat load conditions, the SFP temperature can be maintained at or below the current acceptance criteria of 150 °F.

Concern:

The fuel pool can be drained due to lack of (1) seismic make-up capability to the SFP and, (2) the SFP cooling return pipes are equipped with anti-siphon check valves and do not have the one-half inch holes drilled in them one foot below the normal pool water level. As a result of these events, the fuel assemblies in the SFP will be exposed and will create a heat-up condition, which will subsequently meltdown.

Review Comments:

In the event of a seismic pipe break, there are three alternative sources to provide make-up water to the SFP. These sources include supply of make-up water available from the Condensate Storage Tank, water from the Skimmer Surge Tank and manual use of fire hoses to fill the pool with water. Additionally, liquid level switches monitoring pool water level are provided to detect loss of water and permit refilling of the pool from the above sources.

The function of the anti-siphon check valves is to prevent back flow of water from the pool due to any siphoning effect. The intent of the drilled holes in the same pipe is a secondary device to prevent siphoning of water out of the pool through the pipes. If the holes are not in the pipe and/or the holes become plugged, the anti-siphoning check valves will stop back flow of water out of the pool. In addition, as soon as the water level in the pool drops below a pre-determined safety level, the level switch monitoring device alerts the operator to initiate actions to refill the pool.

To avoid draining of the pool, the pool has been designed to have no penetrations that would permit the pool to be drained below approximately nine feet above the top of the active fuel, which still is a safe storage level. All lines extending below this level are equipped with suitable valving to prevent backflow.

It is believed that suitable measures have been taken by NUSCo to provide sufficient cooling in the SFP for the removal of decay heat from the stored spent fuel and the full core off-load. It is concluded that the safety of the public was never compromised due to full core off-load operations of Millstone 1, under the above stated conditions.

Concern:

With respect to SEP Topic III-7B, as early as 1990, NUSCo knew that the SFP cooling return piping line, which is encased in concrete only is good to 85°F per ANSI B-31.1. No attempt was made to rectify the problem.

Review Comments:

The piping in question is being modified to accommodate thermal expansion. It also was analyzed and it was determined that the return line piping did not exceed the operability criteria as specified in Generic Letter 91-18.

Concern:

NUSCo has knowingly, willingly and flagrantly operated Millstone Unit 1 in violation of License Amendment 39 and 40.

Review Comments:

Amendments 39 and 40 to the Millstone Operating License No. DPR-21 can be interpreted to represent approval of the transfer of the full core off-load, as necessary for maintenance of the Reactor Pressure Vessel and / or for other refueling outage operations. Also, the NRC found the NUSCo calculated spent fuel cooling heat loads in the SFP to be conservative, and the capacity to be sufficient to maintain the SFP outlet water temperature below 125°F.

It is believed that Millstone 1 was operated within the appropriate requirements of the operating license.

5.0 Recommendations

GE recommends completion of the proposed modifications to enhance the failure tolerance of the SFPCC and SDC systems and resolution of any identified configuration or operational weaknesses with these systems. Continued participation in BWR Owners' Group activities and initiatives regarding SFP issues is advisable, along with continued monitoring and evaluation of NRC identified issues.

In addition, clarification of two issues is recommended 1) is the terminology in the UFSAR section 9.1 with regard to the categories of normal and abnormal and their meaning with regard to full core off-loads, and 2) is the use of 140°F and / or 150°F for the bulk temperature of the spent fuel pool cooling water in the analyses, the UFSAR and the Technical Specifications.

6.0 References

1) Letter, Ernest C. Hadley to James Taylor (NRC), *Request for Licensing Actions - 10 C.F.R. § 2.206 - Northeast Utilities - Millstone 1*, dated August 21, 1995

2) NRC letter, 8000130552, *Issuance of Approval to Amendment No. 39 to Millstone 1 Operating License No. DPR-21*, dated June 30, 1977

3) Northeast Nuclear Energy Company letter to NRC, *Modification to Spent Fuel Storage Pool / Millstone 1*, dated July 15, 1976

4) Report MP1-SDC, Revision 2, *Shutdown Cooling Heat Exchanges (M8-54A, B)*, dated January 1, 1995

5) MP1-SFPC, Revision 1, *Spent Fuel Pool Cooling heat Exchangers (M4-12A and B)*, dated August 1, 1993

6) Northeast Utilities letter to NRC, *Information on Proposed Spent Fuel pool Rack Modifications*, dated May 5, 1988

7) NRC letter, *Issuance of Amendment No. 40 to Facility Operating License No. DPR-21 for Millstone Nuclear Power Station, Unit 1*, dated November 27, 1989

8) Millstone Nuclear Power Station - 1, UFSAR, Section 9.1.3, February 1987

9) Millstone Nuclear Power Station - 1, UFSAR, Section 9.1.3, June 1994

10) Northeast Utilities System letter to NRC, *Proposed Technical Specification for Refueling & Spent Fuel Handling, and Additional Shutdown Cooling System Cross-Tie*, dated July 28, 1995

Northeast
Nuclear Energy

Rope Ferry Rd. (Route 156), Waterford, CT 06385

Millstone Nuclear Power Station
Northeast Nuclear Energy Company
P.O. Box 128
Waterford, CT 06385-0128
(203) 444-4300
Fax (203) 444-4277

The Northeast Utilities System

Donald B. Miller Jr.,
Senior Vice President – Millstone

Re: 10CFR50.73(a)(2)(ii)
December 14, 1995
MP-95-358

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Reference: Facility Operating License No. DPR-21
Docket No. 50-245
Licensee Event Report 95-009-02

This letter forwards update Licensee Event Report 95-009-02.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

FOR: Donald B. Miller, Jr.
Senior Vice President – Millstone Station

BY: Gary H. Bouchard
Station Services Director

DBM/KG:ljs

Attachment: LER 95-009-02

cc: T. T. Martin, Region I Administrator
P. D. Swetland, Senior Resident Inspector, Millstone Unit Nos. 1, 2, and 3
J. W. Andersen, NRC Project Manager, Millstone Unit No. 1

NRC Form 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Millstone Nuclear Power Station Unit 1	05000245	1 OF 3

TITLE (4) Incorrect Design Input to Spent Fuel Pool Cooling Piping Analysis

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)		
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR
05	24	95	95	– 009 –	02	12	14	95

OTHER FACILITIES INVOLVED (8)

FACILITY NAME	DOCKET NUMBER
	05000
	05000

OPERATING MODE (9): N

POWER LEVEL (10): 100

THIS REPORT IS BEING SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

	20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)		50.73(a)(2)(viii)
	20.2203(a)(1)		20.2203(a)(3)(i)	X	50.73(a)(2)(ii)		50.73(a)(2)(x)
	20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
	20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)		OTHER
	20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Drexel N. Harris, Nuclear Licensing	(203) 437-5903

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
A	DA	PSP	G080							

SUPPLEMENTAL REPORT EXPECTED (14)

				EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE)	X	NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On May 24, 1995, at 1605, with the plant at 100% power (530°F and 1030 psi), it was determined that portions of the spent fuel pool cooling piping, which currently have maximum operating temperature of 150°F, are actually designed for 85°F. The section of piping extends from the spent fuel pool cooling pumps to the inlet of the spent fuel pool heat exchangers, and from the outlet of the heat exchangers to the spent fuel pool. On May 25, 1995, an operability determination was made of the piping and although the allowable stresses of the original design criteria (USAS B31.1) were exceeded, it was determined that the piping meets the stress allowables of (GL) 91-18. There was also no evidence of piping or support distortion or damage to the portions of piping in question as a result of the system operations at temperatures above 85°F. It was therefore concluded that the piping met the operability limits of GL 91-18.

This event was discovered during analysis work in preparation for spent fuel pool cooling system upgrades and resulted from a failure to reconcile design temperature changes with design bases documents.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Millstone Nuclear Power Station Unit 1	05000245	9[illegible]	– 009 –	02	02 OF 03

TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

I. Description of Event

On May 24, 1995, at 1605, with the plant at 100% power (530° F and 1030 psi), it was determined that portions of the spent fuel pool cooling piping, which currently have maximum operating temperature of 150°F, are actually designed for 85°F. The section of piping extends from the spent fuel pool cooling pumps to the inlet of the spent fuel pool heat exchangers, and from the outlet of the heat exchangers to the spent fuel pool. On May 25, 1995, an operability determination was made of the piping and although the allowable stresses of the original B31.1) were exceeded, it was determined that the piping meets the stress allowables of (GL) 91–18. There was also no evidence of piping or support distortion or damage to the portions of piping in question as a result of the system operations at temperatures above 85°F. It was therefore concluded that the piping met the operability limits of GL 91–18.

This event was discovered during analysis work in preparation for spent fuel pool cooling system upgrades and resulted from a failure to reconcile design temperature changes with design bases documents.

II. Cause of Event

This event is attributable to the failure to reconcile design temperature changes with the design bases documents, including license amendments resulting in the use of an incorrect design input to an analysis of the portion of the spent fuel pool cooling system. The temperature discrepancy was discovered during the current analysis of the piping system that will provide upgrades to the spent fuel pool cooling system piping.

III. Analysis of Event

This event is reportable under 10CFR50.73 (a) (2) ii because portions of the spent fuel pool cooling piping were analyzed using an incorrect design temperature and this condition is outside of the current design bases requirement for the plant. The section of piping extends from the outlet of the spent fuel pool cooling pumps to the inlet of the spent fuel pool cooling HX's, then from the HX outlets to the spent fuel pool. This piping was analyzed to the requirements of USAS B31.1 for internal pressure, deadweight, and thermal expansion.

A portion of the system was evaluated using a maximum operating temperature of 85°F. This temperature was based on field verified actual operating temperature of the spent fuel pool. This appears to have been a judgment made at the time the analysis was performed in 1980. The design basis temperature requirement for the spent fuel pool was stated at various times to be a.) 125°F for a normal refueling offload of a quarter core (amendment 39 to the provisional operating license SER, 1977) b.) 140°F for a full core offload using shutdown cooling in conjunction with spent fuel pool cooling (Amendment 39 SER, 1977) c.) 137°F for normal discharge (Amendment 40 SER, 1989) d.) 140°F for abnormal discharge (Amendment 40 SER, 1989) e.) 150°F was identified as an acceptance criteria limit for the spent fuel pool concrete structure (Amendment 40 SER, 1989).

During the analysis of the piping system upgrades planned for the spent fuel pool cooling system and shutdown cooling system, the discrepancy in design temperature was discovered and an operability determination was initiated. The operability determination demonstrates that piping stresses remain below the material yield stresses and the piping meets the stress allowables of GL 91–18. A walkdown of the system found no evidence of piping or support damage.

Based on the review made and the calculations performed, it was concluded that the spent fuel pool cooling system line in question meets operability limits as set forth in GL 91–18.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Millstone Nuclear Power Station Unit 1	05000245	95	– 009 –	02	03 OF 03

TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

IV. Corrective Action

The following actions and their results were undertaken in response to this event.

1. Yankee Atomic Electric Co. performed a design basis criteria review of selected issues of the entire Spent Fuel Pool Cooling System (SFPCS). The review focused on thermal requirements and on seismic considerations identified in LER 95–006. This review included: (1) original design specifications and standards, (2) Standard Review Plan Guidance, (3) Systematic Evaluation Program requirements, (4) license amendments to provisional operating license and facility operating license and (5) corresponding NRC Safety Evaluation Reports. The review concluded that design basis temperature of the spent fuel pool of 150°F, which corresponds to the acceptance criterion limit identified in the USNRC's SER for Amendment No. 40 should be used as a input for analysis and design. Therefore, all lines that directly communicate with the fuel pool should be evaluated for a temperature of 150°F.

   The reanalysis of the SFPCS (including the Skimmer Surge Tanks) is complete and resulted in required pipe support modifications, the addition of a thermal expansion loop in a vertical 6" pipe and modifications to the Skimmer Surge Tank anchorage. These modifications were completed prior to RFO15.

   In addition to the SFPCS, all other lines that directly connect to the spent fuel pool and reactor cavity were evaluated prior to RFO15. No support modifications were required.

2. All short–term design modifications to restore affected portions of the SFPCS have been identified and were completed prior to RFO 15. These modifications are described in Item 1.

3. Permanent long–term system upgrades to be implemented in conjunction with the upcoming spent fuel pool rerack were assessed and none were identified as being needed. However, design basis reviews will continue throughout the preparation of the amendment for the upcoming spent fuel pool storage expansion.

4. The reanalysis of the SFPCS piping and Skimmer Surge tanks for a 150°F temperature, included where appropriate, the seismic analysis requirements associated with LER 95–006. The required modifications described in Item 1 address both thermal and seismic design basis requirements.

V. Additional Information

None

U.S. Nuclear Regulatory Commission
Operations Center

Event Reports For
01/05/96 - 01/11/96

POWER REACTOR	EVENT NUMBER: 29817

FACILITY: OCONEE	REGION: 2	NOTIFICATION DATE: 01/09/96
UNIT: [1] [2] [ ]	STATE: SC	NOTIFICATION TIME: 17:55 [ET]
RX TYPE: [1] B&W-L-LP,[2] B&W-L-LP,[3] B&W-L-LP		EVENT DATE: 01/09/96
		EVENT TIME: 17:00[EST]
NRC NOTIFIED BY: NOEL CLARKSON		LAST UPDATE DATE: 01/09/96
HQ OPS OFFICER: JOHN MacKINNON		

EMERGENCY CLASS: NOT APPLICABLE	NOTIFICATIONS
10 CFR SECTION:	JACK CRLENJAK RDO
AUNA 50.72(b)(1)(ii)(A) UNANALYZED COND OP	BILL BATEMAN EO
	ED JORDAN AEOD
	RICH BARRETT AEOD
	HENRY BAILEY AEOD

UNIT	SCRAM CODE	RX CRIT	INIT PWR	INIT RX MODE	CURR PWR	CURR RX MODE
1	N	Y	100	POWER OPERATION	100	POWER OPERATION
2	N	N	0		0	

EVENT TEXT

FUEL ASSEMBLY WAS DISCOVERED TO BE SUSPENDED AND UNATTENDED (SINCE 12/14/95) IN THE UNITS 1 & 2 SPENT FUEL POOL (SFP) FUEL HANDLING BRIDGE MAST.

At approximately 1100 on 01/08/96, a fuel assembly was discovered to be suspended and unattended in the Units 1 & 2 SFP fuel handling bridge mast. It was discovered when it was found that the grapple indication on the fuel handling bridge mast indicated that it was grappled to some object and also the load cell indicated the fuel handling bridge had a fuel assembly in it. The fuel assembly was suspended over its respective storage position in the SFP with the lower end fitting approximately flush with the top of the storage location (10' of water above the top of the fuel assembly). Preliminary investigation revealed that this fuel assembly may have been in the suspended position since 12/14/95 (last time the fuel handling bridge mast had been used). Immediate corrective action upon discovery was to lower the fuel assembly back into its SFP location. On 01/09/96 at 1700 hours it was determined that this condition constituted an unanalyzed condition. This event is being conservatively reported as an unanalyzed condition that may have significantly compromised plant safety in the unlikely event that the standby shutdown facility (SSF) was required. For this event to significantly compromise plant safety, an SSF event would have to occur concurrently with the fuel assembly being suspended in the SFP handling bridge mast above the SFP racks. Examples of events requiring SSF activation are fire, flood or sabotage. In these examples the SSF is required to be able to maintain the unit in a hot shutdown condition for 72 hours. During this 72 hour period, water is drawn from the SFP and injected into the reactor coolant system. The sfp inventory depletion during this operation causes sfp level to decrease. This level decrease would uncover the fuel assembly suspended in the SFP bridge mast within 72 hours. It is currently not known what the dose consequences of this fuel assembly uncovering would have been. The current Oconee Nuclear Station licensing basis does not consider a suspended fuel assembly during an event which would require use of the SSF. While this condition is known to be unanalyzed, an analysis is in progress and is required to determine the safety significance of this situation. A significant investigation team will be formed to discover how this event occurred.

Licensee said that a commitment to memory procedure is used when a fuel assembly is just raised/inspected/lowered back into it proper place (fuel assembly is not allowed to be moved from side to side while using a commitment to memory procedure). Licensee did not think that there are any sign offs in the procedure for making sure that the fuel assembly has been placed back in its proper storage space. Licensee also thought that the grapple indication and the load cell indication did not have to be looked at during the end of the procedure to insure that the fuel assembly had been removed from the fuel handling bridge mast.

U.S. Nuclear Regulatory Commission
Operations Center

Event Reports For
03/01/96 - 03/04/96

POWER REACTOR	EVENT NUMBER: 30050

FACILITY: MILLSTONE REGION: 1 UNIT: [1] [ ] [ ] STATE: CT RX TYPE: [1] GE-3,[2] CE,[3] W-4-LP	NOTIFICATION DATE: 03/01/96 NOTIFICATION TIME: 09:20 [ET] EVENT DATE: 03/01/96 EVENT TIME: 09:12[EST] LAST UPDATE DATE: 03/01/96
NRC NOTIFIED BY: D. HARRIS HQ OPS OFFICER: JOHN MacKINNON	NOTIFICATIONS
EMERGENCY CLASS: NOT APPLICABLE 10 CFR SECTION: ADAS 50.72(b)(2)(i) DEG/UNANALYZED COND	BILL RULAND RDO

UNIT	SCRAM CODE	RX CRIT	INIT PWR	INIT RX MODE	CURR PWR	CURR RX MODE
1	N	N	0	COLD SHUTDOWN	0	COLD SHUTDOWN

EVENT TEXT

UNANALYZED HEAVY LOAD PATH FOR MOVING THE GATES OUTSIDE THE SPENT FUEL POOL.

DURING A PRE-JOB BRIEF FOR MOVING THE SPENT FUEL POOL GATES OUT OF THE POOL, IT WAS DETERMINED THAT THE LOAD PATH REQUIRED TO MOVE THE GATES OUTSIDE THE POOL WOULD BE OUTSIDE THE REQUIREMENTS OF MILLSTONE UNIT 1 PROCEDURE MP 790.4, "CONTROL OF HEAVY LOADS." SPECIFICALLY, THE GATE COULD BE SUSPENDED OVER IRRADIATED FUEL AND THE CONSEQUENCES OF A GATE LOAD DROP HAS NOT BEEN ANALYZED.

THE PROCEDURE WAS DEVELOPED BACK IN THE EARLY 1980's IN ACCORDANCE WITH NUREG-0612 (CONTROL OF HEAVY LOADS). THE SPENT FUEL POOL WAS RERACKED IN 1988. THE RERACK MAY HAVE PUT THE RACKS CLOSE ENOUGH TO THE SPENT FUEL POOL WALLS SO THAT TOTAL COMPLIANCE WITH NUREG-0612 (CONTROL OF HEAVY LOADS) WITH THE CURRENT CONTROL OF HEAVY LOADS PROCEDURE (MP 790.4) MIGHT NOT BE POSSIBLE.

THE LICENSEE WILL REVISE THE PROCEDURE TO BE IN COMPLIANCE WITH NUREG-0612.

THE RESIDENT INSPECTOR WAS INFORMED BY THE LICENSEE OF THIS EVENT.

Headquarters Daily Report

MARCH 21, 1996

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	REPORT ATTACHED	NEGATIVE INPUT RECEIVED	NO INPUT RECEIVED
HEADQUARTERS		√	
REGION I		√	
REGION II		√	
REGION III		√	
REGION IV	√		

PRIORITY ATTENTION REQUIRED MORNING REPORT - REGION IV MARCH 21, 1996

Licensee/Facility:

Entergy Operations, Inc.
Arkansas Nuclear 2
Russelville,Arkansas
Dockets: 50-368
PWR/CE

Notification:

MR Number: 4-96-0028
Date: 03/21/96
Resident Inspector

Subject: UNEXPECTED DROP IN SPENT FUEL POOL LEVEL

Discussion:

At approximately 1:05 p.m. on March 20, 1996, Unit 2 operators received a low spent fuel pool level (SFP) alarm in the control room. Earlier, operators had aligned the SFP purification system from the refueling water tank to the SFP to support replacement of an SFP purification system filter. Operators dispatched to investigate found that the SFP filter drain valve was not fully closed by two turns, although the valve position indicator indicated closed. In addition, operators noticed abnormal levels in the radwaste boron management system tank. Upon fully closing the valve the draining ceased. A siphon breaker is installed in the line such that the SFP would not drain below 13 inches above the Technical Specification limit. Approximately 900 gallons drained from the SFP to the radwaste boron management system tank, resulting in a drop in the SFP level of 1 1/2 inches. SFP level remained above Technical Specification limits.

The licensee is reviewing this event.

Regional Action:

The resident inspectors will follow up on this issue.

Contact: S. J. Campbell (501)968-3290
R. L. Nease (817)860-8154

Northeast
Utilities System

Millstone Offices • Rope Ferry Rd., Waterford, CT

P.O. Box 128
Waterford, CT 06385-0128
(203) 447-1791

S

JUL 25 1996

Docket No. 50-245
B15728

Re: 10CFR50.73(a)(2)(ii)

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

This letter forwards supplemental Licensee Event Report (LER) 93-011-02, documenting an event that occurred at Millstone Nuclear Power Station, Unit No. 1 on September 17, 1993. This LER is submitted pursuant to 10CFR50.73(a)(2)(ii).

This supplemental LER corrects information contained in Table 1, which identifies the time between reactor shutdown (all control rods inserted) and the commencement of core offload. A review of the control room logs for refueling outage 7 showed that the date the core offload commenced was October 9, 1980, and not October 10. Hence, the calculated time difference was in error by 24 hours.

Additionally, this supplement identifies the weakness of the 10CFR50.59 safety evaluations for the core reloads between cycle 7 and cycle 11, in that they failed to consider the impact of changing the reload batch size from one-quarter of a core to one-third of a core.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

W. J. Riffer

W. J. Riffer
Director - Millstone Unit No. 1

Attachment: LER 93-011-02

cc: T. T. Martin, Region I Administrator
T. A. Easlick, Senior Resident Inspector, Millstone Unit No. 1
J. W. Andersen, NRC Project Manager, Millstone Unit No. 1

NRC FORM 366 (4-95)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 04/30/98

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Millstone Nuclear Power Station Unit 1	05000245	1 of 6

TITLE (4)
Spent Fuel Pool Cooling Capability

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
09	17	93	93	011	02				FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	N	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)			
		[ ] 20.2201(b)	[ ] 20.2203(a)(2)(v)	[ ] 50.73(a)(2)(i)	[ ] 50.73(a)(2)(viii)
POWER LEVEL (10)	100	[ ] 20.2203(a)(1)	[ ] 20.2203(a)(3)(i)	[X] 50.73(a)(2)(ii)	[ ] 50.73(a)(2)(x)
		[ ] 20.2203(a)(2)(i)	[ ] 20.2203(a)(3)(ii)	[ ] 50.73(a)(2)(iii)	[ ] 73.71
		[ ] 20.2203(a)(2)(ii)	[ ] 20.2203(a)(4)	[ ] 50.73(a)(2)(iv)	[ ] OTHER
		[ ] 20.2203(a)(2)(iii)	[ ] 50.36(c)(1)	[ ] 50.73(a)(2)(v)	Specify in Abstract below or in NRC Form 366A
		[ ] 20.2203(a)(2)(iv)	[ ] 50.36(c)(2)	[ ] 50.73(a)(2)(vii)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Robert W. Walpole, Nuclear Licensing Supervisor	(860)440-2191

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION	MONTH	DAY	YEAR
[ ] YES (If yes, complete EXPECTED SUBMISSION DATE).	[X] NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On September 17, 1993, with the plant at 100 percent power (1030 psig, 530 degrees Fahrenheit), it was determined through engineering analysis that during prior refueling outages the spent fuel pool cooling system, by itself, would have been incapable of maintaining pool temperature below the 150 degree Fahrenheit design limit, under certain conditions. The conditions in question involve the transfer of a full reactor core into the spent fuel pool. In an analysis assuming a full core offload beginning 150 hours after reactor shutdown, and assuming maximum ultimate heat sink temperature (75 degrees Fahrenheit) and a single active equipment failure of the "A" train of Shutdown Cooling System, with no compensatory actions to restore adequate cooling capability, it was determined that the spent fuel pool temperature would exceed the acceptance limit.

NNECO's typical practice during refueling outages has been to perform full core offloads. Additionally, a review of data from previous refueling outages revealed that on nine occasions, a reactor core offload commenced sooner than 150 hours after reactor shutdown.

Refinement of supporting analyses was completed and appropriate schedular and procedural controls were implemented during RFO14.

There were no safety consequences as a result of this event and no safety systems were required to operate as a result of this event.

I. Description of Event

On September 17, 1993, with the plant at 100 percent power (1030 psig, 530 degrees Fahrenheit), it was determined through engineering analysis that during prior refueling outages the spent fuel pool cooling system, by itself, would have been incapable of maintaining pool temperature below the 150 degree Fahrenheit design limit, under certain conditions.

NNECO's refueling practice at Millstone Unit No. 1 is to offload the full core to the spent fuel pool. This practice is not consistent with the "normal" refueling analyzed in the Updated Final Safety Analysis Report (UFSAR). To assess this past condition, NNECO performed an analysis that assumes the transfer of a full reactor core into the spent fuel pool beginning 150 hours after reactor shutdown, at a rate of 10 fuel assemblies per hour. Following completion of fuel transfer, the shutdown cooling system, which is cross-connected to the spent fuel pool cooling system to provide adequate cooling capability, is assumed to fail. Ultimate heat sink temperature of 75 degrees Fahrenheit is assumed consistent with the maximum allowable per Technical Specifications. Results indicated that, under these conditions, the spent fuel pool temperature would exceed 150 degrees Fahrenheit. These results are not consistent with the Millstone 1 UFSAR, and the Safety Evaluation Report for License Amendment 40, which provided for expanded spent fuel pool storage capability in 1988. Based on the above, this event was reported on September 17, 1993, per the requirements of 10CFR50.72, as a condition that was outside the design basis of the plant.

A review of data from previous refueling outages revealed that on nine occasions, a reactor core offload commenced sooner than 150 hours after reactor shutdown.

A further review of the Spent Fuel Pool design history determined that no 10CFR50.59 safety evaluation had been performed on the impact of exceeding the "normal" discharge batch size from one-quarter core to one-third core for the 1980 refueling outage (end of cycle (EOC) 7), until the Spent Fuel Pool rerack project was implemented in 1988 at the end of cycle 11 (NRC Safety Evaluation Report for License Amendment 40). The increased heat load in the Spent Fuel Pool was not evaluated and potentially could have resulted in exceeding the Spent Fuel Pool Cooling system design heat removal rating for normal offloads, as evaluated in the Safety Evaluation Report for License Amendment 39. This increase is bounded by the analysis of the full core offload described above.

There were no safety consequences as a result of this event.

II. Cause of Event

The "normal" refueling sequence described in the Millstone 1 UFSAR assumes discharge of only one third of the core into the spent fuel pool. In practice, Millstone 1 typically performs a full core offload, which, by the UFSAR definition, is considered to be an "emergency" or "abnormal" refueling sequence. The 1988 rerack analysis assumed a single failure for the "normal" refueling event. Additionally, USNRC Standard Review Plan (NUREG-0800) provides analysis criteria for the spent fuel pool rerack project which specifies that a single failure be assumed for the "normal" sequence, but not for the "abnormal" sequence. This SRP was used as guidance in developing the 1988 rerack analysis. Applying this "normal" criterion to the full core offload scenario, with no manual compensatory measures being performed, results in exceeding the design criterion for the spent fuel pool.

TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

This condition represents a discrepancy in ensuring that operations were conducted in accordance with assumptions in the spent fuel pool cooling capability engineering analysis. This is attributable to two factors. First, engineering analysis assumptions were not incorporated into operating procedures or controls, allowing a discrepancy to develop. Second, unit management was insufficiently aware of design basis analyses, and thus did not insure refueling practice conformed to analysis assumptions.

A further contributing factor is the depth of review of the analysis performed in support of the 1988 and 1976/77 spent fuel pool rerack projects, including the review done by the Plant Operations Review Committee (PORC). Neither PORC nor any other review organization identified the discrepancy between actual practice and analysis assumptions.

An additional contributor to this event has been determined to be the inadequacy of the 10CFR50.59 safety evaluation for core reloads between cycle 7 and cycle 11. The change from one-quarter to one-third core was evaluated for reactor impacts, but the evaluation failed to consider the impact of changing the reload batch size on the Spent Fuel Pool.

III. Analysis of Event

This event is being reported per the requirements of 10CFR50.73(a)(2)(ii)(B), as a condition that is outside the design basis of the plant.

A Millstone 1 specific analytical model was developed and analysis was initiated to evaluate the identified discrepancy between typical refueling practice and the licensing analysis. NNECO sought to determine the peak spent fuel pool temperatures that could occur during the event in question. The analysis assumed the following conditions:

1. Full core offload begins 150 hours following plant shutdown.

2. Fuel is transferred into the spent fuel pool at a rate of 10 assemblies per hour.

3. The cooling water supply to the spent fuel pool cooling heat exchangers is 85 degrees Fahrenheit.

4. Concurrent with completion of core offload, the shutdown cooling system, which is manually cross-connected to augment spent fuel pool cooling system capability during refueling evolutions, experiences a system failure.

5. No mitigating compensatory actions are taken.

Results of this analysis indicate that the resulting spent fuel pool bulk temperature would be approximately 212 degrees Fahrenheit. If consideration is given for evaporative cooling of the pool, the resulting temperature would be 186 degrees Fahrenheit. Both of these temperature analyses exceed the 150 degree Fahrenheit design criterion of the spent fuel pool for a normal refueling.

Historically, as shown in Table 1, full core offloads have occurred as regular practice at Millstone 1 and, in many cases, full core offloads have commenced before 150 hours of decay time.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Millstone Nuclear Power Station Unit 1	05000245	93	-- 011 --	02	4 of 6

TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

TABLE 1
Millstone Unit 1
Cycle 1 Through 15 Core Offload Times

RFO	SHUTDOWN (ALL RODS IN)		TYPE OF RELOAD	COMMENCE OFFLOAD			OFFLOAD COMPLETE		
	TIME	DATE		TIME	DATE	HOURS AFTER SHUTDOWN	TIME	DATE	HOURS AFTER SHUTDOWN
1	0750	9/1/72	Offload	1520	9/21/72	487 hrs 30 min	0338	10/4/72	786 hrs 48 min
2	0340	8/31/74	Offload	0050	9/3/74	69 hrs 10 min	0814	9/8/74	196 hrs 34 min
3	0444	9/12/75	Offload	1510	9/17/75	130 hrs 26 min	0100	9/24/75	284 hrs 16 min
4	0105	10/2/76	Offload	1535	10/5/76	86 hrs 30 min	0210	10/11/76	217 hrs 5 min
5	2301	3/10/78	Shuffle	1935	3/15/78	116 hrs 34 min	0630	3/20/78*	223 hrs 29 min
6	0202	4/28/79	Shuffle	2342	5/4/79	189 hrs 40 min	0340	5/19/79	505 hrs 38 min
7	0800	10/4/80	Offload	2353	10/9/80	135 hrs 53 min	2325	10/14/80	255 hrs 25 min
8	1017	9/11/82	Offload	2140	9/15/82	107 hrs 23 min	0338	9/23/82	281 hrs 21 min
9	0857	4/14/84	Offload	0325	4/17/84	66 hrs 28 min	1945	4/21/84	178 hrs 48 min
10	0610	10/26/85	Offload	1919	10/30/85	109 hrs 9 min	0857	11/5/85	242 hrs 47 min
11	0442	6/6/87	Offload	1935	6/9/87	86 hrs 53 min	0529	6/14/87	192 hrs 47 min
12	1238	4/7/89	Offload	1150	4/15/89	191 hrs 12 min	1811	4/20/89	317 hrs 33 min
13	0723	4/7/91	Offload	0147	4/14/91	162 hrs 24 min	1236	4/17/91	245 hrs 13 min
14	2159	1/15/94	Offload	1132	1/28/94	301 hrs 33 min	1028	2/4/94	468 hrs 29 min
15	1432	11/4/95	Offload	1457	11/12/95	192 hrs 25 min	0450	11/19/95	350 hrs 18 min

*Phase 1 of core shuffle

To assess the significance of this matter, NNECO has performed a retrospective assessment of decay heat loads in the spent fuel pool for each refueling outage at the time of the end of fuel movement. The actual heat loads are shown in Table 2. These heat loads are representative of the maximum heat loads in the spent fuel pool. Table 2 also includes a comparison, for each refueling outage, of these maximum heat loads to the available cooling capability. For most refueling outages, the actual heat loads exceeded the design capability of the spent fuel pool cooling system heat exchangers (7.84E +06 BTU/Hr) and the assumed design limit for the "normal" case as applicable at the time. When crediting one train of the shutdown cooling system (which in practice was aligned to the spent fuel pool as needed), total available cooling design capability (29.84E +06 BTU/Hr) was never exceeded. Likewise, the heat load assumed for the "abnormal" case was not exceeded.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)					PAGE (3)
		YEAR		SEQUENTIAL NUMBER		REVISION NUMBER	
Millstone Nuclear Power Station Unit 1	05000245	93	--	011	--	02	5 of 6

TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

This assessment highlights the single failure vulnerability of the spent fuel pool cooling capability for full core offloads that existed prior to refueling outage 14. However, in practice the temperature of the spent fuel pool never exceeded the design acceptance criterion.

NUSCo reviewed the heat load for each of the impacted discharge batches (those greater than one-quarter core), which occurred between 1979 and 1988. The conclusion was that the increased discharge batch size heat load never exceeded the Spent Fuel Pool heat load rating for the normal discharge case.

TABLE 2
Millstone Unit 1
Maximum Decay Heat Loads for Cycle 1 Through 15

RFO	Actual Heat Load (BTU/Hr)	% SFP + SDC Design Capacity	Normal SER Heat Load		Abnormal SER Heat Load	
			BTU/Hr	% Limit	BTU/Hr	% Limit
1	6.28+06	21%	7.84E+06	80%	2.46E+07	26%
2	1.23E+07	41%	7.84E+06	157%	2.46E+07	50%
3	9.64E+06	32%	7.84E+06	123%	2.46E+07	39%
4	1.11E+07	37%	7.21E+06	155%	1.79E+07	62%
5	2.63E+06	9%	7.21E+06	36%	1.79E+07	15%
6	2.28E+06	8%	7.21E+06	32%	1.79E+07	13%
7	1.13E+07	38%	7.21E+06	157%	1.79E+07	63%
8	1.05E+07	35%	7.21E+06	146%	1.79E+07	59%
9	1.35E+07	45%	7.21E+06	188%	1.79E+07	76%
10	1.17E+07	39%	7.21E+06	163%	1.79E+07	66%
11	1.30E+07	44%	7.43E+06	175%	1.79E+07	66%
12	1.09E+07	36%	1.01E+07	146%	1.96E+07	49%
13	1.22E+07	41%	1.01E+07	120%	1.96E+07	55%
14	9.08E+06	30%	1.01E+07	90%	2.20+07	41%
15	1.00E+07	34%	2.20+07	46%	2.20+07	46%

IV. Corrective Action

The following actions were performed in response to this event for RFO14:

1. Cycle-specific procedural and schedular controls were put in place during RFO14 in January, 1994. These controls provided guidance for core decay time and maximum cooling water temperatures, to ensure that the maximum spent fuel pool bulk temperature would remain less than approved limits following the limiting single failure event during the full core offload.

2. The Millstone 1 UFSAR was revised to reflect the information available as of RFO14.

The following actions were performed prior to RFO15:

1. A plant design change was implemented to provide a second train of shutdown cooling to supplement fuel pool cooling, eliminating the single failure vulnerability.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)					PAGE (3)
		YEAR		SEQUENTIAL NUMBER		REVISION NUMBER	
Millstone Nuclear Power Station Unit 1	05000245	93	--	011	--	02	6 of 6

TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

2. A License Amendment was applied for and granted to specifically authorize a revision to the UFSAR description to allow a partial or full core offload during refueling as a normal end-of-cycle event.

3. Pursuant to the license amendment, operating procedures have been implemented to incorporate appropriate operational limitations regarding full core offloads to assure consistency with analysis assumptions.

4. The operating requirements of the spent fuel pool have been addressed, and they are located in the Millstone Unit No. 1 Technical Requirements Manual.

The following actions were performed during RFO15:

1. The effectiveness of PORC was assessed and addressed in NNECO's response to NRC Inspection Report No. 50-245/95-34.

Additional corrective actions taken include improvements to procedures and training, in order to provide better understanding of what constitutes a plant design change, and what facets of those changes require a safety evaluation pursuant to 10CFR50.59.

V. Additional Information

Commitments

There are no commitments contained within this letter. All corrective actions have been completed.

ASSESSMENT OF SPENT FUEL COOLING

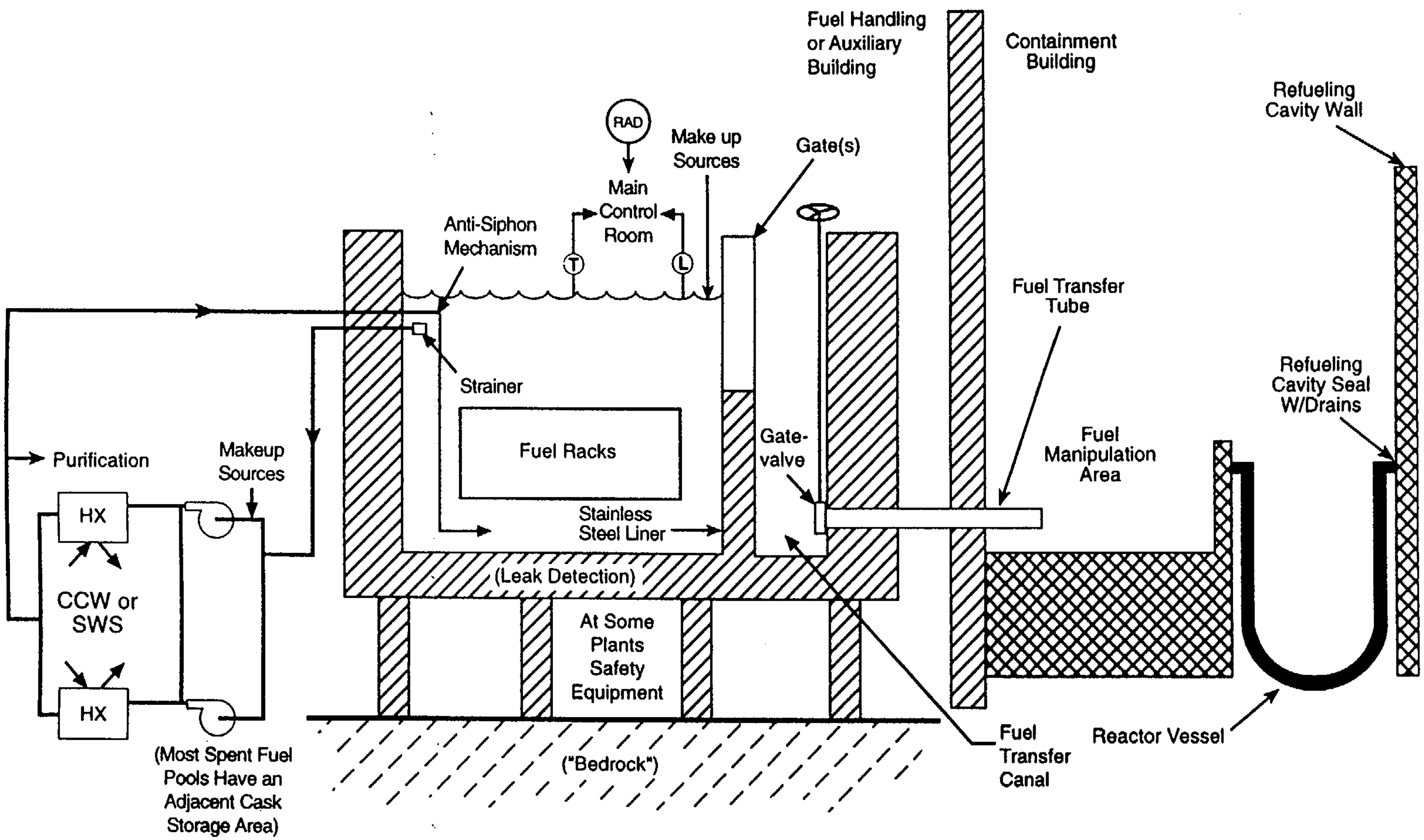
Jose G. Ibarra
Reactor Analysis Branch
Safety Programs Division
Office for Analysis and Evaluation of Operational Data

November 14, 1996

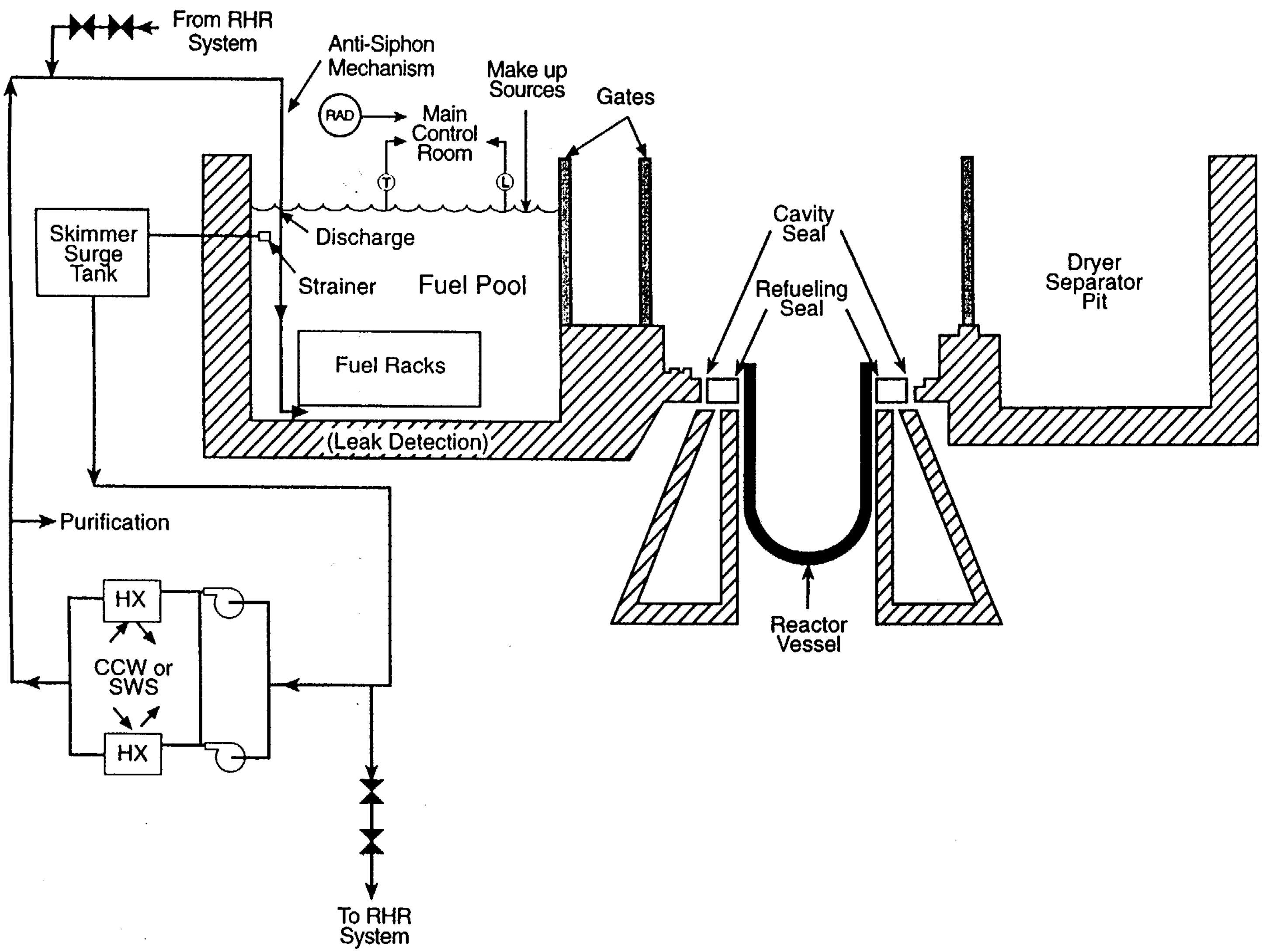
SPENT FUEL COOLING ASSESSMENT

- AEOD study requested by Executive Director for Operations.
- Developed generic configurations to assess loss of spent fuel pool cooling and inventory.
- Assessed 12 years of operational experience.
- Performed site visits to gather information on physical configuration, practices, and procedures.
- Reviewed regulations, standard review plan and regulatory guides.
- Performed assessments of electrical systems, instrumentation, heat loads, and radiation.
- Evaluated risk of losing spent fuel cooling.

PWR SPENT FUEL COOLING SYSTEMS



BWR SPENT FUEL COOLING SYSTEMS



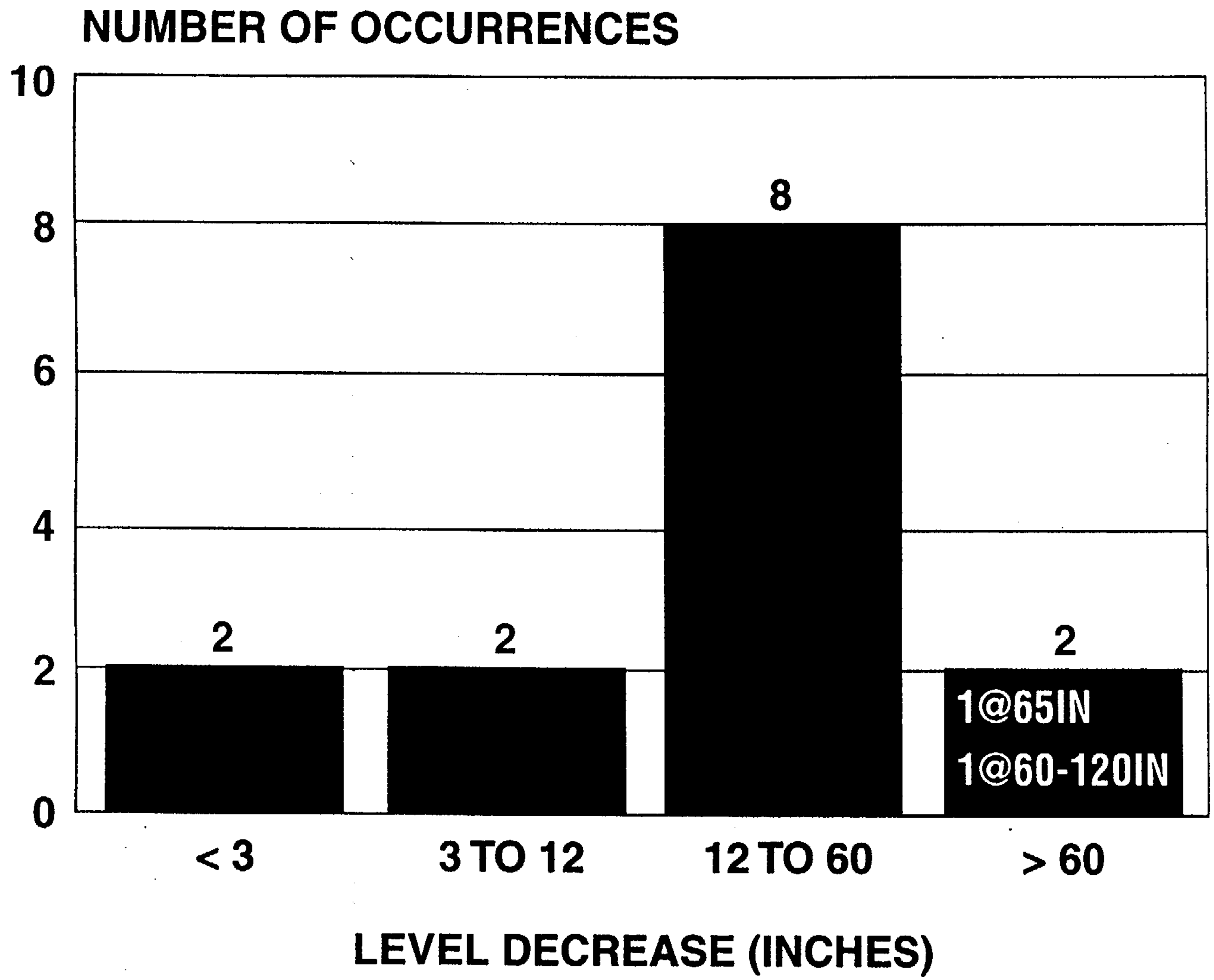
SPENT FUEL POOL EVENTS

TYPE EVENT	ACTUAL	PRECURSOR
<u>SFP Inventory</u>	<u>38</u>	<u>55</u>
Connected Systems	20	12
Gates & Seals	10	8
Structure or Liner	8	35
<u>SFP Cooling</u>	<u>56</u>	<u>22</u>
Cooling Flow	50	20
Heat Sink	6	2

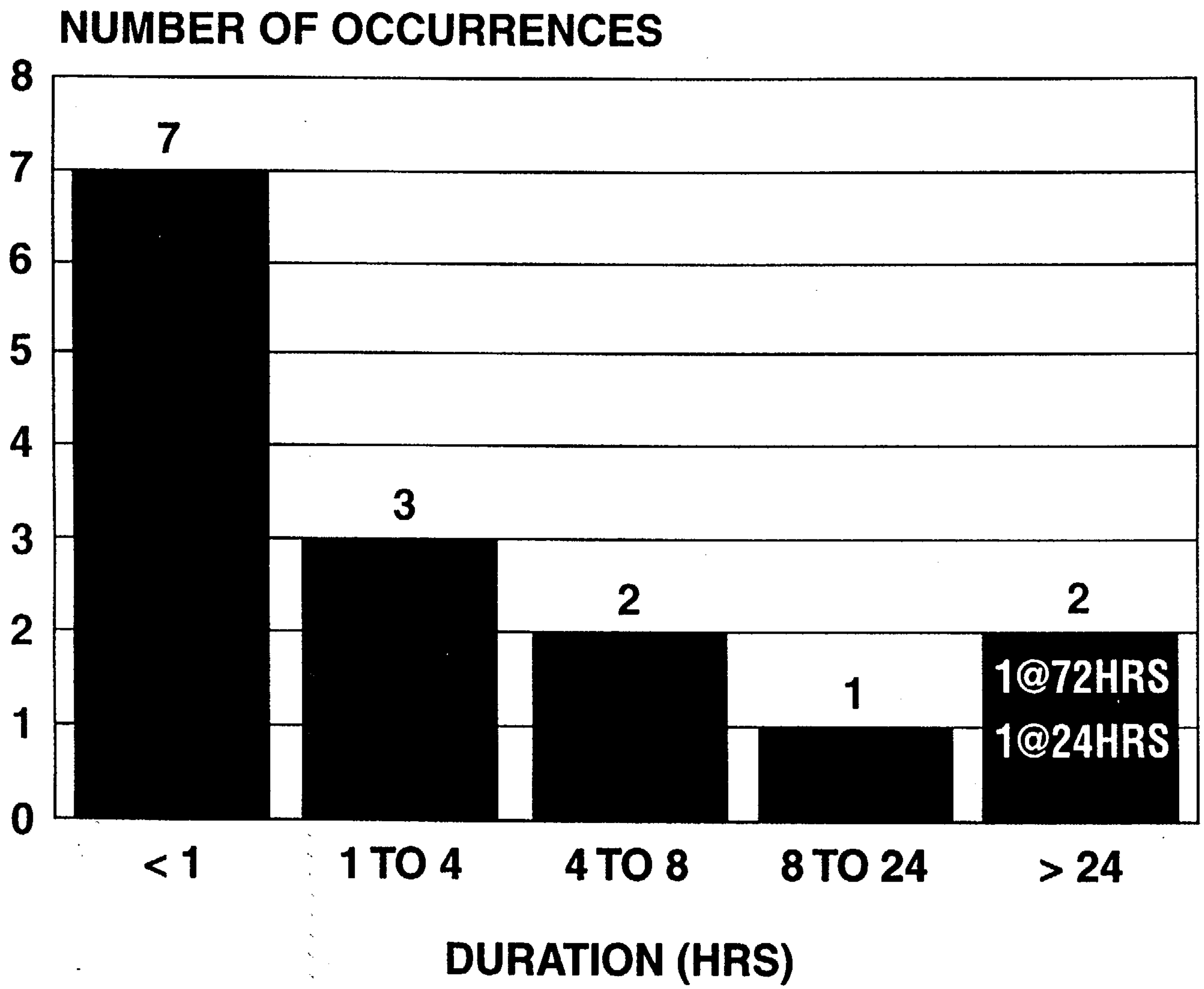
LOSS OF COOLANT INVENTORY EVENTS

TYPE EVENT	ACTUAL	PRECURSOR
Connected Systems	20	12
Configuration Control	16	2
Siphoning	3	1
PWR Transfer Tube	1	1
Piping	0	1
Piping Seismic Design	1	1
Gates & Seals	10	8
Cavity Seals	0	6
Gate Seals	10	2
Pool Structure or Liner	8	35
Liner Leaks	7	1
Load Drops	1	32
Pool Seismic Design	0	2

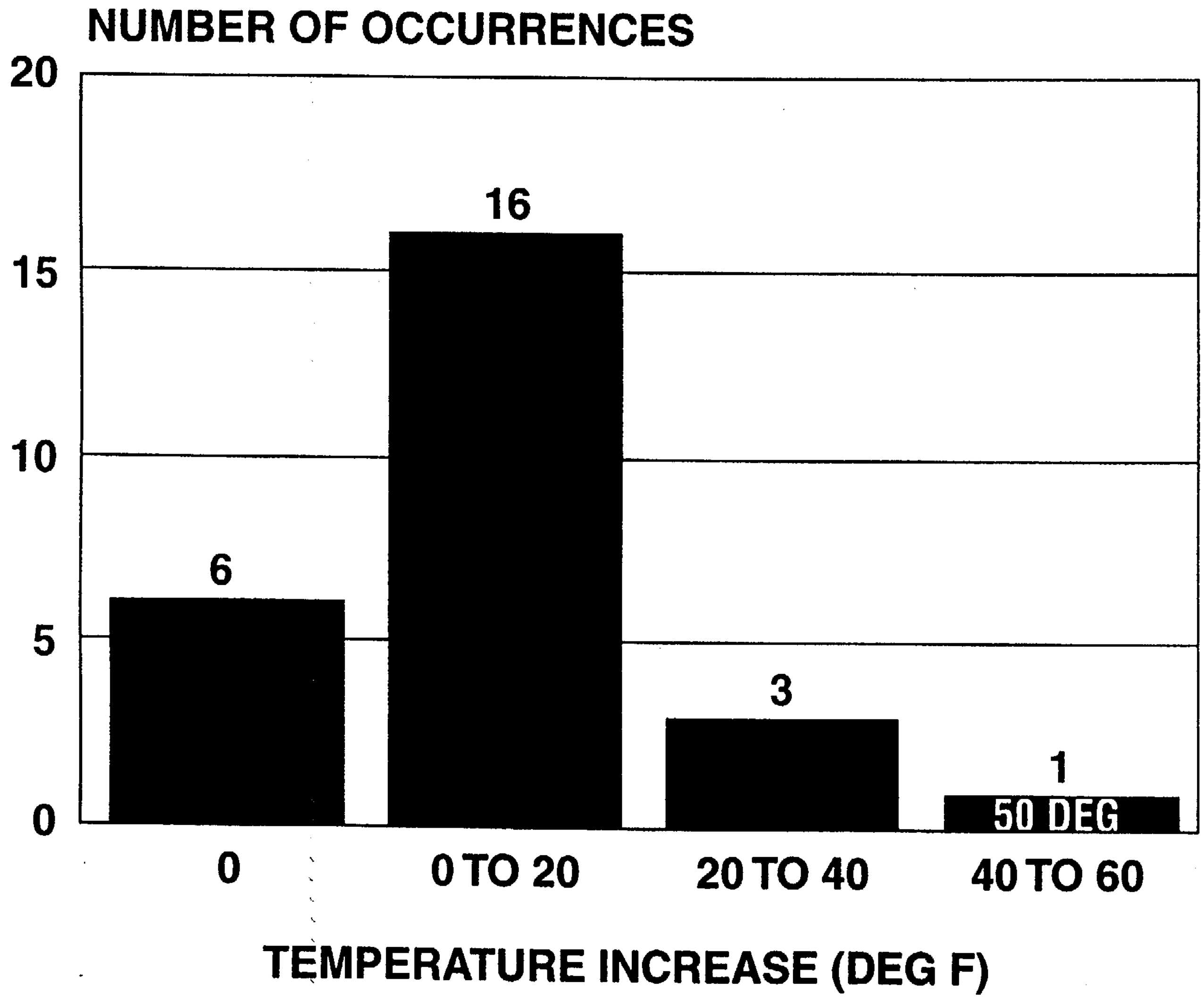
LOSS OF INVENTORY LEVELS



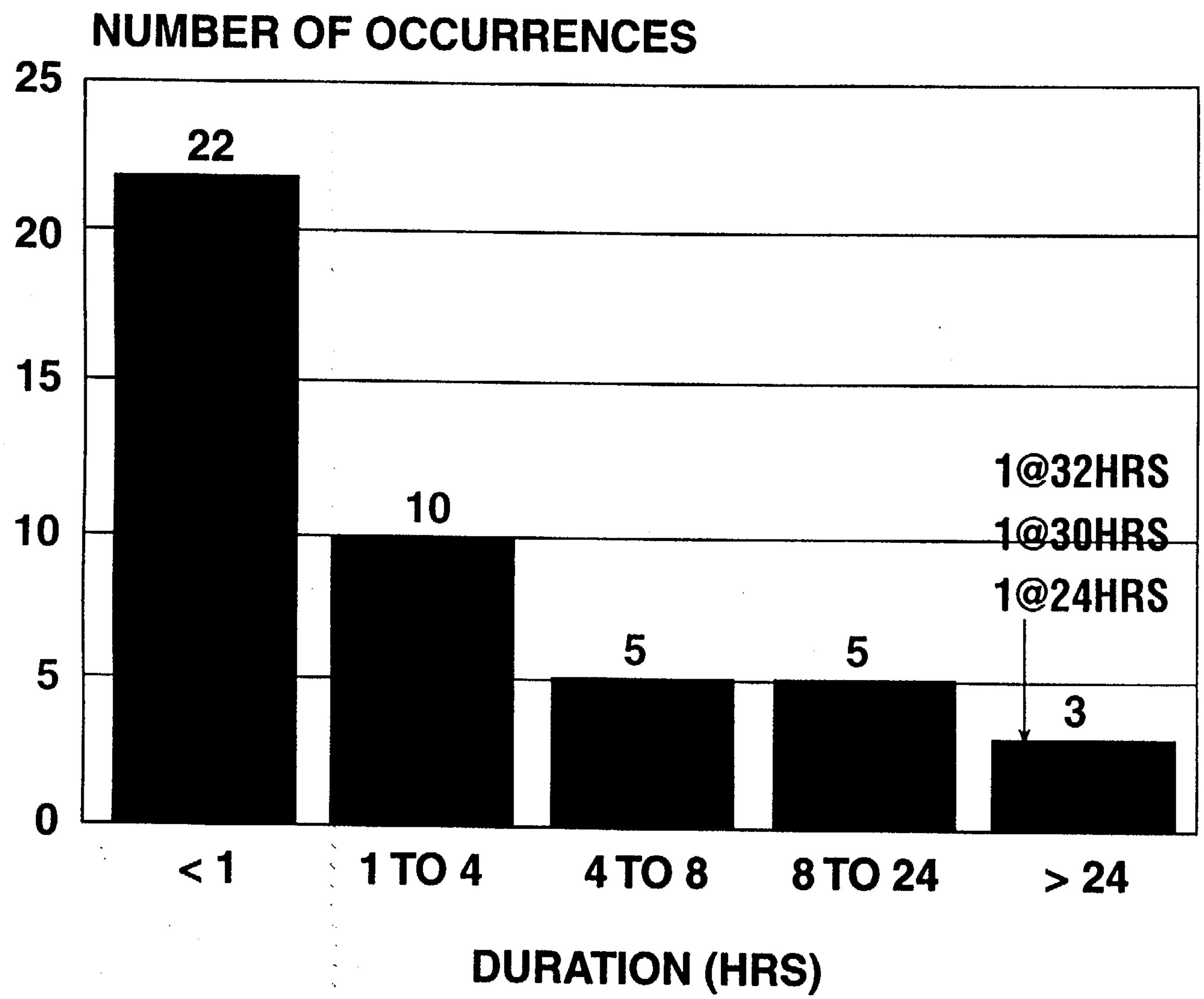
LOSS OF INVENTORY EVENTS



LOSS OF COOLING EVENTS



LOSS OF COOLING EVENTS



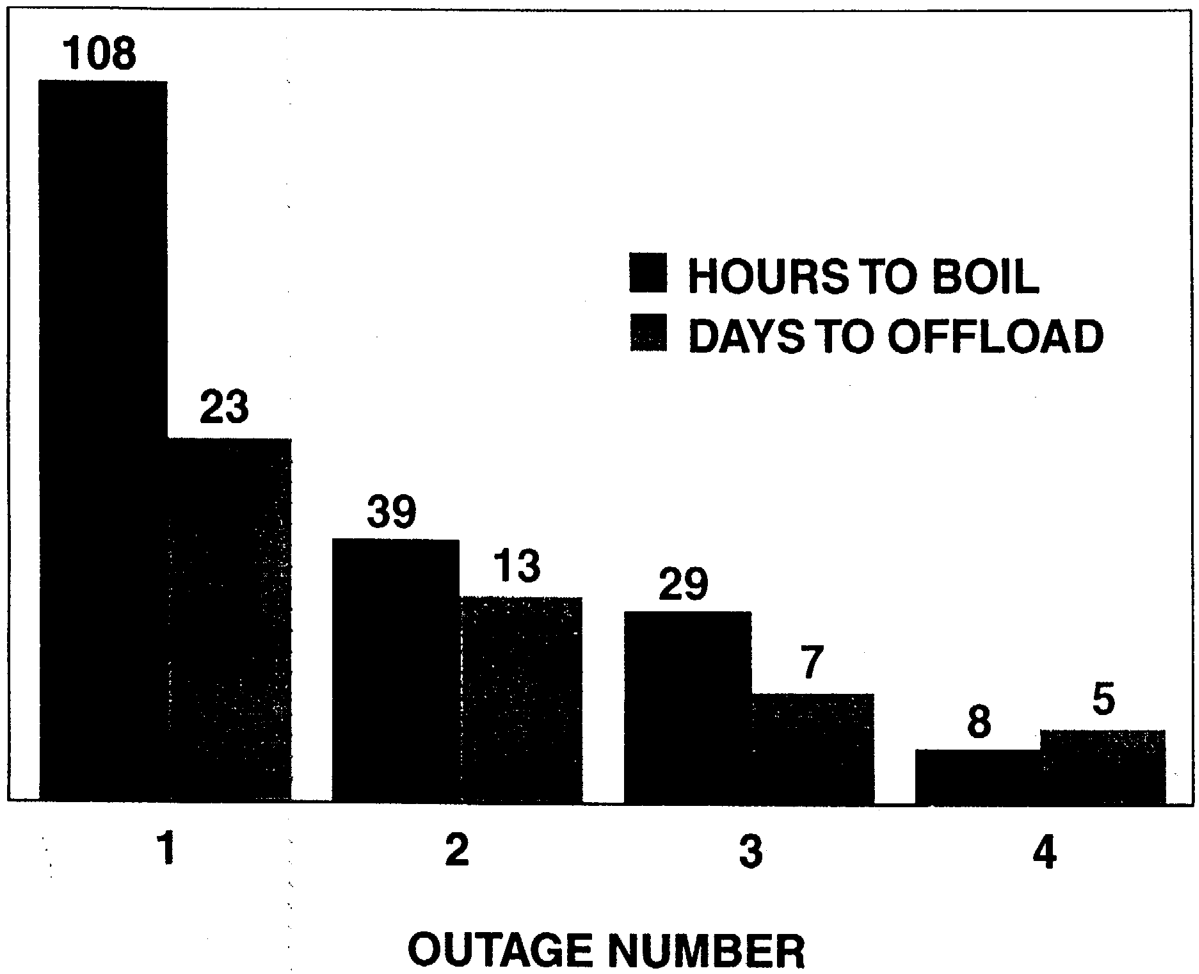
PARTIAL LISTING OF GOOD PRACTICES OBSERVED DURING PLANT VISITS

- Utilization of system diagram prior to all alignment changes
- Including SFP risk during outage planning.
- Classroom and simulator training to prepare for outage.
- User friendly graphs of pool heatup.
- Effective program for feedback of internal and industry operating experience.
- Detailed review at some plants found significant inventory loss vulnerabilities.

REGULATION REVIEW AND ENGINEERING ASSESSMENTS

- Identified applicable guidance and regulations.
- Surveyed 14 plants to determine power supply.
- Surveyed 14 plants to determine instrumentation.
- Assessed radiation levels with varying water levels.
- Performed heat load calculations.

REDUCED TIME TO BOIL AT NINE MILE POINT UNIT 2



NEAR-BOILING FREQUENCIES

	CURRENT INEL WORK	PREVIOUS PNL WORK
Total Near-Boiling Frequencies	5 E-5	2 E-5
LOOP	3 E-5	1 E-5
Inventory Losses	2 E-5	1 E-6

SUSQUEHANNA SPENT FUEL POOL RISK ASSESSMENT

- Showed benefits from:
  - improved instrumentation
  - improved procedures
  - improved training
- Showed vulnerability of operating unit from defueled unit

FINDINGS AND CONCLUSIONS

Likelihood and Consequences

- Consequences of actual events have not been severe.
- Primary cause of events has been human error.
- Relative risk of fuel damage is low compared with other reactor events.
- Highly dependent on human performance and plant design.
- Frequency of coolant loss > 1 foot, 1/100 reactor years.
- Frequency of cooling loss > 20 °F, 2-3/1000 reactor years.

FINDINGS AND CONCLUSIONS (CONT.)

Prevention

- Configuration control improvements can prevent and/or mitigate SFP events.
- Evaluations may be needed at some multiunit sites for potential SFP boiling effects on safe shutdown.

Response

- Attention to time to boil with shorter outages.
- Improved procedures and training may be needed.
- Improvements to instrumentation and power supplies may be needed.

FOLLOW UP

- NRC Information Notice being prepared.
- Study made into a NUREG.
- Report being submitted to Incident Reporting System.
- Working with NRR on implementing recommendations.

PRELIMINARY NOTIFICATION OF EVENT OR UNUSUAL OCCURRENCE PNO-I-97-039

This preliminary notification constitutes EARLY notice of events of POSSIBLE safety or public interest significance. The information is as initially received without verification or evaluation, and is basically all that is known by Region I staff (King of Prussia, Pennsylvania)n on this date.

Facility
Northeast Utilities
Millstone 3
Waterford, Connecticut
Dockets: 50-423

Licensee Emergency Classification
Notification of Unusual Event
Alert
Site Area Emergency
General Emergency
X Not Applicable

Subject: SPENT FUEL POOL HEATUP

At about 8:00 a.m., on June 26, 1997, a licensed operator noticed that the spent fuel pool temperature had increased about 2.5 degrees F since the last reading taken eight hours earlier. A system walkdown revealed that the reactor plant component cooling water (RPCCW) system was lined up to the wrong spent fuel pool heat exchanger. Spent fuel pool water was flowing through the other heat exchanger that was not being cooled by RPCCW. Operators realigned the spent fuel pool system to restore spent fuel pool cooling. The system misalignment and subsequent spent fuel pool temperature increase was not noticed for approximately 28 hours.

Based on subsequent computer temperature data, the spent fuel pool temperature increased from 87 degrees F at 5:30 a.m., on June 25, 1997, to a maximum of nearly 98 degrees F at 9:30 a.m. June 26, 1997. The spent fuel pool temperature remained well below the alarm setpoint of 125 degrees F.

The licensee has assembled an event review team to investigate this occurrence. The QA organization has formed an independent review team to investigate this event as well.

The licensee informed the State of Connecticut of this event. The licensee is planning to issue a press release.

This information has been discussed with the licensee and is current ~~as~~ of 1:00 p.m. on June 27, 1997.

Contact: ARRIGHI, RUSSELL (860)447-3170 LANNING, WAYNE (610)337-5126

June 4, 1998

EA 98-308

Mr. Harold W. Keiser
Executive Vice President
Nuclear Business Unit
Public Service Electric and Gas Company
Post Office Box 236
Hancocks Bridge, NJ 08038

SUBJECT: NRC INTEGRATED INSPECTION REPORT 50-354/98-05
NOTICE OF VIOLATION

Dear Mr. Keiser:

On May 16, 1998, the NRC completed an inspection at your Hope Creek reactor facility. The enclosed report presents the results of that inspection.

Throughout this six week inspection period, your conduct of activities at the Hope Creek facility was generally characterized by safety-conscious operations, sound engineering and maintenance practices, and careful radiological work controls.

During the inspection, four examples of violations of NRC requirements were identified. The violations are cited in the enclosed the enclosed Notice of Violation (Notice) and the circumstances surrounding them are described in detail in the subject inspection report. In the first instance, NRC inspectors identified that the residual heat removal system was not maintained available during a 1990 refueling outage while the reactor core was fully off-loaded for the purpose of augmenting the fuel pool cooling and cleanup system, which was contrary to design basis assumptions.

The NRC identified three additional examples of violations related to a modification that was installed during the Fall 1997 refueling outage and was associated with ventilation system safety related chillers. Specifically, check valves and gas bottle regulators associated with a backup air supply for a pressure control valve were not properly tested and maintained. Further, a design basis assumption inconsistency related to the minimum acceptable inlet temperature for the chillers was recognized by engineering personnel in December 1997, but was not acted upon and corrected until this modification deficiency surfaced.

You are required to respond to this letter and should follow the instructions specified in the enclosed Notice when preparing your response. The NRC will use your response, in part, to determine whether further enforcement action is necessary to ensure compliance with regulatory requirements.

Enclosure 2

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures will be placed in the NRC Public Document Room (PDR).

Sincerely,

Original Signed By:

James C. Linville, Chief
Projects Branch 3
Division of Reactor Projects

Docket No.: 50-354
License No: NPF-57

Enclosures: Notice of Violation
Inspection Report 50-354/98-05

cc w/encl:
L. Storz, Senior Vice President - Nuclear Operations
E. Simpson, Senior Vice President - Nuclear Engineering
E. Salowitz, Director - Nuclear Business Support
A. F. Kirby, III,, External Operations - Nuclear, Delmarva Power & Light Co.
J. A. Isabella, Manager, Joint Generation
Atlantic Electric
M. Bezilla, General Manager - Hope Creek Operations
J. McMahon, Director - QA/Nuclear Training/Emergency Preparedness
D. Powell, Director - Licensing/Regulation & Fuels
R. Kankus, Joint Owner Affairs
A. C. Tapert, Program Administrator
Jeffrey J. Keenan, Esquire
Consumer Advocate, Office of Consumer Advocate
William Conklin, Public Safety Consultant, Lower Alloways Creek Township
State of New Jersey
State of Delaware

APPENDIX A

NOTICE OF VIOLATION

Public Service Electric and Gas Company	Docket No: 50-354
Hope Creek Generating Station	License No: NPF-57

During an inspection conducted on April 5, 1998, through May 16, 1998, violations of NRC requirements were identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," (60FR 34381: June 30, 1995/NUREG-1600), the violations are listed below.

A. 10 CFR 50.59, "Changes, tests and experiments," in part, permits the licensee to make changes to its facility and procedures as described in the final safety analysis report (FSAR) and conduct tests or experiments not described in the safety analysis report without prior Commission approval provided the change does not involve a change in the technical specifications or an Unreviewed Safety Question (USQ). The licensee shall maintain records of changes in the facility and these records must include a written safety evaluation which provides the bases for the determination that the change does not involve a USQ.

FSAR Section 9.1.3.2.3 establishes that the design and operation of the fuel pool cooling and cleanup systems for the decay heat associated with a full core offload is based, in part, on the operation or availability of the residual heat removal (RHR) system to augment the fuel pool cooling and cleanup (FPCC) system.

Contrary to the above, during refueling outage RF03 in December 1990, the licensee did not maintain the RHR system in operation or available to augment the FPCC system which represented a change to the facility as described in the FSAR and did not perform a review of this change to demonstrate that the change did not involve a USQ.

This is a Severity Level IV violation (Supplement I).

B. 10 CFR 50 Appendix B Criterion XI requires, in part, that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service be identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents. The test program shall include, as appropriate, proof tests prior to installation and operational tests during nuclear power plant operation of structures, systems, and components.

Contrary to the above, two examples of inadequate testing requirements associated with a design change modification to the Hope Creek safety-related control area chilled water system chillers were identified as follows:

(1) As of April 7, 1998, a complete proof test prior to installation and an operational test had not been performed to verify that check valves 1KBV-1243 through 1KVB-1250 would provide a relatively leak tight boundary and ensure that the backup safety-related pneumatic supplies for the chiller condenser cooling water pressure control valves would remain available for four hours after a loss of power event.

(2) On April 8, 1998, the backup safety-related pneumatic pressure regulators (1KBPCV-11464, -11466, and -11467) for the chiller condenser cooling water pressure control valves were found set below minimum design requirements. Operational tests had also not been performed to ensure that pressure regulators 1KBPCV-1164 through 1KBPV-1171 would remain properly set in accordance with design requirements.

This is a Severity Level IV violation (Supplement I).

C. 10 CFR Appendix B Criterion XVI (Corrective Action) requires, in part, that measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, and nonconformances are promptly identified and corrected.

Contrary to the above, on December 10, 1997, PSE&G engineers determined that the minimum cooling water inlet temperature for the safety-related control area chilled water system chillers should be changed in a more limiting direction to 70 degrees Fahrenheit from 55 degrees Fahrenheit. On April 9, 1998, the operations department management, still unaware of any necessary change to the minimum allowed cooling water temperature, used 55 degrees Fahrenheit as a basis for determining inoperability when they made a four-hour event notification to the NRC. Hope Creek abnormal operating procedure, *Loss of Instrument Air and/or Service Air, HC.OP-AB.ZZ-0131(Q) - Rev. 14,* and pending change, *HFSAR 97-080,* to the *Hope Creek Updated Final Safety Analysis Report* (UFSAR) also incorrectly stated that 55 degrees Fahrenheit was the minimum cooling water temperature below which the safety-related backup pneumatic supply needed to remain operable. The change in minimum cooling water inlet temperature to a more limiting value was not corrected until May 7, 1998, when guidance was provided to operators specifying the new 70 degrees Fahrenheit minimum cooling water temperature.

This is a Severity Level IV violation (Supplement I).

Pursuant to the provisions of 10 CFR 2.201, Public Service Electric and Gas Company is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555 with a copy to the Regional Administrator, Region I, and a copy to the NRC Resident Inspector at the facility that is the subject of this Notice, within 30 days of the date of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation" and should include for each violation: (1) the reason for the violation, or, if contested, the basis for disputing the violation or severity level, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further violations, and (4) the date when full compliance will be achieved. Your response may reference or include previous docketed correspondence, if the correspondence adequately addresses the required response. If an adequate reply is not received within the time specified in this Notice, an order or a Demand for Information may be issued as to why the license should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

If you contest this enforcement action, you should also provide a copy of your response to the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001, and to the Enforcement Coordinator, USNRC, Region I, 475 Allendale Road, King of Prussia, Pennsylvania 19406-1415..

Because your response will be placed in the NRC Public Document Room (PDR), to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be placed in the PDR without redaction. If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.790(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

Dated at King of Prussia, Pennsylvania
this ____ day of (Month) 1998

UNION OF
CONCERNED
SCIENTISTS

September 3, 1999

Chairman Grate J. Dicus
Commissioner Nils J. Diaz
Commissioner Edward Mgaffigan, Jr.
Commissioner Jeffrey S. Merrifield
United States Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT INADEQUATELY MONITORED SPENT FUEL POOL TEMPERATURE AND OPERATOR RESPONSE TIMES AT PERMANENTLY CLOSED PLANTS

Dear Chairman and Commissioners:

According to the December 28, 1998, issue of *TVA Watts Happening* (enclosed), the spent fuel pool water temperature at Browns Ferry Unit 3 increased approximately 25°F over a two day period. This temperature increase was not detected by the instruments in the control room, which continued to indicate a temperature in the normal range of 85°F to 95°F throughout the heatup. A manual check (i.e., use of a thermometer) determined the spent fuel pool water temperature was actually 109°F.

TVA reported the spent fuel pool water heatup began after Fuel Pool Cooling Pump 3A was taken out of service and replaced by Fuel Pool Cooling Pump 3B. A check valve on the discharge side of Fuel Pool Cooling Pump 3A stuck in the open position. This failure allowed the cooled water leaving Fuel Pool heat Exchanger 3B to flow back through the idle pump to the suction side of the operating pump. Consequently, water was recycling through the heat exchanger instead of returning to the spent fuel pool.

It is apparent from the event narrative that the sensor being used to monitor spent fuel pool water temperature is located in the vicinity of the heat exchanger. This configuration explains why the control room indication remained constant even though the actual spent fuel pool water temperature rose nearly 25°F. From my prior experience on spent fuel pool issues, I know that this configuration is common.

While the actual safety significance of this event is minimal, the potential safety implications are very significant. This operating nuclear power plant experienced degraded cooling of irradiated fuel assemblies that remained undetected for two days (48 hours). The time-to-boil for spent fuel pools can be less than 48 hours under some routine conditions. Because it can take many hours to restore spent fuel pool cooling after the loss or degradation is known, few plants can afford to waste two days of their time-to-boil on merely detecting the problem. Clearly, a sensor that physically measured the temperature of the water *inside* the spent fuel pool would be much safer than one like that at Browns Ferry that monitors the temperature of water *outside* the spent fuel pool.

This event is also significant from the standpoint of nuclear plants that have permanently shut down. I recently attended a workshop on risk-informing decommissioning regulations conducted in Gaithersburg, MD (ironically, it was a two-day workshop or the precise duration of the undetected spent fuel pool problem at Browns Ferry). Mr. Michael Meisner from the Maine Yankee plant and head of an NEI task force on decommissioning criticized the NRC staff because they had assumed the identification and correction of degraded spent fuel pool cooling conditions might last longer than a single shift (12 hours). Mr. Meisner was quite adamant that the NRC had absolutely no basis for assuming that a degraded condition might remain undetected for longer than a day. Clearly, this Browns Ferry event – which occurred at an operating plant receiving much more attention from far more workers than that proposed for permanently shutdown plants – demonstrates beyond any reasonable doubt that the staff's position is indeed justified and Mr. Meisner is simply wrong. The report on spent fuel pool problems presented to the Commission by the then-AEOD staff in November 1996 provides amply other events which prove that the Brown Ferry case was not an isolated one.

As UCS monitors the move towards risk-informed regulation, we continue to be troubled by industry initiatives, such as the extremely non-conservative and ill-advised approach now being contemplated by Maine Yankee management, which toss out r ignore reality. We hope that the NRC staff will be as diligent in guarding against these unwarranted erosions of safety margins as they have been thus far in the spent fuel pool issue at plants being decommissioned.

We also hope that the Commssion, as it guides the NRC down the road to risk-informed regulation, will consider all industry experience – drawing from both good and bad events – before rendering safety decisions.

Sincerely,

David A. Lochbaum

David A. Lochbaum
Nuclear Safety Engineer
Union of Concerned Scientists

Enclosure: as stated

copies: Mr. Gary Holahan
Mr. Michael Masnik
Mr. Ray Shadis
Mr. John Zwolinski

// WATTS HAPPENING

Watts Bar Industry Affairs
Published: *December 28, 1998*

http://knxwbngfp3/ Operating Experience

WATTS HAPPENING

TVA Operating Experience

Watts Bar

Corrective Action Program - When a PER description of condition is revised, SPP-3.1 "Corrective Action Program" states "the revised PER shall be processed the same as a new PER." When a new PER is initiated, Operations receives a report for operability review. Engineering identified that when a PER description of condition is revised, Operations does not receive a report for operability review contrary to SPP-3.1 requirements.

Browns Ferry

Spent Fuel Pool Temperature Increase - Unit 3 fuel pool temperature increased approximately 25 degrees F over a two day period after swapping from 3A fuel pool cooling pump to 3B fuel pool cooling pump. This temperature increase was not detected by the normal control room monitoring temperature element which continued to indicate a temperature in the normal range of 85 to 95 degrees F. A manual reading taken locally determined pool temperature to be 109 degrees F. Investigation determined that the 3B fuel pool cooling heat exchanger outlet was being short cycled through the out of service 3A fuel pool cooling pump due to the pump discharge check valve being stuck in the open position. After the stuck check valve was successfully closed, indicated fuel pool temperature rose to 121 degrees F and a lowering temperature tend was noted.

Sequoyah

Emergency Operating Procedure Deficiency - It was identified that procedure ECA-0.0 "Loss of All AC Power" contains a step which checks for steam generator tube rupture coincident with the loss of power. The step is impossible to execute on Unit 2 due to the fact that none of the Rad Monitors and recorders referenced in the step have power available if the shutdown boards are deenergized.

Industry Operating Experience

Plant Trips	NSSS	Description
Seabrook	W-4-LP	Unit 1 experienced an automatic reactor trip from 100% power due to a 345 kV breaker opening. The breaker opening resulted in a turbine trip followed by the reactor trip. The cause is being investigated. A Pressurizer PORV opened and reseated. One 4160 volt bus did not auto transfer to the reserve auxiliary transformer as designed. (NRC Event Report 35185)

Operating in a Plant Configuration not Described in the FSAR - The Cycle 16 operation at Oyster Creek reached end-of-full-power one month prior to the refueling outage. A safety review was performed in support of the removal of the high-pressure (HP) and intermediate-pressure (IP) feedwater heaters from service to compensate for decreasing reactivity during the coastdown operation. Isolating the extraction steam to the feedwater heaters at a predetermined power level reduces feedwater temperature. The reduced feedwater temperature will increase core inlet subcooling, adding positive reactivity and increasing core thermal power. This process actually decreases overall plant efficiency, however, the net result is an increase in electrical generation when compared to leaving the feedwater heaters in service during coastdown. The safety review documented the effects of reducing the final feedwater temperature at rated thermal power. The decreased feedwater temperature has the potential to affect analyzed plant transients and accidents, potentially impacting the margins to safety limits. The colder feedwater may also affect the feedwater nozzle and other components of the reactor vessel and internals. As part of the effort to support coastdown operation with HP and IP extraction steam removed a review of previous operating history was performed. That review identified an instance during cycle 15 where operation at rated power had occurred with the HP feedwater heaters OOS and feedwater temperature reduced without the required core related analysis to support the operation. This constituted operation in an unanalyzed condition. Procedural controls for reactor operation with feedwater heaters out-of-service (OOS) identified power restrictions based on concerns related to turbine imbalance. The procedures allowed for rated power operation under certain feedwater heater OOS configurations. No consideration was given for core transient/accident analyses required to justify rated power operation under these conditions. **Ensure that operations are within established FSAR and Licensing Basis parameters.** INPO Network OE9513

Worker Injured in Fall - At St. Lucie, a 3-man crew reported to the Unit 1 turbine building mezzanine deck to disassemble a 15-foot high scaffold. One of the workers was climbing a scaffold ladder in preparation for disassembly. The worker was approximately 12 feet off the ground, and in the process of securing his lanyard to the ladder, when he lost his grip and fell. The worker was transported to the on-site medical facility where he exhibited signs of confusion and disorientation. An ambulance was summoned and the individual was transported to a local hospital for observation and treatment. A later report indicated that x-rays revealed a cracked clavicle, a shoulder separation and that the individual had sustained

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
631 PARK AVENUE
KING OF PRUSSIA, PENNSYLVANIA 19406

RECEIVED DEC 23 1976

DEC 20 1976

Mr. Jack A. Lemke
124 Roxbury Road
Niantic, Connecticut 06357

License No. OP-3507-1
Docket No. 55-4617

Dear Mr. Lemke:

This refers to the special inspection conducted by Mr. E. Greenman and other members of our NRC Region I office on November 12-19, 1976 at Millstone Point Unit 1, Waterford, Connecticut of activities authorized by NRC License No. DPR-21 and to the discussions of our findings held by Mr. Greenman with Mr. Kufel and Mr. Ferland of Northeast Nuclear Energy Company on November 15, 1976 and to further discussions of our findings held by Mr. L. Norrholm of this office with Mr. Ferland on November 19, 1976. It also refers to USNRC Region I letters to Northeast Nuclear Energy Company dated November 12 and 15, 1976, and to Northeast Nuclear Energy letter to USNRC, Region I, dated November 15, 1976.

This special inspection of the refueling activities was conducted as a result of the unplanned criticality at Millstone Unit 1, reported by Northeast Nuclear Energy Company on November 12, 1976. The inspection consisted of selective examinations of procedures and representative records, interviews with personnel, and observations by the inspectors.

Based on the results of this inspection it appears that one of your activities was not conducted in full compliance with the requirements of your Operator License No. OP-3507-1 as set forth in the Notice of Violation, enclosed herewith as Appendix A.

This item of noncompliance has been categorized into the levels as described in the attached enclosure dated December 31, 1974. This notice is sent to you pursuant to the provisions of Section 2.201 of the NRC's "Rules of Practice", Part 2, Title 10, Code of Federal Regulations. Section 2.201 requires you to submit to this office within twenty (20) days of your receipt of this notice, a written statement in reply including admission or denial of the item of noncompliance and if admitted, the reasons for the item of noncompliance.

Enforcement action is also being taken against your employer, Northeast Nuclear Energy Company, and your Shift Supervisor at the time of the unplanned criticality. Copies of the correspondence relating to these actions are available in the Public Document Room. Copies are also enclosed for your information.

A copy of this letter and the enclosed Notice of Violation is being transmitted to your employer, Northeast Nuclear Energy Company, holder of NRC License No. DPR-21.

In accordance with Section 2.790 of the NRC's "Rules of Practice", Part 2, Title 10, Code of Federal Regulations, a copy of this letter and your reply will be placed in the NRC's Public Document Room.

Should you have any questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,

James P. O'Reilly

James P. O'Reilly
Director

Enclosures:
1. Appendix A, Notice of Violation
2. Criteria for Determining Enforcement Action and Categories of Noncompliance with AEC Regulatory Requirements--Modifications
3. NRC Region I letters to the Northeast Nuclear Energy Company dated November 12 and 15, 1976
4. Northeast Nuclear Energy Company letter to NRC Region I dated November 15, 1976
5. USNRC letter to Northeast Nuclear Energy Company with enclosures
6. USNRC letter to Lewis T. Crosse with enclosures

cc: D. C. Switzer, President
E. J. Ferland, Plant Superintendent
L. T. Crosse, Licensed Senior Reactor Operator
A. Z. Roisman, Counsel for Citizens Committee for Protection of the Environment

APPENDIX A

NOTICE OF VIOLATION

License No. OP-3507-1
Docket No. 55-4617

Based on the results of an NRC inspection conducted on November 12-19, 1976, it appears that one of your activities was not conducted in full compliance with conditions of your operator's license as indicated below. This is an Infraction.

Operator License No. OP-3507-1 states in part that, "In manipulating the controls of the. . .facility. . .the licensee shall observe the operating procedures and other conditions specified in the facility license or authorization which authorizes operations of the facility. . . ." Appendix A to Provisional Operating License No. DPR-21, Technical Specification Section 6.8.1.b states in part that, "Written Procedures shall be established, implemented and maintained covering. . .refueling operations." Operating Procedure OP 1408, Revision 0, Change 2, dated September 17, 1976, "Administrative Controls for Fuel Loading and Unloading" further established control of shutdown margin testing. Section 3.2.6 of OP 1408 states in part, "withdraw diagonal rod. . . ." Additionally, Reactor Engineering Instruction dated November 6, 1976 required withdrawal of Rod No. 42-19."

Contrary to the above, on November 12, 1976, while manipulating the controls of the Millstone Unit 1 facility to perform the specified shutdown margin test, instead of Control Rod 42-19, Control Rod 46-19 was erroneously selected and withdrawn. An unplanned criticality and automatic reactor trip from high flux on the IRM Channels occurred at 4:49 a.m. following the additional withdrawal of Control Rod 46-23. Between 4:50 and 4:58 a.m., on the same date, further shutdown margin testing was performed without recognition of the previous rod selection error. Control Rod 46-23 was positioned as specified. For the second time, Control Rod 46-19 was then erroneously selected and withdrawn to a predetermined position. Control Rod 46-23 was then additionally withdrawn. The second withdrawal of Control Rod 46-23, while terminated prior to a second automatic reactor trip, did result in an increased reactivity addition requiring immediate insertion of Control Rod 46-23 in order to prevent such an automatic trip.

DCD (CWFN P1-37)
Dist. Clear # IE1C

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
475 ALLENDALE ROAD
KING OF PRUSSIA, PENNSYLVANIA 19406-1415

OCT 21 1993

Docket No. 50-271

Mr. Donald A. Reid
Vice President, Operations
Vermont Yankee Nuclear Power Corporation
RD 5, Box 169
Ferry Road
Brattleboro, Vermont 05301

Dear Mr. Reid:

SUBJECT: NRC AUGMENTED INSPECTION TEAM (AIT) REPORT NO. 50-271/93-81

This letter transmits the AIT report for the inspection led by Mr. J. E. Beall between September 10-14, 1993. This inspection assessed the circumstances, causes, personnel actions, and the safety implications of the fuel handling incidents which occurred on September 3 and 9, 1993. At the conclusion of this inspection, a public exit was held on September 21, 1993, with you and other members of your organization to discuss the preliminary findings of the AIT.

The AIT concluded that errors by operators were the immediate causes of the events. The preponderance of the physical evidence indicated that the grapple had not properly closed on the fuel assembly handle and that the grapple light had not energized resulting in the drop of the assembly on September 3, 1993. The September 9, 1993, event was an inadvertent operator performance error. Human factors weaknesses contributed to the error in lowering rather than raising the fuel assembly.

The AIT concluded that the root cause of the dropped fuel assembly event was a lack of management oversight of fuel handling activities. Weak management oversight had allowed many of the measures intended to prevent a fuel handling accident to become degraded. For example: design changes were not transmitted to allow timely and accurate training on the modifications; training was not effective in that operators were not aware of certain key procedure steps; procedures were not used and were not adhered to; and supervisors did not ensure that procedures were followed. The AIT concluded that management did not communicate its expectations and provide proper oversight of fuel handling activities.

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OCT 21 1993

The decision by the senior reactor operator to continue the planned move of the assembly which had, on September 9, 1993, inadvertently been lowered onto a core component was not in compliance with the licensee's fuel handling procedure. This procedure compliance error had not been identified by licensee line management or by the licensee's event investigation. This was indicative of weakness in corrective actions to the September 3, 1993, incident and continued weakness in management oversight of fuel handling activities.

The intent of this inspection was to determine the cause and any potential generic safety issues. Therefore, no attempt has been made at this time to characterize the findings relative to regulatory requirements. Any enforcement associated with the inspection findings will be transmitted to you under a separate letter.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and the enclosure will be placed in the NRC Public Document Room.

We acknowledge and appreciate your excellent cooperation with our AIT during this period.

Sincerely,

Marvin W. Hodges

Marvin W. Hodges, Director
Division of Reactor Safety

Enclosure: NRC Inspection Report No. 50-271/93-81

OCT 2 1 1993

Mr. Donald A. Reid

cc w/encl:
R. Wanczyk, Plant Manager
J. Thayer, Vice President, Yankee Atomic Electric Company
L. Tremblay, Senior Licensing Engineer, Yankee Atomic Electric Company
J. Gilroy, Director, Vermont Public Interest Research Group, Inc.
D. Tefft, Administrator, Bureau of Radiological Health, State of New Hampshire
Chief, Safety Unit, Office of the Attorney General, Commonwealth of Massachusetts
R. Gad, Esquire
G. Bisbee, Esquire
R. Sedano, Vermont Department of Public Service
T. Rapone, Massachusetts Executive Office of Public Safety
Public Document Room (PDR)
Local Public Document Room (LPDR)
Nuclear Safety Information Center (NSIC)
K. Abraham, PAO (2)
NRC Resident Inspector
State of New Hampshire, SLO Designee
State of Vermont, SLO Designee
Commonwealth of Massachusetts, SLO Designee

bcc w/encl:
Region I Docket Room (with concurrences)
E. Kelly, DRP
J. Shedlosky, DRP

bcc w/encl **(VIA E-MAIL)**:
V. McCree, OEDO
D. Dorman, NRR
W. Butler, NRR

bcc w/encl **(AIT REPORTS ONLY)**:
The Chairman
Commissioner Rogers
Commissioner Remick
Commissioner de Planque
J. Taylor, EDO
T. Murley, NRR
DCD (OWFN P1-37) (Dist. Code #IE10)
A. Chaffee, NRR/DORS/EAB
E. Jordan, AEOD
INPO
P. Boehnert, Chairman, ACRS (AIT Reports Only)
K. Raglin, AEOD (AIT Reports Only)

U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.	50-271
Report No.	50-271/93-81
License No.	DPR-28
Facility Name:	Vermont Yankee Nuclear Power Plant
Inspection At:	Vernon, Vermont
Inspection Conducted:	September 10-14, 1993
Inspectors	G. West, Jr., Engineering Psychologist, NRR T. Shedlosky, Project Engineer, DRP P. Harris, Resident Inspector, DRP A. Burritt, Operations Engineer, DRS
Observer:	W. Sherman, State of Vermont

Team Leader: J E Beall — 9/29/93

J. E. Beall, Team Leader, Engineering Branch, DRS — Date

Approved By: Jacque P. Durr — 10/5/93

J. P. Durr, Chief, Engineering Branch, DRS — Date

Inspection Summary: See Executive Summary

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EXECUTIVE SUMMARY

On September 3, 1993, at about 12:30 p.m., Vermont Yankee personnel inadvertently dropped a fuel assembly approximately eight feet back into its reactor core location during fuel handling activities. At the time, the plant was shutdown in a refueling outage with about one-third of the planned fuel moves already completed. On September 7, 1993, the licensee resumed fuel handling activities after completion of troubleshooting, implementation of corrective actions, and discussions with the NRC. On September 9, 1993, a second fuel handling incident occurred when operators inadvertently lowered a fuel assembly onto a core component. After the second incident was reported, the licensee suspended further fuel handling.

An Augmented Inspection Team (AIT) was dispatched by the NRC to determine the circumstances that led to the events, their causes, safety significance and generic implications. The AIT began its assessments on September 10, 1993, completed its onsite review on September 14, 1993, and presented its preliminary findings in a public exit meeting on September 21, 1993.

Errors by operators were the immediate causes of the events. The preponderance of the physical evidence indicated that the grapple had not properly closed on the fuel assembly handle and that the grapple light had not energized resulting in the drop of the assembly on September 3, 1993. The September 9, 1993, event was an inadvertent operator performance error. Human factors weaknesses contributed to the error in lowering rather than raising the fuel assembly. The team concluded that the fuel handling incidents had been appropriately mitigated by the systems operable and the existing plant configuration, and were, independently, of minor safety significance.

The team concluded that the root cause of the dropped fuel assembly event was a significant weakness in management oversight of fuel handling activities. Weak management oversight had allowed many of the measures intended to prevent a fuel handling accident to become degraded. The AIT found that design changes were not transmitted to allow timely and accurate training on the modifications, training was not effective in that operators were not aware of certain key procedure steps, procedures were not used and were not adhered to, and supervisors did not ensure that procedures were followed. Management did not communicate expectations and provide proper oversight of fuel handling activities.

The decision by the senior reactor operator to continue the planned move of the assembly which had, on September 9, 1993, inadvertently been lowered onto a core component was not in compliance with the licensee's fuel handling procedure. This procedure compliance error was not identified by licensee line management or by the licensee's event investigation. This was indicative of a weakness in corrective actions to the September 3, 1993, incident and a continued weakness in management oversight of fuel handling activities.

DETAILS

1.0 INTRODUCTION

Upon being informed of the fuel handling incidents on September 3 and 9, 1993, at the Vermont Yankee Nuclear Power Station, the NRC Region I Regional Administrator and senior management from the Office of Nuclear Reactor Regulation (NRR) and the Office for Analysis and Evaluation of Operational Data (AEOD) determined that an augmented inspection team (AIT) should be formed to review the circumstances and evaluate the significance of the events. The bases for the determination were the need for the NRC to fully understand the causes of the events and to determine if these were associated with generic issues which required further NRC action. Accordingly, an AIT was selected, briefed, and dispatched to the site on September 9, 1993.

1.1 AIT Scope and Objectives

The charter for the AIT (Attachment I) was finalized on September 9, 1993. The charter directed the team to conduct an inspection and to accomplish the following general objectives:

a. Develop a detailed sequence of events related to both events, from September 3 through 9.

b. Determine the specific circumstances and causes of the apparent operator errors that occurred during refueling operations on September 3 and 9, 1993.

c. Determine and evaluate any changes (specifically Plant Design Change 92-11 implemented in Spring 1993) made in the design, maintenance, testing, or operation of the refueling bridge including hoist, grapple, and operator controls. Also, included was associated training for such modifications.

d. Evaluate the human factors aspects of both events, including: 1) command and control, and communications; 2) human performance factors, such as staffing, overtime, and schedule; and 3) human systems interfaces, such as with console design.

e. Assess the safety significance, including existing damage to the affected fuel assemblies.

f. Determine the adequacy of Vermont Yankee's maintenance and troubleshooting practices for refueling equipment, including vendor interface and control.

g. Evaluate Vermont Yankee's corrective actions and management controls following the September 3rd event, as they relate to the second September 9th event, particularly on the job training.

1.2 AIT Process

During the period September 10 - 14, 1993, the AIT conducted an independent inspection, review, and evaluation of the conditions and circumstances associated with the events. The team inspected fuel handling, equipment, and controls; held discussions and formal interviews both with personnel involved in the event and others performing the same duties; reviewed relevant records, including operator logs, modification packages, and training documents; and evaluated the adequacy of established procedures, personnel training, and management oversight. Attachment 2 is a list of personnel contacted by the AIT.

2.0 DESCRIPTION OF THE EVENTS

On August 27, 1993, the licensee initiated a reactor shutdown and entered a refueling outage. On September 1, 1993, the licensee commenced the first of (about) 700 planned movements of fuel assemblies necessary to refuel the reactor. On September 3, 1993, during fuel move number 233, one fuel assembly became detached from the grapple while being lifted out of the reactor core. The assembly fell an estimated 8 feet back into its original location. Some assembly damage and a small radioactive release occurred.

The licensee suspended fuel handling, notified the NRC, and investigated the event. After the licensee completed their investigation and implemented certain corrective actions, fuel handling was resumed with NRC concurrence.

On September 9, 1993, during fuel move number 388, a fuel assembly was inadvertently lowered, instead of raised, resulting in an apparent impact with another assembly or component in the core. The potentially damaged assembly was then moved to a fuel sipping can, as originally planned. The licensee reported the event, suspended fuel handling activities, and the NRC dispatched an AIT to investigate the fuel handling incidents.

A chronology associated with the fuel handling incidents is contained in Attachment 3.

3.0 PERSONNEL PERFORMANCE AND REFUELING OVERSIGHT

The AIT assessed the preparation, performance and practices associated with core alterations/fuel movement prior to, between and after the two refueling events. The findings of the team are grouped into four categories: Training, Procedures, Personnel Performance and Management Oversight.

3.1 Training

3.1.1 Operator Training Prior to September 3, 1993

Prior to the refueling event on September 3, 1993, all operations personnel received specific refueling training as a part of the normally scheduled continuing training cycle. The training was provided to licensed operators and non-licensed Auxiliary Operators. The training, conducted during the six week period between June 29, 1993 and August 6, 1993 was one and a half hours in length. All operations personnel attended the normally scheduled training except for two who completed a read and sign package on refueling operations as a make-up.

An outline in lesson plan LOT-00-234H, Revision 8 (06/93), was used to guide the training provided on refueling The outline specified a discussion of vessel disassembly, refuel outage tests, planned modifications to the refuel bridge along with a review of the listed licensed operator objectives. Although, one of the objectives required the use and or understanding of procedure 1101, the objective did not provide clear performance expectations regarding the level of understanding. The lesson plan included a memorandum on the scope of PDCR 92-011, "Refuel Platform Upgrades." The memorandum stated that the main hoist controls would be replaced but did not provide any details about the new controls. The figure of the main hoist controls in the lesson plan was not the same as the controls installed at the beginning of the outage. The training occurred prior to the controls being replaced.

Based on interview data, procedure 1101, "Management of Refuel Activities and Fuel Assembly Movement" was discussed in class with emphasis on the administrative limits and duties and responsibilities sections. The body of the procedure was briefly reviewed with particular attention to notes and cautions. The instructor did not specifically discuss the procedure steps or procedural adherence.

Instruction provided to cover the objectives for requalification training, by practice is left to the discretion of the instructor. Although the training provided was generally considered to be adequate based on interviews, there was no documentation of what was covered in class as well as no assurance that the same information was presented from one class to the next.

On The Job Training (OJT) by either demonstration or hands on practice was not provided prior to the September 3, 1993 event. This training was not considered on the basis that it had not been done in the past for requalification training. Although the personnel involved in refueling generally had a significant amount of fuel movement experience, some individuals interviewed felt hands on practice was necessary. Since core alteration fuel movements are done infrequently, the team concluded that it would be a good practice to renew manipulative skills under non safety significant conditions, particularly following the modification of equipment, indications and controls.

Training practices prior to the September 3, 1993, event were weak. This assessment is based on the absence of OJT prior to refueling and the results of interviews. Through interviews, the AIT determined that most of the operators were not aware of certain important procedural requirements and routinely had not performed all the required steps during fuel movement operation.

3.1.2 Reactor Engineer Training Prior to September 3, 1993

Prior to the refuel outage the reactor engineers were provided a number of training sessions. One of these sessions was given using an informal 5 page briefing sheet. The intent of this training was to review refueling goals, various procedural requirements along with refuel contingency plans with the reactor engineers providing fuel movement support. The "Things to remember when moving fuel" section contained a specific discussion that the reactor engineer should verify the grapple close light to be on prior to lifting a fuel bundle.

The AIT did not review all the training and documentation for the instructional briefings provided to the reactor engineers, but the team did confirm that the expectations to verify grapple closure were clearly addressed.

3.1.3 Operator/Reactor Engineer Training Following the September 3, 1993

Training immediately following the September 3, 1993 event was performed as a part of the facility corrective actions. The training consisted of a classroom session, a read and sign package, and hands-on OJT for the operators and reactor engineers.

The classroom training included a review and discussions of procedure 1101, "Management of Refuel Activities and Fuel Assembly Movement," Revision 21, in detail. Although the lesson plan had no specific objectives, it did clearly document the key points to be emphasized and appeared appropriate for the circumstances. The training took approximately 3/4 of an hour to perform and was done prior to the OJT portion.

The read and sign package was provided to operations personnel on the revised procedure 1101, "Management of Refuel Activities and Fuel Assembly Movement," Revision 23.

The OJT consisted of moving the bridge and trolley to coordinates provided in the spent fuel pool, grappling the bundle, performing the grapple engagement verifications including the rotation check under load. All members of the refuel teams performed in their normal positions while practicing the direction and communications required.

Training provided following the September 3, 1993, event was adequate. The team noted, however, that although the controls were operated during the OJT portion of the training, the changes in hoist speed and the effective reversal of switch motion compared to hoist direction were not addressed.

3.2 Fuel Handling Procedure

3.2.1 Procedure Adequacy

The AIT reviewed procedure 1101, "Management of Refuel Activities and Fuel Assembly Movement," Revisions 21, 22, and 23. Although a number of enhancements were made in each of the successive revisions, the team determined Revision 21 of the procedure to be adequate for fuel movement with the main hoist.

The procedure provided expectations that the reactor engineer and the SRO were required to visually verify grapple closure. Additionally, the reactor engineer and SRO verification step was imbedded in the step to close the grapple. The procedure, Revision 23, required a more directed evolution by the SRO as well as establishing additional verifications to ensure proper grappling of a fuel bundle. The additional procedural barriers consisted of SRO verification of proper grapple head orientation prior to engaging the grapple and redundant checks by the reactor engineer and SRO to verify the grapple light was energized following grapple closure.

The procedure, Revision 21, ZZ axis (refueling hoist depth) counter reference values were incorrect. These reference points provide approximate relative height for key components within the spent fuel pool and reactor core. The values had changed based on the installation of a new ZZ axis indication system prior to the refuel outage. Although these levels provided valuable operator information, additional operator aids were in place, updated daily and effectively ensured that the operator had the most up to date reference values.

The AIT identified specific weaknesses with procedure 1101, "Management of Refuel Activities and Fuel Assembly Movement," Revision 21, but concluded that these weaknesses did not significantly contribute to the September 3, 1993 event. Further, Revision 23 of this procedure was assessed as adequate to perform fuel movements with the main hoist and therefore was not a contributing factor in the September 9, 1993 event.

3.2.2 Fuel Handling Steps

Procedure 1101, Revision 23, directed operators withdrawing a peripheral fuel assembly to withdraw it "to just above the core, move toward the center-core area, then fully raise" the assembly. The licensee chose this method over the alternative, raising the assembly fully and then moving toward the center, to reduce radiation levels in the drywell when handling fuel. This allowed fewer restrictions on personnel access to the drywell; such restrictions could increase the time to complete scheduled outage work.

Operators interviewed stated that it was their general practice to move all assemblies using the method directed for peripheral assemblies. Most facilities similar to Vermont Yankee restrict access as necessary and use the "full lift then move to center" approach for moving fuel. The team did not conclude that the licensee's fuel handling steps were inadequate, but did note that the approach used significantly increased the chance that an inadvertent operator error, such as occurred on September 9, 1993, would result in a fuel assembly impacting a core component. Using the more common fuel handling steps, an operator would have over 30 feet of clearance to correct a mistake rather than the two to three feet (or four to six seconds) available to operators.

3.3 Personnel Performance

3.3.1 Reactor Engineers

During the September 3, 1993, dropped fuel assembly event, the reactor engineer was off the bridge of the refueling platform performing other duties on the refueling platform floor. Therefore, the reactor engineer did not follow the facilities Operating Procedure No. 1101, Revision 21, and verify grapple closure which stated: "Move the Grapple Open/Close Switch to the CLOSE position, the SRO and the Reactor Engineering representative visually verify grapple closure."

Operating Procedure 1101, Revision 21, indicated that the reactor engineer shall:

a. watch for and notify the Senior Licensed Operator of any occurrence which may potentially jeopardize safe refueling.

b. ensure compliance with the Fuel Loading Schedule.

c. maintain tag boards on the refuel floor and place target tags on boards for operator reference.

d. assist the Senior Licensed Operator on the refuel floor.

e. maintain the Refueling Log (DP 0420).

The above duties placed conflicting organizational expectations by management on the reactor engineer. When only one reactor engineer was assigned to a refueling shift crew, the individual had duties that were both on the refueling platform bridge and on the refueling platform floor. This conflict was a contributing factor to the dropped fuel assembly event.

The team concluded that, prior to the September 3, 1993, event; the reactor engineers should have been aware of all assigned responsibilities and identified conflicting duties to their supervision.

Reactor engineer performance was good during the September 9, 1993, inadvertent lowering of the fuel assembly event. The reactor engineer quickly observed that the fuel bundle was being lowered rather than raised, called to the refueling platform operator to stop, which probably mitigated the consequences of the event.

3.3.2 Refueling Platform Operators

The refueling platform operator involved in the dropped fuel assembly event reported that he verified grapple closure (i.e., observed that the grapple closed light was on). Moreover, the operator noted that he observed the change in state of the grapple closed light. The report and observation by the operator were inconsistent with the preponderance of physical evidence, which indicated that the grapple was not closed when raised. The team assessed the performance of the refueling platform operators to be poor prior to the September 3, 1993, event; the operators generally were not aware of the requirement to perform a rotational check, which ensured proper engagement, after grapple closure. That rotational check, not done for the dropped assembly, was intended to detect improper grappling and may have prevented the September 3, 1993, event.

The refueling platform operator engaged in the inadvertent lowering of the fuel bundle event committed an error resulting principally from two reasons. First, the human factors aspect of the change to the main crane controls, a "joystick," was a contributing factor (see Section 3.1.1). The operator had received no training on how to use the new joystick. The relationship between the motion of the joystick and control of the grapple was inconsistent with his past experience and reversed the direction of control used in the previous outage. Second, the way the licensee moved fuel, that is, initially raising each assembly 2-3 feet above the core, next moving the bundle to the center of the core, and finally taking the assembly full up, was also a contributor. This pattern provided an additional opportunity to commit an error of this type (see Section 3.2.2).

3.3.3 Senior Reactor Operators

The SRO's performance (specifically, command and control) during the dropped fuel assembly event was unsatisfactory for several reasons. The senior reactor operator failed to verify grapple closure and did not ensure that the reactor engineer, who was off the refueling platform bridge, verified grapple closure. Operating Procedure 1101, Revision No. 21, required the senior reactor operator and the reactor engineer to visually verify grapple closure. Further, the senior reactor operator did not comply with the following step of the subject procedure: "Verify the fuel assembly is grappled by: 1) attempting to carefully rotate the control console one way then the other..." Thus, the senior reactor operator lacked command and control not only because he failed to follow procedure, but also because he did not require the shift crew to follow procedure.

The SRO's performance regarding command and control was also unsatisfactory during the inadvertent lowering of the fuel bundle event. After the fuel bundle was inadvertently lowered, the SRO continued with the movement of the fuel bundle to the spent fuel pool. This action was contrary to Operating Procedure 1101, Revision No. 23, which stated:

- If any off normal condition or evidence of interference or binding of components in the core develops, further operation in that area of the core shall be immediately stopped and the problem investigated and corrected.

- Fuel bundles not properly seated may be reseated. However, no fuel movement or core alterations shall be made to correct a damaged, misloaded or misoriented bundle in the reactor until the R/CE Manager has been notified.

- Halt any activity in the event of an unusual or abnormal occurrence. Once any activity has been halted, it may only be re-initiated by the Shift Supervisor after receiving concurrence from the Operations Manager.

3.4 Management Oversight

3.4.1 Procedure Usage

The facility did not have an administrative procedure or guideline that dictated the level of use required for procedures. Based on interviews, the team was unclear on what the management expectations were for use of procedure 1101, "Management of Refuel Activities and Fuel Assembly Movement." Prior to the September 9, 1993, event, operators did not use procedure 1101 during fuel handling, did not have it on the fuel bridge for reference purposes, and did not know if it was available on the refuel floor. Some operators thought it was appropriate to have the procedure available on the refuel bridge for either step by step referral or to reference it as needed, since there was a lot of information to remember. However, the majority of personnel interviewed thought this was not necessary due to the repetitive nature of fuel movement and due to housekeeping concerns while over the vessel and spent fuel pool.

The AIT determined that clear management expectations did not exist for the required level of use of refuel procedures. The team was concerned that lack of clear management expectations may have created an environment in which refuel procedures routinely were not used.

3.4.2 Communications

During interviews, the AIT determined that the SRO used a hand-held phone to maintain communication with the control room during fuel movement within the core. Additionally, the operator would use binoculars to observe fuel assembly placement. These two factors would challenge the ability to hold and use the procedure. The team considered communications to have been adequate, but noted that the current practice could inhibit future in hand procedure use on the refuel bridge.

3.4.3 Fuel Movement Briefings

During interviews, the AIT determined that pre-job or pre-shift briefs had not been conducted prior to fuel movements. The FSAR stated that "All supervisors shall hold instructional briefings with members of their staffs prior to executing a refuel procedure in the interest of safety and good power station practice," (FSAR Section 13.9.3). The failure of shift supervisors or the refuel floor SROs to perform the briefings, represents a missed opportunity to review and verify the understanding of key administrative and procedural requirements to ensure safe refuelling practices. Based on interviews, the team concluded that cognizant licensee managers had not been aware of the FSAR commitment.

3.4.4 Schedule

Based on interviews, the AIT determined that the refuel teams had felt no undue sense of urgency or pressure from management related to fuel movement. All personnel interviewed stated that there was a clear emphasis on "doing things right and safely." Additionally, there appeared to be no competition between shifts that potentially could have compromised safety.

3.4.5 Refuel Staffing and Overtime

The AIT reviewed the plant staffing assigned to refueling activities and concluded that the refuel staff size was appropriate for the conduct of safe refueling operations. A good initiative was demonstrated by VY to augment the number of senior licensed operators with two individuals from the Training Department. However, the team concluded that the staffing for refueling was inadequate during occasions when only one of the two reactor engineers were on the refuel floor. During these occasions, the remaining engineer was challenged to effectively perform all assigned duties and responsibilities because the engineer's attention was divided between administrative requirements (maintenance of the status board and refuel log) and refuel activities (grapple, hoist, and bridge operations). This conflict contributed to the dropped fuel assembly event (see Section 3.3.1). Following the event, two engineers were assigned to the refuel floor during refueling.

The team reviewed the hours of work for control room and refueling operators and concluded that the use of overtime was within licensee requirements and NRC guidance and did not contribute to either event. The team verified that operators had received at least 8 hours off between shifts and noted that 12-hour breaks were normally scheduled. Operators interviewed by the team stated that they had had adequate rest and the team concluded that there was no evidence that fatigue had contributed to either event. Based on a review of time and attendance records, the team concluded that the licensee had effectively administered operator hours.

3.4.6 Management Oversight of Fuel Handling

The team concluded that there had been little or no management oversight of fuel movement. Operations management stated that there had been no management observations of fuel movement during the current outage. The quality assurance organization had not performed any fuel movement surveillance since 1987.

4.0 REFUELING BRIDGE FUEL HANDLING EQUIPMENT DESIGN CHANGES

The team reviewed the details of design changes made to the refueling platform fuel handling equipment over the past five years. During that period, the licensee completed three separate activities intended to improve the performance of this equipment. The fuel grapple was replaced in 1988 with an improved design with two independent hooks to carry the fuel assembly. This year, modifications were made to improve the overall reliability of the equipment by replacing the hoist and its controls and the other associated components such as the refueling hoist depth (ZZ axis) monitoring and display. Following the September 3, 1993, dropped fuel assembly event, an interlock was added to the hoist upward control circuit.

Modifications to the handling equipment controls had also occurred. Earlier training illustrations depicted a "joystick" control on the operator console to control trolley movement left and right, a second joystick for platform forward and reverse motion and a third for fuel grapple hoist raise and lower. These three vertical joysticks were replaced with three horizontal rotary lever controls. These were operated, for example, by rotating the lever clockwise toward the operator to raise the hoist and counter clockwise away from the operator to lower the load.

4.1 Fuel Grapple Replacement

The original fuel grapple was replaced with a grapple of improved design in April–May 1988. The new grapple had redundant hooks that carried the fuel assembly handle. The two hooks operated independently, each actuated by its own air cylinder. Separate position sensing switches were operated by each of the two hooks. The switch was operated when its associated hook was fully engaged under the fuel assembly handle. The switches were wired in series such that both had to be closed to illuminate the grapple closed lamp located on the hoist operator's console.

The team reviewed the documentation for the grapple replacement contained in Plant Design Change Request (PDCR) No. 87-03, "Fuel Grapple Replacement," and General Electric Company Service Information Letter (SIL) No. 181, "Redundant Hook Grapple Head." The safety evaluation accompanying the design changes reasoned that the redundant hook grapple had two hooks each with a factor of safety equal to or greater than the original single hook. Also, the position sensing microswitches of the new grapple were believed to be more reliable than the proven [illegible] original de[illegible]

The team discussed the grapple design with vendor representatives. The redundant grapple was designed such that a single component failure would not result in dropping a fuel bundle. In regard to load safety factors, under normal conditions, the entire grapple assembly had a safety factor of three, the primary hook a safety factor of 12, the secondary of 8.5 and the shaft on which the hooks pivoted has a safety factor of 21. The redundant hooks were configured such that it should not be possible for both to simultaneously hang up on a fuel bundle. For example, one of the two hooks was 3/16 inch shorter that the other. Additionally, the opening in the grapple head that accepted the fuel assembly bale handle had been machined to provide 1/4-inch clearance across the handle width and 1/8-inch across its thickness. When seated on the fuel assembly handle, the grapple hooks passed under the handle with approximately 0.5 inch clearance. Additionally, the configuration of the grapple closure sensing microswitches and their associated grapple hooks provided a positive indication of grapple closure that was relatively insensitive to changes in switch position adjustment. The team concluded that the grapple design provided a close tolerance fit with the fuel assembly handle, and that the redundant hooks and their associated microswitches provided good assurance of positive grapple operation.

During the 1988 installation, the licensee exercised an option and changed the vendor recommended design not to install an electrical interlock involving hoist upward motion. The design, as proposed by the vendor in SIL 151, provided an interlock to prevent upward hoist motion when loaded to approximately 480 pounds unless both grapple hook position switches were closed indicating that the hooks were fully positioned under the fuel assembly handle. During the design change review process, the licensee decided not to include this electrical interlock because of their experience and the poor performance of the proximity switch in the old grapple. The interlock was not installed because of their concern that the grapple might become inoperable if a microswitch failed.

4.2 Refuel Platform Upgrade

The licensee began the first phase of modifications intended to upgrade the reliability of the refuel platform with PDCR 92-11. The work associated with this change was performed in the period of June 25 through August 19, 1993. The scope of these modifications was determined by a task force formed to review past problems with the platform and provide recommendations to improve reliability and performance. The equipment replaced included the main hoist including the motor, drum, brakes, cable and controls, a new depth indicator including a digital encoder and display, new electronic load cells and display with programmable interlock modules, new air hoses and reels, a new air drive and stainless steel air lines, a new main power cable, a new access ladder and a new remote readout assembly for the digital display.

The team reviewed the design change documents including the safety evaluation, the design change review comments, the completed procedure for shop tests of the hoist and controls, and the installation and test procedure. The design change was routed within the licensee [illegible] departments. These documents described the various aspects of the modifications including their operator interface aspects such as the type and location of new digital displays for main hoist depth and main hoist load. The new hoist and its motor speed controller and associated joystick control were selected to provide the operator with better fine control of hoist motion. The hoist was replaced primarily due to limitations on obtaining spare parts for the old hoist mechanism.

The design change document did not address the hoist direction in relation to joystick movement, nor did it address the difference in control when replacing the rotary motion controller with the joystick and the resultant differences to operations personnel. The installation and test procedure for PDCR 92-11, Step 7.24.2, required operation of the main hoist joystick and verification that the hoist rotated in the proper direction. It was necessary to reverse motor leads to complete this requirement. Proper hoist operation was an attribute that was verified as a third party quality control inspection.

The team concluded that the refuel platform modifications resulted in an overall improvement in the reliability of fuel handling equipment. However, the licensee did not seek design input from operations personnel concerning modification of the operator interface components such as controls and indication. The team also noted that the modification work was not scheduled for completion early enough before the current refueling outage to allow hands-on training of personnel prior to fuel movement.

4.3 Refueling Platform Hoist Interlock Modification

The control circuit for the refueling platform main hoist was modified on September 6, 1993, by the licensee in response to the September 3, 1993, event. The modification added an electrical interlock that prevents upward movement of the hoist when loaded to greater than approximately 450 pounds if both fuel grapple hooks are not fully closed. The installation was accomplished as Temporary Modification (TM) 93-053, and done under Work Order No. 93-07052-04. The interlock installed was similar to that recommended by the vendor for incorporation with the redundant hook fuel grapple (see Section 4.2).

The licensee modified the vendor recommended circuit after finding that a resistor in series with the close grapple relay coil did not allow the relay to pick-up. A "b" contact in the same relay was used to bypass the resistor and, therefore, allowed enough current to pass for the relay to operate. The team concluded that the modification should be transparent to the operator under normal circumstances. The grapple closed indicator lamp and main hoist load cell operated in the same manner as before the modification.

5.0 MAINTENANCE AND TROUBLESHOOTING OF REFUELING EQUIPMENT

The team reviewed records of work requests and work orders for plant refueling equipment. The team found that the licensee had been performing preventive and corrective maintenance, [illegible]

Under Work Order 93-07019-00, an inspection was made of the grapple, its hooks, and their air cylinder operators and position indicating switches on September 5, 1993, two days after the dropped fuel assembly event. Other than air cylinder leaks, all components were determined to be functioning properly. A small air leak was found at the lower air connection to a cylinder, the other was found filled with water. The second cylinder was replaced. The grapple hook position microswitches were found to be set to open with slight outward motion of the grapple hooks. In that condition, the grapple closed light would go out as the main hoist mast was rotated to verify proper engagement with a fuel assembly handle. The AIT was unable to determine the extent, if any, that the leaking air cylinders contributed to the dropped fuel assembly; however, the licensee's investigation findings discounted any contribution by the condition of the air cylinders to the event.

Work Order 93-07133-00 was written to adjust the microswitches and eliminate the intermittent loss of grapple closure indication. This was performed on September 8, 1993. Although the team did not identify any concerns with technician performance, the team noted that the work was accomplished without the benefit of a written procedure.

The licensee was unable to reproduce an incomplete grapple configuration with the grapple light energized. The team concluded that the preponderance of the physical evidence indicated that the grapple light had not been energized during the September 3, 1993, event. The team noted, however, that the incomplete grapple configurations tested by the licensee on smooth bar stock had not demonstrated sufficient gripping or pinching force to allow a partial lift of a fuel assembly. The licensee stated that the assembly handle geometry, which had a chamfered underlip, would have allowed more lift force to be transferred. The team considered the licensee's troubleshooting, completed about one week before the AIT investigation, to have been adequate.

6.0 HUMAN FACTORS REVIEW

The AIT reviewed the human factors aspects of both events regarding (a) human-system interfaces with the refueling platform console and (b) human performance, including command and control. In addition, human engineering deficiencies relative to the refueling platform console that were not necessarily related to the two events were also identified.

6.1 Human-System Interfaces

During the previous outage the control for the grapple was a lever switch. To raise the grapple, the lever switch would be pushed forward (see Figure 1). Prior to the current outage (during August 1993), the lever switch was changed to a joystick. Although training received information on this design change No. 92-11 (June 1993), refueling platform operators (RPOs) received no training on how to use the new joystick. Further, the classroom training that was taught in preparation for the outage used a figure that erroneously depicted the lever switch as the control for the grapple.

The licensee indicated that the joystick was installed based on Occupational Safety and Health Administration (OSHA), American Society of Mechanical Engineers (ASME), and Crane Manufacturers Association of America (CMAA) guidelines. OSHA 1910.179 section 3(iv) states: "As far as practicable, the movement of each controller handle shall be in the same general directions as the resultant movements of the load." CMAA Specification 70-1988, Section 5.7.3, states: "The movement of each master switch handle should be in the same general direction as the resultant movement of the load, except as shown in Figures 5.7.3a and 5.7.3b, unless otherwise specified." ASME B30.2-1990, Section 2-1.13.3(d), states: "The movement of each manual controller or master switch handle should be in the same general direction as the resultant movement of the load, except as shown in Figs. 6 and 7."

The licensee interpreted the above guidelines as a control panel in the horizontal plane. The licensee considered the control console (shown in Figure 2) for the refueling platform to be an incline plane design (i.e., about 30 degrees). Thus, the licensee installed and labelled the joystick with the resultant load in the direction that would be appropriate for a fully vertical plane design (i.e., forward motion is up and backward motion is down, as shown in Figure 1). An alternate rationale would be to consider forty five degrees to be the dividing line for deciding whether guidelines should be applied for horizontal or vertical plane design. In either case refueling platform operators should have been trained regarding the control change for the refueling platform grapple.

6.2 Human Factors Weaknesses

The walkthrough, interview results, and desk top review of photographs relative to the operation of the refueling platform console identified several additional human factors weaknesses [illegible] the guidelines [illegible] NUREG-[illegible]. The weaknesses are presented in Attachment 4.

6.3 Conclusion

The team concluded that the human factors weakness associated with the joystick controls change was a significant contributor to the September 9, 1993, fuel assembly inadvertent lowering event. The licensee stated at the September 21, 1993, public exit that the joystick controls had been changed as one of the licensee's corrective actions. The licensee indicated that forward motion was now "down" and backward motion was "up" as appropriate for a control panel in the horizontal plane.

7.0 LICENSEE CORRECTIVE ACTIONS

Following the September [illegible], 1993 event, the licensee conducted maintenance and trouble shooting of the hardware involved (see Section 5.0), conducted additional training (see Section [illegible]), and revised the fuel handling procedure (see Section 3.2). The licensee also added a grapple interlock (see Section 4.[illegible]) and required oversight by QA during the [illegible]

8.0 SAFETY SIGNIFICANCE

The team reviewed the radiation surveys, air sample records, and chemistry results following both fuel handling incidents. No evidence was found of any increases following the September 9, 1993 event. Some small increases were identified to have occurred after the September 3, 1993 event. These increases are summarized in Attachment 5. The team concluded that the radiological safety significance of the events was low. The team noted, however, that the observed gaseous activity increases suggested some damage probably occurred to one or more fuel rods.

The team reviewed the videotapes taken of the two assemblies involved in the events and the central core structures potentially impacted during the September 9, 1993 event. There was no indication of any damage other than to the two assemblies, but the full extent of damage from the September 3, 1993 event could not be assessed without removing the dropped assembly. The damage visible in the videotapes was of low safety significance.

A fuel handling accident is a design basis accident and, as such, is addressed in Section 14.6.4 of the Vermont Yankee FSAR. Secondary containment integrity and operability of key ventilation systems are prerequisites for handling fuel and were confirmed by the team to have been in place during both events. The team concluded that the fuel handling incidents had been appropriately mitigated by the systems operable and the existing plant configuration, and were of minor safety significance separately.

The team was concerned, however, that the weaknesses in management oversight had allowed many of the measures intended to prevent a fuel handling accident to become degraded. Design changes were not properly transmitted to allow timely and accurate training on the modifications. Training was not effective in that operators were not aware of certain key procedure steps. Procedures were not used and were not adhered to. Supervisors did not ensure that procedures were followed. Management did not effectively communicate expectations and provide proper oversight of fuel handling activities.

9.0 CONCLUSION

The AIT concluded that the preponderance of the evidence indicated that the September 3, 1993, event resulted from operator error. An improperly grappled fuel assembly became detached and fell about eight feet back to its original location with some minor damage to the assembly. The root cause of the event was a weakness in management oversight of fuel handling activities as evidenced by lack of knowledge of and adherence to the fuel handling procedure by operators, engineers, and supervisors.

The team concluded that mistaken lowering of a fuel assembly onto another core component on September 9, 1993, was an inadvertent operator performance error caused, in part, by human factors weaknesses. The continuance of movement of the affected assembly was a procedure compliance error. This procedure compliance error was not identified by licensee line management or by the licensee's event investigation. This was indicative of a weakness in corrective actions to the September 3, 1993, incident and a continued weakness in management oversight of fuel handling activities.

10.0 MANAGEMENT MEETINGS

Licensee management was informed of the scope of this AIT during an entrance meeting on September 10, 1993. The team briefed licensee management of the team's observations routinely and at the conclusion of onsite review on September 14, 1993.

A public exit meeting was conducted on September 21, 1993, at 10:00 a.m., at the Vernon, Vermont, town hall with licensee representatives identified in Attachment 2 to discuss the preliminary inspection findings. The slides presented at the public exit meeting are contained in Attachment 6. The licensee acknowledged the inspection findings and provided the results of their assessment of the event and the short term and long term corrective actions.

AIT MEMBERS

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AGENDA

- AIT CHARTER
- FACILITY DESCRIPTION
- CHRONOLOGY
- FINDINGS

AIT CHARTER

- DEVELOP CHRONOLOGY OF THE EVENTS
- DETERMINE THE CAUSE(S)
- DETERMINE PLANT RESPONSE
- DETERMINE THE ADEQUACY OF LICENSEE RESPONSE
- DETERMINE GENERIC IMPLICATIONS OF THESE EVENTS

CHRONOLOGY

- DESIGN CHANGE TO REFUELING [illegible] 'IDGE DEVELOPED
- PRE-OUTAGE TRAINING CONDUCTED
- MODIFICATION COMPLETED (CONTROLS REWIRED)
- REFUELING OUTAGE BEGAN
- FUEL ASSEMBLY DROPPED, FUEL HANDLING SUSPENDED
- LICENSEE REVIEW AND CORRECTIVE ACTIONS COMPLETED, FUEL HANDLING RESUMED
- FUEL ASSEMBLY INADVERTENTLY LOWERED ONTO CORE, THEN MOVED TO PLANNED LOCATION
- FUEL HANDLING SUSPENDED, AIT DISPATCHED

LICENSEE CORRECTIVE ACTIONS

- Stop Fuel Handling
- Additional training, including OJT
- Enhanced procedure for fuel moves
- Grapple inspection, testing
- Air system troubleshooting, repairs
- After September 9 incidents, again stopped fuel handling
- Initiated management team investigation

CAUSE

SEPTEMBER 3, 1993 FUEL DROP EVENT

- BREAKDOWN IN MANAGEMENT CONTROL AND OVERSIGHT OF FUEL HANDLING OPERATIONS

  - REQUIRED SHIFT BRIEFINGS NOT HELD

  - OPERATORS NOT KNOWLEDGEABLE OF ALL PROCEDURE REQUIREMENTS

  - PROCEDURES NOT USED

  - PROCEDURES NOT ADHERED TO

  - REACTOR ENGINEER RESPONSIBLE FOR GRAPPLE CLOSURE VERIFICATION ASSIGNED CONFLICTING COLLATERAL DUTIES

  - LITTLE OR NO INDEPENDENT OVERSIGHT DURING FUEL HANDLING

CAUSE

September 9, 1993 Fuel Handling Errors

1. Lowering assembly onto core was an inadvertent operator performance error with two major contributors

- Controls modification
  - No operator input into change
  - No training on wiring reversal
  - Increase in speed (30 vs previous 20 feet per minute)

- Fuel Shuffling procedure
  - Fuel lifted 2-3 feet out of core, then moved horizontally to core center
  - Vertical motion then initiated from 2-3 feet above core
  - Little time to correct error (4-6 seconds)
  - Lower drywell radiation levels represent a risk tradeoff

2. Continuance of fuel move was a procedure adherence error

- SRO believed he could authorize completion of fuel move

- Procedure states any unusual or off normal occurrence required halting movement and notifying the shift supervisor

- Further movement required operations manager permission

- Neither the licensee line management review nor the management team investigation identified SRO action as procedure non-compliance

CONCLUSIONS

- September 3 event caused by poor management control and oversight
  - widespread procedure non-compliance
  - preponderance of the evidence suggests refuel platform operator performance error rather than a hardware failure
  - Ltle or no management presence during fuel handling activities

- Initial September 9 incident was caused by inadvertent operator error
  - human factors contributor
  - fuel handling methodology contributor

- Second September 9 incident was caused by failure to follow procedures

  - weakness in corrective actions from September 3 event
  - continued weakness in management oversight

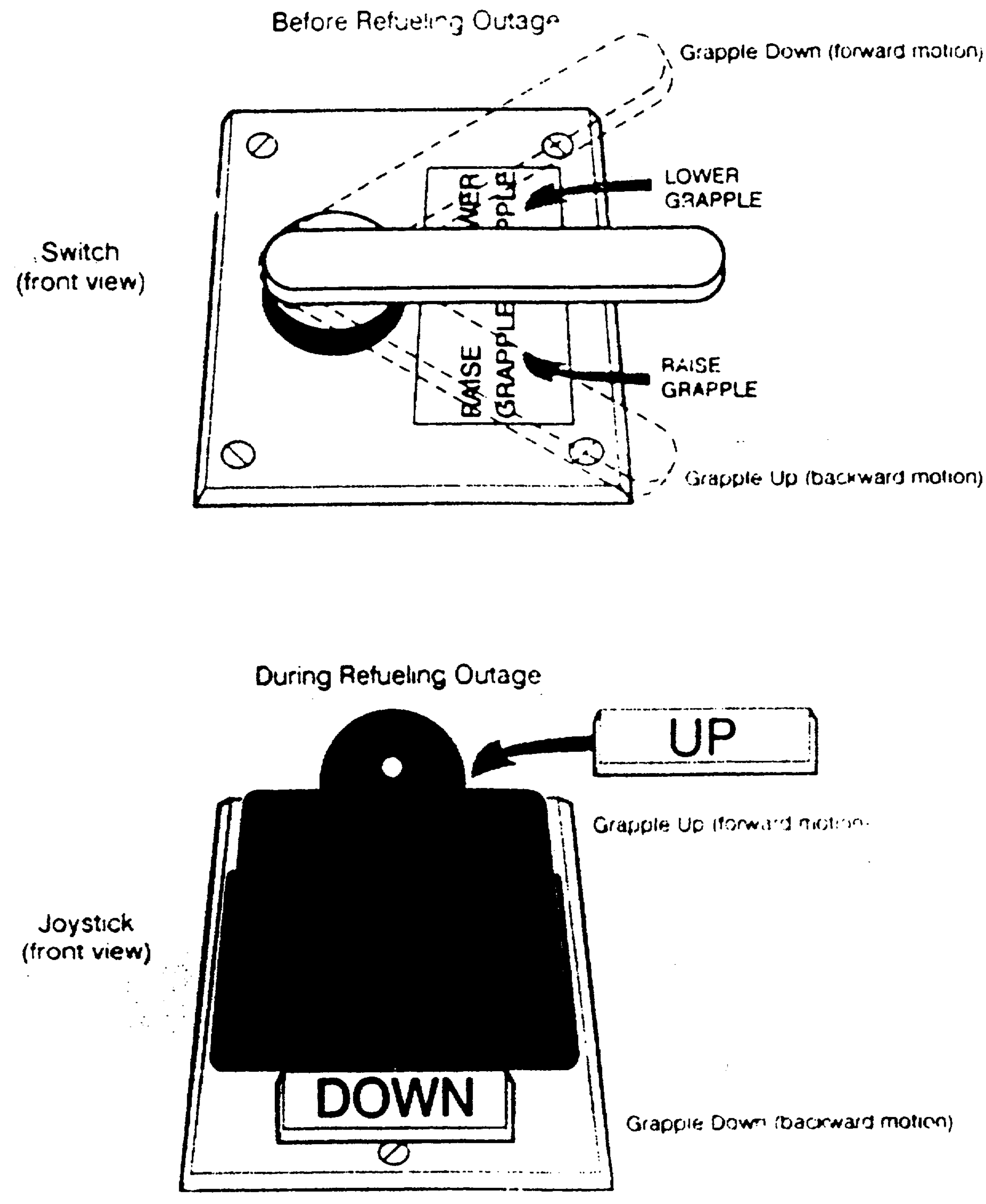
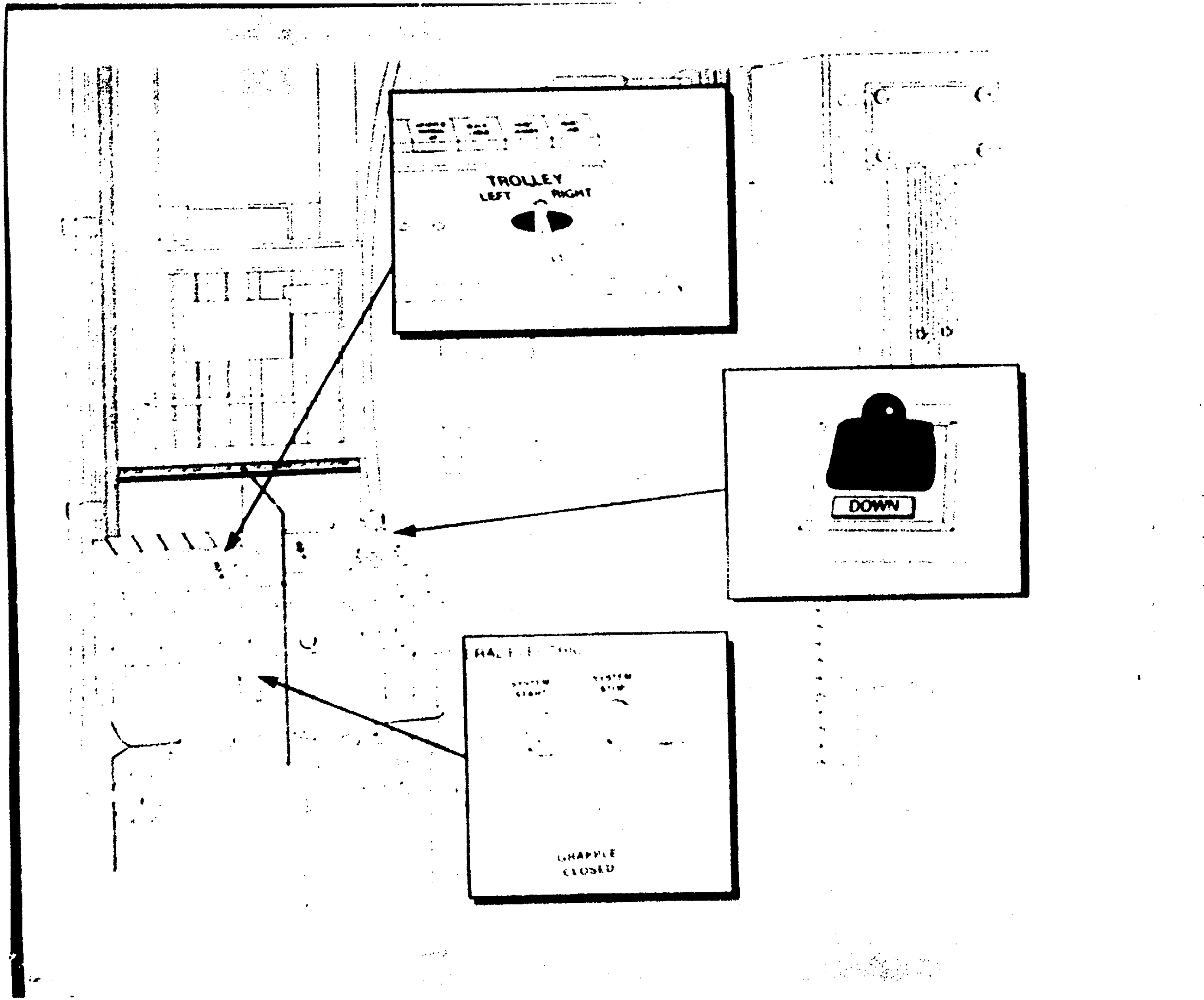


Figure 1. Grapple Up and Down Control


TROLLEY
LEFT
RIGHT
DOWN
GRAPPLE
CLOSED

ATTACHMENT 1

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
475 ALLENDALE ROAD
KING OF PRUSSIA, PENNSYLVANIA 19406 1415

September 9, 1993

Docket No. 50-271

MEMORANDUM FOR: Richard W. Cooper, II, Director, Division of Reactor Projects
Wayne M. Hodges, Director, Division of Reactor Safety
Charles W. Hehl, Director, Division of Radiation Safety and Safeguards

FROM: Thomas T. Martin, Region Administrator

SUBJECT: AUGMENTED INSPECTION TEAM CHARTER FOR REVIEW OF THE FUEL HANDLING EVENTS AT VERMONT YANKEE

Due to fuel handling incidents on September 3 and 9, 1993 at Vermont Yankee, I have determined that an Augmented Inspection Team (AIT) inspection should be conducted to verify the circumstances and evaluate the significance of these events.

The Division of Reactor Safety (DRS) is directed to conduct the AIT with James Beall as the Team Leader. Further, DRS, in coordination with the Division of Reactor Projects, is responsible for the timely issuance of the inspection report, the identification and processing of potentially generic issues, and the identification and completion of any enforcement action warranted as a result of the team's review.

Enclosed is the charter for the Augmented Team delineating the scope of this inspection. The inspection shall be conducted in accordance with NRC Management Directive (MD) 8.3, NRC Inspection Manual 0325, Inspection Procedure 93800, Regional Office Instruction 1010.1, Revision 2 and this memorandum. The bases for this inspection, per MD 8.3, are: (1) the staff's need to fully understand the causes of the events; and, (2) the staff's need to determine if there are potential generic issues worthy of staff action associated with these events. Preliminary information indicates that each event was caused by human error due, in part, to deficiencies in training, procedure clarity, and possibly human factors.

Thomas T. Martin
Regional Administrator

Enclosures:
1. Augmented Inspection Team Charter
2. Team Membership

cc w/encls:
J. Taylor, EDO
J. Sniezek, OEDO
T. Murley, NRR
J. Partlow, NRR
J. Calvo, NRR
C. Rossi, NRR
D. Dorman, PD I-3, NRR
F. Miraglia, NRR
C. McCracken, NRR
F. Rosa, NRR
W. Russell, NRR
J. Richardson, NRR
A. Thadani, NRR
B. Grimes, NRR
J. Roe, NRR
E. Jordan, AEOD
D. Ross, AEOD
V. McCree, OEDO
W. Kane, DRA, RI
R. Cooper, DRP, RI
W. Lanning, DRP, RI
J. Linville, DRP, RI
J. Beall, DRS, RI (Team Leader)
E. Kelly, DRP, RI
C. Hehl, DRSS, RI
S. Shankman, DRSS, RI
H. Eichenholz, SRI - Vermont Yankee
W. Butler, PD I-3, NRR
W. Ruland, DRS, RI
J. Durr, DRS, RI
W. Hodges, DRS, RI
L. Bettenhausen, DRS, RI
K. Abraham, PAO, RI
M. Miller, SLO, RI

ENCLOSURE 1

VERMONT YANKEE NUCLEAR POWER STATION

FUEL HANDLING ERRORS

AUGMENTED INSPECTION TEAM (AIT) CHARTER

The general objectives of this AIT are to:

1. Develop a detailed sequence of events related to both events, from September 3 through 9.

2. Determine the specific circumstances and causes of the apparent operator errors that occurred during refueling operations on September 3 and 9, 1993.

3. Determine and evaluate any changes (specifically Plant Design Change 92-11 implemented in Spring 1993) made in the design, maintenance, testing, or operation of the refueling bridge including hoist, grapple and operator controls. Also included is associated training for such modifications.

4. Evaluate the human factors aspects of both events, including: (a) command and control, and communications; (b) human performance factors, such as staffing, overtime, and schedule; and, (c) human-systems interfaces such as with console design.

5. Assess the safety significance, including existing damage to the affected fuel assemblies.

6. Determine the adequacy of Vermont Yankee's maintenance and troubleshooting practices for refueling equipment, including vendor interface and control.

7. Evaluate Vermont Yankee's corrective actions and management controls following the September 3rd event, as they relate to the second September 9th event, particularly on-the-job training.

8. Explicitly excluded from this charter are the recovery plans and radiological precautions associated with eventual removal of the first dropped assembly.

9. Prepare a report documenting the results of this review for signature by the Regional Administrator within 30 days of the completion of the inspection.

ENCLOSURE 2

VERMONT YANKEE AIT MEMBERSHIP

James Beall, AIT Leader, Division of Reactor Safety (DRS), Region I (RI)

Arthur Burritt, Operations Engineer, DRS, RI

Thomas Shedlosky, Project Engineer, DRP, RI

Paul Harris, Resident Inspector, Vermont Yankee

G. West, Engineering Psychologist, NRR

Other NRC personnel, consultants, or contractors will be engaged in this AIT, as needed.

ATTACHMENT 2

Persons Contacted

Vermont Yankee Nuclear Power Corporation

*L. Amirault	Training Instructor, SRO
J. Bazenas	Reactor Engineer (General Electric)
M. Benoit	RE&C Manager
*B. Bateau	Engineering Director
J. Cihak	Reactor Engineer
*R. Clark	Quality Assurance Director
D. Leere	Operations Training Instructor
D. Faulstich	Senior Program Manager (General Electric)
*E. Harms	Operations Training Supervisor
N. Jennison	ACRO
M. Newsom	Reactor Engineer (General Electric)
K. Oliver	SCRO
B. Perry	Auxiliary Operator
B. Pichette	SCRO
R. Ramsdell	Human Factors Engineer
*D. Reid	Vice President, Operations
R. Sharma	ACRO
A. Stetak	ACRO
R. Sojka	Operations Support Manager
M. Tessier	Electrical Engineer
*R. Wanczyk	Manager, Vermont Yankee Station

United States Nuclear Regulatory Commission

*R. Cooper	Director, Division of Reactor Projects
H. Eichenholz	Senior Resident Inspector

*Denotes those present at the exit meeting on September 21, 1993, attended by the public and news media. The team also held discussions with other licensee management, operations, maintenance, engineering and quality assurance personnel.

ATTACHMENT 3

Sequence of Events
for the
Vermont Yankee Refueling Incidents

Event 1: On September 3 at 12:23 p.m., during fuel move 233 fuel assembly LYN-831 inadvertently uncoupled from the fuel grapple while being removed from the core position 21-14 to sipping can #1. LYN 831 fell approximately 8 feet back into core position 21-14.

Event 2: On September 9 at 4:10 a.m., during fuel move 388, fuel assembly LYV-667 was in transport from core position 25-34 to fuel sipping can #2, and was inadvertently lowered into the double blade guide inser .d in the center of the core. LYU-667 was then moved to fuel sipping can #2.

3/4/88	--	The single J-hook grapple was replaced by the double J-hook design
8/[illegible]/93	--	Joy Stick Modification installed
8/27/93	1:00 p.m.	Commenced reactor shutdown
	6:10 p.m.	RMS - Startup
	10:[illegible] p.m.	All rods inserted, RMS - Refuel
8/28/93	5:3[illegible] a.m.	Rx coolant temp < 212 degrees F
8/29/93	1:20 p.m.	Rx vessel head removed
	6:00 p.m.	Commenced CRD maintenance
8/31/93	6:45 p.m.	Secondary Containment Capability check
	9:45 p.m.	Commenced refueling interlocks per OP 4102

9/1/93	6:40 a.m.	Completed refueling interlocks
	9:40 a.m.	Commenced fuel moves. NRC questions the lack of a 50.59 for a TM installed on CR 26-39 position indicator.
	11:55 a.m.	Refueling secured pending analysis of TM 93-49.
	6:55 p.m.	VY removed TM 93-49 and replaced the connector pin for CR 26-39. Corrective maintenance completed for CR 26-39.
	7:35 p.m.	Completed refueling interlocks.
	7:58 p.m.	Commenced fuel moves
9/2/93	1:40 p.m.	Refueling secured, air hose uncoupled
	2:35 p.m.	Commenced fuel moves, following repair of the air hose
	10:50 p.m.	Refueling secured, due to a broken power cable for the grapple.
9/3/93	3:15 a.m.	Commenced fuel moves, following repair of the power cable.
	12:23 p.m.	1 YN 831 uncoupled from the fuel grapple and dropped back into core position 21-14. A shudder was felt by Operators 1A and 1B on the refuel bridge. The SCRO heard a hissing noise. The grapple closed light was energized. No bubbles or debris were observed. No radiation detectors alarmed. The RP tech replaced the iodine cartridge with a new one. The bridge and grapple was left in the "failed" position. The refuel floor was evacuated.
	12:34 p.m.	Slight rise in stack gas 1 and 2 rate, 80 cpm to 140 cpm (peak at 180 cpm); Air sample at the 318 ft indicated very slight increase in Xe-133 (0.1 mr/hr). Drywell radiation surveys were normal. Hourly sampling of reactor cavity water indicated increase in Xe-133, Cs-134, and Iodine.
	12:34 p.m.	Drywell evacuated
	12:50 p.m.	Unusual Event declared, based on OP 3125, General Criteria.

	1:45 p.m.	Licensee and NRC conference call: assembly reseated in the proper position and orientation; minor stack gas increases had been observed in the past during sipping and refueling; no previous dropped fuel bundles at site; refuel floor continuous air sampler saw no increase in background or airborne.
	2:05 p.m.	PRO submitted, event notifications completed
	2:50 p.m.	Secondary containment capability check performed
	3:15 p.m.	Licensee management meeting; PORC reviewed the event. Initial inspections revealed no damage to the grapple. One fuel channel dog ear was bent the other was loose on the fuel assembly upper spider. All chemistry samples were trending down and expected to be normal within 5-6 hours. Iodine at 3E-3 dacs; root cause in progress.
	3:30 p.m.	Unusual Event terminated, based on OE 3125 criteria and a PORC recommendation.
	3:40 p.m.	NRC Operations Center notified
	7:06 p.m.	Licensee and NRC conference call; both parties agreed to hold refueling until NRC inspection of initial moves was performed.
9.5.93	[illegible]:25 p.m.	Secondary containment capability check performed
		10:35 p.m. TM 93-53 installed on the refuel grapple
	1:30 a.m.	Completed refueling pre-requisites
	2:25 p.m.	Commenced fuel moves
9.8.93	1:10 a.m.	Refueling secured, rotation of the grapple operator console causes "grapple closed" to go out. Emergency work order initiated.
	3:40 a.m.	Commenced refueling, corrective maintenance completed, ... grapple limit switches adjusted.
	9:55 a.m.	Refueling secured, outage manager concerned about inadequate documentation of the maintenance performed on the [illegible] switch

	11:20 a.m.	Refueling authorized by licensee management following review of the limit switch corrective maintenance.
	1:00 p.m.	Commenced fuel moves
	1:50 p.m.	Refueling secured due to noise in the take-up reel. Work order initiated.
	2:40 p.m.	Commenced fuel moves, completed troubleshooting of the take up reel. Licensee unable to identify the cause of the noise.
9/9/93	4:10 a.m.	Refueling secured due to the inadvertent lowering of fuel bundle LYV-667. During transfer of the bundle to the sipping container the grapple would not ungrapple. A hand-held mechanical tool also couldn't open the grapple. The fuel bundle was repeatedly reseated, without freeing the grapple and the control room was notified. The grapple finally operated and an emergency work order was initiated.
	4:45 a.m.	LYV-667 in sip can, commenced grapple, bail, and double blade guide inspections.
	5:30 a.m.	Plant management halted fuel handling activities.
	10:30 a.m.	Licensee secured all refueling activities
		Licensee task team chartered to review both refueling incidents with the plant manager as the team leader.
9/10/93		NRC Augmented Inspection Team arrived onsite.

ATTACHMENT 4

HUMAN FACTORS WEAKNESSES

1. Visibility of Label for Joystick: The up label for the joystick control of the grapple was obscured by the rubber base of the joystick, if visible at all, with the refueling platform operator positioned directly in front of the control. This weakness was inconsistent with NUREG-0700 guidelines: "Concealment - Labels should not be covered or obscured by other units in the equipment assembly" [guideline 6.6.2.4(b)], and "Controls - Labels should be visible to the operator during control actuation" [guideline 6.6.2.4(c)].

2. Grapple Open/Close Switch: The grapple open/close switch had a metal cover over it which obscured its open and close label. This weakness was inconsistent with NUREG-0700 guideline 6.6.2.4(c): "Controls - Labels should be visible to the operator during control actuation."

3. Demarcation Line: A black 1-inch wide demarcation line had been drawn with a magic marker between the system stop light and the grapple closed light. This weakness was inconsistent with NUREG-0700 guideline 6.6.6.2: "Permanence - Lines of demarcation should be permanently attached." The team noted that the licensee had installed a permanent plastic demarcation line as part of corrective action.

4. Grapple On/Off Light Switch: A danger tag was taped over the grapple on/off light switch and its label because the light attached to the grapple was burned out. This tag obscured the adjacent right label (i.e., frame) for the monorail switch. This weakness was inconsistent with the following NUREG-0700 guidelines: 6.6.5.1(h): "Adjacent Devices - Tag-outs should not obscure any adjacent devices or their associated labels" [guideline 6.6.5.1(h)], and "Obscuration - Tag-outs should not obscure the label associated with the non-operable device" [guideline 6.6.5.1(e)].

5. Grapple Closed Light Indication: The grapple closed light was located vertically on the front, lower area of the left panel of the refueling platform console. Whereas, the grapple open and close switch was located vertically on the front, upper area of the right panel of the console. This weakness was inconsistent with the following NUREG-0700 guidelines: "Sequence - controls and displays [indications] which are use together during a normal task sequence should be grouped together" [guideline 6.8.2.1(a)]; "Frequency of Use - frequently used controls and displays should be arranged to reduce search time and minimize the potential for during use" [guideline 6.8.2.1(b); "Functional Considerations - functionally related controls and displays should be grouped together when they are use together to perform tasks related to a specific function..." [guideline 6.8.2.1(c)]; and "Proximity - a visual display that will be monitored during control manipulation should be located sufficiently close that an

operator can read it clearly and without parallax from a normal operating posture" [6.9.1.19(a). With regard to the last guideline, interview results indicated that in a normal standing position it was difficult to see (i.e., parallax) the grapple closed light indication without shifting to a lowered position.

The label for the grapple closed light was placed below the indication light. This weakness was inconsistent with NUREG-0700 guideline 6.6.2.1(a): "Normal Placement - Labels should be placed above the panel element(s) they describe."

6. Monorail Frame and Console Switch: The labels for the monorail switch (either frame or console) were positioned diagonally. This weakness was inconsistent with NUREG-0700 guideline 6.6.2.3: "Horizontal Orientation - (1) Labels should be oriented horizontally so that they may be read quickly and easily from left to right. (2) Although not normally recommended, vertical orientation may be used only where space is limited. Improperly oriented labels can lead to confusion and cause delays in location and identification of important controls and/or displays."

7. Verbal Communication: As a result of background noise, verbal communication on the refueling platform was degraded at distances of approximately 6-8 feet or greater from speaker and listener using normal voice levels. This weakness was inconsistent with NUREG-0700 guideline 6.1.5.5(a): "Background Noise - Background should not impair verbal communication between any two points in the primary operating area. Verbal communications between these points should be intelligible using normal or slightly raised voice levels." Further, NUREG-0700 guideline 6.1.5.5(b) indicates: "Limit - Background noise levels should not exceed 65 dB(A)." No noise level measurements were obtained during the onsite inspection.

8. Weaknesses Identified From Interviews: During interviews with refueling platform personnel, the following weaknesses were identified: (1) it was difficult to impossible to simultaneously hold a hand-held telephone in one hand and binoculars in another hand, while reading a procedure; (2) no procedure on the bridge; the units for grapple load (pounds) and depth indication or z-indication (inches) were not provided on digital displays; (3) operators complained about the trolley left/right switch because the switch was pushed in the opposite direction than the trolley travels which was confusing and could lead to operator error; (4) operators indicated that the position of the pointer for the forward/reverse control for the refueling platform control was unrelated to its direction; (5) the up and down joystick control for the grapple could be inadvertently actuated when attempting to manipulate the platform forward and reverse control to its left.

ATTACHMENT 5

September 3, 1993: Post-Event Radiological Summary

Radiation Levels

Drywell (DW) 318 ft level	- $\leq$15 mr/hr (background)
Reactor Building (RB) 345 ft level	- $\leq$5 mr/hr (background)
Reactor Water Clean Up room	- normal
Spent Fuel Pool Cooling	- normal

Instrument	Initial	Peak	Post-peak	Trend
Stack Gas 1	30 cpm	140 cpm	110 cpm	down
Stack Gas 2	80 cpm	225 cpm	180 cpm	down
RB ventilation exhaust				
Gaseous	250 cpm	460 cpm	310 cpm	down
Particulate	200 cpm	none	NA	NA

Surface Contamination Levels

RB 345 ft - < 5k dpm/100 cm^2 (normal during refueling operations)
RB 345 ft - no detectable alpha

Reactor Cavity Water Chemistry (uCi/ml)

Isotope	8:10 a.m.	12:45 pm.	1:45 p.m.	2:45 p.m.	3:45 p.m.
Iodine 133	7.7E-5	7.7E-5	8.6E-5	7.7E-5	8.3E-5
Xenon 133	5.6E-5	4.4E-3	2.1E-3	1.8E-3	1.4E-3
Cesium 134	6.0E-5	1.4E-4	1.0E-4	1.2E-4	1.1E-4

NUCLEAR REGULATORY COMMISSION

VERMONT YANKEE

AUGMENTED INSPECTION TEAM EXIT

SEPTEMBER 21, 1993

PURPOSE OF AN AIT

A. LOWEST LEVEL OF NRC INCIDENT INVESTIGATION PROGRAM FOR RESPONSE TO OPERATIONAL EVENTS

B. CONDUCT A TIMELY AND THOROUGH INSPECTION WITH THE EMPHASIS ON FACT-FINDING

C. COLLECT AND ANALYZE THE FACTS TO DETERMINE CAUSE(S) OF THE EVENT

D. ASSESS THE SAFETY SIGNIFICANCE OF THE EVENT

E. AN AIT DOES NOT DETERMINE WHETHER NRC RULES WERE VIOLATED OR RECOMMEND ENFORCEMENT ACTION

F. AN AIT DOES NOT ADDRESS THE APPLICABILITY OF GENERIC CONCERNS TO OTHER PLANTS

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the matter of Lewis T. Crosse RFD #1, Old Stonington Road Box 148 Stonington, Connecticut 06378	)))))	Senior Operator License License No. SOP-1355-3 Docket No. 55-3274

ORDER TO SHOW CAUSE AND ORDER SUSPENDING LICENSE

I

Lewis T. Crosse, RFD #1, Old Stonington Road, Box 148, Stonington, Connecticut, ("the licensee") is the holder of Senior Operator License No. SOP-1355-3 ("the license") issued by the Nuclear Regulatory Commission ("the Commission"). The license authorizes the licensee to direct the licensed activities of licensed operators at, and to manipulate all controls of, Millstone Point Nuclear Power Station, Unit No. 1, Power Reactor, Facility License No. DPR-21 located at Waterford, Connecticut. This license was originally issued on June 17, 1970, and has been renewed periodically, the most recent renewal being November 9, 1976. The present expiration date of the license is November 9, 1978.

II

On November 12, 1976, the Northeast Nuclear Energy Company, holder of License No. DPR-21, reported to the USNRC, Region I, that an unplanned criticality and subsequent reactor trip had occurred.

III

As a result of the report, an inspection was conducted between November 12 and November 19, 1976. It was determined from this inspection that after the withdrawal of a control rod by the control operator which resulted in an inadvertent reactor criticality and reactor trip, the licensee directed a second withdrawal which was identical to the first test without recognition of the prior error and which required manual rod insertion to preclude a subsequent event and trip. In addition, the licensee failed to observe operating procedures and other conditions specified in Facility Operating License No. DPR-21. The items of noncompliance which were discovered during the inspection are set forth in the attachment.

IV

From the foregoing, it is apparent that the licensee has exhibited a disregard for the Commission's regulations. In view of this, and in the interest of the public health and safety and the health and safety of the employees directed by the licensee, it has been determined that no prior notice as provided in 10 CFR 2.201 and 10 CFR 55.40 is required and that pursuant to 10 CFR 2.202(f) License No. SOP-1355-3 should be suspended effective immediately.

V

In view of the foregoing and pursuant to the Atomic Energy Act of 1954, as amended, and the regulations in 10 CFR 2, 50, and 55, IT IS HEREBY ORDERED THAT:

1. The licensee show cause, in the manner hereinafter provided, why License No. SOP-1355-3 should not be revoked permanently. The licensee may, within twenty days of receipt of this Order, file a written answer to this Order under oath or affirmation and may also request a hearing within said twenty day period.

Any answer filed shall specifically admit or deny the inspection findings identified in Section III, above and the items of noncompliance attached to this Order. If a hearing is requested, the Commission will issue an order designating the time and place of hearing. Upon failure of the licensee to file an answer within the time specified, the Director for the Office of Inspection and Enforcement will, without further notice issue an Order revoking License No. SOP-1355-3.

In the event the licensee files a timely answer and requests a hearing within the time specified, the issues to be considered at such hearing shall be (1) whether the licensee was in noncompliance with the Commission's regulations and conditions of License SOP-1355-3 as specified in Section III, above, and in the Notice of Violation Appendix A, enclosed with this Order; and (2) whether License No. SOP-1355-3 should be permanently revoked.

In the event the licensee files a timely answer to this Order To Show Cause why License No. SOP-1355-3 should not be revoked, the licensee must describe, in detail, actions planned or completed by Northeast Nuclear Energy Company to satisfy the following:

a. An augmented program for your retraining and requalification in licensed activities.

b. Recertification by Northeast Nuclear Energy Company.

2. Pending further Order of the Commission, License No. SOP-1355-3 is suspended effective immediately, and accordingly the licensee shall cease and desist from further licensed activities until successful reexamination by the NRC or until license revocation if such order is issued after answer to the Order to Show Cause.

FOR THE NUCLEAR REGULATORY COMMISSION

/S/

Ernst Volgenau
Director
Office of Inspection
and Enforcement

Dated at Bethesda Maryland
this 20th day of December
1976.

APPENDIX A

NOTICE OF VIOLATION

Lewis T. Crosse
License No. SOP-1355-3

Senior Operator License No. SOP-1355-3 requires that you observe the Operating Procedures and other conditions specified in Facility Operating License No. DPR-21.

Certain of your activities conducted under Senior Operator License No. SOP-1355-3 and listed below, appear to be in noncompliance with your license requirements in that you failed to adequately direct licensed activities, observe the operating procedures, administrative procedures and other conditions of Facility Operating License No. DPR-21.

A. Senior Operator License No. SOP-1355-3 states in part, "In directing the licensed activities of licensed operators... the licensee shall observe the operating procedures and other conditions specified in the facility license which authorized operation of the facility..." Technical Specifications for Facility License No. DPR-21, Section 6, establishes procedural requirements, and Section 6.8.1.b states in part, "Written procedures shall be established, implemented, and maintained covering... refueling operations." Operating Procedure No. 1408, Revision 0, dated September 16, 1976, established Administrative Controls for Fuel Loading and Unloading.

   Contrary to the above, subsequent to the withdrawal of an erroneously selected control rod No. 46-19 by a control operator, which was not performed in accordance with OP 1408 and resulted in an inadvertent criticality and reactor trip at 0449 a.m. on November 12, 1976, you directed and supervised a second identical test, without recognition of the prior error and which required manual rod insertion to preclude a subsequent event and trip.

B. Procedure No. 106/2106/3106-01 Revision 0 dated September 30, 1975 states "Operators shall believe installed plant instrumentation to be correct unless proven faulty by direct comparison with other instruments monitoring the same variable; or proven faulty by instrument functional testing or calibration."

   Contrary to the above, after a reactor trip had occurred on November 12, 1976, nuclear instrumentation was not proven faulty nor was functional testing or calibration performed, and you proceeded to further direct the manipulation of reactor controls affecting reactivity changes, on the premise that the instrument indications were not meaningful.

C. Operating Procedure No. 502, Revision 4, dated August 26, 1975, entitled "Emergency Shutdown," and Station Order SO-6.01, Revision 4 "Nuclear Power Facility Communications Controls," required that you promply notify higher management of events such as automatic action of the reactor protection system and unplanned criticalities.

Contrary to the above, on November 12, 1976 you did not promptly inform higher management of the unplanned criticality and automatic reactor trip which occurred at 4:49 a.m. Your notification to management was not made until approximately 7:30 a.m. when higher management reported for the start of the day's work.

NUREG/CR-5771
BNL-NUREG-52294
RK, 1A

Probability and Consequences of Misloading Fuel in a PWR

Manuscript Completed: August 1991
Date Published: August 1991

Prepared by
D. J. Diamond, C.-J. Hsu, V. Mubayi

Brookhaven National Laboratory
Upton, NY 11973

Prepared for
Division of Systems Research
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555
NRC FIN L1131

ABSTRACT

This report documents the results of a study into the frequency and consequences of misloading fresh fuel assemblies during the reloading of a pressurized water reactor. The consequences that were considered included: i) loss of required shutdown margin, ii) inadvertent criticality, and iii) worker exposure within the plant given inadvertent criticality. Neutronic calculations were performed for different patterns of fresh fuel clustered together in a Combustion Engineering reactor. The fresh fuel considered had a high U-235 content and was assumed to be loaded without control element assemblies. The frequencies of misloading fresh fuel assemblies into these clustered patterns were calculated taking into account operator errors and equipment malfunctions that could occur during an offload/reload sequence. The study has improved our understanding of how difficult it is to misload fuel and has quantified the loss of shutdown margin and the frequency of occurrence for specific misloadings as well as the doses that might result from an inadvertent criticality.

EXECUTIVE SUMMARY

In order to extend the length of fuel cycles, utilities operating pressurized water reactors (PWRs) have been increasing the enrichment of reload fuel. The fresh reload assemblies may be highly reactive when they do not contain control element assemblies (CEAs) or many burnable poison rods. If they are placed in certain loading configurations this could lead to a loss of shutdown margin below the required 5% or, in the extreme, to an inadvertent criticality.

Analysis is required to confirm that the refueling boron concentration is sufficient to maintain the required shutdown margin for the final core layout. However, this concentration may not be sufficient to assure that the shutdown margin will be maintained for all intermediate fuel assembly positions that could arise in the course of a normal refueling. Furthermore, if multiple operator errors occur during refueling, there is the possibility that the configuration of fuel will lead to a loss of shutdown margin or perhaps an inadvertent criticality.

The U.S. Nuclear Regulatory Commission (NRC) alerted PWR owners to this potential problem and requested that licensees assure (via specific actions) that any intermediate fuel assembly configuration maintains the required shutdown margin. The NRC also requested that Brookhaven National Laboratory analyze different refueling configurations in order to understand the potential for losing shutdown margin and for inadvertent criticality. The analysis that was then done consisted of both deterministic and probabilistic calculations. The former was to determine the change in shutdown margin under different configurations and the latter to determine the expected frequency of occurrence of abnormal configurations. Another objective of the study was to determine the exposure to workers if there was an inadvertent criticality.

Cycle 9 of Calvert Cliffs 2 (a Combustion Engineering (C-E) plant) was chosen to be modeled because of its use of reload fuel with a high U-235 enrichment (4.3 w/o). NODE-P2, a three-dimensional nodal code, was used to do the calculation of k-effective and hence shutdown margin.

The properly loaded core with all CEAs present and with 2300 ppm of boron as specified by the Technical Specifications was calculated to be 13% shutdown. This is considerably greater than the required 5% primarily because it has generally been the philosophy of C-E plants to calculate the shutdown margin assuming all CEAs are removed and because the 2300 ppm requirement is very conservative.

Calculations with misloaded fuel in the center of the core were done conservatively assuming that the misplaced fuel is fresh fuel with no burnable poison rods. The reduction in shutdown margin when 1, 3, 5, and 9 fresh assemblies were misloaded was -0.5%, -4.0%, -7.8% and -12.6%, respectively. These results mean that it would take at least a cluster of 5 assemblies (in a cross configuration) to lose the required shutdown margin and a cluster of 9 assemblies (approximately) to have an inadvertent criticality in that core. It also indicates that if the original shutdown margin was only 5% it might only take a cluster of 5 assemblies to have an inadvertent criticality.

Calculations done with a reduced boron concentration (1800 ppm) quantified the expected result that fewer fresh assemblies would be required to cause a problem, i.e., either an inadvertent criticality or loss of the required shutdown margin. Conversely, calculations with a reduced assembly enrichment (4.08 w/o) quantified the result that more fresh assemblies would be required to cause a problem in this situation. Another set of calculations with CEAs removed from all fuel assemblies showed how fewer fresh assemblies would cause a problem. Calculations were also done with clusters placed symmetrically in each of the four quadrants to show that the effect is considerably less than 4 times the effect of a single cluster.

In order to understand what errors would have to occur to lose the required shutdown margin or have an inadvertent criticality, and what the frequency of occurrence of these events might be, a probabilistic assessment was carried out. Refueling at Calvert Cliffs is done by first off-loading the entire core into the spent fuel pool.* CEA repositioning is done in the pool and then the fresh and burned fuel assemblies, with CEAs if required, are returned to the reactor where they are loaded according to a pattern which allows for proper monitoring.

The probabilistic assessment was done considering three patterns of misloaded fresh fuel without CEAs. One pattern was a group of 4 fuel assemblies placed in a 2x2 array, another pattern was a group of 5 assemblies in a cross configuration, and the third pattern was a group of 9 assemblies placed in a 3x3 array. It was assumed that there would be a problem if one of these patterns was loaded anywhere in the core except at a location adjacent to the baffle. The results were that the probability of forming a cluster of 4 or 5 fresh assemblies was 1.2E-6 and 1.1E-7, respectively and the probability of forming a cluster of 9 fresh assemblies was much smaller.

Based on the deterministic calculations it can be conservatively assumed that a cluster of at least 4 fresh fuel assemblies is needed to lose the required shutdown margin and hence, the frequency of this event is expected to be 1.0E-6/YR. This result might be different if core shuffling were used rather than an offload/reload scheme as modeled here. The result was determined to be sensitive to certain human error probabilities used in the analysis and this sensitivity was quantified. Errors made in using intermediate positions were found to be important as well as errors associated with picking up or storing assemblies in the spent fuel pool. Other dominant human errors were those made in verifying fuel assembly and CEA positions prior to refueling.

* Refueling had previously been done using shuffling.

The frequency of an inadvertent criticality was obtained taking into account that neutron monitors should pick up the approach to criticality and that the operator could then stop the refueling and/or add boron. If the problem was noticed at an early stage the operator could easily take action, however, if the operator did not become aware of the problem until the last misplaced assembly was being lowered into the core the probability of operation action would be relately low. The expected frequency of an inadvertent criticality caused by a cluster of 9 fresh assemblies was calculated to be insignificant relative to that for the situation with 5 fresh assemblies clustered. If the conservative assumption is made that it only takes 5 fresh assemblies to cause an inadvertent criticality then the frequency increases to 9.1E-9/YR. This number is of course sensitive to the same human error probabilities discussed with regard to losing the required shutdown margin.

The probabilistic analysis could be extrapolated to other plants using similar offload/reload procedures only if the loading pattern was similar to that chosen for study herein. It does not apply to a shuffling scheme. Furthermore, it should be noted that in the future, plants that currently do a complete offloading may switch to a shuffling scheme if, for example, spent fuel pool space becomes limited.

If there is an inadvertent criticality then workers will be exposed to radiation. Calculations were done to determine the direct dose to workers from γ-rays and neutrons as well as the dose due to fission products escaping from damaged burned fuel which surrounds the fresh assemblies.

The direct dose was found to be insignificant using the conservative assumption that the power level could reach over 3000 MW (which would not be possible because of boiling that would occur at lower powers). However, the indirect dose due to fission products was calculated to be significant. The calculation was carried out assuming that 12 burned assemblies surrounding a 3x3 array of fresh fuel suffered fuel damage and released fission products from the gap. Inhalation and immersion doses were calculated for the important radionuclides assuming an evacuation time of 5 minutes. The doses to workers in the containment building would be more than 200 rem under these assumptions and there is a significant potential for early health effects. If only half the fuel rods in the assemblies were affected or if the evacuation time was halved then health effects are less likely. No calculation of dose to the public was carried out under the assumption that containment isolation systems would be effective.

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ACKNOWLEDGEMENTS

The authors were pleased to be able to get information on refueling procedures from Mr. C. M. Mitchell of JBF Associates, Inc. Mr. R. A. Holland (formerly) of Baltimore Gas and Electric provided general information and the ROCS calculation used as a benchmark for the core neutronics model. The authors appreciate the comments of Dr. J. Boccio of Brookhaven National Laboratory. Also appreciated is the support and comments of Drs. F. Odar (Project Manager) and L. Kopp of the U. S. Nuclear Regulatory Commission. Lastly, the authors are indebted to Ms. K. Ryan for her help in preparing this report.

1. INTRODUCTION

1.1 Objectives

The objective of this project was to analyze different refueling configurations that may be possible in a pressurized water reactor (PWR) in order to understand the potential for losing the required shutdown margin and, in the extreme, for an inadvertent criticality. The different configurations would be the result of making errors in the refueling process, including the use of improper intermediate positions. The analysis was to include calculations of shutdown margin with different configurations of fresh fuel clustered together with the effect of fuel enrichment and boron concentration also considered. For configurations in which the required shutdown margin was lost or criticality was obtained, a probabilistic analysis was to be done to estimate the frequency of occurrence of these events. Another objective was to calculate the consequences, in terms of doses to individuals, assuming an inadvertent criticality.

1.2 Background

1.2.1 The Potential for Losing Shutdown Margin

In order to extend the length of fuel cycles, utilities operating PWRs have been increasing the enrichment of U-235 in reload fuel. For example, the two unit Calvert Cliffs plant is operating with fuel having an enrichment as high as 4.3 w/o. Fresh reload assemblies have a large reactivity worth when they have a high enrichment and when they do not contain control rod assemblies or many burnable poison rods. If they are clustered together during refueling, in a configuration different from the final loading pattern, there is the potential for losing the shutdown margin required by Technical Specifications (usually 5%) and, in the extreme case, for an inadvertent criticality.

Utilities perform analysis to determine the refueling boron concentration which is sufficient to maintain the required shutdown margin for the final core layout. In addition, many utilities add requirements which increase the boron concentration beyond the minimum required. In some utilities the required shutdown margin is 6%, in some it is 5% plus that obtained with an additional 100 ppm of soluble boron and in some it is that obtained with a specified boron concentration which is expected to give more than 5% shutdown margin. Another conservatism, added in some plants that require control rods to always be present in fuel assemblies where they belong, is to assume that the rods are missing when the calculation of shutdown margin is carried out. However, depending on the approach taken, the actual shutdown margin may not be sufficient if certain intermediate positions are used during refueling.

Intermediate positions refers to the loading of fuel assemblies on a temporary basis into locations which are not the location in the final core loading map. This may be necessary for a number of reasons. If a fuel assembly is bowed and there is difficulty in removing it then adjacent assemblies may be temporarily removed to an intermediate location. If a bowed assembly is difficult to seat then it may be temporarily located until the position is boxed (i.e., assemblies present at all four faces in order to facilitate the seating), and the boxing may be done by moving assemblies on a temporary basis. Intermediate positions may also be used to support leaning assemblies, to accommodate a fuel inspection or to move a source which is contained within a particular assembly. Another use of intermediate positions is to optimize the movements needed to complete a refueling by shuffling fuel. The use of intermediate positions to optimize the refueling is not done in plants that perform a complete offload before refueling.

A concern arises if fuel assemblies with a high reactivity worth are being loaded and either unanalyzed intermediate positions are being used or mistakes are made in the loading of fuel. There is then the possibility that the configuration of fuel will lead to a loss of shutdown margin below the required amount or, in the extreme, to an inadvertent criticality.

This concern became apparent to Baltimore Gas and Electric (BG&E) after the fifth refueling of Calvert Cliffs 2. During that refueling 22 fuel assemblies were placed into unanalyzed non-conservative locations as a result of the shuffling scheme being used. The assumption had been made that the boron concentration was adequate for any configuration. That assumption had validity when the fuel enrichment was low but for Cycle 5 the fuel enrichment increased to above 4 w/o and there was the potential for losing the required shutdown margin.

After recognizing this problem BG&E, in March 1989, reported [1.1] the situation to the U.S. Nuclear Regulatory Commission (NRC) under 10CFR21 ("Reporting of Defects and Noncompliance"). Furthermore, BG&E took corrective actions: 1) They requested the fuel vendor [(Combustion Engineering (C-E)] to give them a list of the reactivity worth of fuel bundles in terms of k_∞ and 2) They changed their refueling procedures for the next loading to ensure that a fuel bundle would not be placed in an intermediate position during core alterations without first verifying its potential reactivity. Their intent was to require that fuel only be positioned in intermediate core locations which would contain fuel of equal or greater reactivity in the final core configuration.

Combustion Engineering, alerted to the problem, also took action via an Infobulletin to C-E plant owners in March 1989. Their recommendations included that either all intermediate positions be analyzed to assure adequate shutdown margin is maintained or that fuel only be positioned in intermediate core locations which would contain fuel of equal or greater reactivity in the final core configuration.

1.2.2 NRC Response

The NRC alerted PWR owners to the potential for losing the required shutdown margin, first with an Information Notice [1.2] in May 1989 and then with a Bulletin [1.3] in November 1989. In the Information Notice the NRC passed on to both PWR and boiling water reactor (BWR) plants the information they had received from BG&E and C-E, including the fact that simple calculations done at BG&E had shown the potential for an inadvertent criticality if a number of highly reactive fuel assemblies were clustered together and no credit was taken for control element assemblies or burnable poisons.

In the Bulletin the NRC only addressed PWR owners*. They requested that operating plants do the following:

"All PWR licensees are requested to assure that adequate shutdown margin is maintained during all refueling operations. This should be accomplished through the following actions:

1. Assure that any intermediate fuel assembly configuration (including control rods) intended to be used during refueling is identified and evaluated to maintain sufficient refueling boron concentration to result in a minimum shutdown margin of approximately 5%.

2. Assure that fuel loading procedures only allow those intermediate fuel assembly configurations that do not violate the allowable shutdown margin and that these procedures are strictly adhered to.

3. Assure that the staff responsible for refueling operations is trained in the procedures recommended in Item 2 above and understand the potential consequences of violating these procedures. This training should include the fundamental aspects of criticality control with higher enriched fuel assemblies."

As required by the Bulletin, all PWR owners have responded to the NRC by letter saying whether they have taken, or will take the actions requested above. Some of these responses included details of how the refueling procedures deal with the question of shutdown margin when intermediate positions are used. Some plants followed the C-E recommendation (cf Section 1.2.1). Other strategies for dealing with intermediate positions included: 1) the intermediate position must be next to the baffle, 2) it must have either a water hole (empty assembly position) or baffle on all four sides, 3) if the assembly is fresh it must have either control rods or more than 20 burnable poison rods. At least one utility had done calculations to show that any fuel assembly in an intermediate position that would be worth <u>more</u> than the assembly planned for that position would require an additional 100 ppm of boron and if this were true for a second assembly then an additional 200 ppm of boron would be required before the second intermediate position could be used. One scheme was to avoid more than 4 fresh fuel assemblies in a cluster. (This approach might be sufficient for that particular plant but as will be seen in Section 2 might not be sufficient for other plants.)

* The potential problems with misloading BWR fuel are sufficiently different so that NRC is treating the issue separately.

The NRC also requested that Brookhaven National Laboratory analyze different refueling configurations in order to understand the potential for losing shutdown margin and for inadvertent criticality due to either the use of intermediate positions or misloading of fuel.

1.3 Scope of Study and Outline of Report

The study consisted of three different types of analysis. The deterministic analysis is explained in Section 2. A neutronic model is used to calculate shutdown margin as a function of different parameters. The model is discussed in Section 2.1 and the results are discussed in Section 2.2. The results include the configurations which would result in losing the required shutdown margin or an inadvertent criticality. These configurations were then studied probabilistically in order to estimate the frequencies of occurrence as explained in Section 3. After some general remarks the key fault trees are explained in detail in Sections 3.2 and 3.3. Section 3.4 explains how these trees were quantified and the results are discussed in Section 3.5. This includes a discussion of a sensitivity study to examine human reliability in more detail. Some concluding remarks are given in Section 3.6. Section 4 explains the analysis of the dose to workers assuming an inadvertent criticality. This dose can be the result of radiation coming directly from the core (discussed in Section 4.1) or indirectly from fission products released from the affected fuel. References are found in Section 5.

2. SHUTDOWN MARGIN WITH DIFFERENT CONFIGURATIONS

2.1 Calculational Method

Calculations of shutdown margin were carried out using the three-dimensional reactor core code NODE-P2 [2.1]. This code models the neutronics with one energy group and a nodal method where k_∞ and M^2 are the basic neutronic data for each bundle. NODE-P2 has successfully been applied by many PWR licensees. Cycle 9 of Calvert Cliffs 2 [2.2] was chosen to be modeled with the code because of its use of reload fuel with a high U-235 enrichment (4.3 w/o).

The basic neutronics data needed for each fuel assembly found in Cycle 9 were generated using CASMO [2.3]. CASMO is a multigroup, two-dimensional, transport theory code for burnup calculations of light water reactor fuel assemblies. The code has been extensively validated. The data were generated for fresh fuel containing either 0, 4, 8, or 12 burnable poison rods (containing B_4C) and for four types of burned fuel from Cycle 8, each with a different enrichment and/or number of burnable poison rods. Each burned fuel assembly was assumed to be burned to the average exposure for that fuel type. Table 2-1 gives the enrichment, number of burnable poison rods, and burnup of each assembly type. An additional fresh fuel assembly for which data were generated had an enrichment of 4.08 w/o and was used to determine the effect of enrichment on shutdown margin.

To simplify the data generation two fuel types representing only 5 assemblies in Cycle 9 were not explicitly represented. A single burned assembly from Cycle 7 was represented as one of the bundles from Cycle 8 (Type J) and 4 erbia bearing demonstration assemblies were represented as fresh fuel with 4 burnable poison rods.

The assembly data were calculated with and without the presence of a CEA. There are two types of CEAs, each axially zoned differently with rods containing B_4C, Ag-In-Cd, Al_2O_3, or stainless steel. The assumption was made that all CEAs were identical and contained B_4C rods since more than 85% of the rods are of this composition. Part length assemblies were neglected. The data were calculated at a temperature of 311 K (100°F) and at boron concentrations of 1500 and 2000 ppm.

Table 2-1 gives the k_∞ calculated with CASMO for the assemblies with no control element assembly (CEA) at a boron concentration of 1500 ppm. The results show that even with 12 burnable poison rods a fresh fuel assembly can have a large reactivity worth.

The core layout for Cycle 9 as modeled is shown in Figure 2-1. There is octant symmetry. The number of burnable poison rods is noted on the figure for fresh fuel. The burned fuel, in the shaded locations, is identified by the fuel type designation given in Table 2-1. For both fuel types the presence of a CEA is noted. Figure 2-2 gives the distribution of k_∞ (as calculated by CASMO) with this layout at a boron concentration of 1500 ppm. The distribution of k_∞ with CEAs removed is also given on the figure.

The NODE-P2 model contains free parameters, related to the transport kernel and albedos, that are best obtained by normalizing results to measurements. Since none were available, the parameters were adjusted to get agreement with a more rigorous model; specifically the ROCS code. Using fuel assembly k_∞ values edited from a ROCS calculation of Calvert Cliffs 2, Cycle 5 with misloaded fuel, NODE-P2 calculations were done with parameters adjusted to get agreement with the k-effective and radial power (i.e., relative fission rate) distribution from ROCS. The results are given in Figure 2-3. The agreement is good especially considering the fact that the power distribution varies over two orders of magnitude.

The calculational uncertainty in k-effective for misloaded configurations of Cycle 9 is not known precisely because NODE-P2 has not been validated against measurements. Nevertheless, the model is reasonable and should be particularly useful for calculating changes in shutdown margin or relative values of k-effective. Based on an uncertainty analysis that was done for the ROCS code, the comparison of ROCS with NODE-P2, and a desire to be conservative, it is estimated that NODE-P2 may be calculating a k-effective that is 1-2% low.

2.2 Discussion of Results

The NODE-P2 calculations of k-effective for the Cycle 9 reloading were done with a boron concentration of 2300 ppm unless otherwise specified. The Technical Specifications at Calvert Cliff require either 5% shutdown or 2300 ppm whichever is more restrictive. With the nominal configuration for Cycle 9 this means that the reactor would be shut down with a margin of 13%. This is considerably greater than the required 5% primarily because it has generally been the philosophy of C-E plants to calculate the shutdown margin assuming all rods are out of the core. A calculation with all rods removed was done to show that the shutdown margin is reduced by 5.6% if there are no CEAs present.* The shutdown margin for Cycle 9 is also high because the requirement for 2300 ppm of boron is more conservative than the 5% requirement.

Calculations with misplaced fuel were done conservatively assuming that the misplaced fuel is fresh fuel with no burnable poison rods. Taking into account the reduced

* Note that this worth is less than expected at hot shutdown conditions with lower boron concentration.

worth of fresh fuel with burnable poisons would increase the number of assemblies needed for a problem to exist. The fresh fuel is assumed to not contain a CEA as that would reduce the reactivity worth of the assembly to that expected for highly burned fuel.

The change in shutdown margin with four different configurations of fresh fuel is given in Figure 2-4. The change depends not only on the reactivity worth and number of fresh bundles and their configuration, but also on the worth of the fuel that has been displaced.

These results are shown graphically on Figure 2-5 which plots k-effective vs the number of fresh fuel assemblies clustered together at the center of the core. The base case (middle curve) is for nominal conditions with 4.3 w/o fresh fuel and a boron concentration of 2300 ppm and corresponds to the results shown in Figure 2-4.. Along the x-axis are diagrams showing the actual configuration of the misloaded fuel assumed for the calculation. From the base case curve it is seen that the required shutdown margin is lost when 5 or more assemblies are clustered and criticality is almost possible when 9 assemblies are grouped together. Taking into account the uncertainty in the calculation discussed above, it will be assumed that for this loading only a 2x2 array is necessary to lose the required shutdown margin and a 3x3 array is necessary for an inadvertent criticality.

The results show that for this reload it would take 9 misloaded assemblies to have an inadvertent criticality. However, if the core initially only had the required shutdown margin of 5% then there is the possibility of an inadvertent criticality when only 5 assemblies are misloaded. This follows from the fact that the change in shutdown margin with this configuration is 7.8% (see Figure 2-4).

Figure 2-5 also shows results for a case in which the boron concentration is only 1800 ppm. The reactivity worth of boron varies from -0.9 to -1.2%/(100 ppm) and the curve with the reduced boron concentration is displaced from the base case by approximately 5% reactivity.

Also shown on Figure 2-5 is the effect of U-235 enrichment. The lower curve was obtained using fuel with 4.08 w/o enrichment. The results show that the effect of enrichment increases as more fresh fuel is loaded.

Figure 2-6 shows similar results to those shown in Figure 2-5 except that the calculations were done with all CEAs removed. As expected, the results are displaced by 5.6% in reactivity giving much more conservative results. The changes in shutdown margin are smaller with fewer neutron absorbers in the core. This can best be seen by comparing the changes in shutdown margin given in Figure 2-7 with no CEAs with the changes given in Figure 2-4 with CEAs present.

In Figure 2-8 the change in shutdown margin is given for different configurations that assume quadrant symmetry and with the condition that no CEAs are present. It is expected that errors leading to these configurations are possible during a shuffling when fuel moves may be done symmetrically rather than during an offload/reload. The effect of having the same pattern of either 1, 3, or 5 clustered fresh fuel bundles in all four quadrants is not four times the effect of the same clustering at the center of the core as shown in Figure 2-7. This is because different fuel bundles are being displaced when different locations are used and because the coupling of the effect between quadrants is limited when there are gaps between the clusters; especially in this case where these gaps contain burned fuel and a high boron concentration.

Figure 2-8 also shows the effect of moving the cluster to the core periphery. For the case of a cluster of 5 fresh fuel assemblies the effect on shutdown margin is reduced by almost one-half.

An inadvertent criticality would require the misloading of many assemblies and, as will be shown in Section 3 this is highly unlikely. However, if this were to occur any power excursion would be autocatalytic initially. This is because in a highly borated condition the moderator feedback is positive.

In order to quantify this effect, calculations of k_∞ were carried out for a 4.08 $_{w/o}$ fresh fuel assembly with no CEA or burnable poison. The calculations were done at a boron concentration of 2000 ppm with a base case temperature of 100°F. Over an additional 100°F the moderator temperature reactivity coefficient was +8.6 pcm/°F. Over the range from 0 to 40% void fraction the void reactivity coefficient is +130 pcm/(% void). The Doppler reactivity coefficient is relatively small (-1.6 pcm/°F) and hence, as the power increased and the moderator heated and voided, the excursion would become more severe.

Table 2-1

Fuel Types in Calvert Cliffs 2 Cycle 9
(1500 ppm Boron, No Control Rods)

Fuel Type	Enrichment w/o	Burnable Poison Rods	Burnup MWD/MT	k_∞
J	4.05	0	30	0.895
K	4.08	0	0**	1.186
K	4.08	0	15	1.023
K/	4.08	8	21	0.953
K*	4.08	12	21	0.943
L	4.30	0	0	1.202
LX	4.30	4	0	1.157
L/	4.30	8	0	1.109
L*	4.30	12	0	1.065

** Not in Cycle 9 but used in analysis

Figure 2-1

DISTRIBUTION OF FUEL WITH BURNABLE POISON RODS (BP) AND CONTROL ELEMENT ASSEMBLIES (CEA)
Calvert Cliffs Unit 2 Cycle 9 Refueling

							J	0 BP
				J	0 BP CEA	K	4 BP CEA	K/
			J	4 BP CEA	K	12 BP CEA	K-	K/ CEA
		J	4 BP CEA	K/	12 BP CEA	K-	8 BP CEA	K-
	J	4 BP CEA	K/	4 BP CEA	J	8 BP CEA	K/	J
	0 BP CEA	K	12 BP CEA	J	12 BP	K/	12 BP	K-
	K	12 BP CEA	K-	8 BP CEA	K/	K/ CEA	K/	8 BP CEA
J	4 BP CEA	K-	8 BP CEA	K/	12 BP	K/	8 BP CEA	K-
0 BP	K/	K/ CEA	K-	J	K-	8 BP CEA	K-	J CEA

Figure 2-2

K-INF DISTRIBUTION NOMINAL (AND UNRODDED)
Calvert Cliffs Unit 2 Cycle 9 Refueling With 1500 ppm Boron

							.89	1.20
				.89	.99 (1.20)	1.02	.95 (1.16)	.95
			.89	.95 (1.16)	1.02	.87 (1.07)	.94	.78 (.95)
		.89	.95 (1.16)	.95	.87 (1.07)	.94	.92 (1.11)	.94
	.89	.95 (1.16)	.95	1.15	.89	.92 (1.11)	.95	.89
	.99 (1.20)	1.02	.87 (1.07)	.89	1.07	.95	1.07	.94
	1.02	.87 (1.07)	.94	.92 (1.11)	.95	.78 (.95)	.95	.92 (1.11)
.89	.95 (1.16)	.94	.92 (1.11)	.95	1.07	.95	.92 (1.11)	.94
1.20	.95	.78 (.95)	.94	.89	.94	.92 (1.11)	.94	.74 (.89)

Figure 2-3

POWER DISTRIBUTION BENCHMARK

Calvert Cliffs 2 Cycle 5 Refueling

	Keff
ROCS	0.985
NODEP-2	0.982

						0.01 0.00	0.00 0.00	
				0.08 0.06	0.05 0.05	0.03 0.03	0.02 0.01	0.02 0.01
			0.09 0.08	0.34 0.29	0.22 0.23	0.15 0.15	0.08 0.05	0.11 0.07
		0.09 0.07	0.28 0.28	0.71 0.81	.0.96 1.08	0.70 0.80	0.34 0.34	0.22 0.22
	0.05 0.04	0.30 0.24	0.67 0.73	3.01 3.01	4.04 4.08	3.07 3.10	0.78 0.88	0.67 0.59
	0.08 0.04	0.23 0.20	0.86 0.78	4.10 4.07	4.93 5.16	4.12 4.12	0.99 1.12	0.40 0.43
	0.08 0.03	0.46 0.37	0.93 0.98	3.72 3.63	4.36 4.26	3.19 3.14	0.78 0.86	0.30 0.30
0.02 0.01	0.11 0.10	0.59 0.45	0.85 0.87	2.42 2.19	0.91 1.07	0.78 0.83	0.57 0.47	0.13 0.13
0.02 0.02	0.10 0.08	0.35 0.33	1.25 1.07	2.00 1.76	0.66 0.63	0.32 0.30	0.13 0.13	0.11 0.07

Figure 2-4
Change in Shutdown Margin (SDM) With Fresh Fuel Clusters (Calvert Cliffs 2, CY9 Refueling)

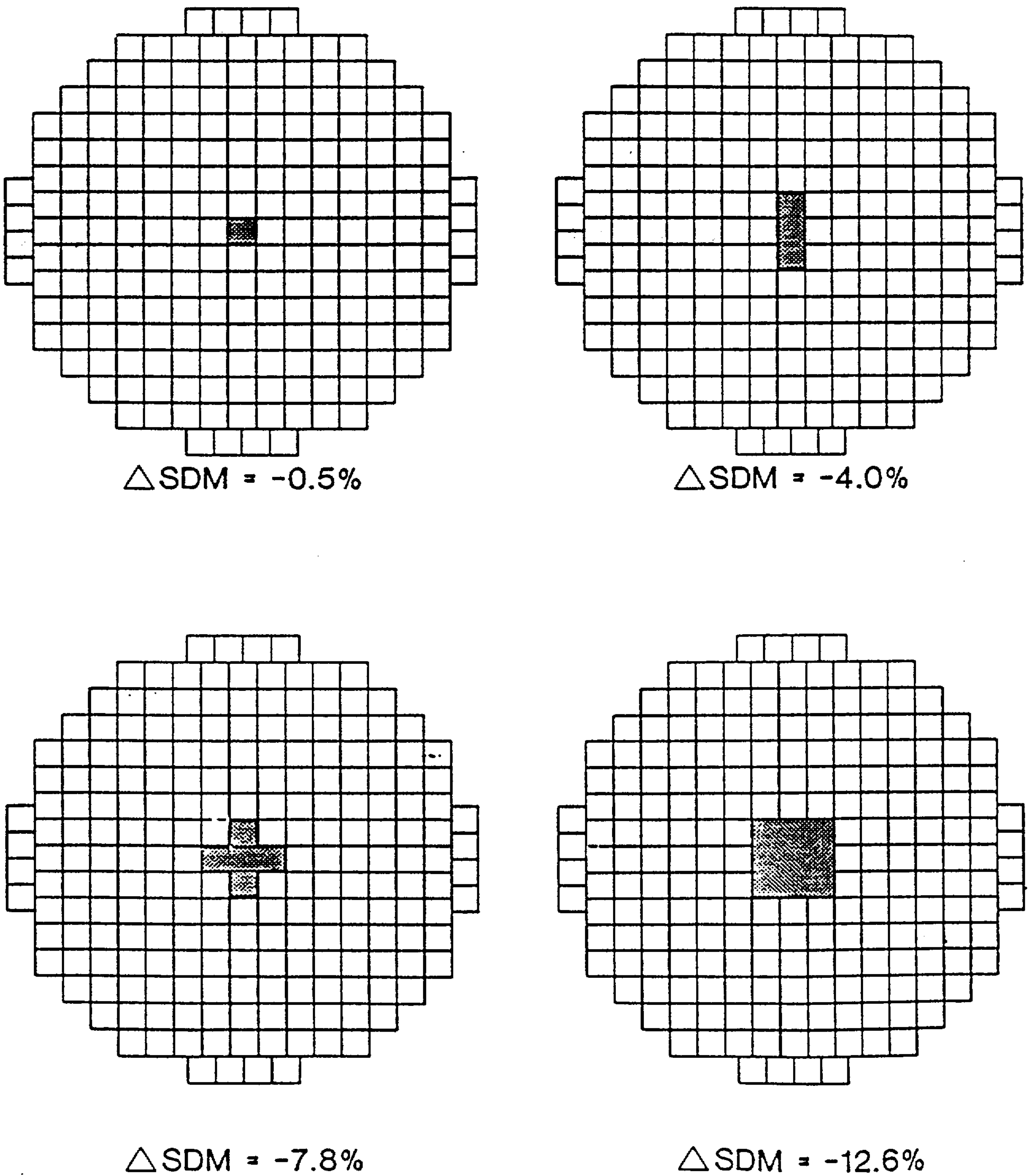


Figure 2-5
K-EFFECTIVE FOR MISLOADED CONFIGURATIONS

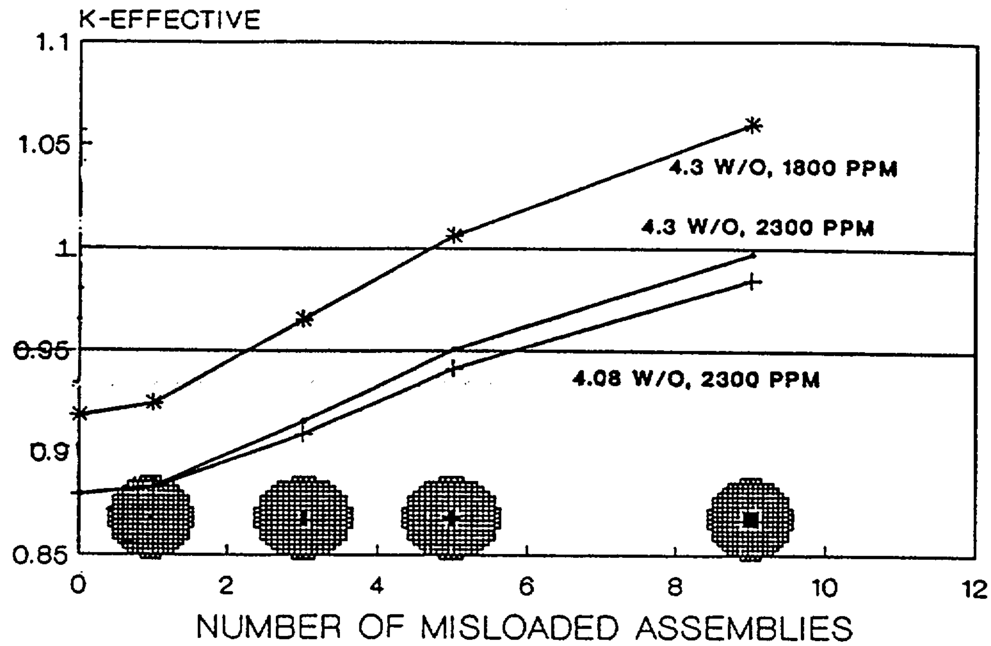


Figure 2-6
K-EFFECTIVE FOR MISLOADED CONFIGURATIONS
ALL CEAs REMOVED

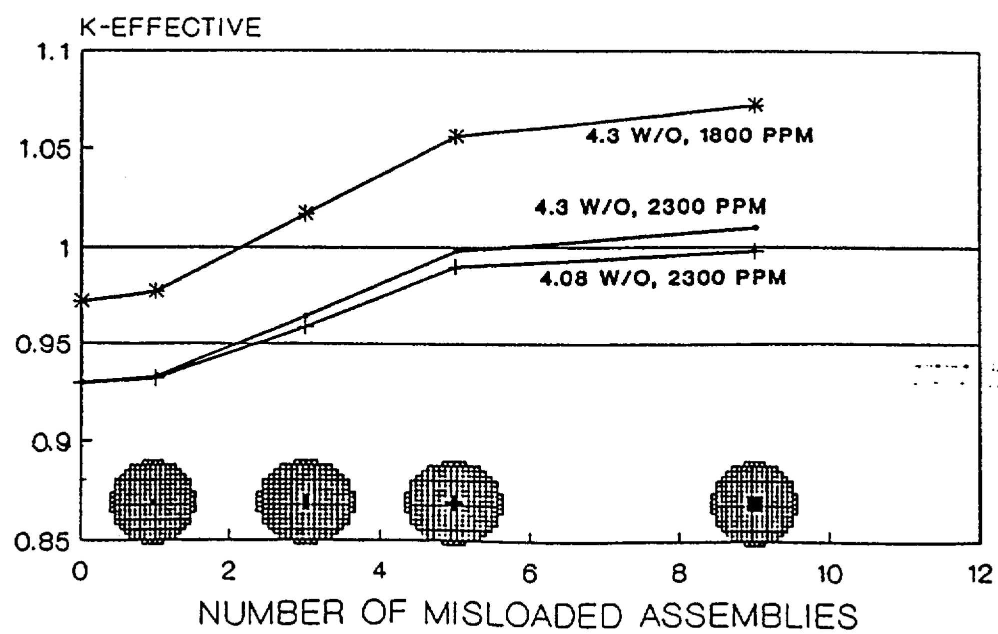
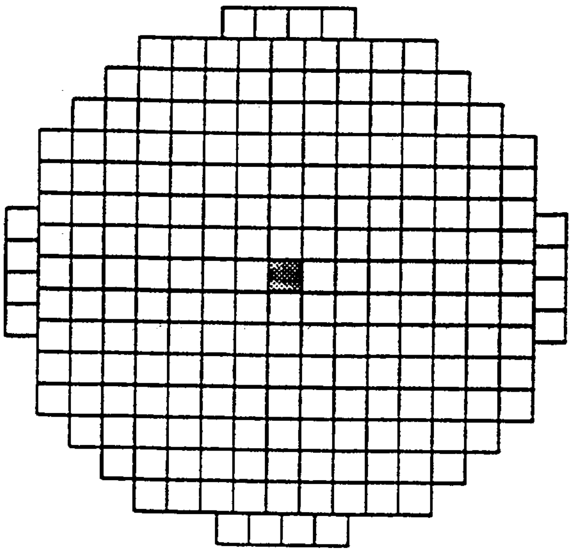
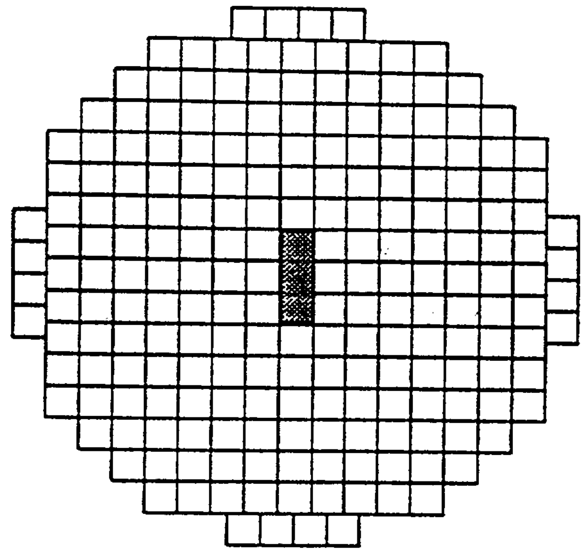


Figure 2-7

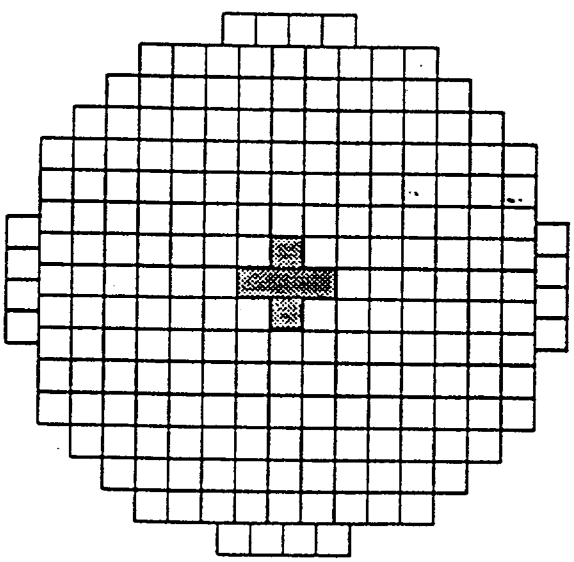
Change in Shutdown Margin (SDM)
With Fresh Fuel Clusters (All CEAs Removed)



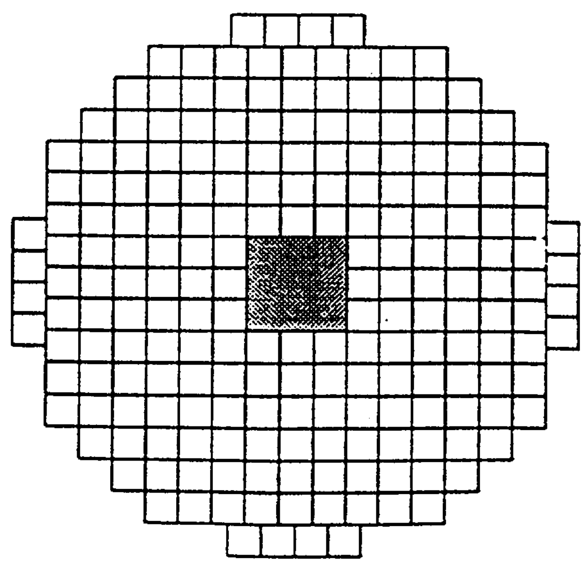
ΔSDM = -0.2%



ΔSDM = -3.5%



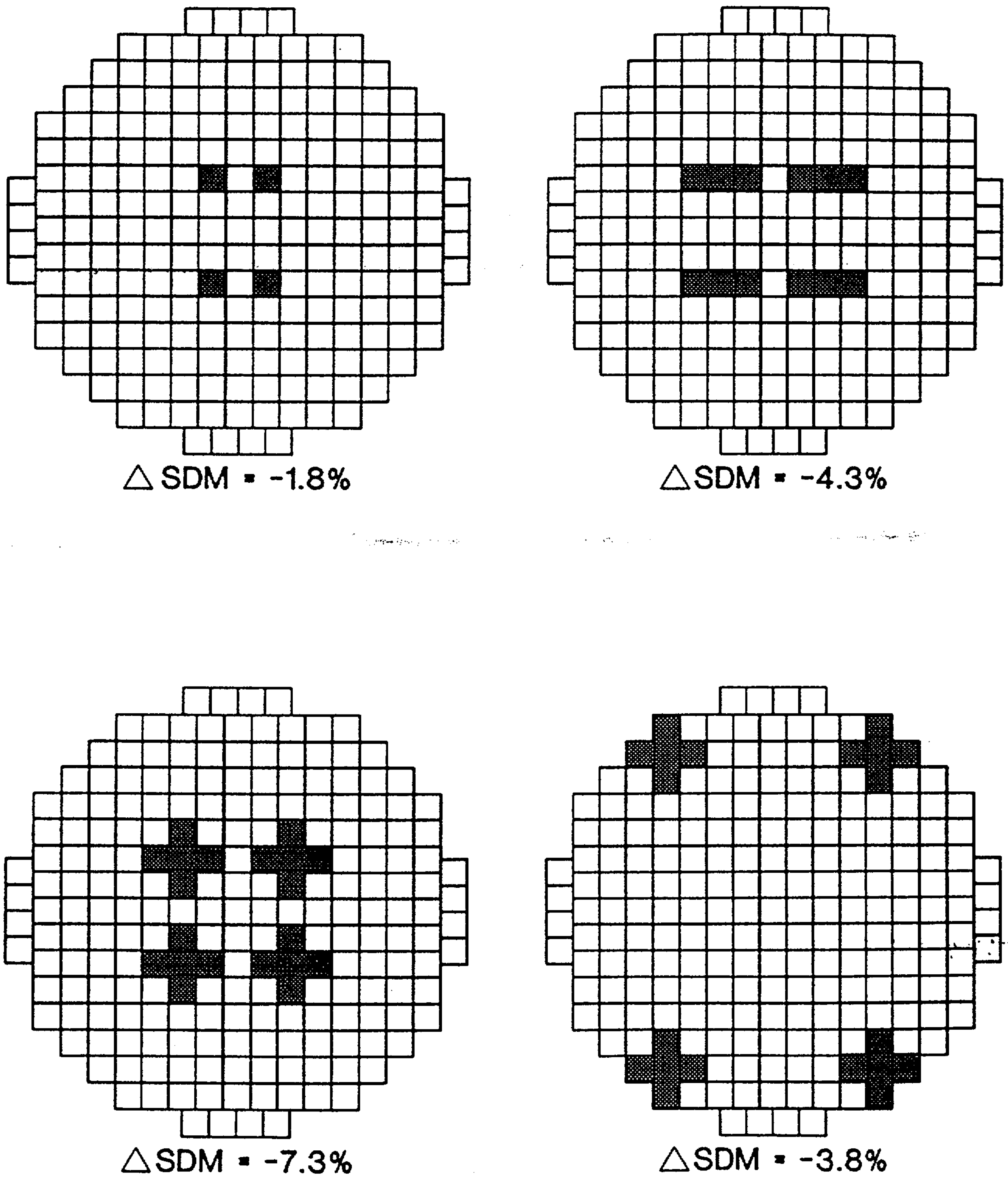
ΔSDM = -7.0%



ΔSDM = -8.1%

Figure 2-8

Change in Shutdown Margin (SDM)
With Fresh Fuel Clusters (All CEAs Removed)



3. PROBABILISTIC ASSESSMENT OF MISLOADED CONFIGURATIONS

3.1 Introduction

The probabilistic assessment of the frequency of losing shutdown margin (and perhaps becoming critical) during PWR refueling was based on the current loading procedures for Calvert Cliffs Unit 2 (October 1990). These procedures entail a complete offload rather than shuffling; a change made subsequent to the reporting to NRC of potential problems with intermediate positions during shuffling (cf Section 1.2.1). The present scheme can be summarized as follows:

Each fuel assembly in the core is first removed and placed in the spent fuel pool. This involves first moving the fuel out of the core with the refueling machine. It is placed in the upender which turns it from a vertical to horizontal orientation. The upender cart is then moved through the fuel transfer canal into the spent fuel building where an upender turns it back into a vertical orientation. The spent fuel handling machine then lifts it and places it into position in the spent fuel pool. After the fuel has been moved into the pool the spent fuel handling machine is used to remove CEAs from assemblies where they are not needed and place them into assemblies that are to contain a CEA for the next cycle. A single operator has responsibility for operating this machine. When all movement in the spent fuel pool is completed a check is made of all positions to assure that the correct fuel assembly and CEA are present.

The reloading starts with the control room operator specifying the fuel assembly location in the spent fuel pool for transfer back into the reactor containment building. The spent fuel handling machine operator moves to that grid location and the assembly is moved through the fuel transfer canal into containment. The refueling machine operator also receives instructions from the control room operator specifying the core grid location for that assembly. The refueling machine operator reads the bridge and trolley indices corresponding to that grid location and moves the machine accordingly. A second operator checks the movement and the control room operator is kept appraised of the location of the assembly. The fuel assembly is lowered into position and when the assembly is in place the tag board which records the location of each fuel assembly is updated and count rates are observed. Changes to this sequence that affect the destination of an assembly (e.g., temporarily moving to an intermediate location) cannot be made by the operator without proper approval.

The final core layout for Cycle 9 of Unit 2 showing the positions of both fresh and burned fuel assemblies, with and without CEAs, is depicted in Figure 3-1 (cf Figures 2-1 and 2-2). The core contains a total of 217 fuel assemblies of which 92 are fresh and the remainder (125) burned fuel assemblies. The different types of fresh fuel assemblies are listed below using the nomenclature of Figure 3-1.

Type of Fresh Fuel Assembly	Number of Assemblies
FU	8
FUB	12
FC	8
FCB	64
Total	92

The fuel assemblies of FU or FUB type do not have CEAs. They are uncontrolled and have the potential of causing loss of shutdown margin if improperly arranged as a group. Note that a fuel assembly of FC(FCB) type will become FU(FUB) type if a CEA fails to be attached to it due to CEA repositioning errors.

Based on the shutdown margin calculations for different configurations, discussed in Section 2, it was determined that significant margin will be lost if a group of four fuel assemblies of FU or FUB type are placed in a 2x2 array, or a group of five fuel assemblies of FU or FUB type are placed in a cross-shaped configuration. These clusters would be important anywhere in the core except at locations adjacent to the baffle where neutron leakage would reduce the reactivity.

The first step in the probabilistic analysis was to calculate the probability that a given core location may be loaded with a fuel assembly of either FU or FUB type. Note that there are 48 core locations (shaded area in Figure 3-1) which are adjacent to the baffle. These locations need not be analyzed, because no significant loss of shutdown margin would occur if they are loaded with fuel assemblies of FU or FUB type. Note also that all the FU and FC types of fuel assemblies are meant to be loaded in these areas. There are an additional twelve core locations, that do not have to be considered because they are supposed to be loaded with fuel assemblies of FUB type. The probability that such a location contains fresh fuel is close to unity and does not have to be determined using fault tree analysis. The number of core locations misloading a fuel assembly is, therefore, 217-48-12 = 157.

Based on the final loading pattern shown in Figure 3-1, these 157 core locations can be divided into two groups, one to be loaded with fuel assemblies of FCB type, and the other with burned fuel assemblies. For convenience, they are denoted as core locations of "A" type and "B" type, respectively. Fault trees were constructed to calculate the probability that a core location of "A" type or "B" type will be mistakenly loaded with an uncontrolled fresh fuel assembly, i.e., a fuel assembly of either FU type or FUB type. The probability of forming a cluster of such misloaded locations that might lead to loss of shutdown margin is then calculated. Intuitively, the probability of loading a core location of "A" type with an uncontrolled fuel assembly can be expected to be larger because location "A" is designated for loading a fresh fuel assembly of FCB type. Errors in CEA repositioning alone can cause location "A" to be loaded with an uncontrolled fuel assembly, even though the loaded fuel assembly has a correct ID number. Loading a location of "B" type with an uncontrolled fuel assembly, on the other hand, necessarily requires misloading of a fuel assembly with an incorrect identification (ID) number, since it is designated for loading a burned fuel assembly.

3.2 Fault Trees for Misloading a Fuel Assembly Location

3.2.1 Loading FU(FUB) Fuel into a Core Location of "A" Type

The fault trees, shown in Figures 3-2a through 3-2e, were developed to calculate, given a core location of "A" type is loaded with a fuel assembly, the probability that the loaded fuel assembly is a fresh assembly without a CEA attached. To simplify the analysis, a few major assumptions are made. They are: (1) Errors associated with a) loading or unloading a fuel assembly in a core location, b) storing a burned or fresh fuel assembly in a spent fuel pool location, and c) repositioning of a CEA between the fuel assemblies, are considered to occur completely randomly, independent of the order of procedures, or the positions of the fuel assembly or the CEA, or the empty space in the core, or in the spent fuel pool; (2) Loading or unloading errors which may be detected by means of inconsistency during the loading or unloading operations are neglected; and (3) No common-cause errors, human or non-human, are considered.

With regard to the first assumption note that if a fuel assembly is misloaded into a wrong location, it is conceivable that this wrong location is likely to be in the vicinity of the correct location. This is because a loading error can occur simply as a result of mispositioning the refueling machine. In this study, however, it is assumed that any empty location in the core has an equal chance of becoming the wrong location. The second assumption is very conservative. If a loading error has been committed by placing a wrong fuel assembly into a core location, it is almost certain that the error will be eventually detected when it comes to load the same location with a correct fuel assembly. Due to the complex nature of the possibility for such an occurrence, no credit was given to the detection of errors during the process of loading or unloading fuel assemblies or during CEA repositioning. For the third assumption, it should be noted that since movement of a fuel assembly from one position to another over the core usually requires more than one operator to be involved, the probability of making common-cause human errors is considered very small. Movement of a fuel assembly or a CEA over the spent fuel pool, on the other hand, is usually done by a single operator following procedural instructions. Any common-cause human errors, including those which may be committed by a single spent fuel machine operator, however, were not considered in this analysis.

The fault trees shown in Figures 3-2a through 3-2e were developed based on the following core unloading and loading scenarios: Initially, the core is entirely filled with burned fuel assemblies. The core is unloaded by removing all the burned fuel assemblies and placing them in proper spent fuel pool locations. Repositioning of the CEAs among the fuel assemblies (FAs), fresh or burned, is then carried out in the spent fuel pool. After the CEA repositioning is completed, the empty core is loaded, one FA at a time, following a pattern that involves loading the core row-wise from one end of the core to the other. One pattern for loading the core like this would follow the numbering of assemblies as shown in Figure 3-1.

Before describing the detailed structure of the fault trees, it is necessary to define a few terms. Throughout this report, a fuel assembly with a correct ID number implies that the fuel assembly is the one specified in the unloading or loading procedures. A fuel assembly without a correct ID number is called a "wrong" fuel assembly. Also, if the core or the spent fuel pool location does not correspond to the one specified in the procedures, it is called a "wrong" location.

To gain a clear perspective of the overall structure of the fault trees, it is worthwhile examining the possible outcomes of an event involving the movement of a fuel assembly from one location to another following a specified procedure. In general, there are four possible outcomes when a fuel assembly is moved by the operators from the spent fuel pool to the core based on a written procedure. They are:

1. A FA with a correct ID number is picked up at the spent fuel pool, and loaded into the correct core location.

2. A FA with a correct ID number is picked up at the spent fuel pool, but loaded into a wrong core location.

3. A FA with a wrong ID number is picked up at the spent fuel pool, but loaded into the correct core location.

4. A FA with a wrong ID number is picked up at the spent fuel pool and loaded into a wrong core location.

Outcome 1 is the ideal case that can be expected to occur most frequently. In structuring the fault trees, attention was focussed on outcomes 2, 3, and 4. In general, the probability of outcome 4 can be expected to be very much smaller compared to that of outcome 2 or 3, since two mistakes have to be made rather than one.

Referring to the fault tree shown in Figure 3-2a, the fault tree top event, a new FA without a CEA is loaded into location A, can occur if a new FA without a CEA, having a correct or incorrect ID number, is loaded into a location of "A" type. If the FA loaded into location A has a correct ID number, an error must have been committed during the course of CEA repositioning, since a location of "A" type is supposed to be loaded with a new fuel assembly with a CEA attached. The logic of failing to attach a CEA to a FA as required by procedures during CEA repositioning is further depicted by the sub-fault tree, A1 (see Figure 3-2e).

For a complete offload/reload refueling operation, the CEAs are usually repositioned in the spent fuel pool. The CEA repositioning is performed by personnel qualified to operate the spent fuel handling machine. Unlike the situation when loading the core, CEA movements in the spent fuel building are made by a single operator. Personnel errors that can occur during CEA repositioning are modeled in the sub-tree, A1. Failures to verify the position of each CEA using a video camera and monitor prior to refueling the reactor are also modeled.

If a new FA without a CEA and having a wrong ID number is loaded into location A, two events must be concurrently true: (a) a FA with a wrong ID number is loaded into location A; and (b) the wrong FA happens to be new and without a CEA. The logic for each of these two events is further developed in the fault tree. For convenience of discussion, attention is first directed to event (b). In order for the wrong FA loaded into location A to be new and without a CEA, the FA has to be either (i) of FU or FUB type, or (ii) of FC or FCB type, but without the CEA having been attached during CEA repositioning operations. The logic of these events is delineated in the fault tree branch under the heading of "The wrong FA is new and without CEA". Note that the probability of the event, "The FA is of FU or FUB type", or the event, "The FA is of FC or FCB type", varies as the loading operation progresses since the number of fuel assemblies which have been loaded into the core, and that which remain to be loaded changes at each loading step. In other words, the probabilities of these two events change depending on the core location being loaded.

Next, for the event (a) to be true, it must be either (i) a FA intended for another core location is mistakenly loaded into a particular core location of A type during the course of loading the core locations other than the particular location being considered, or (ii) a wrong FA has been picked up at the spent fuel pool and loaded into core location A when the step of loading the particular core location is taken. These two types of errors can be considered to correspond respectively to the outcomes 2 and 3 mentioned above. They are discussed separately in the following subsections.

A) A Fuel Assembly Intended for Another Core Location is Mistakenly Loaded into Location A.

The logic of this event is developed in the sub-fault tree B1 see (Figure 3-2b). This type of error occurs not during the course of loading the particular location being considered, but during the course of loading locations other than the one being considered. For the reactor core being considered, there are 217 core locations which must be loaded with fuel assemblies during the entire refueling operation. The refueling procedure, which specifies the initial location and final destination of each of the 217 fuel assemblies, therefore, consists of 217 basic refueling steps. For each core location, there is only one correct fuel assembly. It is possible, however, that during the multiple refueling steps, a fuel assembly intended for another location is mistakenly loaded into core location A. Note that the wrong fuel assembly could be mistakenly loaded into any empty core location. The event, "The wrong core location is location A" appearing in the sub-tree B1, under an "And" gate, is used to restrict the location to the particular location being considered. The probability of this event, therefore, varies depending on the core location and the order of refueling.

To discuss possible causes of loading a fuel assembly into a wrong core location, a brief description is first given on how the refueling operation is actually carried out in a typical Combustion Engineering plant.

Any movement of a fuel assembly during refueling is directed by a control room operator, who maintains overall coordination of the refueling activities. As required by plant Technical Specifications, the control room operator maintains direct communication with anyone moving the fuel. Core alterations are suspended whenever these communications cannot be maintained. Based on the refueling procedures, the control room operator directs the fuel handling operator to move the refueling machine to a specific grid location, which is expressed in terms of a letter and number (e.g., A-5). The refueling machine operator has a printed table of grid locations vs. index locations. He looks up the grid location, then positions the machine based on the appropriate index location, read to the nearest one hundredth of an inch (e.g., bridge = 400.12 in., trolley = 295.63 in.). The refueling machine operator verifies the machine's position with the control room operator. If fuel is to be removed, the control room operator instructs the refueling machine operator to pick up the fuel assembly. At this time, the control room operator signs the procedure step and updates the tag board indicating that a fuel assembly is located in the fuel handling machine mast. In the reverse situation where fuel is inserted, the same process of instructions from the control room operator and verifications from the fuel machine operator is performed. After the fuel assembly is inserted, the procedure is signed off and the tag board is updated.

The potential for mistakenly loading a fuel assembly into a wrong core location exists in several areas. The refueling machine operator could incorrectly position the machine as a result of communication errors with the control room operator, looking up a wrong index location, or mentally transposing the bridge and trolley locations. The logic is delineated in the sub-fault tree, E1 (see Figure 3-2b). Note that failures in verifying the machine position by a second bridge operator prior to removing or inserting an assembly is also modeled. The control room operator could also commit errors by misreading the procedures and thereby giving wrong instructions to the fuel handling machine operator. One contributing factor to this type of error is failure of the control room operator to update the procedure and tag board. It should also be remarked that the control room operator is allowed to change the order of portions of the refueling procedures without additional approval. This change of orders involves skipping forward or backward in the procedures and performing a number of sequential steps to accomplish an assembly transfer. The operator, however, is not permitted to alter the order of individual steps, as this could lead to misloading of an assembly.

These possible human errors by the control room operator are modeled in the sub-fault tree, G1 (see Figure 3-2d). Another possible, but less likely cause of mis-positioning the refueling machine is the mechanical malfunction of the machine itself, which is modeled in the sub-fault tree, F1 (see Figure 3-2b).

B) A Fuel Assembly with a Wrong ID Number is Picked up at the Spent Fuel Pool and Loaded into Location A.

One possible cause of picking up a wrong FA at the spent fuel pool is through human errors committed by the spent fuel machine operator. Unlike the movement of fuel assemblies over the core, which requires supervision and direction of a control room operator, FA or CEA movements in the spent fuel building is usually done by a single operator. This is because these activities do not directly affect the reactivity of the core.

The personnel involved in the process are not required to be in direct communication with the control room, and a senior reactor operator need not be present on location. Generally speaking, positioning the spent fuel machine is less complicated than positioning the refueling machine. It can be accomplished by aligning the bridge and trolley index pointers to the appropriate letter or number position markers. The spent fuel machine operator, however, must read the procedural instructions alone and carry out the movements accordingly. Possible errors that can be committed by the spent fuel machine operator are modeled in the fault tree.

Another possible cause of picking up a wrong FA at the spent fuel pool is that, although the spent fuel machine operator did move the machine to the correct spent fuel pool position, a wrong FA has been stored in the specified location without being detected. Prior to loading the core, the position of each FA and CEA is verified using a video camera and monitor. The position of each FA and CEA is checked to ensure that it matches the position documented in the refueling procedures. Any discrepancies uncovered are resolved prior to initiating core loading. Possible human errors that can lead to failure of position verification are modeled in the sub-fault tree denoted by C1 (see Figure 3-2c). Note that the wrong FA stored in the specified pool location can be either fresh or burned. If the wrong FA is fresh, a storage error must have been committed when new fuel assemblies are stored. Normally, new fuel assemblies are shipped by truck to the site in new fuel casks. The casks are moved to the fuel storage area where the assemblies are removed and inspected by qualified personnel. After the inspection, each new FA is placed in an appropriate spent fuel pool location. Since only one operator is required for FA movements in the spent fuel pool, the new FA could be stored in a wrong place if the procedures are incorrect or the operator carries out the written instructions incorrectly.

If the wrong FA stored in the specified spent fuel pool location is a burned one, a storing error must have been made when the burned FAs were unloaded from the core and stored in the spent fuel pool, or when shufflings in the pool, if required, are made subsequent to the unloading. This logic is depicted by the sub-fault tree, D1 (see Figure 3-

2d). The storing error can occur if a wrong FA is picked up at the core during the unloading and stored in the specified spent fuel pool location. Possible causes of picking up a wrong fuel assembly at the core during unloading are essentially identical to those discussed above for loading the core. They include: (i) control room operator specifies a wrong core location; (ii) refueling machine operator incorrectly positioned the machine without being detected; and (iii) refueling machine is incorrectly positioned due to mechanical errors. They are modeled respectively by sub-fault trees, G1, E1 and F1.

Another possible cause of the storing error at the spent fuel pool is that, although a correct FA was picked up at the core, it was mistakenly stored in a wrong pool location. Note that the wrong pool location can be any empty location in the pool. Since attention is being focussed on a particular pool location, it is phrased in the fault tree as "an old fuel assembly intended for another spent fuel pool location is mistakenly stored in the specified pool location." The logic is depicted by the sub-fault tree, H1 (see Figure 3-2e), which shows that human errors attributable to the spent fuel machine operator are mainly responsible for the occurrence of this event.

3.2.2 Loading FU(FUB) Fuel into a Core Location of "B" Type

Supporting logic for mistakenly loading a fresh fuel assembly with no CEA attached into a core location of "B" type is delineated in the fault trees shown in Figures 3-3a and 3-3b. Since core locations of "B" type are supposed to be loaded with burned fuel assemblies, the fresh fuel assembly mistakenly loaded into a location of B type necessarily has an incorrect ID number. The structure of the fault trees is otherwise essentially identical to that described in Section 3.2.1 above.

3.3 Fault Trees for Losing Required Shutdown Margin

To estimate the probability of losing the required shutdown margin during a PWR refueling operation, a fault tree (in abbreviated form) was developed as shown in Figure 3-4. As discussed in Section 3.1, if a cluster of four (2x2 array) or five (cross configuration) fresh fuel assemblies without CEAs are placed anywhere in the core, except at locations adjacent to the baffles, the required shutdown margin is assumed to be lost. To calculate the probability of losing the required shutdown margin, therefore, it is necessary to identify all the possible combinations of core locations to form such clusters. Referring to Figure 3-1, for example, a combination of four core locations, such as locations 8, 9, 18, 19 and a combination of five core locations, such as locations 17, 28, 29, 30 and 43 form a cluster of a 2x2 array and a cluster of a cross configuration, respectively. There are a total of 140 combinations possible to form a cluster consisting of a 2x2 array, and 129 combinations possible to form a cluster consisting of a cross configuration. For simplicity, they are represented by the single events, CLUS4 and CLUS5 respectively in the fault tree shown in Figure 3-4.

As can be seen from Figure 3-4, event CLUS4 is related, through an "AND" gate, to four input events, CLOCA, CLOCB, CLOCC and CLOCD, which denote that core locations, designated by core location indices "a", "b", "c" and "d", are respectively are loaded with a fresh fuel assembly without a CEA. For the example of a 2x2 array cited above, the core location indices can be taken as a=8, b=9, c=18, d=19. The fault-tree event, CLUS4, therefore, is an abbreviated expression used to encompass all of the 140 possible 2x2 array combinations, with the core location indices, "a", "b", "c" and "d" specified to correspond to the core locations for each combination that can lead to loss of required shutdown margin.

Similarly, the event CLUS5 is related, through an "AND"" gate, to five input events, CLOCR, CLOCS, CLOCT, CLOCU and CLOCV, which denote that core locations designated by core location indices "r", "s", "t", "u" and "v", respectively are loaded with a fresh fuel assembly without a CEA. For the example of a cross configuration cited above, the core location indices can be taken as r=17, s=28, t=29, u=30 and v=43. The fault-tree event, CLUS5, therefore, is an abbreviated expression used to represent all of the 129 possible combinations to form a problem cluster of five fuel assemblies, with the core location indices, "r", "s", "t", "u" and "v" specified to correspond to the core locations for each of the combinations.

Since each combination of core locations that can form a problem cluster is unique and mutually exclusive, the probability of forming a cluster of four or five so as to lose the required shutdown margin (i.e., the probability of the fault tree top event, SDMLOST) can be computed by adding together the probabilities of all the possible combinations. This logic is delineated by the fault tree shown in Figure 3-4.

The probability of each combination (i.e.,the probability of CLUS4 or CLUS5) can be calculated by multiplying together the probabilities of the relevant input events, i.e., the probabilities of CLOCA, CLOCB, CLOCC and CLOCD if the cluster consists of a 2x2 array, and the probabilities of CLOCR, CLOCS, CLOCT, CLOCU and CLOCV if the cluster consists of a cross configuration. This approach is consistent with the assumption made in this study that each of these input events occur independently of one another.

The probabilities of the input events, CLOCA, CLOCB, CLOCC or CLOCD, can be obtained by quantifying the fault trees shown in Figures 3-2 (a through e) and 3-3 (a and b), by first discerning whether the core location is of A-type or of B-type. If the core location of interest is of A-type, the probability of the input event is taken to be the probability of the top event, TOPA, of the fault tree shown in Figure 3-2a. On the other hand, if the core location of interest is of B-type, it is taken to be the same as the probability of the top event, TOPB, of the fault tree shown in Figure 3-3a.

It should be remarked that, even with core locations of the same type, the probability of being mistakenly loaded with a fresh fuel assembly without a CEA could be somewhat different depending on where in the loading pattern one was focusing. This is primarily due to the existence in the fault trees (see Figures 3-2a, 2b and 2e) of four basic events, FATYPE2, FATYPE3, ALOC and SPLOC, whose probabilities vary depending on the order of loading the fuel assemblies. In other words, the probabilities of these four basic events are core-location dependent. A detailed explanation of how to compute the probabilities of these four basic events, by taking into consideration the order of refueling, will be given in Section 3-4. A simplified approach is discussed below.

Evaluating the probabilities of the input events, such as CLOCA, or CLOCB, requires quantifying the fault trees shown in Figures 3-2a through 3-2e or Figures 3-3a and 3-3b for each of the relevant core locations. This, however, would entail considerable effort. The burden of quantifying the fault trees, however, can be significantly alleviated by employing, for each of the above four basic events (whose probabilities actually depend on core location), a probability averaged over the entire 217 core locations, or a subset of core locations. The frequency of losing the required shutdown margin was calculated by quantifying the fault trees using, for each of those four basic events, its probability obtained by averaging over the entire core locations. Similar fault tree quantifications were also performed by using the probabilities obtained by averaging over every twenty core locations. The latter approach was found to yield results that are about 20% lower than those obtained using the former approach, indicating that the former approach is relatively conservative. This study, therefore, used for each of those four basic events, its probability obtained by averaging over the entire 217 core locations.

The simpler approach taken in quantifying the fault trees has one important implication that must be specifically noted. Since only one probability averaged over the entire 217 core locations is used for each of the four basic events (whose probabilities are actually core location dependent), all core locations of A-type, regardless of where the actual core location is, will have the same probability of being mistakenly loaded with a fresh fuel assembly without a CEA. Similarly, all core locations of B-type will have the same probability of being misloaded with a fresh fuel assembly without a CEA. This important consequence greatly simplifies the process of evaluating the frequency of losing the required shutdown margin.

As explained previously, the event, CLUS4, appearing in the fault tree shown in Figure 3-4 represents all the possible combinations of four core locations to form a problematic cluster of a 2x2 array. To calculate the probability of CLUS4, therefore, each of the combinations must be identified with regard to its four associated core locations and the type of each core location (i.e., whether it is of P_A-type or P_B-type). Since all core locations of the same type have the same probability of being misloaded with a fresh fuel assembly without a CEA, the probability of each combination can be conveniently expressed in terms of the product of two probabilities P_A and P_B, which denote respectively the probability associated with location A-type and B-type. For example, the combination consisting of a=8(A), b=9(B), c=18(B), and d=19(B), shown in Figure 3-4, indicates that location 8 is of A-type and locations 9, 18 and 19 are of B-type. The probability of this combination is, thus, $P_AP_B^3$. Similarly, the probability of the combination consisting of core locations 33, 34, 47 and 48 can be expressed as $P_A^2P_B^2$. This computational approach also applies to the evaluation of the probability of CLUS5, which represents all the possible combinations of five core locations to form a cluster of a cross configuration.

There are twelve core locations which are designated to be loaded with a fuel assembly of FUB type. For these twelve core locations, therefore, the probability of being loaded with a fresh fuel assembly without a CEA is, for practical purposes, unity. Of the 140 possible combinations of forming a cluster of a 2x2 array, 48 combinations contain a core location of this type. For example, as shown in Figure 3-1, the combination consisting of core locations 41(FUB), 42(B), 56(B), and 57(A), belongs to this category, since location 41 is designated to be loaded with a fuel assembly of FUB type. The probability of this combination can thus be simplified to $P_A P_B^2$. The event, CLUS3, shown in Figure 3-4, is included to illustrate this point.

It can be expected that combinations belonging to this category tend to dominate the total frequency of losing required shutdown margin, since their probabilities can be expressed as products of the probabilities of three events rather than four.

A similar situation exists for the combinations of five core locations to form a cluster of a cross configuration. Of the 129 possible combinations, four contain two FUB type core locations, while 53 contain one FUB type core location. Examples are the combination consisting of core locations 45(B), 59(FUB), 60(B), 61(FUB) and 76(A) for the former, and the combination consisting of core locations, 30(A), 43(A), 44(B), 45(B), and 59(FUB) for the latter.

3.4 Quantification Of The Fault Trees

The fault trees shown in Figures 3-2a through 3-2e and Figures 3-3a and 3-3b were quantified using the SAICUT code [3.1]. It is noteworthy that, of the 38 basic events appearing in the fault trees developed to estimate the probability of misloading core locations of A-type, 30 basic events involve human errors. The probabilities of all the basic events used in quantifying the fault trees are summarized in Tables 3-1(a) and 3-1(b). The human error probabilities given were derived from data in Swain and Guttmann [3.2]. The results obtained by using the basic event probabilities listed in Tables 3-1(a) and 3-1(b) are referred to as the base case results, to distinguish them from those obtained in the sensitivity studies to be presented later.

The fault tree comprises four basic events (FATYPE2, FATYPE3, ALOC and SPLOC) whose probabilities depend on the order of refueling. The calculation of these probabilities was done approximately as discussed in Section 3.3. In the following, the more rigorous approach is given. Since prior to loading the core, all the fuel assemblies, fresh or burned, are stored in the spent fuel pool, calculations of the probabilities of basic events FATYPE2 and FATYPE3 are relatively straightforward. To calculate the probability of the event, FATYPE2 (the FA is of FU or FUB type), for example, it is known that the spent fuel pool initially contains 20 fuel assemblies that are of FU or FUB type. With the final loading pattern and the order of loading the core clearly depicted by Figure 3-1, the probability of FATYPE2 at each loading step can be readily calculated by assuming that each of these fresh assemblies can be picked up randomly with equal probability. Similarly, estimation of the probability of the event, FATYPE3 (the FA is of FC or FCB type), can be done by noting that, prior to the loading operation, a total of 72 fuel assemblies of these types are stored in the spent fuel pool.

Estimation of the probabilities of the basic events ALOC (the core location is location A), and SPLOC (the wrong spent fuel pool location is the specified pool location), on the other hand, is more intricate. Note, first of all, that event ALOC, in conjunction with the event AINTEND (a FA intended for another core location is loaded into a wrong core location), is used to calculate the probability that any core location might be accidently loaded with a fuel assembly intended for another core location. As mentioned previously, since there are 217 core locations to be loaded, the loading procedures consist of 217 steps, each step involving moving a fuel assembly from the spent fuel pool to a designated core location.

Suppose that the Nth refueling step involves moving a fuel assembly from a spent fuel pool location to core location N. During refueling steps 1 through (N-1) there is a possibility that a fuel assembly intended for core locations 1 through (N-1) may be accidently loaded into location N, due to, for example, errors committed by the refueling machine operator. Once the core location N is occupied by a fuel assembly, correct or wrong, no further error of this type need be considered for core location N, since the error would be detected. For the Nth core location, therefore, only loading steps 1 through (N-1) need be analyzed for this type of loading error. Note that at the Nth loading step, there are (217-N) empty core locations where the fuel assembly can be misloaded with an equal probability of 1/(217-N). The probability of the basic event ALOC, therefore, is a function of the loading steps and, hence, the core locations. Since the misloading can take place at any of the (N-1) loading steps, it is reasonable to take an average over the (N-1) loading steps. The probability, P, of the event ALOC can thus be expressed as

$$\overline{P_N} = \frac{1}{N-1} \cdot \sum_{n=1}^{N-1} \frac{1}{(217-n)} \qquad N \geq 2 \qquad (1)$$

Equation 1 can be used to calculate the probability of ALOC at each core location. However, an averaged value (=7.6E-03) over the entire 217 core locations was actually used, as discussed in Section 3.3. A similar equation for computing the probability of the basic event SPLOC can also be derived. Since no detailed unloading procedures are specified in this study, the probability of SPLOC was taken to be the same as that of ALOC.

3.5 Discussion of Results

3.5.1 Base Case

The results of fault tree quantifications for the base case using the SAICUT code are summarized in Table 3-2. The probability of the fault tree top events, TOPA and TOPB, were calculated to be 3.38E-03 and 2.81E-03 respectively. Substituting these values into the expressions for all the possible combinations (cf Figure 3-4), the probability of forming a cluster of 4 fuel assemblies (2x2 array) or of 5 assemblies (cross configuration) so as to lose the required shutdown margin was calculated to be 1.2E-06 and 1.1E-07, respectively.

The more severe but less likely case of forming a cluster of nine fuel assemblies (3x3 array) was also considered in the analysis. As discussed in Section 2, this configuration is of particular significance, because if such an array is formed anywhere in the core, it can result in not only loss of shutdown margin, but also inadvertent criticality. There are a total of 129 such combinations possible including those which contain one baffle core location at one corner of the 3x3 array. The probability of forming a cluster of nine fuel assemblies anywhere inside the core was found to be negligibly small (3.3E-17).

Some of the dominant cut sets generated by the SAICUT code are listed in Table 3-3, along with their probabilities. Picking up a wrong fuel assembly (of FU or FUB type) at the spent fuel pool and loading it into a correct core location or errors introduced by intermediate positioning of a fuel assembly (of FU or FUB type) in the core, dominate the frequency of the top event, TOPA. Failures to attach a CEA to the correct fuel assembly during CEA repositioning, in combination with failures to detect the error prior to fuel loading operation are also important. Another dominating scenario involves storing a fresh assembly of FU or FUB type in a wrong pool location and, with the error undetected prior to fuel loading, the fuel assembly is loaded into a correct core location. In general, cut sets comprising control room operator errors or refueling machine operator errors are less dominant because multiple errors have to be committed in order for the top event TOPA to occur.

Applying the industry average frequency of 0.75 refuelings per reactor year and assuming that the required shutdown margin will be lost when at least 4 fresh assemblies without CEAs are clustered together, the expected frequency of losing the required shutdown margin can be calculated to be 1.0E-6/YR for the loading being studied.

To estimate the frequency of an inadvertent criticality that could result from misloading of the fuel assemblies, a simple event tree was constructed, as shown in Figure 3-5. One of the top events in the tree is the misloading of fresh fuel into a cluster that results in the complete loss of the shutdown margin (SDM). From the deterministic analysis described in Section 2, approximately 9 fresh fuel assemblies clustered together could lead to an inadvertent criticality. Based on the probability of this occurring (3.3E-17) the frequency of an inadvertent criticality is negligible. In Section 2 it was also seen that only

5 fresh fuel assemblies would be necessary for an inadvertent criticality if it is assumed that the initial shutdown margin is the minimum required. Hence, the probability for this type of cluster is used in the event tree in order to estimate a conservative frequency.

The event tree models the possible failure of ex-core source range neutron flux monitors that are used during refueling to monitor the subcritical neutron multiplication. The nuclear instrumentation has a visual indication in the control room as well as an audible (loudspeaker) indication in the containment and control room. They would normally be capable of detecting an increase in count rate before criticality was reached and the operator would be able to take preventive actions, such as stopping the refueling or adding additional boron. The largest increase in count rate would occur when the last fuel assembly needed for criticality is being lowered into the core. If the operator had taken no action until this time he/she might not be able to respond quickly enough to prevent the inadvertent criticality. Hence, the human error probability (0.1) is relatively high. Using the event tree and the probability of forming a cluster of 5 fresh assemblies the resulting frequency for an inadvertent criticality is 9.1E-9/YR (i.e., IC-1 + IC-2).

3.5.2 Sensitivity Studies

Since the fault trees developed in this study contain a large number of human error related basic events and since the human error probabilities assigned to these basic events are subject to uncertainty, sensitivity studies were performed exclusively with regard to the human error probabilities. Two different approaches were taken in examining the sensitivity of the results to the variation of human error probabilities. In the first approach, all the human error probabilities used in the fault tree quantification were multiplied by a constant factor, such as 0.5 or 2, without changing the probabilities of basic events unrelated to human errors. In the second approach, human-error related basic events are grouped into several categories based on their similarity in nature and the actions involved. The fault trees were then quantified by only increasing the human error probabilities of the basic events belonging to the same category by a factor of ten, without altering the probabilities of the remaining basic events.

The results of the sensitivity studies for the cases where all the human error probabilities are multiplied by a constant factor are summarized in Table 3-4. It can be seen that the results are rather sensitive to the changes in human error probabilities. When all the human error probabilities are increased by a factor of 2, for instance, the frequency of losing the required shutdown margin increases by an order of magnitude from 1.0E-6/YR to 1.1E-5/YR. The frequency of an inadvertent criticality based on a cluster of 5 fresh fuel assemblies also increases by an order of magnitude from 9.1E-9/YR to 1.2E-7/YR. Although the frequency of an inadvertent criticality based on a cluster of 9 fresh fuel assemblies increases more than two orders of magnitude, it is still quite insignificant relative to frequencies that are usually of concern. When all the human error probabilities are increased by a factor of 5, the frequency of an inadvertent criticality based on a cluster of 5 fresh assemblies further increases significantly to 5.6E-6/YR. For the more extreme case where all the human error probabilities are increased by a factor of ten, the frequency of inadvertent criticality (1.0E-8/YR) based on a cluster of 9 assemblies becomes comparable to the base case frequency for a cluster of 5 assemblies.

To better understand what types of human errors dominate the frequency of misloading the fuel assemblies, the human-error related basic events were grouped into several categories, as shown in Table 3-5. For example, errors committed by the control room operator and those committed by the refueling machine operator during core loading or unloading are each grouped together as a single category. The human error related basic events characterizing each category are also given in Table 3-5. The fault tree quantifications were performed, one category at a time, by multiplying the probabilities of those basic events that characterize the category by a factor of ten. The results show clearly the importance of three categories of human errors. They are: spent fuel pool machine operator errors in picking up (or storing) a fuel assembly from (or in) the spent fuel pool, errors due to intermediate positioning of fuel assemblies, and verification of the locations of fuel assemblies and of CEAs prior to beginning fuel loading operation. Errors committed by both the refueling machine operator and the control room operator, or mechanical problems with the refueling machine caused by human errors appear to be less important. These trends are, in general, consistent with those observed from the dominant cut-sets discussed earlier.

3.6 Conclusions

Based on the probabilistic analysis presented in this section it may be concluded that, for a PWR plant that uses offload/reload procedures similar to those for Calvert Cliffs, the frequency of losing the required shutdown margin during refueling operations is approximately 1.0E-6 events per reactor year. This estimate is based on the assumption that refueling procedures only allow intermediate positioning of fuel assemblies (FAs) that are analyzed in advance (i.e., NRC Bulletin No. 89-03 [1.3] is followed) and that other plant procedures are followed as well. To extend the analysis for Cycle 9 to other fuel loadings it is also necessary to assume that there would be a similar number of fresh fuel assemblies without CEAs loaded within the core interior (i.e., not on the core periphery). Intermediate positioning of FAs, human errors associated with picking up or storing FAs in the spent fuel pool and verification of both FA positions and CEA positions prior to refueling operation were identified to be dominant contributors to the frequency of misloading FAs.

Results of this analysis are quite sensitive to the human error probabilities (HEPs) assigned in the fault tree quantifications. Doubling all the HEPs, for example, caused an increase in the estimated frequency of loss of shutdown margin by an order of magnitude. However, assuming the validity of the base case HEPs, this result is expected to be an upper bound because of the conservative assumption that errors made in one stage of the refueling procedures will not be detected at a latter stage.

The frequency of an inadvertent criticality due to refueling errors was estimated to be insignificant for Cycle 9 of Calvert Cliffs. This is because the large shutdown margin requires the misloading of 9 fresh fuel assemblies in a 3x3 array and this necessitates many errors for it to occur. With relatively pessimistic assignment of HEPs the frequency might increase to 9.9E-9/YR. If a plant was refueled with only a minimum shutdown margin, the occurrence of an inadvertent criticality caused by the misloading of only 5 fresh fuel assemblies might have a frequency of 9.1E-9/YR (with nominal HEPs).

Table 3-1(a)

Human Errors Modeled in the Fault Trees for TOPA

Event Name	Description of Basic Event	Probability Used In Base Case Calculation
ANEWOP1	Spent fuel machine operator read the bridge and trolley locations incorrectly when storing new assemblies.	8.0E-3
ANEWOP2	Spent fuel machine is incorrectly aligned to letter or number position markers when storing new assemblies.	1.0E-2
ANEWPR	Errors in the procedure for storing new assemblies.	1.0E-3
CALIBER	Refueling machine position is calibrated incorrectly.	3.0E-3
CALINDX	Index point calibration check is not done by the operator.	1.0E-2
CEAERR1	Spent fuel machine operator read the bridge and trolley locations incorrectly during CEA repositioning.	8.0E-3
CEAERR2	Spent fuel machine is incorrectly aligned to letter or number position markers during CEA repositioning.	1.0E-2
CEAPR	Errors in CEA repositioning procedures.	1.0E-3
COINTM	A FA with wrong ID number is loaded into location A due to errors introduced by intermediate positioning etc.	8.0E-3
GRID	Refueling machine operator misunderstood core grid location due to communication errors.	3.0E-3
LUNLDPR	Errors in loading (or unloading) procedures.	1.0E-3
LUNLD1	Control room operator updated the procedure incorrectly leading to specification errors during loading or unloading.	8.0E-3
LUNLD2	Control room operator read a wrong core grid location during loading or unloading.	8.0E-3
LUNLD3	Tag board was updated incorrectly leading to specification errors during loading or unloading.	8.0E-3

Table 3-1(a) (Continued)

Human Errors Modeled in the Fault Trees for TOPA

Event Name	Description of Basic Event	Probability Used In Base Case Calculation
LUNLD4	Control room operator misread the procedure due to changes in the order of portion of the procedures during loading or unloading.	1.0E-2
RMOP1	Refueling machine operator reads index value from a wrong line.	5.0E-3
RMOP2	Refueling machine operator transposes bridge and trolley locations.	5.0E-3
SPICK1	Spent fuel machine operator read the bridge and trolley locations incorrectly (during core loading).	8.0E-3
SPICK2	Spent fuel machine is incorrectly aligned to letter or number position markers (during core loading).	1.0E-2
SPICKPR	Errors in refueling procedures (for core loading).	1.0E-3
SPINTM	A FA with wrong ID number is stored in the specified spent fuel pool location due to errors introduced by intermediate positioning etc.	5.0E-3
SPSTOR1	Spent fuel machine operator read the bridge and trolley locations incorrectly during core unloading.	8.0E-3
SPSTOR2	Spent fuel machine is incorrectly aligned to letter or number position markers during core unloading.	1.0E-2
SUBSEQ	Shufflings subsequent to core unloading causes a wrong FA to be stored in the specified spent fuel pool location.	5.0E-3
VERICE1	No verification of CEA location is done before fuel loading operation.	1.0E-2
VERICE2	Errors committed during verification of CEA location using remote video camera.	2.0E-2

Table 3-1(a) (Continued)

Human Errors Modeled in the Fault Trees for TOPA

Event Name	Description of Basic Event	Probability Used In Base Case Calculation
VERIFA1	No verification of FA position is done before fuel loading operation.	1.0E-2
VERIFA2	Errors committed during verification of FA position using remote video camera	2.0E-2
VERISE1	No verification of machine position is done by a second machine operator.	8.0E-1
VERISE2	Verification of machine position incorrectly done.	5.0E-3

Table 3-1(b)

Basic Events (Not Related to Human Errors)

Modeled in the Fault Trees for TOPA

Event Name	Description of Basic Event	Probability
ALOC	The wrong location is location A.	7.6E-3
FATYPE1	The loaded FA is of FCB type.	1.0
FATYPE2	The FA is of FU or FUB type.	9.6E-2
FATYPE3	The FA is of FC or FCB type.	3.2E-1
SMOOTH1	A FA can be loaded or unloaded smoothly at the wrong core index location.	2.0E-1
SMOOTH2	Mispositioning of refueling machine corresponds to a FA location.	1.0E-1
SPLOC	The wrong spent fuel pool location is the specified pool location.	7.6E-3
TRANSD	Malfunction of position transducer causes the machine to be mispositioned.	2.2E-3

Table 3-2

Results of Base Case Calculations

Description of Calculated Items	Calculated Probability (or Frequency)
Probability of TOPA (A fresh FA without CEA is loaded into location A)	3.4E-3
Probability of TOPB (A fresh FA without CEA and with wrong ID is loaded into location B)	2.8E-3
Probability of forming a cluster of 4 (2x2 array)	1.2E-6
Probability of forming a cluster of 5 (a cross configuration)	1.1E-7
Probability of forming a cluster of 9 (3x3 array)	3.3E-17
Frequency of losing required shutdown margin (/YR)	1.0E-6
Frequency of inadvertent criticality (IC)	
IC-1 (see Fig. 3-5) based on a cluster of 5 FAs (/YR)	8.4E-9
IC-2 (see Fig. 3-5) based on a cluster of 5 FAs (/YR)	6.5E-10
IC-3 based on a cluster of 9 FAs (/YR)	<1.0E-10

Table 3-3

List of Dominant Cut-Sets for Fault Tree Top Event TOPA

Rank	Probability of Cut-Set	Cut-Sets
1	9.6E-4	FATYPE2 * SPICK2
2	7.7E-4	FATYPE2 * SPICK1
3	7.7E-4	COINTM * FATYPE2
4	2.0E-4	CEAERR2 * FATYPE1 * VERICE2
5	1.6E-4	CEAERR1 * FATYPE1 * VERICE2
6	1.0E-4	CEAERR2 * FATYPE1 * VERICE1
7	9.6E-5	FATYPE2 * SPICKPR
8	8.0E-5	CEAERR1 * FATYPE1 * VERICE1
9	2.0E-5	CEAPR * FATYPE1 * VERICE2
10	1.9E-5	ANEWOP2 * FATYPE2 * VERIFA2

Description of Basic Events:

ANEWOP2: Spent fuel machine is incorrectly aligned to letter or number position markers when storing new fuel assemblies.
CEAERR1: Spent fuel machine operator read the bridge and trolley locations incorrectly.
CEAERR2: Spent fuel machine is incorrectly aligned to letter or number position markers when repositioning CEAs
CEAPR: Errors in CEA repositioning procedures
COINTM: A FA with wrong ID is loaded into location A due to errors introduced by intermediate positioning etc.
FATYPE1: The loaded FA is of FCB type
FATYPE2: The FA is of FU or FUB type
SPICK1: Spent fuel machine operator read the bridge and trolley locations incorrectly.
SPICK2: Spent fuel machine is incorrectly aligned to letter or number position markers when picking up a FA from spent fuel pool.
SPICKPR: Errors in refueling procedures
VERICE1: No verification of CEA location is done before fuel loading operation
VERICE2: Errors committed during verification of CEA location using remote video camera
VERIFA2: Errors committed during verification of FA position using video camera

Table 3-4

Results of Sensitivity Studies on Human Error Probabilities (HEP)

	Probability of Forming a Cluster of 4	Probability of Forming a Cluster of 5	Probability of Forming a Cluster of 9	Frequency of Loss of the Required Shutdown Margin (/YR)	Frequency of Inadvertent Criticality Based on a Cluster of 5 (/YR)	Frequency of Inadvertent Criticality Based on a Cluster of 9 (/YR)
Base Case	1.2E-6	1.1E-7	3.3E-17	1.0E-6	9.1E-9	2.7E-18
HEP x 0.5	1.3E-7	1.1E-8	1.7E-19	1.1E-7	9.1E-10	1.4E-20
HEP x 2	1.3E-5	1.4E-6	9.4E-15	1.1E-5	1.1E-7	7.8E-16
HEP x 3	6.1E-5	6.7E-6	3.4E-13	5.1E-5	5.5E-7	2.8E-14
HEP x 5	4.9E-4	6.6E-5	4.7E-11	4.2E-4	5.3E-6	3.8E-12
HEP x 10	1.4E-2	3.1E-3	1.2E-7	1.3E-2	2.5E-4	9.9E-9

Table 3-5

Results of Sensitivity Studies Based on Human Error Categories (with HEP x 10)

Description of Human Error Category	Relevant Human Error Basic Events	Probability of Forming a Cluster of 4	Probability of Forming a Cluster of 5	Probability of Forming a Cluster of 9	Frequency of Loss of the Required Shutdown Margin (/YR)	Frequency of Inadvertent Criticality Based on a Cluster of 5 (/YR)	Frequency of Inadvertent Criticality Based on a Cluster of 9 (/YR)
Base Case	—	1.2E-6	1.1E-7	3.3E-17	1.0E-6	9.1E-9	2.7E-18
Spent fuel machine operator errors in picking up or storing FA in spent fuel pool	SPICK1, SPICK2, SPSTOR1, SPSTOR2	3.1E-4	3.2E-5	1.3E-11	2.6E-4	2.7E-6	1.0E-12
Errors in procedures	SPICKPR, ANEWPR, UNLDPR, CEAPR	2.9E-6	2.7E-7	2.5E-16	2.4E-6	2-3E-8	2.1E-17
Refueling machine operator errors during loading or unloading	VERISE1 (=1.0) VERISE2, GRID, RMOP1 RMOP2	1.3E-6	1.2E-7	4.1E-17	1.1E-6	1.0E-8	3.4E-18
Intermediate positioning of FA during loading or unloading	COINTM, SPINTM, SUBSEQ	5.1E-5	4.8E-6	1.9E-13	4.2E-5	4.0E-7	1.5E-14
Mechanical errors of refueling machine caused by human errors	CALIBER, CALINDX	1.2E-6	1.1E-7	3.3E-17	1.0E-6	9.1E-9	2.7E-18
Control room operator errors during loading or unloading	UNLD1, UNLD2 UNLD3, UNLD4	3.2E-6	2.9E-7	3.0E-16	2.6E-6	2.5E-8	2.5E-17
Verification of FA or CEA positions prior to fuel loading	VERIFA1, VERIFA2, VERICE1, VERICE2	8.7E-6	1.0E-6	4.3E-15	7.3E-6	8.3E-8	3.6E-16
Spent fuel machine operator errors in repositioning CEAs	CEAERR1, CEAERR2	2.6E-6	3.1E-7	2.8E-16	2.2E-6	2.5E-8	2.3E-17
Spent fuel machine operator errors in storing new FAs	ANEWOP1, ANEWOP2	1.9E-6	1.7E-7	9.2E-17	1.6E-6	1.4E-8	7.6E-18

F - Fresh Fuel
U - No CEA
C - With CEA
B - With burnable poison rods

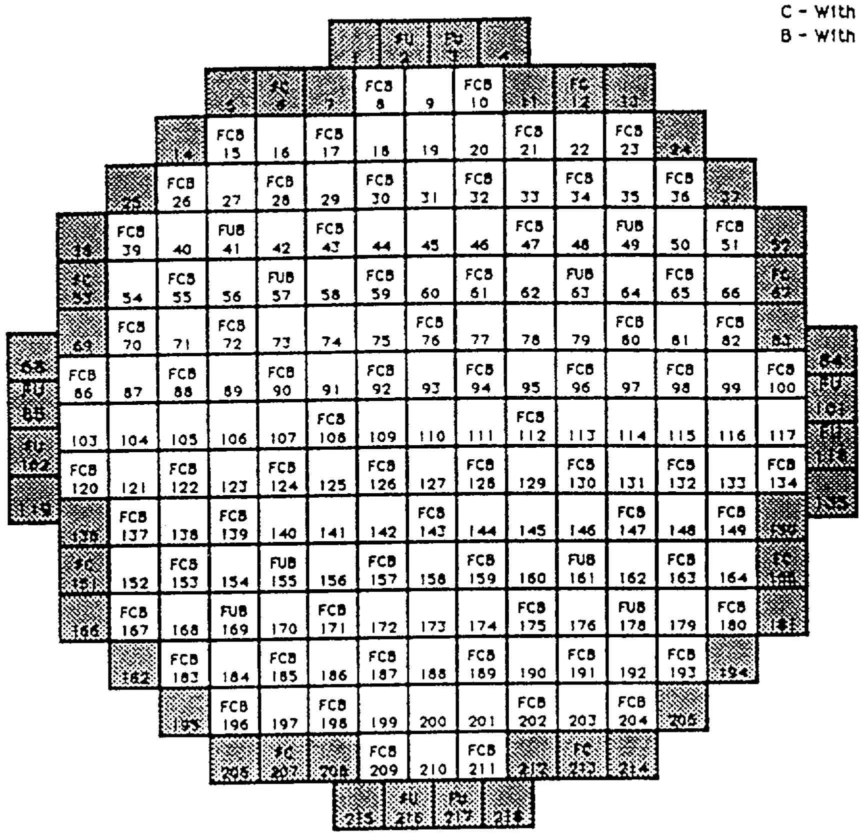


Figure 3-1 Fresh Fuel Locations in Cycle 9

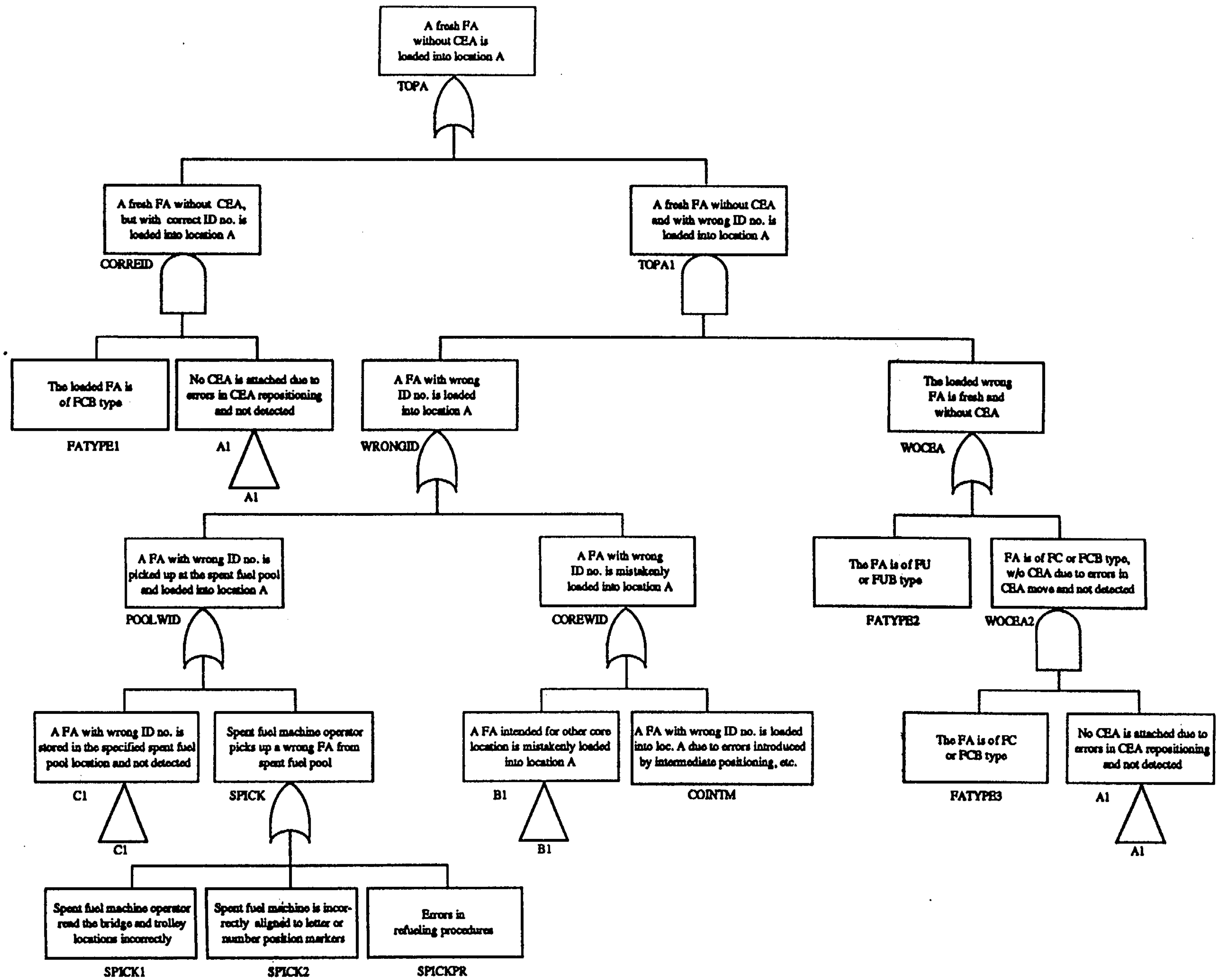


Figure 3-2a
Fault Tree for Loading Fresh Assembly into Type A Location

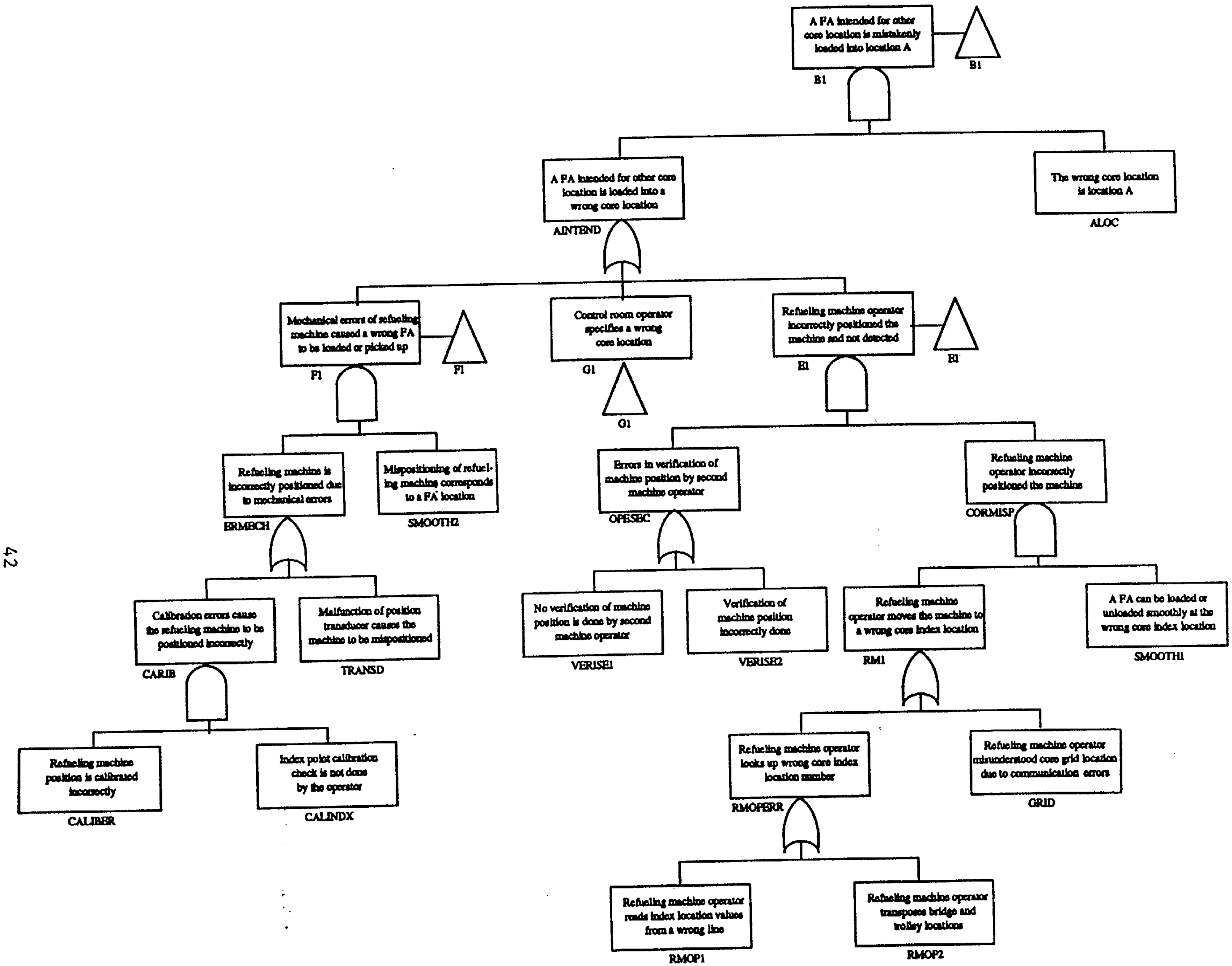


Figure 3-2b
Fault Tree for Loading Fresh Assembly into Type A Location

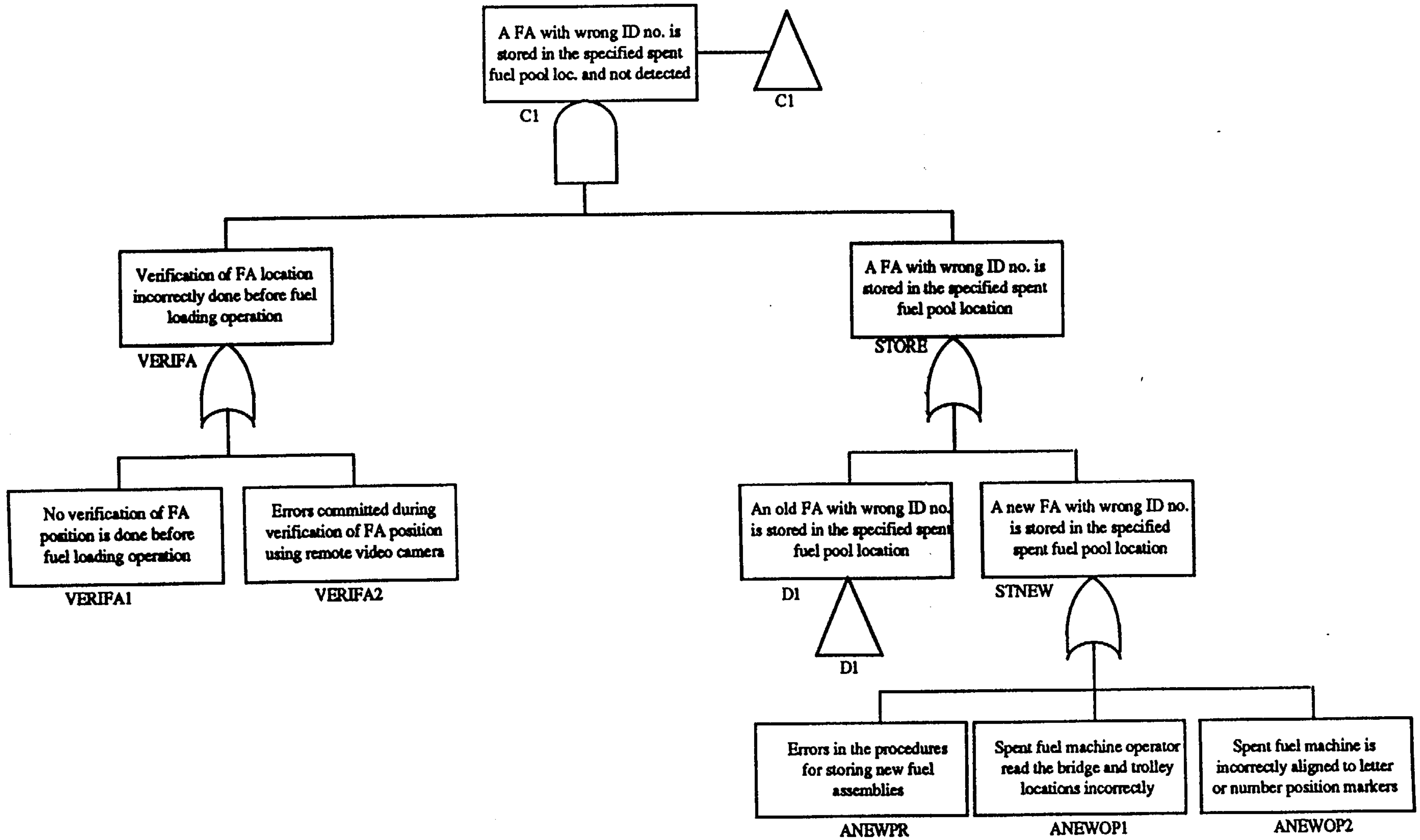


Figure 3-2c
Fault Tree for Loading Fresh Assembly into Type A Location

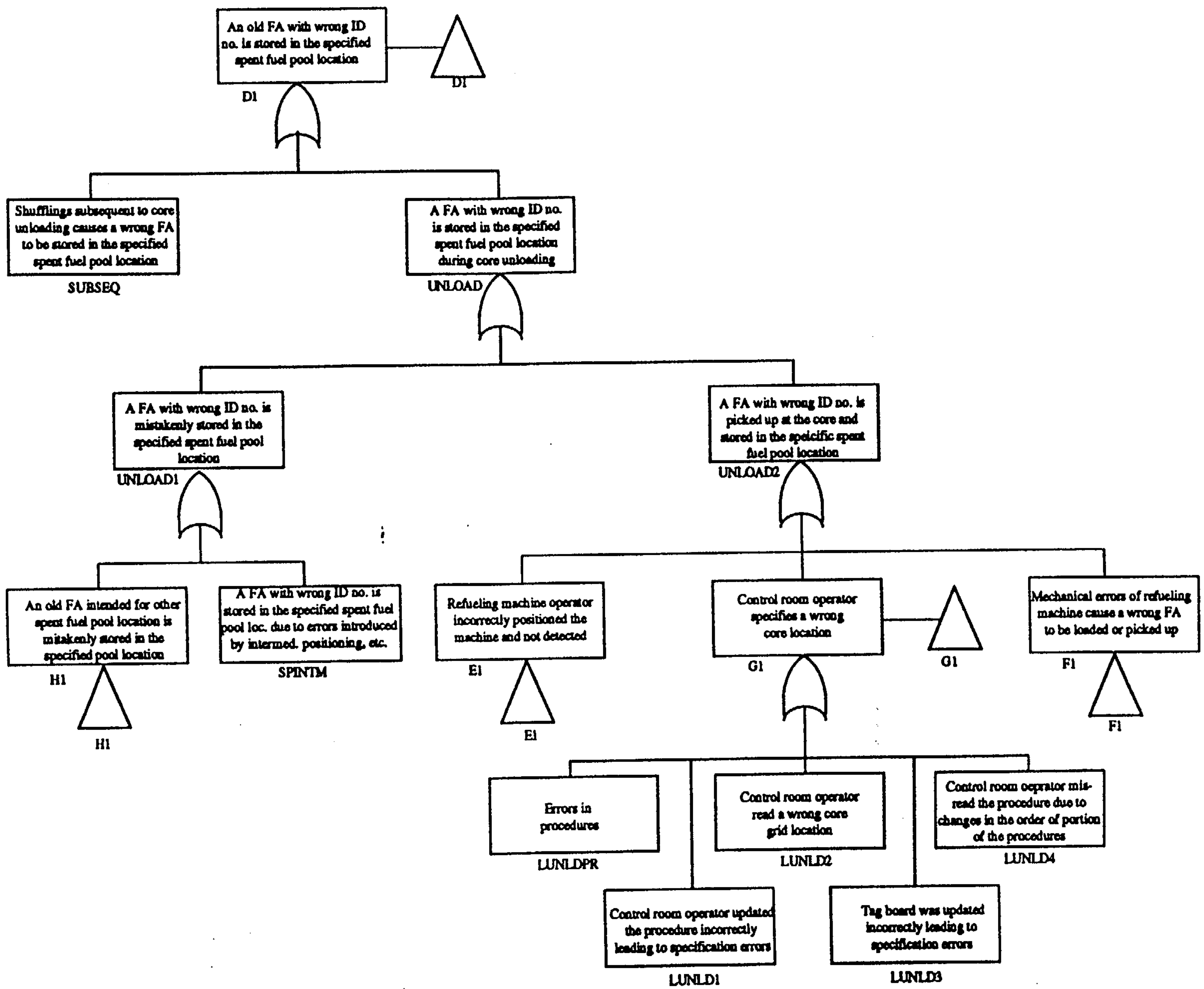


Figure 3-2d
Fault Tree for Loading Fresh Assembly into Type A Location

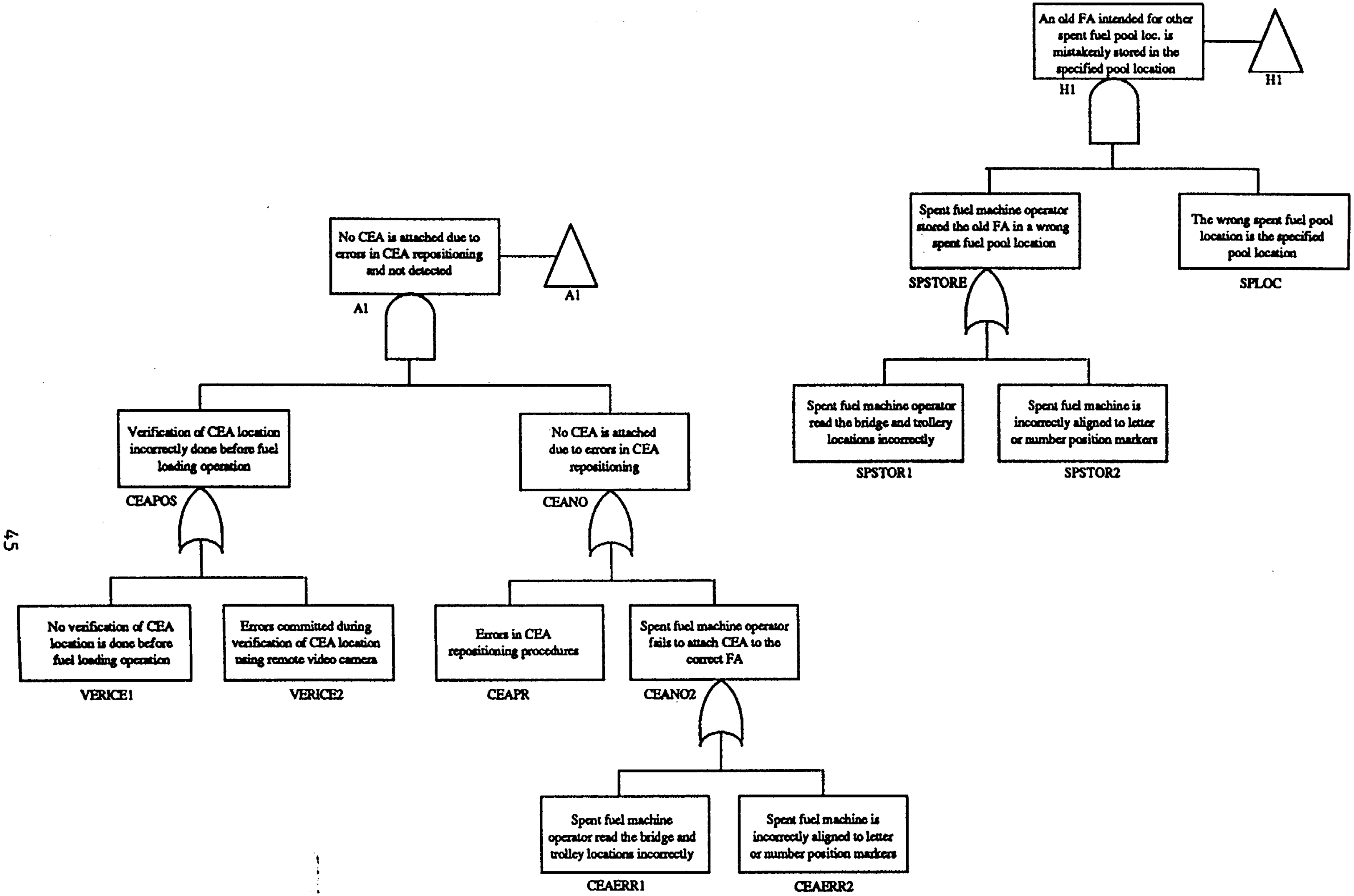


Figure 3-2e
Fault Tree for Loading Fresh Assembly into Type A Location

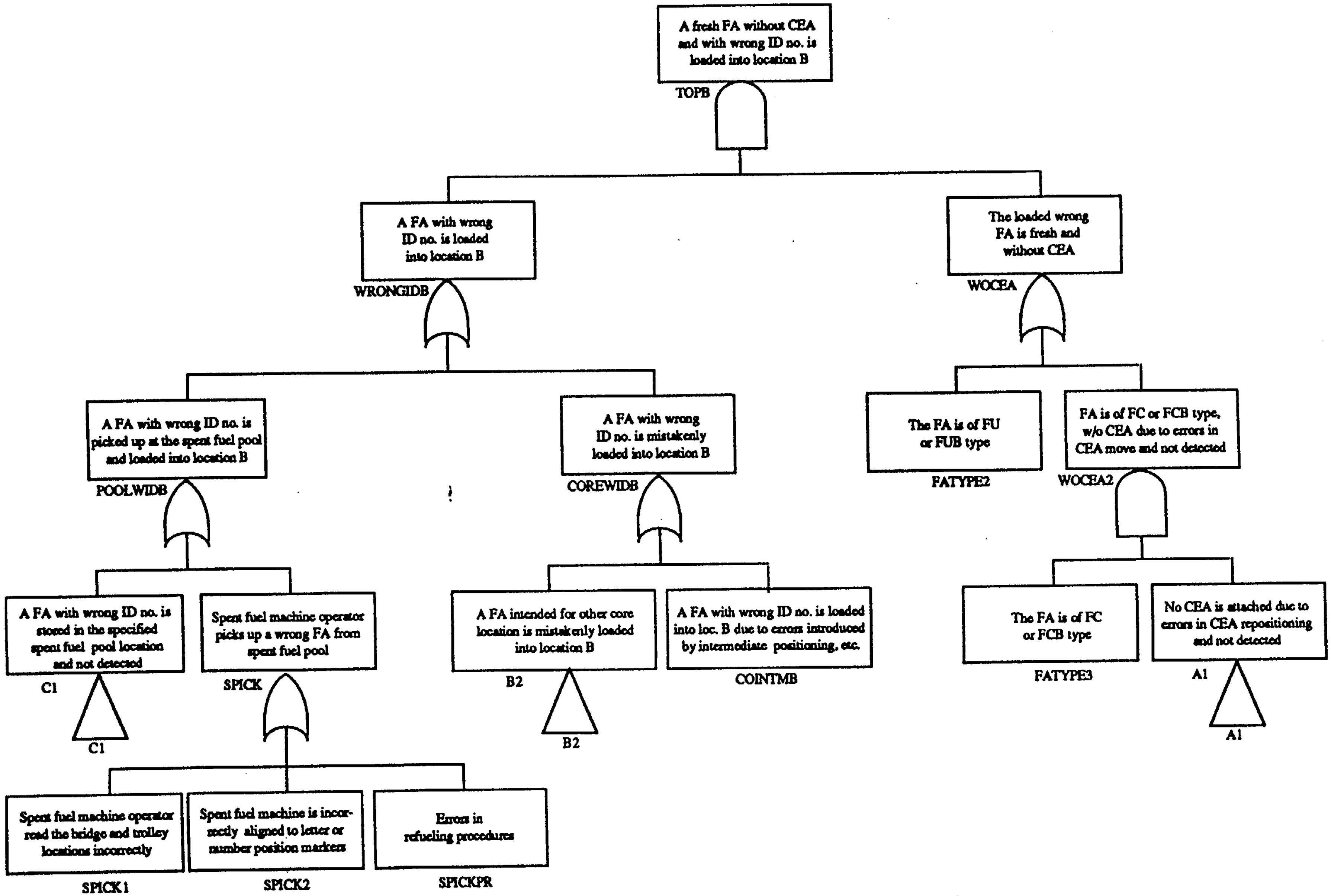


Figure 3-3a
Fault Tree for Loading Assembly into Type B Location

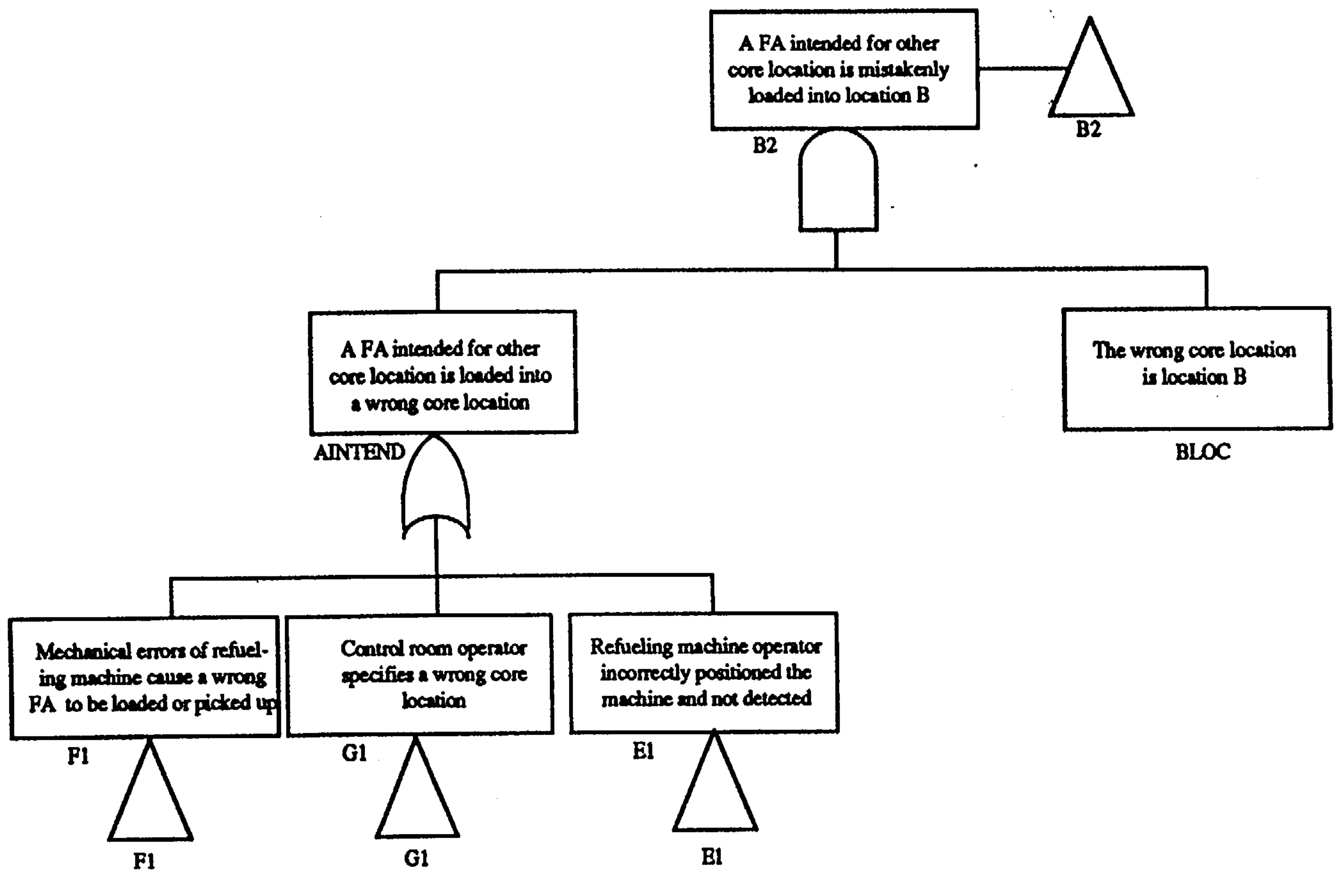
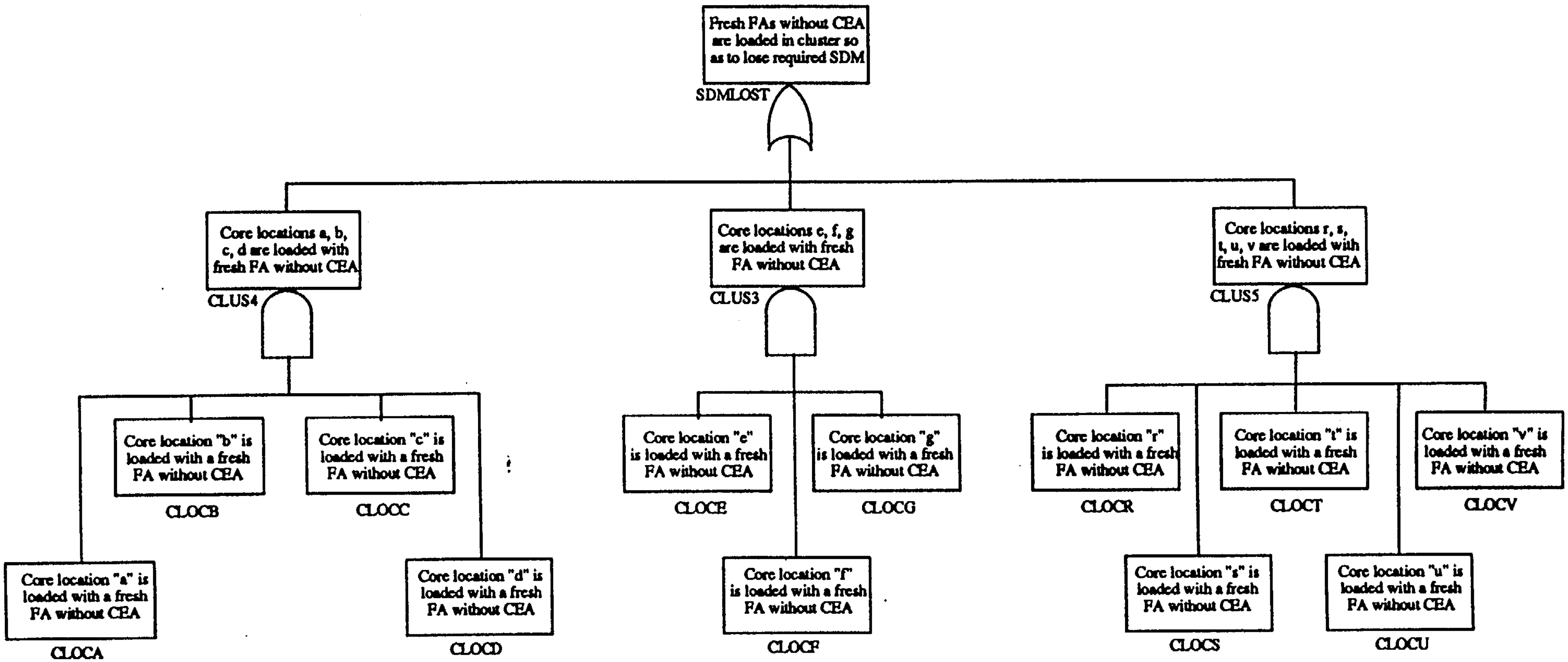


Figure 3-3b
Fault Tree for Loading Fresh Assembly into Type B Location



a = 8 (A)
b = 9 (B)
c = 18 (B)
d = 19 (B)

$P = P_A P_B^3$

a = 33 (B)
b = 34 (A)
c = 47 (A)
d = 48 (B)
etc.

$P = P_A^2 P_B^2$

e = 42 (B)
f = 56 (B)
g = 57 (A)
etc.

$P = P_A P_B^2$

r = 17 (A)
s = 28 (A)
t = 29 (B)
u = 30 (A)
v = 43 (A)
etc.

$P = P_A^4 P_B$

Legend

P = Probability of top event (CLUS4, CLUS3, or CLUS5)
P_A = Probability of event TOPA (see Figure 3-2)
P_B = Probability of event TOPB (see Figure 3-3)

Figure 3-4
Fault Tree for Misloading Fuel into Unacceptable Cluster

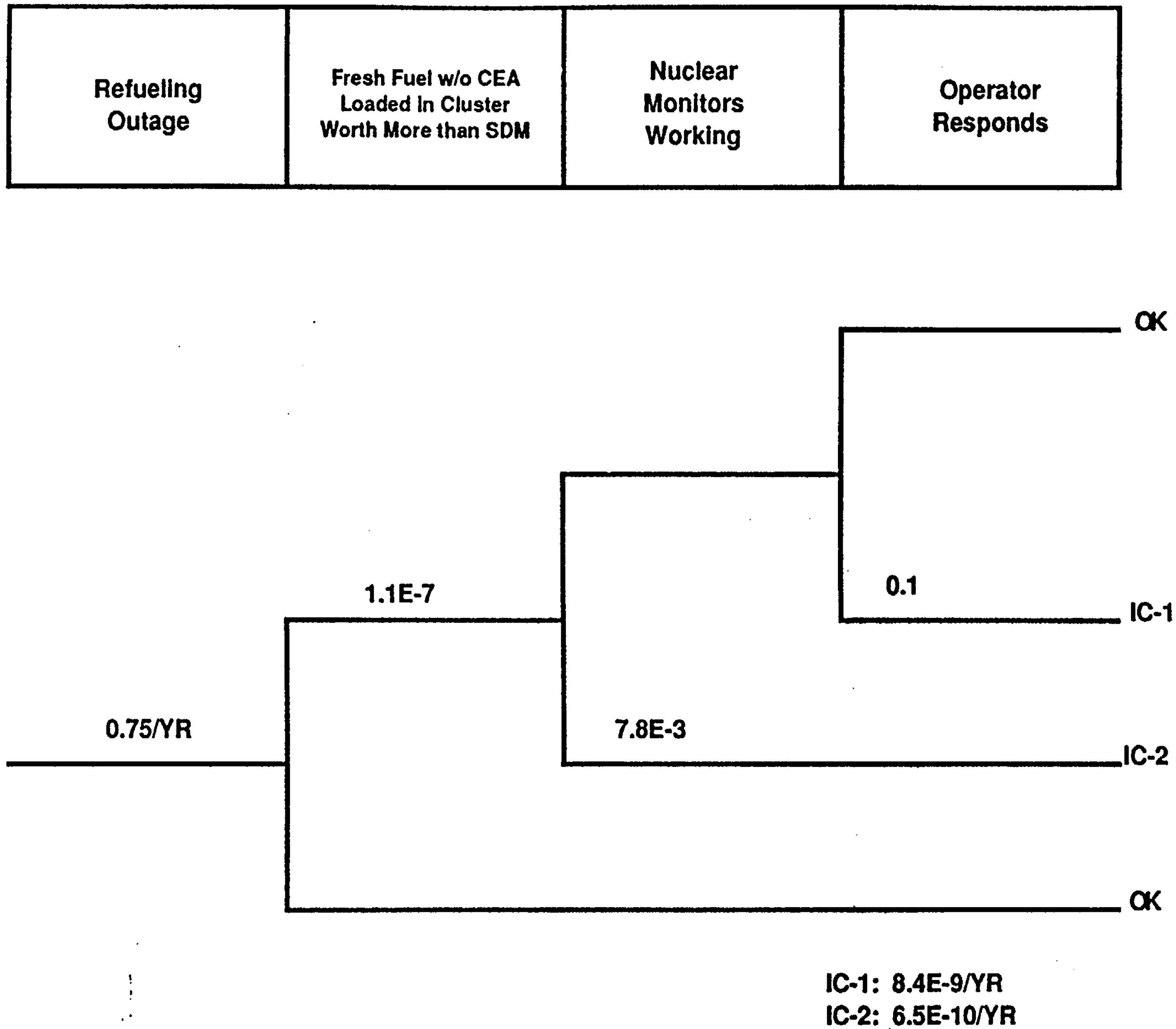


Figure 3-5 Event Tree for Inadvertent Criticality

4. DOSES DUE TO AN INADVERTENT CRITICALITY

4.1 Direct Doses

Doses to workers within containment from an inadvertent criticality can result from direct exposure to γ-rays and neutrons generated in the core or indirect exposure to fission products that may be released from the fuel. The direct γ-dose has been calculated, approximately, based on well-known techniques [4.1]. The assemblies are assumed to be under water to a depth of 32 ft (approximately 10 m). The problem then is to calculate the γ-ray fluence and hence the dose rate at a point above the water shield which represents, approximately, the position at which the workers (who are loading the fuel assemblies) are located.

Due to the shield thickness and the fact that the source (i.e., the reactor core) is embedded entirely inside the water shield, it is appropriate to treat the core as a point isotropic source for the purpose of the dose calculation. The exposure rate in air, $\dot{X}$, at a point just above the surface of the shield is given by the expression (for a point isotropic source),

$$\dot{X} = 0.0659 E_m \left(\frac{\mu_a}{\rho}\right)^{air} \frac{S(E_m)}{4\pi R^2} B_\rho(\mu R)\, e^{-\mu R} \qquad mR/hr$$

and the dose rate, $\dot{D}$, at the same point is:

$$\dot{D} = 0.0576\, E_m \left(\frac{\mu_a}{\rho}\right)^{air} \frac{S(E_m)}{4\pi R^2}\, B_\rho(\mu R)\, e^{-\mu R} \qquad mrad/hr$$

where,

R = thickness of shield (cm)

E_m = mean energy of the photon, MeV

$\left(\frac{\mu_a}{\rho}\right)^{air}$ = linear absorption coefficient in air for photon of energy E_m (cm^2/g)

μ = mass attenuation coefficient of photon of energy E_m in water (cm^{-1})

$S(E_m)$ = source strength of photons of energy E_m (# photons/sec)

$B_\rho(\mu R)$ = dose build-up factor for water (point isotropic source)

A conservative value of the fission rate = 10^{20} fissions/sec was chosen for calculating the source strength, S, for this scoping exercise. This corresponds (at 200 MeV/fission) to a power level of ≈ 3200 MW which is close to the nominal operating power. A hypothetical misloading accident would be expected to lead to lower power levels in a localized region of the core but as shall be shown, the power level is not important as the dose will be shown to be small.

Table 4-1 shows the approximate energy group distribution of prompt and decay γ-rays from U-235 fission [4.2] and the corresponding source strength, $S(E_m)$, as a function of γ-ray energy assumed for this calculation.

Table 4-2 shows the values for the mass attenuation coefficient of water for various γ energies and the calculated values of μR assuming a shield thickness R of 10 meters.

The build-up factor $B_\rho(\mu R)$ at the various photon energies was calculated using both the Taylor form and the Berger form. The Taylor form is given by:

$$B_\rho(\mu R) = A_1\, e^{-\alpha_1 \mu R} + A_2\, e^{-\alpha_2 / \mu R}$$

$$\text{where } A_2 = 1 - A,\ A_1 = A$$

and the Berger form is:

$$B_\rho(\mu R) = 1 + a\mu R\, e^{b\mu R}$$

Values of the various coefficients and the corresponding calculated values of $B_\rho(\mu R)$ at various γ energies are given below in Table 4-3.

We see that the differences between the build-up factors calculated using the Taylor and Berger forms are not much, except at large values of the argument where the γ-fluence is completely insignificant.

Table 4-4 shows the values of the linear absorption coefficient and the calculated values of the γ-ray fluence, the exposure rate, and the dose rates at the various γ energies. The total dose rate is about 0.03 mrad/hr which is well below any limit of concern.

The direct neutron dose can be calculated following the methods recommended in Reference 4.1; however, the same reference (page 7-82) states that "For thick water shields beyond 120 cm, the γ-ray dose predominates ..." Since the γ-dose is insignificant, the direct neutron dose for a water shield of thickness $\approx$ 1000 cm will be even smaller so its explicit calculation is unnecessary.

4.2 Doses Due to Released Materials

Following an inadvertent criticality, it is hypothesized that an overtemperature condition will occur in a number of fuel assemblies leading to a disruption of the cladding on the fuel rods. This can or will lead to a release of the radionuclide inventory contained in the gap between the fuel pellets and the cladding. Most of this inventory consists of volatile fission gases which would then be transported through the water and the open vessel head into the containment. Containment interlock hatches are closed during the refueling operations. The gases released into containment would, ordinarily, flow along paths determined by the containment ventilation system and into the stack filters and then into the environment. However, on receipt of a high radiation signal, the containment would be isolated and the ventilation system would shut down to prevent the released material from being transported to the outside environment. Thus, the release of any radionuclide inventory during the accident will be confined to the containment alone; the only doses of concern from released material are those to the workers inside containment and not to the general public.

The total amount of activity released to the containment depends upon:

a. the core inventory,
b. the number of assemblies involved in the accident,
c. the gap inventory of fission products,
d. the decontamination factors (DFs) provided by the water above the core.

Guidance on the gap inventory of fission products and the decontamination factors provided by the water was obtained from Regulatory Guide 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" [4.4]. Although this Regulatory Guide deals with an accident in the storage pool involving a mechanical disruption of the cladding*, its recommendations on the gap fractions and pool DFs are pertinent to this scoping calculation since, in both cases, only disruption of the cladding and release of the gap inventory is involved. The pertinent recommendations of Regulatory Guide 1.25 are as follows:

---

*"An illustrative accident sequence consists of the dropping of a fuel assembly resulting in breaching of the fuel rod cladding, release of a portion of the volatile fission gases from the damaged fuel rods, absorption of water soluble gases in and transport of soluble and insoluble gases through the water ..." (Regulatory Guide 1.25).

1. "The accident occurs at a time after shutdown identified in the Technical Specifications as the earliest time fuel handling operations may begin. Radioactive decay of the fission product inventory during the interval between shutdown and commencement of fuel handling operations is taken into account." (In the present case the fueling operations begin 3-5 days after shutdown; hence, 4 days after shutdown is taken as the time at which the accident occurs for the purpose of estimating the core inventory.)

2. "The maximum fuel rod pressurization is 1200 psig."

3. "The minimum water depth between the top of the damaged fuel rods and the fuel pool surface is 23 ft."

   (The Guide states that "for release pressures greater than 1200 psig and water depths less than 23 ft, the iodine decontamination factors (DFs) will be less than those assumed in the guide and must be calculated on an individual case basis using assumptions comparable in conservation to those of this guide." The assumed water depth in the present problem is 32 feet so the condition is satisfied with an extra margin which may compensate for any fuel rod pressurization in excess of 1200 psig.)

4. "All of the gap activity in the damaged rods is released and consists of 10% of the total noble gases other than Kr-85, 30% of the Kr-85, and 10% of the total radioactive iodine in the rods at the time of the accident."

5. "The iodine gap inventory is composed of inorganic species (99.75%) and organic species (0.25%)."

6. "The pool decontamination factors for the inorganic and organic species are 133 and 1, respectively, giving an overall effective decontamination factor of 100 (i.e., 99% of the total iodine released from the damaged rods is retained by the pool water)."

7. "The retention of noble gases in the pool is negligible (i.e., decontamination factor of 1)."

These recommended values from the Regulatory Guide were used for calculating the releases to the containment from the inadvertent criticality.

The core inventory of radionuclides, 4 days after shutdown, was obtained from a calculation (using the ORIGEN 2 code) of the time-dependent inventory of radionuclides at the Surry plant [4.5]. This calculation assumed a core divided into three regions with average burnups of 11,000, 22,000, and 33,000 MWD/MTU, respectively.

Although the criticality excursion arises from the potential mispositioning of fresh fuel assemblies in the core during refueling, the releases of materials from the cladding disruption can only come from the older, irradiated assemblies surrounding the fresh fuel, since the new fuel has not undergone any burn-up and, therefore, cannot have accumulated any fission product inventory in the gap. Based on the analysis in Section 2, it was determined that a maximum of 12 older, irradiated fuel assemblies on the periphery of a misloaded 3x3 array of fresh fuel could potentially suffer a cladding disruption from the criticality event. Table 4-5 shows the total core inventories [4.5] the inventories of 12 assemblies, and the releases to the containment of the isotopes of the noble gases (Kr, Xe) and iodine based on the Regulatory Guide 1.25 recommendations outlined above.

Doses to the workers can arise from two pathways:

1. Inhalation doses due to breathing the contaminated air (it is assumed that the workers during refueling operations are not wearing respirators)

2. Immersion doses from standing in the contaminated air (analog of the "cloudshine" dose)

The dose rate depends on the concentration of activity of various released nuclides at the receptor point and the corresponding dose conversion factor.

For a scoping calculation, the simplest assumption to make for an approximate calculation of inhalation and immersion dose is to assume a uniform concentration inside the containment for the time of interest (say, a few minutes) for calculating the worker dose. This assumption will probably be nonconservative for the workers on the platform above the vessel where the concentration shortly after the release is likely to be higher. On the other hand, the assumption is likely to be conservative for workers in other parts of the containment engaged in maintenance operations.

The inhalation dose rate to the k^{th} body organ of a worker located at a point r inside the containment from radionuclide i is:

$$\dot{D}_i^{inh,k}(r,t) = X_i(r,t) \cdot DCF_k^{inh}(i) \cdot BR$$

where,

$\dot{D}_i^{inh,k}$ is the inhalation dose rate to organ k from the i^{th} radionuclide (rem/sec)

$X_i(r,t)$ is the instantaneous concentration of the i^{th} radionuclide at the point r (Ci/m^3)

$DCF_k^{inh}(i)$ is the dose conversion factor to organ k for the i^{th} radionuclide from inhalation (acute), (rem/Ci)

BR is the breathing rate (m^3/sec).

The immersion dose rate, using the same notation, is:

$$\dot{D}_i^{imm,k}(r,t) = X^i(r,t) \cdot DCF_k^{imm}(i)$$

where the dose conversion factor $DCF_i^{imm}(i)$ has the units rem-m^3/Ci-sec. The total dose to the organ k is obtained by summing over all released radionuclides.

If the concentration is assumed to be uniform and constant for the purpose of the dose calculation,

$$X_i\ (r,t) \equiv X_i = R_i/V$$

where R_i is given in Table 4-5 for various radionuclides i and V is the free volume of the containment which for the Calvert Cliffs plant is 2×10^6 ft^3. The dose conversion factors have been taken from the DOSDATA file of the latest version of the MACCS Code [4.7].

Inhalation doses were calculated using the "standard man" breathing rate of 3.5×10^{-4} m^3/sec. The dose rates for the immersion and inhalation pathways to three organs (whole body, red marrow, and thyroid) from individual radionuclide as well as the total are shown in Table 4-6.

The whole body <u>immersion</u> dose rate of about 0.1 rem/sec is dominated by Xe-133 which contributes about 70% and I-132 which contributes about 20% of the total dose rate. If it is assumed that it will take a minimum of 5 min (300 sec) for the workers to evacuate the loading platform and they continue to be exposed to the contaminated air at a constant exposure rate for this time, the total immersion whole body dose would be $\approx$ 30 rem.

The whole body <u>inhalation</u> dose rate of about 0.7 rem/sec is completely dominated by I-131 and is over seven times larger than the immersion dose rate. The I-131 isotope accounts for over 95% of this. Under the same conditions of evacuation, i.e. 5 minutes, the whole body inhalation dose would be over $\approx$ 200 rem which is about half of the LD_{50} dose of $\approx$ 450 rem. The thyroid inhalation dose rate is very large, about 24 rem/sec and 97.5% of this is again contributed by I-131.

If this accident is credible, the inhalation doses to the workers, who are assumed not to be wearing respirators, would be likely to lead to injury and some fatalities. If certain assumptions are relaxed then the question of fatalities becomes more speculative. For example, if the number of fuel rods releasing fission products were only half the number of rods in the 12 affected fuel assemblies or if the evacuation time was only two and a half minutes, then the dose received by workers would be 1/2 that calculated above.

Table 4-1

Energy Spectrum of γ-rays and Source Strength

Group #	Energy Interval, MeV	Mean Energy, E_m, MeV	# γ/Fission			Source Strength # γ/sec
			Prompt	Decay	Total	
1	0 - 1	0.5	5.2	3.2	8.4	8.4E+20*
2	1 - 3	2.0	1.8	1.5	3.3	3.3E+20
3	3 - 5	4.0	0.22	0.18	0.40	4.0E+19
4	5 - 7	6.0	0.025	0.021	0.046	4.6E+18

* Read as 8.4×10^{20}

Table 4-2

Calculated Values of Linear Attenuation, μR, for Various Photon Energies

γ-Energy, E_m, MeV	μ/ρ (cm^2/g)	μR
0.5	0.0966	96.6
2.0	0.0493	49.3
4.0	0.0339	33.9
6.0	0.0275	27.5

Table 4-3

Calculated Values of the Buildup Factor $B_\rho(\mu R)$

E_m	Taylor Form*					Berger Form+		
(MeV)	A	$-\alpha_1$	α_2	μR	B_ρ	a	b	B_ρ
0.5	100.85	0.1269	-0.1093	96.6	1.74E+7	1.4386	0.1772	3.78E+9
2.0	12.61	0.0532	0.0193	49.3	169.0	0.8229	0.0346	224.0
4.0	11.16	0.0254	0.0303	33.9	22.79	0.5801	0.0024	22.33
6.0	8.39	0.0182	0.0416	27.5	11.48	0.4633	-0.0109	10.44

*Coefficient values from Reference 4.2
+Coefficient values from Reference 4.3

Table 4-4

Calculated Values of γ-ray Fluence and Dose Rate

E	$(\mu_a/\rho)^{air}$	γ-fluence	Exposure Rate	Dose Rate	
MeV	cm^2/g	$\#\gamma/cm^2$-sec	(mR/hr)	mrad/hr	mrem/hr
0.5	0.0297	1.3E-21	1.2E-24	≈0	≈0
2.0	0.0238	1.7E-6	5.3E-9	≈0	≈0
4.0	0.0194	1.4E-1	7.2E-4	6.3E-4	6.3E-4
6.0	0.0172	4.8E+0	3.3E-2	2.9E-2	2.9E-2

(For converting rad to rem, the quality factor Q=1 for γ-rays)

Table 4-5

Material Released to Containment from Inadvertent Criticality

Isotope	Core Inventory (Ci)	Inventory of 12 Assemblies (Ci)	Gap Fraction	DF	Release to Containment R_i, (Ci)
Kr-85	4.670E+5	3.184E+4	0.3	1.0	9.552E+3
Kr-85m	7.100E+0	4.841E-1	0.1	1.0	4.841E-2
Xe-131m	7.681E+5	5.237E+4	0.1	1.0	5.237E+3
Xe-133	1.032E+8	7.036E+6	0.1	1.0	7.036E+5
Xe-133m	1.962E+6	1.338E+5	0.1	1.0	1.338E+4
Xe-135	2.461E+5	1.678E+4	0.1	1.0	1.678E+3
Xe-135m	9.295E+2	6.338E+1	0.1	1.0	6.338E+0
I-130	8.489E+3	5.788E+2	0.1	100.0	5.788E-1
I-131	5.168E+7	3.524E+6	0.1	100.0	3.524E+3
I-132	4.456E+7	3.038E+6	0.1	100.0	3.038E+3
I-133	6.106E+6	4.163E+5	0.1	100.0	4.163E+2
I-135	5.801E+3	3.955E+2	0.1	100.0	3.955E-1

Table 4-6
Dose Rates from the Released Radionuclides

Radionuclide	Immersion Dose Rate (rem/sec)			Inhalation Dose Rate (rem/sec)		
	Whole Body	Red Marrow	Thyroid	Whole Body	Red Marrow	Thyroid
Kr-85	1.44E-4	5.32E-5	6.47E-5	2.57E-5	1.53E-5	1.53E-5
Xe-133	7.11E-2	3.35E-2	7.89E-2	2.83E-3	2.71E-3	2.22E-3
Xe-135	1.21E-3	1.02E-3	1.29E-3	1.21E-5	9.82E-6	9.01E-6
I-131	3.84E-3	3.34E-3	4.09E-3	7.14E-1	5.04E-3	2.34E+1
I-132	2.04E-2	1.81E-2	2.24E-2	7.14E-3	9.70E-4	1.20E-1
I-133	7.32E-4	6.39E-4	7.81E-4	1.50E-2	2.59E-4	4.63E-1
I-135	1.87E-6	1.72E-6	2.11E-6	2.99E-6	2.01E-7	7.62E-5
TOTAL	9.74E-2	25.67E-2	1.08E-1	7.39E-1	9.00E-3	2.40E+1

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11. ABSTRACT (200 words or less)

This report documents the results of a study into the frequency and consequences of misloading fresh fuel assemblies during the reloading of a pressurized water reactor. The consequences that were considered included: i) loss of required shutdown margin, ii) inadvertent criticality, and iii) worker exposure within the plant given inadvertent criticality. Neutronic calculations were performed for different patterns of fresh fuel clustered together in a Combustion Engineering reactor. The fresh fuel considered had a high U-235 content and was assumed to be loaded without control element assemblies. The frequencies of misloading fresh fuel assemblies into these clustered patterns were calculated taking into account operator errors and equipment malfunctions that could occur during an offload/reload sequence. The study has improved our understanding of how difficult it is to misload fuel and has quantified the loss of shutdown margin and the frequency of occurrence for specific misloadings as well as the doses that might result from an inadvertent criticality.

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A Scoping Study

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Abstract

This report documents the results of a scoping study of rapid dilution events in pressurized water reactors. It reviews the subject in broad terms and focuses on one event of most interest. This event could occur during a restart if there is a loss-of-offsite power when the reactor is being deborated. If the volume control tank is filled with water at a low boron concentration then a slug of this water could accumulate in the lower plenum. This would be the result of the trip of the reactor coolant pumps leading to relatively low flow conditions and the restart of the charging pumps on emergency power. The concern is that this diluted slug will rapidly enter the core after a reactor coolant pump is restarted and this could cause a power excursion leading to fuel damage. This problem was studied probabilistically for three plants and the important design features that affect the core damage frequency were identified. This analysis was augmented by an analysis of the mixing of the diluted water with the borated water already present in the vessel. The mixing was found to be significant so that neglect of this mechanism in the probabilistic analysis leads to very conservative results. Neutronic calculations for one plant were carried out to understand the effect of nuclear design on the consequences of the event.

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Executive Summary

A rapid boron dilution event in a pressurized water reactor is postulated to occur when two requirements are met. The first requirement is that unborated or diluted water enters the reactor cooling system (RCS) during a period when there is very little circulation and the assumption is made that this water collects in a part of the system. The second requirement is that a reactor coolant pump (RCP) is started so that the slug of relatively diluted water passes rapidly through the core with the potential to cause a power excursion and fuel damage.

Although there are several scenarios that qualify as rapid boron dilution events, the one of most concern in this study occurs during a reactor restart. Analysis of this event without accounting for mixing of the diluted water entering the RCS results in a significant core damage frequency. However, if mixing is taken into account it becomes possible to have core damage only under the most extreme set of core initial conditions.

The reactor restart scenario occurs during the period when the reactor is being deborated according to normal procedures so that criticality can be achieved. A loss of offsite power (LOOP) is the initiating event. When this occurs there is reactor trip (the shutdown banks would be withdrawn during deboration) and trip of the charging pumps and the RCPs. When emergency power is brought on line the RCPs are not able to start but the charging pumps will start. It is assumed that the volume control tank (VCT), which supplies the suction for the charging pumps, is filled with highly diluted water. This water continues to be pumped into the RCS if the operator takes no action to switch the suction to a borated source. Since the RCPs are not running, if the natural circulation flow rate is low, the first requirement for a rapid boron dilution is met, i.e., there is the potential for a slug of diluted water to accumulate in the RCS, in this case most likely in the lower plenum.

The second requirement, that an RCP start, is fulfilled after offsite power is restored. It is assumed that the operators will start the RCP in order to resume the restart procedure. When this occurs it is assumed that the slug passing through the core adds sufficient reactivity to overcome the shutdown margin and cause a power excursion. Furthermore, the concern is that the power excursion is sufficient to cause fuel damage.

A probabilistic analysis had been done for this event for a European PWR. The estimated core damage frequency was found to be high enough so that corrective actions were taken. A system was installed so that the suction of the charging pumps would switch to the highly borated refueling water storage tank (RWST) when there was a trip of the RCPs. This was felt to reduce the estimated core damage frequency to an acceptable level.

In order to see if the core damage frequency might be as high in U.S. plants, a probabilistic assessment of this scenario was done for three plants. The plants chosen, Oconee, Calvert Cliffs, and Surry, represent a sample from the three U.S. reactor vendors, Babcock & Wilcox, Combustion Engineering, and Westinghouse. The estimated core damage frequency based on a scoping analysis was 2.8E-5/yr, 2.0E-5/yr, and 9.7E-6/yr for the three plants, respectively. These numbers are relatively high compared to desireable goals, but they are only the result of a scoping analysis and include many assumptions.

Although there were several conservative assumptions made in the analysis it should also be noted that there were several conditions present at these plants which might not be relevant at other plants and would make the core damage frequency *higher* at those other plants. One of these is the initiating frequency for a LOOP which is lower at U.S. plants than at plants in some other countries.

Another condition is related to plant design and the size of the source of unborated water during the event. In this analysis it was assumed that the volume of diluted water available was the volume within the VCT from the normal level to the low level at which point highly borated water from the RWST would automatically start to fill the VCT. In each of the three plants studied, the pump from the source of primary makeup water was tripped after the LOOP and remained tripped until full power was restored. If the design was such that this pump was connected to the emergency bus then the source of unborated water would be greatly increased and the core damage frequency would be increased. Since there are PWRs in Europe where this is the case it may also be true within the U.S. Therefore, some plants may have a higher vulnerability to this event than those chosen for study.

The two most important conservative assumptions in this analysis are:

- The mixing of the injectant is insignificant
- Fuel damage occurs when the slug passes through the core

The first assumption was found to be conservative by performing an analysis using a mixing model that had been developed to treat the mixing of streams of water at different tempertures. The mixing is significant when the injectant first enters the cold leg, when it enters the downcomer, and then as it moves down the downcomer into the lower plenum. The calculations were done for Calvert Cliffs and Surry. The reactor conditions assumed were that the RCS was initially stagnant and that the temperature of the injectant was either 100F° or 290F° lower than the initial temperature of the RCS. If there was significant loop flow due to natural circulation this would enhance the mixing.

The results of the mixing analysis were that the boron concentration in the lower plenum was not expected to be lower than 1080 ppm or 900 ppm for Surry and Calvert Cliffs, respectively, assuming in both cases that the boron concentration in the RCS at the time of the LOOP was 1500 ppm. This means that the reactivity addition would correspond to a change of only 400-600 ppm rather than the 1500 ppm that was theoretically possible.*

The second major conservative assumption in the probabilistic analysis is that sending a slug of diluted water through the core will cause fuel damage. In reality the effect of the slug will depend

* The extent of mixing means that the volume of diluted water created is much larger than the initial volume available from the VCT. Hence when the RCP is restarted the slug will remain in the core for a longer period of time than would be the case with no mixing. This will not have a strong impact on the initial power burst and the potential for immediate fuel damage, but would be important in understanding fuel behavior over a longer period.

on the reactivity addition caused by the diluted slug, the volume and geometry of the slug, the initial shutdown margin, and the thermal-hydraulic feedback.

A scoping analysis of the neutronics was done to show how these factors affect the consequences of the event and in particular whether catastrophic fuel damage might occur. This was defined as damage that could change fuel geometry and was determined by a local fuel enthalpy criterion of 280 cal/g. No consideration was given to other fuel damage mechanisms. A model of the Calvert Cliffs plant was used although some of the results have general applicability.

It was found that the slug boron concentration would have to be less than 430 ppm (i.e., a dilution of 1070 ppm) for catastrophic fuel damage to occur if the shutdown margin was 4% and the Doppler feedback was relatively strong. If the shutdown margin is smaller or if the core has a smaller Doppler feedback, then a smaller dilution would cause a problem. The Doppler feedback plays a very important role and varies significantly during a cycle and for different cycles so that it can have an important effect on the results.

The shutdown margin is made up of the worth of the shutdown bank which enters the core after the LOOP and the shutdown margin that existed prior to that. The shutdown bank worth typically varies from 2% to 5% depending on the plant. The pre-LOOP shutdown margin depends on when during the startup deboration that the LOOP occurs. If it occurs at the start of this period then the shutdown margin will be larger than if it occurs toward the end when the core boron concentration is closer to the value needed for criticality. The probabilistic analysis assumes that the core damage is equally likely anytime during the normal deboration period and, therefore, neglects this effect. For the Calvert Cliffs case the 4% used for the shutdown margin was assumed to be the effect of the shutdown bank only.

Based on the Calvert Cliffs neutronics calculations of shutdown bank worth and Doppler feedback, and mixing calculations indicating a slug boron concentration of 900 ppm, catastrophic fuel damage would not be expected. However, if the magnitude of the Doppler coefficient was half of that used, and the worth of the shutdown banks was only 2% then the result would be close to the fuel damage criterion. Furthermore, if the temperature of the slug was low, this would add to the severity of the excursion due to the positive effect of coolant temperature feedback.

It is important to note that these results will be a function of plant design. Every plant may be vulnerable to some form of rapid dilution event. Plants that use a diluted VCT to deborate may be vulnerable to the reactor restart scenario examined in detail in this study. For those plants the probability that this event leads to fuel damage will be a function of many design factors. Of particular importance are the volume and boron concentration of the VCT, the pumping rate of the charging flow and its orientation at the cold leg, and the reactivity worth of the shutdown banks and Doppler feedback.

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1 Introduction

1.1 Objectives

The general objective of this work is to improve the understanding of rapid boron dilution events in pressurized water reactors (PWRs). This is to help the U.S. Nuclear Regulatory Commission (NRC) to determine if any action should be taken to reduce the expected frequency of such events. This objective is met by examining different scenarios and performing scoping probabilistic and deterministic analysis. The probabilistic work quantifies the frequency of occurrence of one sequence which has been of particular interest to the NRC. This is done for plants of each of the U.S. reactor vendors. The deterministic work consists of neutronics and thermal-hydraulics calculations. The neutronics calculations are of the reactivity effect of different dilution slugs and the resulting power excursion. The thermal-hydraulics calculations are of mixing to understand whether unborated water introduced into the reactor coolant system (RCS) can remain as a slug of sufficient dilution to cause a problem.

1.2 Background

Boron dilution events have always been of concern in PWRs. A slow inadvertent dilution due to malfunction of the chemical and volume control system (CVCS) is a design-basis event which satisfies stringent acceptance criteria. The question of whether additional failures beyond the CVCS malfunction might lead to inadvertent criticality and fuel damage has also been addressed in the past. More recently this type of event and many other possible dilution scenarios have been surveyed in a study for the NRC [1.1]. That study noted that more scenarios were being postulated in different countries and that additional work would have to be done in the future to determine the importance of these events.

It is convenient to separate these beyond-design-basis dilution events into three types according to how they cause the power to rise in the core. For one type, the power excursion is caused by a relatively slow, uncontrolled dilution in which the boron concentration in the core changes slowly but steadily throughout the entire core. This type of event requires a large volume of diluted water. It is relatively easy to analyze as the power increase will be determined by a linear reactivity addition, mitigated by feedback effects, until stopped by operator action or the melting of fuel.

A second type of excursion occurs when pumps are off and diluted water accumulates in the lower plenum of the vessel to the extent that the bottom of the core becomes critical and power increases. This power increase causes an increase in the natural circulation flow rate which draws the diluted water up from the bottom of the vessel into the core. Without consideration of thermal-hydraulic feedback, this is an autocatalytic power excursion which is more rapid than the first type above.

The third type of power excursion is caused by a slug of diluted water rapidly entering the core. Because it is a slug, less diluted water is required than in the first type of dilution in which the diluted water mixes uniformly with the water in the RCS. It is this type of event that is currently of interest to the NRC [1.2] and that is the subject of this study. In order to have such an event it is first necessary to introduce diluted water into the RCS during a period when there is minimal circulation so that the water can collect in one place. The slug of diluted water can then be passed rapidly through the core by the startup of one or more reactor coolant pumps (RCPs) or the blowdown of one or more accumulators. This type of event is also different from the other two in that it has the potential of causing catastrophic fuel damage, i.e., rapid changes in fuel geometry, rather than relatively slow fuel melting.

The rapid dilution event (as well as the second, autocatalytic, type of excursion) would occur when the reactor coolant pumps are not running as would be the situation during a shutdown period or immediately following reactor trip. The shutdown period is also the time when the core might be most vulnerable to this type of event because control rods are already inserted and, therefore, reactor trip would not be possible to mitigate the power excursion.

1.3 Scope and Organization of this Report

This study is both an overview of rapid dilution events and a detailed analysis of one particular event. In Section 2 a review is given of scenarios which could lead to a slug of water with a low boron concentration passing through the core. One of these events, the reactor restart scenario, is of particular interest because in one country in Europe remedial action has been taken to help prevent the event. Hence, specific probabilistic and deterministic analysis was carried out for this scenario.

The analysis is considered of a scoping nature because it uses simple models and only considers a limited number of plant conditions and designs. The probabilistic analysis is described in Section 3. The analysis is carried out for the Oconee, Calvert Cliffs, and Surry plants, representing each of the U.S. reactor vendors. One of the important assumptions in the analysis is that there is insufficient mixing when a source of diluted water is introduced into the RCS so that that slug has the potential to remain intact and pass through the core when an RCP is started. This assumption is tested by performing an analysis of mixing that is described in Section 4. The analysis is for one particular plant and uses straightforward empirical models.

Section 5 describes the neutronic analysis that was carried out to understand the potential consequences of a diluted slug passing through the core. The analysis includes static calculations of reactivity for different slug geometries and dilutions and dynamic calculations of the power excursion that would be expected if these slugs passed through the core. Although the modeling is based on the reactor restart scenario, much of the analysis would be applicable to other scenarios which lead to a slug of diluted water passing through the core.

A summary of results and conclusions is given in Section 6 and Section 7 contains references.

2 Rapid Dilution Scenarios

2.1 Reactor Restart Scenario

One of the sequences that the industry has been aware of for a long period of time was recently studied in Europe. The result of a preliminary probabilistic analysis indicated that the frequency of occurrence of the event might be large in those plants and hence, the NRC issued an information notice [2.1].

The scenario occurs at the end of a shutdown period when the plant is being brought back to a critical configuration. The normal deboration is done when the reactor coolant system (RCS) is at hot, pressurized conditions and the shutdown banks are withdrawn.

The initiating event for this scenario is a loss of off-site power (LOOP). When that occurs there is a reactor trip and a trip of the reactor coolant pumps. The charging pumps would also stop but it can be assumed that emergency power is brought on line quickly and the charging pumps are energized from the emergency bus. These pumps will continue to pump from the volume control tank (VCT), emptying the diluted water that is present into the RCS. Assumptions are made that the operator takes no action to switch to a boration mode, and that the VCT contains a relatively large volume of water which is at a boron concentration that is much less than that originally in the RCS. This diluted water may be colder than the water in the RCS. When it gets pumped into the cold-leg it is assumed that there is minimal mixing so that the water can collect as a diluted slug at the bottom of the reactor pressure vessel (RPV). The probability of minimal mixing is enhanced if the event takes place after a long refueling when the decay heat level is low and consequently the amount of natural circulation in the RCS is relatively low.

Under these circumstances, if off-site power is recovered, it is likely that the operator will start an RCP in order to continue the process that was interrupted by the LOOP. This will send the slug of diluted water through the core and it is assumed that the reactivity added will be sufficient to overcome the existing shutdown margin and cause a power excursion leading to fuel damage.

When this event was studied in Europe as part of a probabilistic risk assessment (PRA) for operating and shutdown conditions, a relatively high frequency for a LOOP was used and other assumptions regarding inaction of the operator and the mixing of the water were assumed to hold so that the core damage frequency was found to be high. As a result of this rough estimate corrective action was recommended and a program of analysis and experimentation was initiated. The latter is to examine the effect of mixing which if present would eliminate the creation of the slug. The corrective action was a hardware change which would switch the suction of the charging pumps to the refueling water storage tank (RWST) when there was an RCP trip. Since the boron concentration in the RWST is very high this would eliminate the possibility of this accident. Taking into account the reliability of this new system would significantly reduce the expected frequency of occurrence.

2.2 Other Scenarios Involving Startup of an RCP

There are several other sequences that have been postulated which involve the startup of an RCP after a period in which diluted water has accumulated somewhere in the RCS. By necessity these sequences occur during shutdown. They have been studied in varying degrees of detail for one Westinghouse plant [2.2] and the following summary is based on that study.

A sequence studied by the French [2.3, 2.4] starts with a loss of power and failure of equipment involved in the startup procedure. This could lead to boiling in the core. If the auxiliary feedwater is operable this water is assumed to condense in the steam generator. The condensate which is unborated could accumulate in the cross over leg so that when the situation returns to normal and the RCPs are started, a diluted slug could be sent through the core.

Another source of diluted water in the cross over leg is secondary water. If steam generator tubes are cut either on purpose or inadvertently during steam generator modifications, and no repairs are made before the secondary side is brought back into service, then leakage of unborated water into the primary will occur when the secondary is filled. There were two such events [2.5] reported for the period from June 1969 to January 1981 which were found to cause an overall reduction in boron concentration rather than a localized diluted slug. These dilutions were both detected at an early stage and resulted in less than a 100 ppm change in RCS boron concentration.

A sequence studied in great detail by Swedish workers [2.6, 2.7] is one initiated by a steam generator tube rupture (SGTR). The plant is initially at hot zero power or if at power, it is shut down immediately. It is assumed that the RCPs are tripped due to a loss of power or some other cause. If the operators use a backfill cooldown procedure then unborated water from the secondary will enter the RCS. If this water does not mix but is assumed to collect in a stagnant part of the loop then if an RCP is started there is the possibility that the slug will enter the core and cause a power excursion.

Calculations performed in Sweden showed that the boron concentration in the core could go from 850 ppm to a minimum of 163 ppm in 10 seconds. These same calculations did not show any immediate fuel damage due to the energy deposition. However, the calculations are claimed to be inconclusive and further analysis is needed. As a result of this analysis Westinghouse, in 1990, recommended to the Westinghouse Owners Group that changes be made to the Emergency Response Guidelines regarding the procedures after a SGTR.

Other sources of unborated water during shutdown that could cause a problem if a slug collects and an RCP is started include the RCP seal water flow or a leaking thermal barrier or the water used to clean the refueling cavity after refueling. The cavity is hosed down with unborated water to remove radioactivity. An event involving more than 12 hours of inadvertent dilution from an unattended hose has occurred [2.8] causing a change in RCS boron concentration of 340 ppm.

2.3 Opening of Loop Stop Valves

A situation that is similar to a pump restart is the opening of a loop stop valve when pumps are running. Calculations had been done by Westinghouse [2.9] to determine the consequences of a startup of an inactive unborated loop without consideration of how the loop became diluted. All rods were assumed to be initially out of the core and hence, the worth of the scram reactivity would be considerable. In the worst case considered, where they also assume that the temperature of the water in the isolated loop is relatively cold, they calculated that approximately 3% of the fuel experiences clad rupture and <0.5% melt completely. However, insufficient information was presented to know what was the worth of the control rods and it is not possible to say that the calculations bound all possible consequences.

2.4 Blowdown of a Diluted Accumulator

If an accumulator has become diluted then if there is an inadvertent blowdown of that accumulator there is the potential for a diluted slug to pass through the core. The blowdown is postulated to occur during shutdown when the RCS is at atmospheric pressure and the accumulator is at operating pressure (625 psia). There is a motor operated valve that isolates the accumulator during shutdown. If this valve is not deenergized according to procedures, then there is possibility that it can open allowing the accumulator fluid to enter the cold loop and flow through the downcomer and into the core.

There are several ways in which the accumulator can become diluted. In a study at Brookhaven National Laboratory [2.10] it was determined that the most likely cause was back-leakage during operation at end-of-cycle. With a low boron concentration in the RCS and leakage through the check valves, the accumulator boron concentration could change dramatically if monitoring instrumentation was defective or operators did not respond properly.

A detailed probabilistic analysis of this type of event was carried out for a Westinghouse plant and showed that the expected frequency was insignificant. However, since that study an accumulator dilution has occurred which indicates that the most likely source of diluted water might be demineralized water that has been added for testing. This occurred in a French plant in July 1991. The unborated water that had been used for testing was left in the accumulator and eventually 350 ft^3 of this water flowed under gravity into the RCS. Although the discharge of the accumulator did not occur suddenly and the dilution of the RCS did not have any consequences, it was an important precursor and also indicates that the most likely source of diluted water might be due to maintenance rather than back-leakage.

3 Probabilistic Analysis

3.1 Introduction

In this section an estimate is made for the frequency of a rapid dilution event which could lead to core damage. The analysis is for the reactor startup scenario as described in Section 2.1. It is carried out for the Oconee, Calvert Cliffs, and Surry plants which were designed by Babcock & Wilcox, Combustion Engineering, and Westinghouse, respectively. This enables one to understand not only the important systems and operator actions with regard to this scenario, but also to identify any major differences that might exist between plants designed by each of the U.S. PWR reactor vendors. The specific plants selected for study were chosen because of the availability of information.

For each plant a summary description of the important systems for this type of event is first presented. This consists of a section describing the systems through which the unborated or diluted water might enter the reactor coolant system (RCS), and a section describing the relevant electrical systems. After this, the probabilistic analysis for each plant is presented in subsections on timing, modeling, and quantification. The quantification is done separately for refueling and nonrefueling outages. A summary section at the end presents the core damage frequencies for each plant and a discussion of important assumptions used in the analysis.

3.2 System Description - Oconee Station

3.2.1 Makeup and High Pressure Injection System

The makeup and dilution of the RCS is accomplished at Oconee using the High Pressure Injection (HPI) system. The relevant portions of the HPI and related systems are shown in Figures 3.1, 3.2 and 3.3. In normal operation a small amount of coolant is bled off from the RCS through the letdown and is directed to the purification demineralizers. The letdown (upper left on Figure 3.1) is cooled by the letdown coolers and can be isolated using several valves (HP-3[a], HP-4, HP-5 and HP-6). The output from the demineralizers is directed through a three way valve (HP-14, upper right on Figure 3.1) into the letdown storage tank (LDST) or into the deborating system where it is normally collected and stored in one of the RC bleed holdup tanks (upper left on Figure 3.3). Another RC bleed holdup tank holds demineralized water for dilution purposes.

Reactor coolant may directly enter the letdown storage tank through the three way control valve (HP-14) or from the RC bleed holdup tank by operating the RC bleed transfer pump 1A. The other RC bleed transfer pump, 1B, is used to supply fresh demineralized water during the deboration operation, transferring deborated water to the letdown storage tank.

The letdown storage tank serves as a surge tank and normal suction source for the HPI pumps (lower center on Figure 3.2). Another source of suction for the HPI pumps is the borated water storage tank (BWST).

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[a] Figures show valves without the "HP" designation.

The normal makeup flow is supplied by one of the HPI pumps and controlled automatically by a control valve (HP-120) to maintain the level in the pressurizer. During normal operation the three way valve (HP-14) is in "normal" position directing all letdown flow into the letdown storage tank. The valve may be placed in the "bleed" position to direct the letdown flow to the deborating demineralizer or RC bleed holdup tank. HP-14 is automatically placed in the "normal" position when there is low level in the letdown storage tank.

During startup of the reactor, the operator has to deborate the RCS from the shutdown boration level to achieve criticality. The dilution requires adding a predetermined amount of demineralized water to the RCS through the letdown storage tank. The deboration procedure, given in Table 3.1* directs the operator to determine the amount of demineralized water or batch size that is needed. The operator then sets the totalizer/batch controller (flow meter and integrator) to the desired setting and opens the makeup control valve HP-15 (middle right on Figure 3.1).

The control valve remains open until the totalizer indicates the end of the batch size and an automatic signal closes the valve. Even though the totalizer/batch controller is started and control valve HP-15 is opened, makeup to the letdown storage tank is prevented until the makeup isolation valve HP-16 is opened. Once the transfer path is established, the RC bleed transfer pump is started to add the desired quantity of demineralized water.

The rate of addition of deborated water may be as much as or less than the letdown flow. It could range from 45 to 90 gpm and normally is about 70 gpm. The volume of the batch size is generally larger than the volume of the letdown storage tank which requires the operator to position the HP-14 three way valve to the "bleed" position. Consequently, the boron concentration in the letdown storage tank may decrease as the demineralized water is being added by the RC bleed transfer pump. If the transfer rate is slower than the makeup rate through the HPI pumps then the three way valve has to be in an intermediate position to maintain the letdown storage tank level.

As a result of this process the following system conditions may be obtained:

1. The letdown storage tank volume is diluted to low boron concentrations by adding demineralized water

2. Depending on the demineralized water transfer rate or dilution rate, the boron concentration may be as low as 0-200 ppm (transfer equals makeup rate) or may range to a maximum of about 50% of the RCS boron concentration, i.e., 1000-1200 ppm (transferred demineralized water is mixed with letdown).

3. The water level in the letdown storage tank is maintained at an intermediate position (between high/low) during the deboration operation.

* Oconee Nuclear Station Operating Procedure OP/3/A/1103/04, "Soluble Poison Concentration Control," Duke Power Co., approved Feb. 28, 1989.

Table 3.1
Procedure Used to Dilute LDST

NORMAL MAKE-UP TO THE LDST

1.0 Initial Conditions

1.1 Determine the amounts of borated and unborated water to be used in the batch makeup.

2.0 Procedure

2.1 Set the desired batch size.

2.2 Ensure 3HP-15 (Makeup Control) is reset; control knob to OPEN and Toggle Switch to START.

2.3 Open 3HP-16 (Makeup Isolation).

2.4 Start the desired Bleed Transfer pump.

2.5 Open its respective discharge valve:
3CS-46 (Bleed Transfer Pump 3A Discharge).
OR
3CS-56 (Bleed Transfer Pump 3B Discharge).

2.6 If BLEED is required, ensure 3CS-26 (Letdown to BHUTs) and 3CS-41 (Bleed Tank 3A Inlet) are open.

2.7 If more volume is required than available in the LDST, position 3HP-14 (LDST Bypass) to BLEED.

2.8 When Batch size is reached:
2.8.1 Close 3HP-16 (Makeup Isolation).

2.8.2 Stop the selected Bleed Transfer pump and close its respective discharge valve:
3CS-46 (Bleed Transfer Pump 3A Discharge).
OR
3CS-56 (Bleed Transfer Pump 3B Discharge).

2.8.3 Ensure 3HP-14 (LDST Bypass) is in the NORMAL position.

2.8.4 Clear and reset 3HP-15 (Makeup Control).

3.2.2 Electrical System

The boron dilution accident of interest is initiated by a loss of off-site power during the deboration process. For this reason the specific features of the electrical distribution and supply system play an important role and must be discussed in some detail.

The line diagram of the Oconee electrical system is presented in Figure 3.4. The external grid connects to the 230 and 525 kV switchyards which are interconnected by a 230/525 kV autotransformer. One of the two buses (yellow bus) at the 230 kV switchyard plays a fundamental role in supplying power to the plant auxiliaries should the switchyard become isolated from the external grid.

The 230 kV switchyard and the yellow bus are also connected to a two-unit hydro station (Keowee Hydro) through an overhead line that provides emergency backup power. The hydro units perform the role played by diesel-generators at other plants. If there is a switchyard isolation (loss of grid), the 230 kV yellow bus will be reconnected to the hydro stations and be available to supply power to all station startup transformers.

The startup transformers CT-1, 2 and 3 (see Figure 3.4) can supply most of the unit auxiliaries, including the reactor coolant pumps (RCPs) which are connected to the 6.9 kV buses. If the switchyard is unavailable, then emergency power from the hydro station is provided through an underground connection to transformer CT-4, which supplies power to essential auxiliary equipment connected to the 4 kV buses. In this case, the RCPs cannot be utilized and may be restarted only after the switchyard or grid is recovered.

The important features of the electrical system may be summarized as follows:

1. Loss of grid events (not weather related) will not affect the availability of the 230 kV switchyard and the RCPs may be restarted at any time after the hydro units provide backup power.

2. Weather related loss-of-off-site-power events, or switchyard trouble, deenergizes the 6.9 kV buses and the RCPs may be restarted only after the recovery of the off-site grid or switchyard.

3.3 Probabilistic Analysis - Oconee Station

3.3.1 Accident Sequence Timing

The outcome in a boron dilution scenario is strongly dependent on the timing of events or the time evolution of the expected responses. In order to properly model and estimate the risk due to the accident scenario, the time behavior of the various events must be established with reasonable certainty.

Probabilistic Analysis

The deboration process itself is rather time consuming due to the small makeup and letdown flow relative to the total RCS volume. For the purpose of this analysis, it is estimated that the RCS contains about 75,000 gallon of water and the average makeup/letdown flow is about 70 gpm. The initial boron concentration at the start of the deboration is on the order of 2000 ppm and the final boron level is assumed to be 1500 ppm. The change in the RCS boron concentration (C) can be calculated using:

$$\frac{dC}{dt} * V_{RCS} = W_M * C_M - W_L * C_L \tag{3.1}$$

where V_{RCS} is the volume of the RCS, C_M and C_L are the boron concentrations of the makeup and letdown, respectively, and W_M and W_L the makeup and letdown flow rates, respectively. The initial boron concentration will exponentially be diluted to the final boron concentration as a function of time. Solving the above equation using $W_M = W_L$, $C_M = 0$, and $C = C_L$ gives a time span of 5 hours for the dilution process from 2000 to 1500 ppm.

The average length of deboration was estimated by the Oconee station operators to be 8-12 hours. This is consistent with the above calculation since the rate of deboration is dependent on the actual actions occurring during the startup process, and these are expected to necessitate a deboration slower than theoretically possible. Hence, the analysis will set the deboration time span as 8 hours.

The maximum amount of diluted primary grade water that can be injected into the RCS when the RCPs are not running (after a loss of off-site power or LOOP) is the available water in the letdown storage tank, which is assumed to be diluted to a very low boron concentration. The total volume between the HI/LO levels is approximately 420 ft^3. The water level is expected to be around midlevel, maintained by regulating the dilution flow and the position of the three-way valve, HP-14. Therefore, after a LOOP event the amount of diluted water volume is assumed to be approximately 250 ft^3 or 1870 gallons. Once the low level is reached, the suction source for the HPI pumps shifts to the BWST and highly borated water is pumped into the RCS. Assuming that the makeup flow is about 70 gpm, the time interval before the switchover to the BWST is about 20-25 minutes once the HPI pump is restarted after the LOOP event.

The probability for conditional core damage P(CCD) is defined in order to determine the time-dependent probability that there is core damage once the charging pumps start to pump diluted water into the RCS while there is no longer forced circulation. Without having the benefit of the mixing and neutronic calculations discussed in Sections 4 and 5, respectively, and in order to complete a scoping analysis, a simplistic approach is taken. For the situation after refueling, it is assumed that P(CCD) varies between zero and one depending on the amount of diluted water that enters the system. The value of zero is expected at the beginning of this time period when no diluted water has entered under the relatively stagnant flow conditions. The value of one is associated with the assumption that if the full diluted volume of the letdown storage tank (between HI/LO levels) is injected into the cold leg, a sufficiently diluted region will accumulate in the lower plenum so that fuel damage with the restart of an RCP is certain. Hence, the probability P(CCD) is assumed to increase linearly with time from zero to one in the time period of 25 minutes during which the diluted water in the letdown storage tank is pumped into the RCS.

After the suction source switches to the BWST the potential for core damage decreases since borated water is injected and presumably mixes with the diluted water. It is assumed that after about an additional 15 minutes P(CCD) decreases (linearly) to zero and there is no longer the possibility of a rapid dilution event occurring if the RCPs are restarted.

The time dependence of P(CCD) is shown on Figure 3.5. The bottom curve on the figure is for startups other than after a refueling. After a refueling (especially if it coincides with a long shutdown), the decay heat is relatively low because of the replacement of so many fuel assemblies (typically one-third of the core). However, after a shutdown without refueling which is most likely to occur in the middle of the cycle, the decay heat is expected to be much larger. It is expected to be sufficient so that the natural circulation flow rate is considerably higher than after a refueling. If the natural circulation flow rate is sufficiently large then after injection of diluted water there may be sufficient mixing to reduce the probability that there will be core damage with the restart of an RCP. This is taken into account by decreasing P(CCD) by a factor of 0.5 as is shown in Figure 3.5.

It is important to recognize in the curves shown in Figure 3.5 that t=0 corresponds to the time when the charging flow is reestablished through any of the power sources available. It does not correspond to the beginning of the LOOP. If the hydro units are available, then t=0 is the same as the initial time of the accident, however, for scenarios when the hydro units fail to provide power initially, t=0 corresponds to the recovery of either the hydro units or off-site power.

The restart of the RCPs after the LOOP is also modelled as a function of time. Once a power source is available either from off-site or from the hydro units, the operators are expected to start the RCPs. The preferred method of operating at this stage of the startup is to keep forced circulation in the RCS. Once the power is available, certain procedures have to be followed before the RCPs can actually be started. According to plant operational personnel it is expected that after about 30 minutes the RCPs would be running.

The model, therefore, assumes that the cumulative probability of restarting the RCPs increases from zero to one in the thirty minute time period after recovery of power. This is shown in Figure 3.6. Again the time t=0 does not correspond to the occurrence of the LOOP but rather, in this case, to the availability of high capacity electrical power, i.e., the off-site grid or the hydro unit through the 230 kV switchyard.

3.3.2 Accident Sequence Modelling

The event tree shown in Figure 3.7 was developed to evaluate the different accident sequences; in particular those leading to core damage (CD) due to rapid dilution as marked on the diagram. The first top event (ILOOP) is the accident initiator, i.e., the loss of off-site power event during the start up period after the plant was placed in a shutdown condition. The shutdown itself may be divided into two different categories: a) shutdown when refueling is done and b) shutdown without refueling. As explained in Section 3.3.1, this is important because of the relationship between decay heat levels and the probability of the diluted water from the letdown storage tank mixing with the higher concentration water in the RCS.

The top event SWYRD questions the type of LOOP event. One type (represented by the top branch) is grid related without affecting the availability of the 230 kV switchyard. This is important since the hydro station, through the switchyard, is capable of providing electrical power backup to all emergency sources including the RCPs. However, if the switchyard is affected, or the overhead connection to the hydro station is unavailable (represented by the bottom branch), then the RCPs may not be started, but the HPI pumps (charging flow) may have electrical power through the underground cables from the hydro station.

The top event HYDRO questions the availability of the hydro station given a demand. If the hydro station fails to start (bottom branch), it may be recovered and this is modelled by top event NR-HYDRO or non-recovery of the hydro station. For this top event the success path (top branch) represents the recovery of the hydro station.

The HPI pumps are powered from the emergency buses and their availability (and the expected charging flow) is questioned by top event CHG. The top branch signifies that they are available.

The top event NR-SWYRD questions the recovery either of the switchyard or the off-site grid, either of which would reestablish electric power to the RCPs. For this top event the success path represents the recovery of the power supply.

The top event CCD questions whether there is core damage given the amount of diluted water that has entered the RCS. The bottom branch represents the possibility that this occurs.

The last top event questions the status of the RCPs and whether or not the operator started one of them. The top branch represents the successful restart of the RCPs.

The most important top events in the accident event tree are time dependent and a conventional static approach is inappropriate to model the complete sequences. The time dependence may be included in the event tree by assuming that each top event is a time functional and the end-states are also dependent on time. This may be considered a process, where the event tree is being asked and evaluated at each time step $[t,t+\Delta t]$ and the final values are summed or integrated over the respective time period. The actual numerical evaluation of these integrals will be discussed in Section 3.3.3.

There are five sequences which involve potential core damage through a rapid boron dilution scenario. These are shown in Figure 3.7. The other sequences are either safe conditions or other types of accident sequences, such as station-black-out, which are not the subject of this analysis. The following is a short summary and description of each CD sequence:

Sequence 1 — Given the LOOP event the switchyard remains operational. The hydro station starts up and provides a backup source of power for the unit, including the RCPs. The charging flow is automatically reestablished by the

HPI pumps. The operator decides to start up the RCPs and depending on the elapsed time, core damage may occur.

Sequence 2 — Again, after a LOOP event the switchyard remains available, but the hydro station fails to start. When the hydro station recovers, the charging flow is immediately started and after a while the operator may start the RCPs leading to a reactivity excursion if a sufficient amount of diluted water is available in the RCS.

Sequence 3 — The switchyard is also affected by the LOOP event, however, the hydro station is able to provide emergency power through the underground connection. The charging flow is started, but the RCPs may be started by the operator only after the switchyard recovers.

Sequence 4 — Both the switchyard and the hydro station are initially unavailable. The charging flow is started after the hydro station recovers and when the switchyard is able to recover, the RCPs may also be started by the operator.

Sequence 5 — This is the same as Sequence 4 except that the switchyard recovers earlier than the hydro station and both the charging flow and the RCPs may be powered from the grid.

3.3.3 Accident Sequence Quantification

3.3.3.1 Refueling Outage

The frequency of the initiating event, ILOOP, is based on plant-specific data available in a recent update of the plant probabilistic risk assessment (PRA) [3.1]. The total rate of loss of off-site power from all causes is 9.0E-2/yr, which consists of two parts. Seventy percent of these events are such that the switchyard is unavailable or the overhead connection to the hydro station is affected. The remaining 30% of the cases are simple grid losses which do not affect the switchyard or the hydro station.

The refueling outage frequency is about 0.6/yr and the average duration of the startup dilution is 8 hours. It is assumed that the core damage frequency will be independent of when during the deboration the LOOP occurs. This is a conservative assumption as during the early phase of the deboration the initial shutdown margin will be large and the probability that the diluted water can cause a power excursion will be reduced. ILOOP is the product of the frequency of a LOOP (per hour), the duration of the deboration (hours), and the frequency of refueling, and hence, ILOOP= 4.93E-5/yr.

The probability of the top event SWYRD is simply the fraction of LOOP events which affect the switchyard and this was established in the plant specific PRA as 70%. Hence, P(SWYRD)=0.7. The probabilities for HYDRO and CHG are also based on the plant PRA and are

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P(HYDRO)=9.3E-3 and P(CHG)=8.4E-4. Both of these values were derived by examining the specific fault-trees representing the various failure modes.

The time dependence is modelled in the event tree by calculating the contribution of each sequence in each time period $[t, t+\Delta]$ and then summing to the total time for which there is core damage potential. The probability at a given time P(CCD) is obtained using the distributions shown in Figure 3.5. The probability per unit time for P(RCPRST) is from Figure 3.6. Since Figure 3.6 is a cumulative probability distribution it is the derivative of that curve with respect to time, i.e., (1/30) per minute, which is equal to P(RCPRST).

The evaluation of the time integral of the CD sequences involves a convolution integral (e.g., $\int dt\{P_1(t) \int dt'[P_2(t')...]\}$) with the appropriate probabilities. In the following, for simplicity, the short form $\int dt\{...\}$ will represent this type of integration.

For Sequence 1 the appropriate integral is:

$$CDF(S1) = \int dt\{ILOOP*[1-P(SWYRD)]*[1-P(HYDRO)]*[1-P(CHG)]*P(CCD)*P(RCPRST)\}.$$

In this sequence the integral is over the period 0 to 30 minutes as the RCPs are expected to be running by the end of this period. The integral of the time dependent portion, $\int dt[P(CCD)*P(RCPRST)]$, is evaluated from 0 to 25 minutes using the ascending part of the distribution shown in Figure 3.5 and from 25 to 30 minutes using the descending part of the distribution[a]. The numerical value is 0.56. Hence, the result for Sequence 1 is that CDF(S1)=8.2E-6/yr.

Sequence 2 is similar to the previous one except that the hydro station fails to start, but recovers to supply emergency power. Hence,

$$CDF(S2) = \int dt\{ILOOP*[1-P(SWYRD)]*P(HYDRO)*[1-P(NR\text{-}HYDRO)]*[1-P(CHG)] *P(CCD)*P(RCPRST)\}.$$

In this sequence, there are two time periods to consider. One is the time related to the recovery of the hydro unit and the other is the time associated with the start of the charging flow, RCPs and core damage potential. The latter period does not start until there is recovery and hence, the integral $\int dt[P(CCD)*P(RCPRST)]$ can be evaluated independently of the question of recovery of the hydro station. Using the results for Sequence 1 the value of this integral is 0.56. The integral $\int dt[1-P(NR\text{-}HYDRO)]$ can be assumed to be unity as it is expected that over a long period of time there would be recovery. Hence, the result for Sequence 2 is that CDF(S2)=7.0E-7/yr.

Sequence 3 represents the failure of the high capacity electrical power supply either through the loss of the main grid and the failure of the 230 kV switchyard, or the loss of the grid and the overhead supply line from the hydro station. In either case, the power supply to the RCPs are lost and there

[a] The integral is not carried out to 40 minutes because P(RCPRST) is zero after 30 minutes.

is no forced circulation. However, the underground connection to the hydro station remains intact and the emergency power supply is available for all essential equipment including the HPI pumps providing the charging flow to the RCS. The core damage frequency in this case is

$$\mathrm{CDF(S3)} = \int dt\{\mathrm{ILOOP}*P(\mathrm{SWYRD})*[1-P(\mathrm{HYDRO})]*[1-P(\mathrm{CHG})]*[1-P(\mathrm{NR\text{-}SWYRD})] *P(\mathrm{CCD})*P(\mathrm{RCPRST})\}.$$

The period of vulnerability is the first 40 minutes, since after that time sufficient borated water has been taken from the BWST to eliminate the possibility of a core damage event. During this 40-minute period the switchyard or the off-site power connection may recover, and consequently the RCPs may be restarted.

The recovery rate of off-site power, or the switchyard, will be assumed to be the same and constant through the entire period. Based on operating data for the industry [3.2], at the end of 60 minutes the total recovery is about 15%. Thus the recovery rate is [1-P(NR-SWYRD)]=(.15/60) per minute.

The time dependent portion of CDF(S3) may be written as

$${}_0\int^{40} dt\{[1-P(\mathrm{NR\text{-}SWYRD})]*{}_t\int^{t+30} dt' P(\mathrm{CCD})*P(\mathrm{RCPRST})\}$$

This integral is evaluated using the distribution for P(CCD) shown in Figure 3.5 and taking into account that the limit on the second integral (t+30) cannot go beyond 40 minutes. The result for this expression is 0.0415 and hence, CDF(S3)=1.42E-6/yr.

In Sequence 4, both the switchyard and the hydro fail and recover with a constant recovery rate. It is assumed that in this sequence the hydro units will recover before the switchyard or the grid. This implies that the charging flow is established first and then the RCPs may be restarted after the switchyard or the grid recovers.

The core damage frequency in this case is:

$$\mathrm{CDF(S4)} = \int dt\{\mathrm{ILOOP}*P(\mathrm{SWYRD})*P(\mathrm{HYDRO})*[1-P(\mathrm{NR\text{-}HYDR})]*[1-P(\mathrm{CHG})]*[1-P(\mathrm{NR\text{-}SWYRD})] *P(\mathrm{CCD})*P(\mathrm{RCPRST})\}.$$

Over a long period (e.g., 24 hours) the hydro unit is expected to recover. Hence, the non-recovery probability is neglected to simplify the calculation. Given the recovery of the hydro unit the sequence progresses as in Sequence 3. Once the charging flow is reestablished by the startup of the HPI pumps, there is a period of vulnerability of 40 minutes, if the RCPs are started during this time. Hence, the time dependence of CDF(S4),

$$\int dt\{[1-P(\mathrm{NR\text{-}HYDR})]*[1-P(\mathrm{NR\text{-}SWYRD})]*P(\mathrm{CCD})*P(\mathrm{RCPRST})\},$$

has the same numerical value as the integral evaluated for Sequence 3, i.e., 0.0415. This results in CDF(S4)=1.33E-8/yr.

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In Sequence 5 the switchyard, or the off-site grid, recovers earlier than the hydro station. The charging flow and the RCPs may be restarted as soon as the off-site power becomes available. The sequence is essentially identical with Sequence 2 except that the off-site power used in Sequence 2 is replaced by the hydro units through the overhead lines.

The core damage frequency in this case is:

$$CDF(S5) = \int dt\{ILOOP*P(SWYRD)*P(HYDRO)*[1-P(CHG)]*[1-P(NR\text{-}SWYRD)]*P(CCD)*P(RCPRST)\}.$$

The time dependent portion appearing under the integral is effectively the same as in Sequence 2 and was evaluated as 0.56. Hence, CDF(S5)=1.80E-7yr.

The CDF for this type of sequence after a refueling is the sum of the CDF for each of the above five sequences and is equal to 1.05E-5/yr.

3.3.3.2 Non-Refueling Outage

If the accident is assumed to occur after a drained or non-drained outage that does not involve refueling the event tree shown in Figure 3.7 is still applicable. The primary difference between the refueling and non-refueling outages is that the latter occurs with a relatively higher frequency since refueling is only done at about 18 months intervals. The operating history of the U.S. PWR population indicates that the average frequency of non-refueling outages is about 2/yr, which includes both drained and non-drained outages.

The plant specific LOOP frequency was found to be 9.02E-2/yr. The corresponding initiating frequency for the boron dilution event during a start up after a non-refueling outage, again assuming a dilution interval of 8 hours, is 1.64E-4/yr.

Another difference is the larger amount of decay heat after a non-refueling outage. This could considerably enhance the natural circulation rate in the RCS thereby increasing the probability that a slug of diluted water will mix with the borated water before the RCPs are turned on again. This has been taken into account (as explained in Section 3.3.1 and shown in Figure 3.5) by reducing the conditional probability for core damage following the restart of an RCP by a factor of 0.5.

The other variables are assumed to be the same as used for the case after refueling. The five sequences leading to core damage were requantified using the relevant data and resulted in the following results: CDF(S1')=1.37E-5/yr, CDF(S2')=1.16E-6/yr, CDF(S3')=2.36E-6/yr, CDF(S4')=2.21E-8/yr, and CDF(S5')=2.99E-7/yr. All the sequence frequencies increased by about 70% relative to the refueling case. The initiating frequency is about a factor of 3 higher, however, P(CCD) was lowered by a factor of 0.5.

The CDF for this type of dilution sequence after a non-refueling outage is the sum of the CDF for each of the above five sequences which is equal to 1.75E-5/yr.

3.4 System Description - Calvert Cliffs Station

3.4.1 Chemical and Volume Control System

The Chemical and Volume Control System (CVCS) is designed to perform various functions, the most important being: a) control of RCS volume (letdown and makeup), b) removal of corrosion and fission products, and c) boric acid concentration control. The CVCS consists of two major subsystem, the letdown and charging system and the makeup system. Simplified block diagrams of these systems are presented in Figures 3.8 and 3.9.

The normal reactor coolant letdown from one cold leg first passes through two letdown stop valves (upper left in Figure 3.8), then through the regenerative heat exchanger where its temperature is reduced by transferring heat to the makeup flow entering the RCS. Both letdown stop valves fail closed if instrument air pressure is lost (as in a LOOP event). The letdown flow then passes through the excess flow check valve and flows through the letdown control valve. The control valve is operated by a signal from the pressurizer level control system to maintain constant level in the pressurizer.

The temperature of the letdown is further reduced in the letdown heat exchanger (cooled by component cooling water) for the proper operation of the ion exchangers. A temperature controller on the outlet of the heat exchanger senses the letdown flow temperature and if it reaches a high level it shifts the three-way ion exchanger bypass valve to the bypass position preventing the hot liquid from entering the ion exchangers. A pressure control valve is also provided on the outlet of the letdown heat exchanger to prevent the fluid from flashing.

The letdown flow then passes through the purification filters, the ion exchangers and the letdown strainer before entering the Volume Control Tank (VCT). There is a three-way inlet valve (CVC-500 in Figure 3.8) that can be operated manually or automatically. In automatic mode the position of the inlet valve is controlled by the level in the VCT and for high level it directs the excess flow to the liquid waste processing system. Normally, the valve is aligned to direct letdown flow into the VCT.

The VCT is used to accumulate letdown water and RCP leak off, to receive makeup water from the makeup system, and to provide positive suction head for the charging pumps. The level in the VCT is controlled by a level controller, which at high level (110") shifts the inlet control valve position to bypass, at 90" starts automatic makeup from the makeup system and at 87.5" alarms as LO level.

If the level in the VCT drops to 3", the suction of the charging pumps is aligned to the Refueling Water Storage Tank (RWST) by closing the outlet valve (CVC-501) and opening the valve connecting to the RWST (CVC-504). The VCT supplies water to the charging pumps, which provide the makeup flow to the RCS. Three charging pumps are provided and normally, one pump is selected for operation. Each pump is capable of supplying 44 gpm makeup flow, which is returned to the RCS through the regenerative heat exchanger.

Probabilistic Analysis

The makeup system (see Figure 3.9) provides a predetermined amount of demineralized water and/or boric acid to the RCS. The system can be put in automatic, borate, dilute, or manual operating mode. For the purpose of this analysis only the dilute mode is discussed.

The dilute mode is used to decrease the boric acid concentration of the RCS. In this mode, the boric acid supply is not used and the makeup consists of demineralized water. The demineralized water is pumped by the reactor coolant makeup pumps (RCMP) through a flow element, a flow control valve and the makeup stop valve before it enters the VCT.

The actual dilution process is accomplished in the following steps:[a]

1. Operator determines that there is space available in the waste processing system for the diverted letdown. Charging and letdown is aligned for normal operation.

2. Total amount of demineralized water to be added is calculated by determining the difference between the desired and existing boron concentration (change in ppm) and relating it to the water volume to be added.

3. Makeup flow controller is set to the desired flow rate consistent with the number of operating charging pumps. The charging or makeup rate (normally 44 gpm - one pump) may be increased, if so desired. The makeup flow controller is shifted to auto position to start the makeup process.

4. Letdown is diverted to the waste processing system, if high level is reached in the VCT.

The main features of the dilution process relevant for the dilution reactivity accident are the following:

1. The VCT is diluted to low boron concentrations by adding demineralized water. The letdown flow is diverted allowing the VCT volume to be replaced by demineralized water.

2. The rate of makeup is matched to the charging rate and consequently the water level is maintained at normal level (95-105") in the VCT.

3. The VCT low level alarm (87.5") is substantially higher than the 3" level where the makeup source shifts to the RWST.

3.4.2 Electrical System

The main features of the electrical system at the Calvert Cliffs power plant are summarized below to the extent which is relevant to the boron dilution accident. The reactivity accident is postulated

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[a] Calvert Cliffs Nuclear Power Plant Operating Procedure OI-2B, Revision 10, "CVCS Boration, Dilution & Makeup Operations," Baltimore Gas and Electric Co., approved Sept. 24, 1985.

to occur during the RCS dilution when there is a LOOP event followed by the startup of an RCP. The availability of the electrical power supply to the charging pumps and the RCPs is the most important consideration with respect to the electrical system.

The simplified block diagram of the Calvert Cliffs electrical power system is shown in Figure 3.10. The major sources of electrical power are provided by off-site and on-site sources. The normal supply is through the 500 kV main buses, which are connected to the electrical grid by two 500 kV transmission lines. In addition, the two generators of the two units (Unit 1 & 2) are also connected to the main buses.

The 500 kV buses are connected to the 13 kV buses (see Figure 3.10) through the station service transformers which can also be directly supplied from off-site power by connecting them to a 69 kV transmission line. The 13 kV electrical buses are directly connected to the RCP motors and also energize the safety related 4 kV buses. The RCP electrical supply is, therefore, separated from all safety related loads (4 kV buses) and upon loss of the 500 kV/69 kV transmission line connection, the RCPs are without any major source of electrical power.

The 4 kV emergency electrical buses have a second source of power provided by the emergency diesel generators. Given a LOOP event, the breakers connecting the diesel generators to the emergency 4 kV buses close. The emergency diesel generators start and begin accepting loads in a predefined automatic sequence as determined by the load sequencer.

The charging pumps are connected to their electrical supply (which is powered by the diesel generators) 10 seconds after the generator breaker closes (Step 2 of the sequencer) and consequently, the charging pumps will continue to supply the makeup flow. In step 6 or about 30 seconds later, the instrument air compressors are also connected back to the safety buses. The letdown line is isolated upon loss of instrument air pressure, but this is unlikely given the relatively quick (in 30 seconds) restart of the air compressors.

The main features of the electrical system with respect to the boron dilution accident are the following:

1. The RCPs are powered directly from the 13 kV buses which are lost during a LOOP event (loss of electrical grid, two 500 kV and a 69 kV transmission line). There is no additional source of backup power source to the 13 kV buses. The RCPs may be restarted only if off-site power recovers.

2. The charging pumps and the instrument air compressors are sequentially loaded to the diesel generators and after a LOOP event this equipment is restarted in about 30 seconds.

3.5 Probabilistic Analysis - Calvert Cliffs Station

3.5.1 Accident Sequence Timing

In order to assess and adequately model the outcome of a potential dilution accident, the timing associated with the various events and operator actions as well as the corresponding probability values must be estimated and included in the modeling as was done for Oconee in Section 3.3.1. Two of the important considerations are the amount of diluted water available for injection into the RCS and the time period of the dilution process.

The total time for the dilution may be estimated by using Equation 3-1. The normal letdown and makeup rate $W_L = W_M$ is about 40 gpm with one charging pump in operation. Assuming that two pumps are put in operation for a faster dilution rate (84 gpm), the length of time to dilute the RCS from 2000 to 1500 ppm would be about 4.5-5 hours. Since the actual time varies greatly depending on the specific circumstances, it will be assumed (as was done for Oconee) that the dilution time is 8 hours.

The VCT level is maintained between 90-110" (Normal) and the switchover to the RWST occurs at 3" (LO/LO) and the water volume corresponding to 100" (Normal-LO/LO) is about 2900 gallon. Depending on the rate of charging, this volume may be injected into the RCS in about 30-50 minutes.

For Calvert Cliffs the sequence would most likely proceed as follows: As dilution proceeds at an average rate of 84 gpm, a LOOP event occurs. The RCPs coast down and the diesel generators start up establishing charging flow. The operator is likely to reduce the charging rate and tries to recover off-site power. The LOOP procedure[a] directs the operator to borate the RCS if a cooldown is expected. The boration may be accomplished either by using the boric acid addition system or simply supplying makeup water from the RWST.

In order to maintain the possibility of quick recovery and continuation of the start up procedure, rather than borate it is preferable to continue the makeup at a slower rate for a short period of time. Therefore, it is expected that the makeup from the VCT continues and the letdown is diverted to the VCT. The dilution from the makeup system would automatically stop, since the reactor coolant makeup pumps are powered from non-safety buses and are not connected to the diesel generators.

The amount of diluted water contained in the VCT, about 2900 gallons, is expected to be injected into the RCS at a rate which may last 50-70 minutes. It is assumed, based on the results discussed in Section 5, that after about 2000 gallons of diluted water are injected into the RCS, the passage of a diluted slug through the reactor core would lead to core damage. No credit is taken for mixing that might occur as discussed in Section 4.

[a] Calvert Cliffs Nuclear Power Plant Emergency Operating Procedure EOP-2, Revision 1, "Loss of Off-Site Power," Baltimore Gas and Electric Co., approved Feb. 10, 1988.

The conditional core damage probability, P(CCD), is assumed to vary linearly from zero to one in a time period of 40 minutes after injection starts, if the event occurs after a refueling. Since the amount of diluted water is about 2900 gallon, the value would remain at this level (P(CCD)=1) for about 20 more minutes. After the start of the boration or switchover to the RWST the probability of a dilution accident and consequently the value of P(CCD) would decline and reach zero at 80 minutes. Figure 3.11 presents the function P(CCD) as a function of time[a]. As was done for Oconee, a second distribution with half the probability is also shown on the figure to represent the situation when the startup is not after a refueling. This takes into account the higher decay heat and the greater likelihood that the diluted water will mix before the RCPs are restarted.

The cumulative probability distribution for the startup of the RCPs is assumed to be the same as for the Oconee calculation (Section 3.3). After recovery from a LOOP event, the operator will try to restart the RCPs in a 0-30 minute time frame. This is shown in Figure 3.6.

3.5.2 Accident Sequence Modelling

Figure 3.12 shows the boron dilution event tree developed for the Calvert Cliffs Station. The first top event, ILOOP, is the loss of off-site power initiator and represents the loss of the electrical grid and/or the two 500 kV and the 69 kV transmission lines.

The next top event, DSL, questions the availability of the emergency diesel generators which would provide backup power for the safety systems, but not for the RCPs. The diesel generators may fail to start, but could recover after a certain period of time and this is modelled in the top event NR-DSL or non-recovery probability of the diesel generators. Note that the top branch (i.e., success) under this event represents recovery of the diesel generators.

The charging pump availability is examined at the top event CHG. The recovery of the off-site power is an important event and P(NR-LOOP) expresses the probability of non-recovery in a given time interval and is the lower branch (or failure path) on the tree.

The last two top events are related to the condition of the diluted slug and its potential effect on the reactor core. CCD is conditional core damage given that the diluted water has entered the RCS. The RCPRST top event reflects the probability of restarting the RCPs after the LOOP event recovered.

There are three sequences marked in Figure 3.11 involving core damage potential. The other sequences are other unrelated scenarios that are not discussed here, since they do not involve dilution accidents. The three sequences are summarized as follows:

Sequence 1 After a LOOP event the diesel generators start and the charging flow is automatically reestablished. As soon as off-site power is recovered the operator restarts the RCPs in a time frame of about 30 minutes. The

[a] This is Option A. An Option B will be discussed in Section 3.5.3.1.

charging flow is reduced to 44 gpm to extend the time window for LOOP recovery before borating the RCS.

Sequence 2 — After the LOOP event the diesel generators fail to start, but recover sooner than the off-site power and charging flow is automatically restarted. After off-site power recovers the RCPs may start and core damage may result.

Sequence 3 — This is similar to Sequence 2 except the off-site power recovers earlier than the diesel generators. Both charging and the RCPs may be started leading to the dilution event.

3.5.3 Accident Sequence Quantification

3.5.3.1 Refueling Outage

The initiating frequency, ILOOP, is based on the loss of off-site power frequency data obtained from industrial experience. The loss of off-site power frequency is .11/yr or 1.26E-5/hr. The average yearly frequency of a refueling outage is 0.6/yr and the average length of a dilution during start up is about 8 hours. Hence, ILOOP=6.03E-5/yr.

The time independent probability values P(DSL) and P(CHG) were obtained from Reference 3.3. The value P(DSL)=1.44E-2 represents the unavailability of two diesel generators and P(CHG)=3.3E-3 reflects the unavailability of the three charging pumps.

The calculation of core damage frequency is done similarly to that for Oconee given in Section 3.3.3.1, i.e., with convolution integrals. The core damage frequency for Sequence 1 is:

$$CDF(S1) = \int dt\{ILOOP^*[1-P(DSL)]^*[1-P(CHG)^*[1-P(NR\text{-}LOOP)]^*P(CCD)^*P(RCPRST)\}$$

The terms ILOOP, P(DSL) and P(CHG) are independent of time but the time dependence of the other terms must be taken into account. The charging flow restarts after the LOOP event and there is a period of vulnerability of 80 minutes (see Figure 3.11). Hence, the portion of the integral where time dependence is considered is:

$$\int_0^{80} dt\{[1-P(NR\text{-}LOOP)]^* \int_t^{t+30} dt' P(CCD)^*P(RCPRST)\}$$

The above expression is evaluated using the distributions shown in Figures 3.6 and 3.11. P(CCD)=(t'/40) in the interval [0,40], =1 in the interval [40,60], and =(80-t')/20 in the interval [60,80] and P(RCPRST)=(1/30) per minute. The probability of recovering off-site power is [1-P(NR-LOOP)]=(0.15/60) per minute and is based on data for 15% recoveries in one hour. The result of the evaluation is a probability of 0.12. Hence, CDF(S1)=7.11E-6/yr.

In Sequence 2 the diesel generators (DGs) fail initially, but at least one DG recovers before the off-site power recovers. The core damage frequency is:

$$CDF(S2) = \int dt\{ILOOP*P(DSL)*[1-P(CHG)]*[1-P(NR\text{-}DSL)]*[1-P(NR\text{-}LOOP)]*P(CCD)*P(RCPRST)\}.$$

This sequence may be evaluated using the results for Sequence 1 and noting that over a long period (24 hours) the diesel generators would eventually recover. This implies that $\int dt[1-P(NR\text{-}DSL)]$ is approximately one and the other time dependent terms are independent of the diesel recovery rate. The last three terms in the integral for CDF(S2) are the same as for Sequence 1 and hence, CDF(S2)=1.04E-7/yr.

In Sequence 3 the off-site power recovers sooner than the diesel generators. The charging flow is immediately reestablished, but the RCPs are restarted during a 30-minute time period. At the end of the 30 minutes, the RCPs are running and total mixing is assumed. The core damage frequency is:

$$CDF(S3) = \int dt\{ILOOP*P(DSL)*[1-P(NR\text{-}LOOP)]*P(CCD)*P(RCPRST)\}.$$

As was the case with the recovery of DGs in Sequence 2, in this sequence it can also be assumed that the probability of recovery of off-site power is unity over a 24 hour period so that $\int dt[1-P(NR\text{-}LOOP)]$ is taken as one. The last two terms are evaluated:

$$\int_0^{30} dt P(CCD)P(RCPRST)$$

since the RCPs would have started by the end of this 30-minute period. For this integral P(CCD)=t/40. This results in CDF(S3)=3.30E-7/yr.

The total core damage frequency for a refueling outage for this type of rapid dilution scenario is the sum of the frequencies for the three different sequences or 7.54E-6/yr.

In the above analysis it was assumed that the makeup rate after a LOOP was reduced to 40 gpm from the normal 84 gpm in order to extend the time period before it might be required to initiate boration of the RCS. The effect of this assumption is examined by assuming that the makeup rate remains at 84 gpm. The effect on P(CCD) is shown as Option B in Figure 3.13. The maximum value is now reached at 25 minutes and the period of vulnerability for the accident is reduced to 60 minutes from the 80 minutes with the smaller makeup flow.

With the different makeup rate the core damage frequencies for the three sequences become 4.74E-6/yr, 6.92E-8/yr, and 5.04E-7/yr, respectively. The total core damage frequency is 5.31E-6/yr which is a reduction of 30% and reflects the decreased period of vulnerability.

3.5.3.2 Non-Refueling Outage

For a non-refueling outage the approach is similar to that explained in Section 3.3.3.2. The initiating frequency is higher than for a refueling outage because there are two of these outages per year. Hence, ILOOP=2.01E-4. The other change is the reduction in the conditional probability for core damage given the injection of the diluted water and the restart of an RCP. This is because of the higher natural circulation flow rate assumed for these outages. This is reflected in the time dependent probability P(CCD) shown in Figure 3.11. Th. esult with the assumption of 40 gpm for the makeup flow is: CDF(S1')=7.11E-6/yr, CDF(S2')=1.04E-[illegible]/yr, and CDF(S3')=3.30. This results in a total core damage frequency of 1.25E-5/yr.

If the assumption is made that the makeup flow rate is not changed from the 84 gpm expected before the LOOP (Option B), then the core damage frequencies for the three sequences become 7.87E-6/yr, 1.17E-7/yr, and 8.37E-7/yr, respectively, and the total becomes 8.82E-6/yr.

3.6 System Description - Surry Station

3.6.1 Chemical and Volume Control System

At the Surry plant it is the chemical and volume control system (CVCS) which controls makeup and letdown and boron concentration. Figures 3.14 and 3.15 show the boron control and letdown/makeup control subsystems of the CVCS, respectively.

To carry out the RCS deboration, the quantity of primary grade water required, along with the rate of addition, is first determined from tables and graphs[a] and set on the batch integrator control. A sample sheet from this procedure is shown in Table 3.2. When boron dilution is initiated, both the primary grade water flow control valve (FCV-1114A on Figure 3.14) and the primary grade makeup stop valve (FCV-1114B) open to establish the flow to the Volume Control Tank (VCT), with the primary water supply pump running.

The boric acid flow control valve (FCV-1113A) is closed so that only primary grade water can enter the VCT through the VCT spray nozzle. The primary makeup stop valve (FCV-1113B) is also closed to prevent the primary grade water from going directly into the charging pump suction header. This is a precaution against a sudden increase in reactivity although bypassing is allowed during the early phase of a xenon transient. When the amount of injected primary grade water reaches the value set on the batch integrator, the makeup valves shut automatically. It is important to note that the rate of addition of primary grade water is determined by the operator and is generally less than the charging rate.

The VCT has an internal volume of 300 cubic feet and normal operating pressure and temperature of 15 psig and 105°F, respectively. The spray nozzle flow is normally about 120 gpm. The VCT level control valve (LCV-1115A), located upstream of the VCT, is a solenoid-operated control valve which is positioned by instrument air to maintain the VCT level at less than 85%. When the VCT level reaches a preset value, the VCT level control valve will begin to direct letdown flow to the Boron Recovery System. At a VCT level of 85%, all letdown flow is diverted to the Boron Recovery System. In the event that the VCT level falls to 13%, the charging pump will automatically shift its suction from the VCT to the Refueling Water Storage Tank (RWST).

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[a] Surry Power Station Operating Procedure 1-OP-8.3.2, "Blender - Dilute," Virginia Electric Power Co., Aug. 19, 1988.

Table 3.2
Extract from Surry Dilution Procedure

INITIALS

5.0 Procedures

______ 5.1 Initial conditions satisfied.

______ 5.2 Precautions and limitations noted and satisfied.

______ 5.3 Determine the rate and magnitude of primary grade water, to get the desired dilution (see nomograph).

______ 5.4 Place the MAKEUP MODE CONTROL switch to the "STOP" position.

______ 5.5 Set the primary grade flow controller (FC-114A) for desired flow rate and set primary grade water integrator (YIC-114A) to desired quantity.

______ 5.6 Place the MAKEUP MODE SELECTOR switch to "DILUTE".

______ 5.7 Place the MAKEUP MODE CONTROL switch to "START".

______ 5.8 Verify the following actions taken place:

Dilute	FCV-1113A	FCV-1113B	FCV-1114A	FCV-1114B
Mode	Closed	Closed	Controlling	Open

______ 5.9 Dilution will be automatically stopped when the integrator (YIC-114A) setpoint is reached.

______ 5.10 When the dilution operation is complete, refer to 1-OP-8.3.1 for returning blender to automatic.

During the RCS deboration, the primary grade water exiting from the VCT is injected into cold-leg B by a charging pump at a rate of approximately 120 gpm. About 20% of this flow is diverted to the RCPs for use as seal water. The letdown flow from cold-leg A, in the meantime, is partially diverted to the Boron Recovery System. As explained above, the primary grade water addition is usually less than the charging rate and in order to maintain a constant level in the VCT some letdown flow must be diverted to the Boron Recovery System. The water level in the VCT is maintained between 60-85%. This implies that during the period of about eight hours required for the RCS deboration at Surry, about 60-85% (corresponding to approximately 1800 gallons) of the VCT volume is filled with highly diluted primary grade water.

When the charging pump stops due to loss of off-site power, the letdown orifice isolation valve (HCV-1200B) shown in Figure 3.15 will close automatically, isolating the letdown flow. Unless the operator reopens the valve by resetting the handswitch to OPEN position, it will remain closed even after the charging pump is restarted by emergency diesel power. At the moment the off-site power is lost, it can be assumed that the VCT still contains about 1800 gallons of primary grade water. Since power to the primary water supply pump is also lost, no more primary grade water is pumped into the VCT. From this point on, two somewhat different scenarios are conceivable depending upon whether the letdown flow is quickly restored following the restart of the charging pump.

Probabilistic Analysis

If the letdown orifice isolation valve (HCV-1200B) is not quickly reopened, the relatively cold (120°F) primary grade water will be injected into the RCS by the charging pump without being heated properly at the regenerative heat exchanger due to the absence of letdown flow on one side of the heat exchanger. As the water level in the VCT decreases, the VCT level control valve (LCV-1115A) will gradually realign to allow any letdown flow to enter the VCT. Since no letdown flow is entering the VCT, the VCT water level will continue to fall until it reaches the level of 13%, at which point the suction of the charging pump will automatically switch from the VCT to the RWST. At a flow rate of 120 gpm, it will take about 15 minutes for the VCT level to drop to the 13% level. The amount of primary grade water discharged from the VCT into the RCS during this period is approximately 1600 gallons.

If the charging pump continues to run, this volume of primary grade water will be followed by intake from the RWST with a boron concentration of 2300 ppm. It is likely, however, that the letdown flow will be reestablished before the VCT level falls to the 13% level. Since the letdown flow is isolated, the charging flow introduced into the RCS will cause the pressurizer level to gradually increase. The charging flow control valve (FCV-1122) is controlled by a signal from the pressurizer level instrumentation to maintain a prescribed pressurizer water level. It will automatically close if the level setpoint is reached. If this happens, the operator is likely to quickly reestablish the letdown flow so that the charging flow can be maintained. The subsequent scenario will be similar to that which will be described below for another series of events.

Assuming that the letdown orifice isolation valve (HCV-1200) is reopened by the operator soon after the restart of the charging pump, the VCT water level will still fall initially, causing the VCT level control valve (LCV-1115A) to adjust its position to admit the letdown flow to the VCT. The boron concentration in the letdown flow at this point could range from 1500 to 2300 ppm, depending on when during the deboration the LOOP occurs. Before this letdown flow gets recirculated into the RCS there will be about 1800 gallons of (almost) primary grade (PG) water injected since all the PG water remaining in the VCT can be drained by the charging pump. It will take about 15 minutes for the charging pump to inject the 1800 gallons of PG water into the RCS.

3.6.2 Electrical System

A simplified block diagram for the Surry emergency electrical system is presented in Figure 3.16. The RCPs are connected to the non-safety related buses 1A, 1B and 1C. These buses are not supported by any secondary backup source upon losing electrical power from a loss of off-site power event. The charging pumps, however, are connected to either safety related bus 1H or 1J and these are powered in a LOOP event by the respective diesel generators DG1 or DG3.

3.7 Probabilistic Analysis - Surry Station

3.7.1 Accident Sequence Timing

It is apparent from the forgoing discussions that, in addition to the accumulation of a slug of PG water in the primary system, the off-site power has to be recovered and one RCP has to be started up in order for this type of reactivity accident to happen. The times at which these causative events occur relative to each other appear to play an important role in determining the probability and severity of such an accident. As noted previously, it takes about 15 minutes to inject all the PG water left in the VCT into the cold leg. If the off-site power is recovered in about 15 minutes after it is lost, and one of the RCPs is restarted immediately, the likelihood that the accident will occur is maximized. Since the injection of PG water is followed by that of either RWST water (which contains 2300 ppm of boron) or letdown water (which contains at least 1500 ppm of boron), the slug of water which has settled at the bottom of the lower plenum will eventually mix with the borated water reducing the probability of a dilution accident.

Assuming that the letdown water contains 1700 ppm of boron when it is injected into the RCS following the exhaustion of PG water in the VCT, and that complete mixing will occur in the lower plenum, about 10 minutes is sufficient to raise the boron concentration in the lower plenum to the level that is no longer a threat to criticality. In other words, if off-site power is not recovered for more than 25 minutes and if during this period, the charging pump (which was restarted by diesel power soon after loss of off-site power) continues to inject water from either the VCT (PG water followed by letdown water) or the RWST, the frequency of occurrence of this accident will become vanishingly small.

The time behavior of the conditional core damage probability or P(CCD) is modelled as indicated in Figure 3.17. It increases linearly to a value of one in 15 minutes. After this point the probability decreases linearly to zero in another 10 minutes. After this time there is no longer the possibility of this sequence occurring.

The RCP may be restarted after recovery from the LOOP event and this is modelled as it was for Oconee and Calvert Cliffs. The time dependence of the cumulative probability is shown in Figure 3.6 and indicates that the RCPs are expected to be started within 30 minutes of recovery from the LOOP.

3.7.2 Accident Sequence Modeling

To perform the probabilistic analysis for this accident scenario, the event tree shown in Figure 3.18 was developed. This event tree was applied for both refueling and nonrefueling outages. As was seen in the analysis for Oconee and Calvert Cliffs, the top events that change with outage are ILOOP and CCD.

The second top event, DSL, in Figure 3.18 questions whether the emergency diesel generators are available and provide back up power to the emergency safety buses. The third top event, NR-DSL, questions the recovery of the diesel generators. If the diesel generators are available, then the charging flow may be started, if at least one of the three charging pumps are available and its respective electrical supply bus is powered. This function is represented by top event CHG.

The recovery of off-site power is questioned next in top event NR-LOOP. This is a prerequisite for restarting the RCPs, since the diesel generators do not have sufficient capacity. Once the RCPs can be started the top event CCD questions the potential for core damage. The restart of the RCPs is questioned in the top event RCPRST.

Probabilistic Analysis

The event tree as indicated in Figure 3.18 is a time dependent tree, since some of the top events strongly depend on the time and this must be taken into account to evaluate the final CDF. There are three sequences marked as leading to core damage (CD) and these indicate where the boron dilution reactivity excursion leads to core damage. The following is a short description of each CD sequence:

Sequence 1	After a LOOP event the diesel generators start and the charging flow is automatically reestablished. As soon as off-site power is recovered the operator will restart the RCPs in a time frame of 30 minutes.
Sequence 2	After the LOOP event the diesel generators fail, but recover sooner than the off-site power and charging flow is immediately restarted. After off-site power recovers the RCPs may start and core damage may result.
Sequence 3	This is similar to Sequence 2 except the off-site power recovers earlier than the diesel generators. Both charging and the RCPs may be started leading to a reactivity accident.

3.7.3 Accident Sequence Quantification

3.7.3.1 Refueling Outage

The initiating event frequencies were calculated based on (Surry) plant specific data. For refueling outages the frequency is 0.6/yr and the failure rate of off-site power is 2.85E-5/hr. Assuming that the RCS deboration requires about eight hours, ILOOP=6.03E-5/yr.

The time independent probabilities P(DSL) and P(CHG) were obtained from Reference 3.3. P(DSL)=3.7E-03 represents the unavailability of two diesel generators and P(CHG)=4.0E-03 reflects the unavailability of the three charging pumps.

The calculation of core damage frequency is done similarly to that for Calvert Cliffs given in Section 3.5.3.1. The core damage frequency for Sequence 1 is:

$$CDF(S1) = \int dt\{ILOOP*[1-P(DSL)]*[1-P(CHG)]*[1-P(NR\text{-}LOOP)]*P(CCD)*P(RCPRST)\}.$$

The first three terms ILOOP, P(DSL) and P(CHG) are independent of time and only the last three terms must be evaluated by considering their time dependence. Effectively, the charging flow starts after the LOOP event and there is a period of accident vulnerability of 25 minutes (see Figure 3.17). The time dependent portion may be written:

$$\int_0^{25} dt\{[1-P(NR\text{-}LOOP)]_t \int_t^{25} dt' P(CCD)*P(RCPRST)\}$$

The above expression is evaluated taking into account the functions given in Figures 3.6 and 3.17. P(RCPRST)=(1/30) per minute. P(CCD)=t'/15 in the interval [0,15] and =(25-t')/10 in the interval [15,25]. The value of [1-P(NR-LOOP)], or the probability of LOOP recovery, is (0.15/60) per minute based on a 15% recovery rate at the end of one hour. Evaluating the integral in the subintervals leads to a probability of 5.92E-2. Hence, CDF(S1)=3.55E-6/yr.

In Sequence 2 the diesel generators fail initially, but at least one DG recovers before the off-site power recovers. The CDF is evaluated from:

$$CDF(S2) = \int dt\{ILOOP*P(DSL)*[1-P(CHG)]*[1-P(NR\text{-}DSL)]*[1-P(NR\text{-}LOOP)] *P(CCD)*P(RCPRST)\}.$$

This sequence may be evaluated using the results of Sequence 1 by noting that over a long period (24 hours) the diesel generators are expected to recover. This implies that $\int dt[1-P(NR\text{-}DSL)]$ is approximately one. The other time dependent terms are independent of when the diesels recover. For this sequence the last three terms are the same as for previous sequence and CDF(S2)=1.32E-8/yr.

In Sequence 3, the off-site power recovers sooner than the diesel generators. The charging flow is immediately reestablished, but the RCPs are restarted over a 30-minute period. At the end of the 30 minutes, the RCPs are running and total mixing is assumed. The core damage frequency is:

$$CDF(S3) = \int dt\{ILOOP*P(DSL)*[1-P(NR\text{-}LOOP)]*P(CCD)*P(RCPRST)\}.$$

The integrated probability for off-site recovery is similar to that for diesel recovery, i.e., eventually all LOOP events recover within 24 hours so that $\int dt[1-P(NR\text{-}LOOP)]$ is about one. The last two terms are the only time dependent terms and they are evaluated from 0 to 25 minutes using the functions given in Figures 3.6 and 3.18. The final result is that CDF(S3)=9.37E-8/yr.

The total core damage frequency is obtained by summing the frequencies for the three sequences and is 3.66E-6/yr.

3.7.3.2 Non-Refueling Outage

The frequency of the accident during a non-refueling outage is calculated taking into account the different initiating frequency and a different probability for conditional core damage (as seen in Figure 3.17). The frequency of a drained maintenance outage is 1.2/yr and a non-drained maintenance outage is 0.6/yr. Using these numbers and the estimated frequency for a loss of off-site power results in ILOOP=2.01E-4/yr.

The results for the same three sequences are now: CDF(S1')=5.89E-6/yr, CDF(S2')=2.19E-8/yr, and CDF(S3')=1.56E-7/yr. The total core damage frequency for this type of accident in this type of outage is 6.08E-6/yr.

3.8 Summary

The results of the analysis for the three plants are given in Table 3.3 which shows not only the expected core damage frequency (CDF), but also the initiating frequency of these events. The latter was of particular interest because in the analysis done in Europe (cf Section 2.1), the initiating frequency was quoted as being an order of magnitude higher. The CDF is similar for all three plants and in the range considered significant.

Table 3.3
Summary of Important Frequencies

	OCONEE		CALVERT CLIFFS[a]		SURRY	
	INIT FR /YR	CDF /YR	INIT FR /YR	CDF /YR	INIT FR /YR	CDF /YR
REFUELING	4.93E-5	1.05E-5	6.03E-5	7.54E-6	6.03E-5	3.66E-6
NON-REFUELING	1.64E-4	1.75E-5	2.01E-4	1.25E-5	2.01E-4	6.08E-6
TOTAL		2.80E-5		2.00E-5		9.74E-6

[a] Option A

These results are dependent on plant design and various assumptions used in the analysis. The most important assumptions are summarized below. Note that some of them result in overestimating the core damage frequency.

1. The dilution time during startup is 8 hrs. The consequences of the event are independent of when during this period the loss of off-site power occurs. In reality, an event occurring early during this period will have more shutdown margin to overcome and is, therefore, expected to have less of an effect than an event occurring near the end of the normal dilution procedure.

2. No credit is given for the operator to take action and stop the charging flow from the VCT after the LOOP. Although dilution while the shutdown banks are inserted or the RCPs are stopped, (as would be the case after a LOOP) is not a normal procedure, it is assumed that since the operator knows that the flow from the primary water makeup pump has ceased that no other action is deemed warranted. An action that could be taken by the operator would be to switch the charging pump suction to the RWST.

3. For all three plants the dilution is done with flow from the VCT. It should be noted that in some plants the suction for the charging flow comes directly from the primary grade makeup water source and once the PG water pump is tripped there is no longer the potential for adding unborated water to the RCS.

   Oconee: The dilution rate is about the same as the letdown and the volume in the letdown storage tank is diluted to low boron levels (0-200 ppm). The available volume in the letdown storage tank is about 1900 gal.

Calvert Cliffs: The dilution rate is matched by the letdown flow rate. The volume control tank (VCT) is eventually diluted to a very low boron concentration. The available volume for injection into the RCS is about 2900 gal.

Surry: The dilution rate is generally lower than the charging rate, and the VCT may get diluted to a very low boron concentration (0-100 ppm). The dilution flow is always directed to the VCT and bypassing is allowed only during xenon transients. The available volume for injection into the RCS is about 1600 gal.

4. For all three plants the potential for an accident is limited by the amount of diluted water in the VCT as the supply of primary grade water is stopped by a PG makeup pump. However, there are plants where this pump is connected to the emergency bus and the probability of an accident will be increased if primary grade water continues to be pumped into the VCT. This appears to be the case for some plants in France and Sweden and is another reason why the problem may be more serious there. The question of whether the makeup pump trips or continues to run has to be evaluated on a unit by unit basis.[a]

5. Refueling outage: The conditional core damage probability is linearly changing between zero and one, corresponding to the amount of diluted water injected into the RCS. Mixing and switchover to a borated source reduces the probability of core damage from one to zero over a short period of time.

   Non-refueling outage: The probability for conditional core damage varies between zero and one-half to account for the potentially higher natural circulation rate and mixing.

   For any outage the probability of core damage used is expected to be conservative because it does not account for any mixing that may occur (cf Section 4).

6. If off-site power, or another adequate power source, is available, the reactor coolant pumps (RCPs) will be started over a 30-minute interval.

[a] At the Ringhals plant in Sweden of three units designed by Westinghouse, two have the PG water pump connected to an emergency bus.

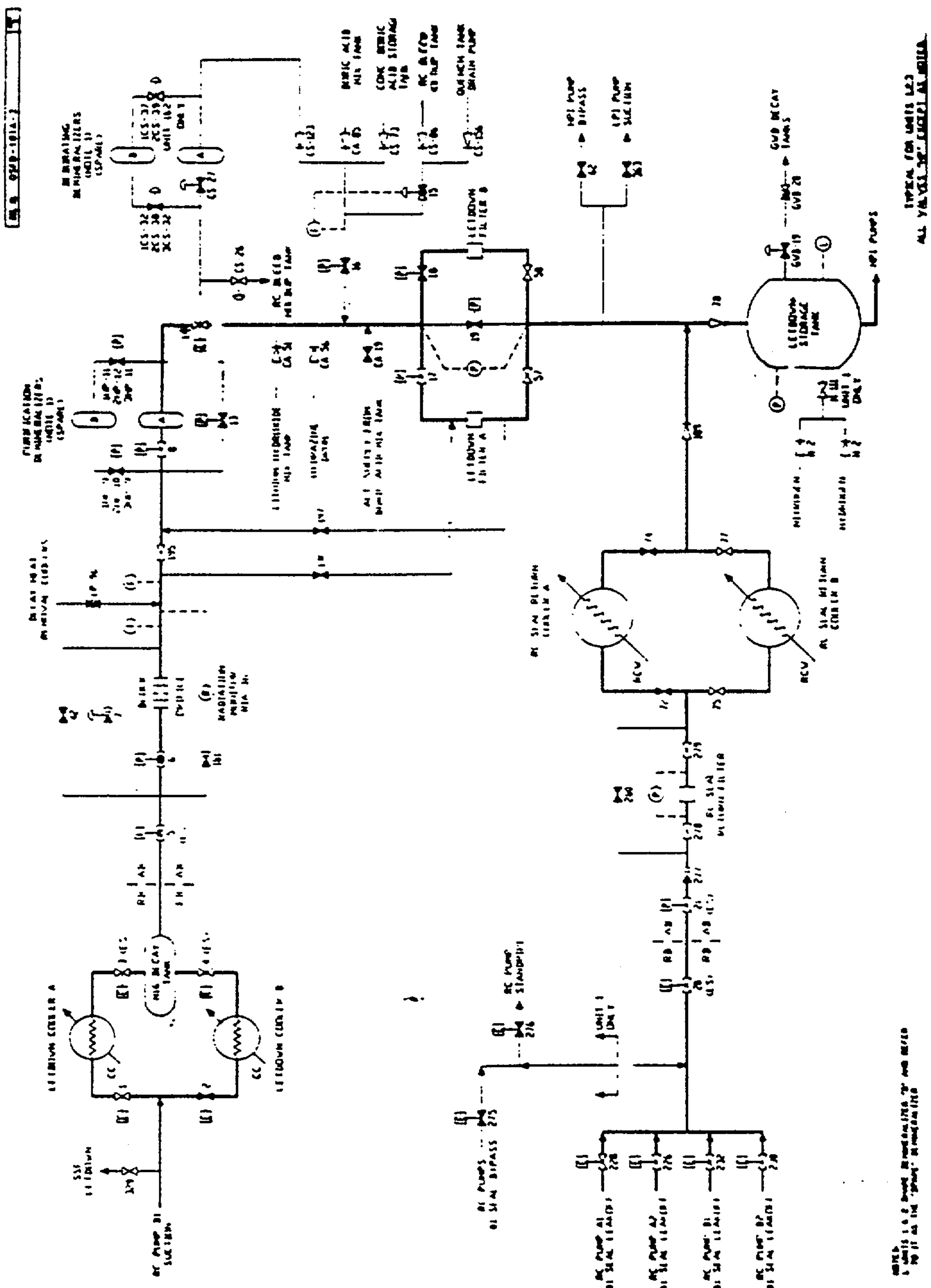

Figure 3.1 High Pressure Injection System - Oconee

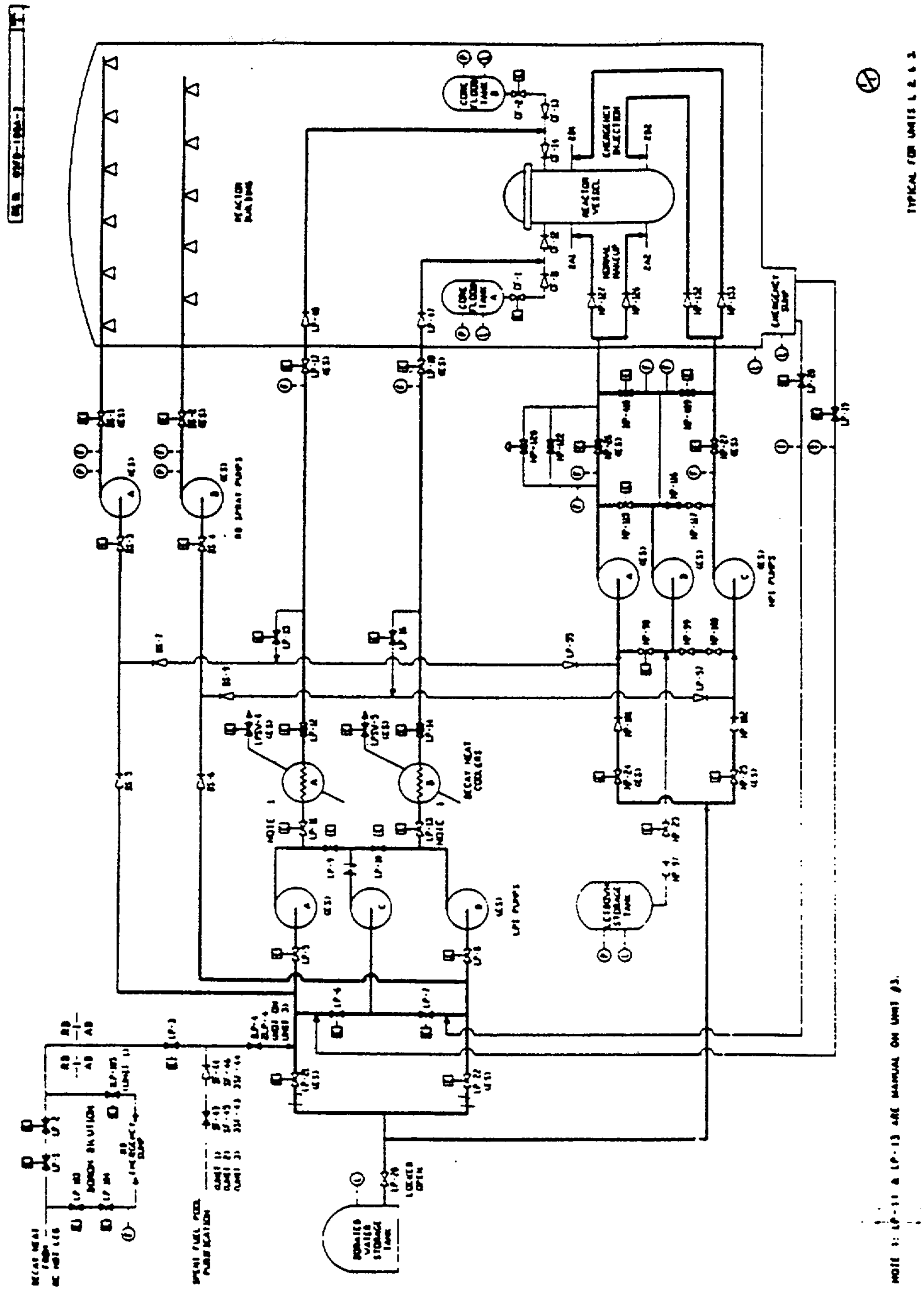

Figure 3.2 ECC and RB Spray Systems - Oconee

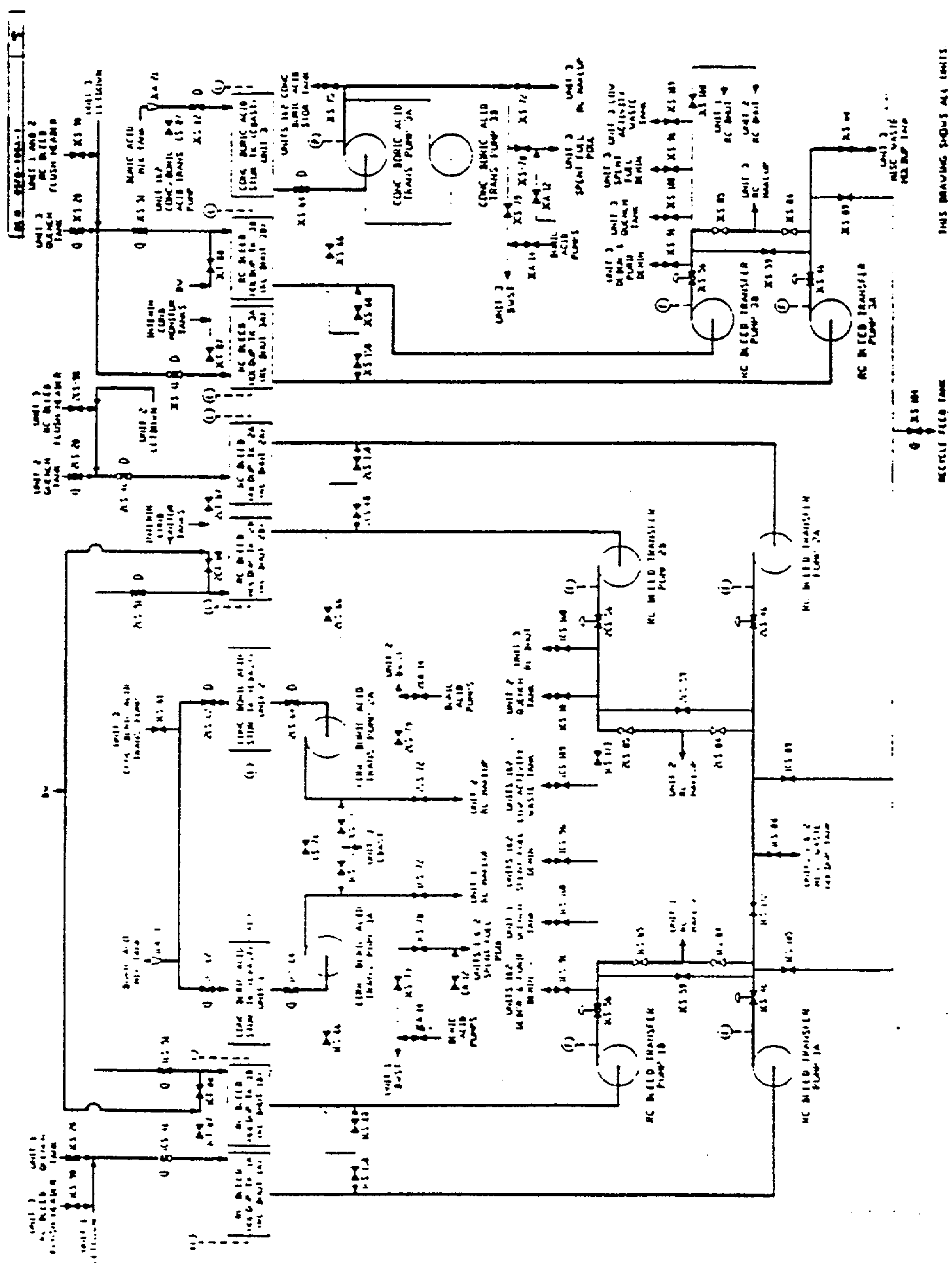

Figure 3.3 Coolant Treatment System - Oconee

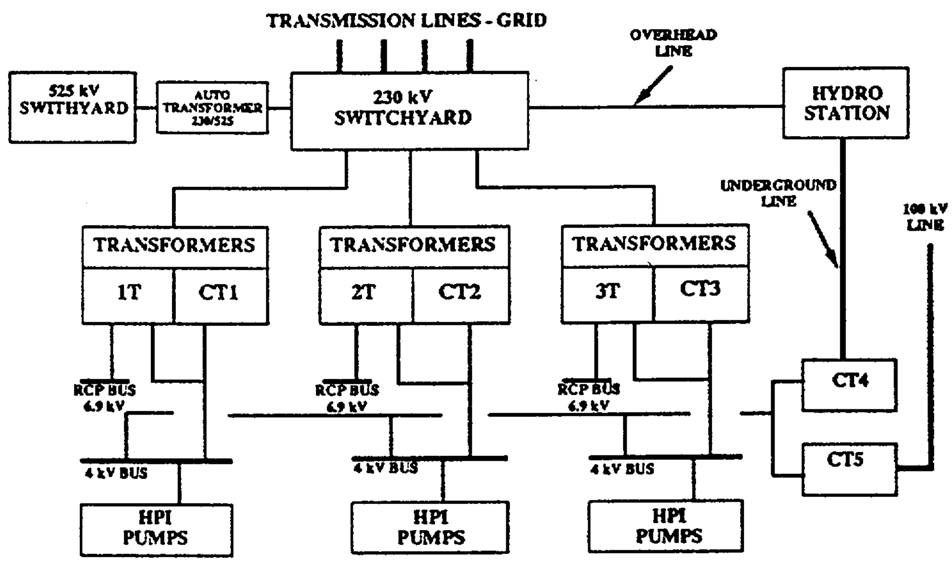

Figure 3.4 Electrical System - Oconee

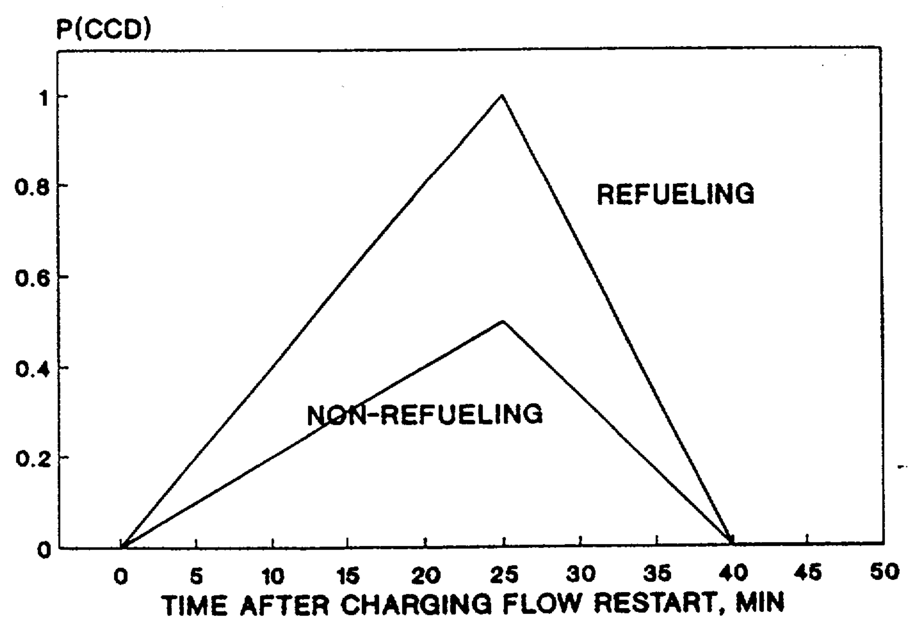

Figure 3.5 Conditional Core Damage Probability - Oconee Plant

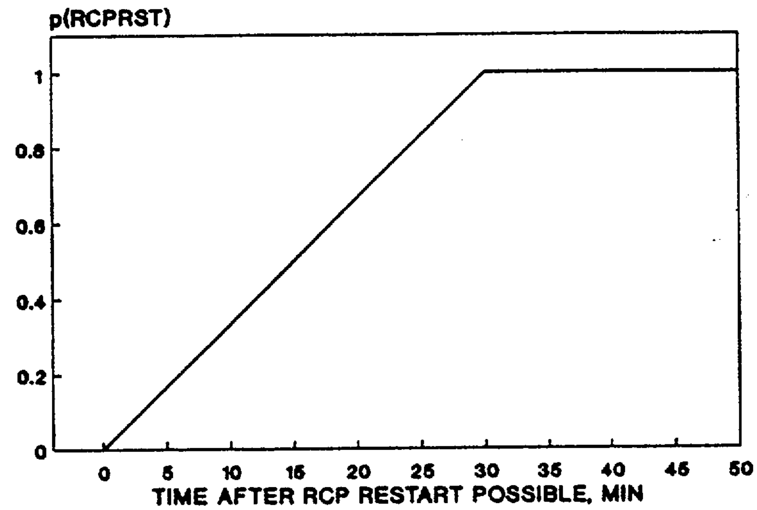

Figure 3.6 Cumulative Probability for RCP Restart

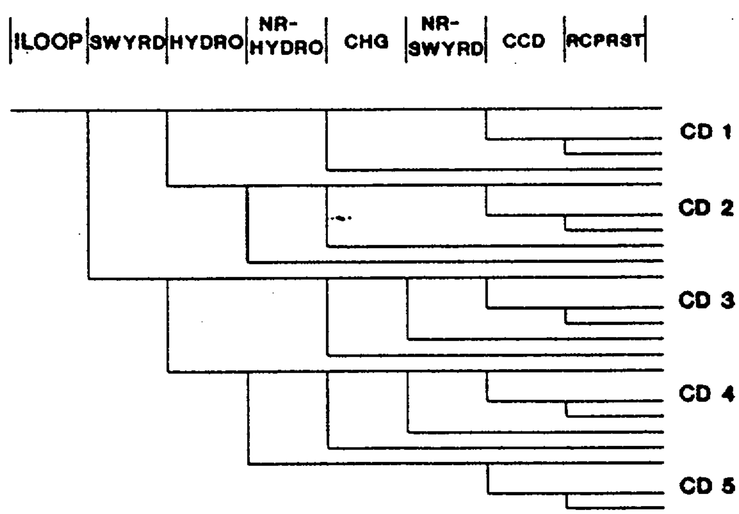

Figure 3.7 Boron Dilution Event Tree - Oconee

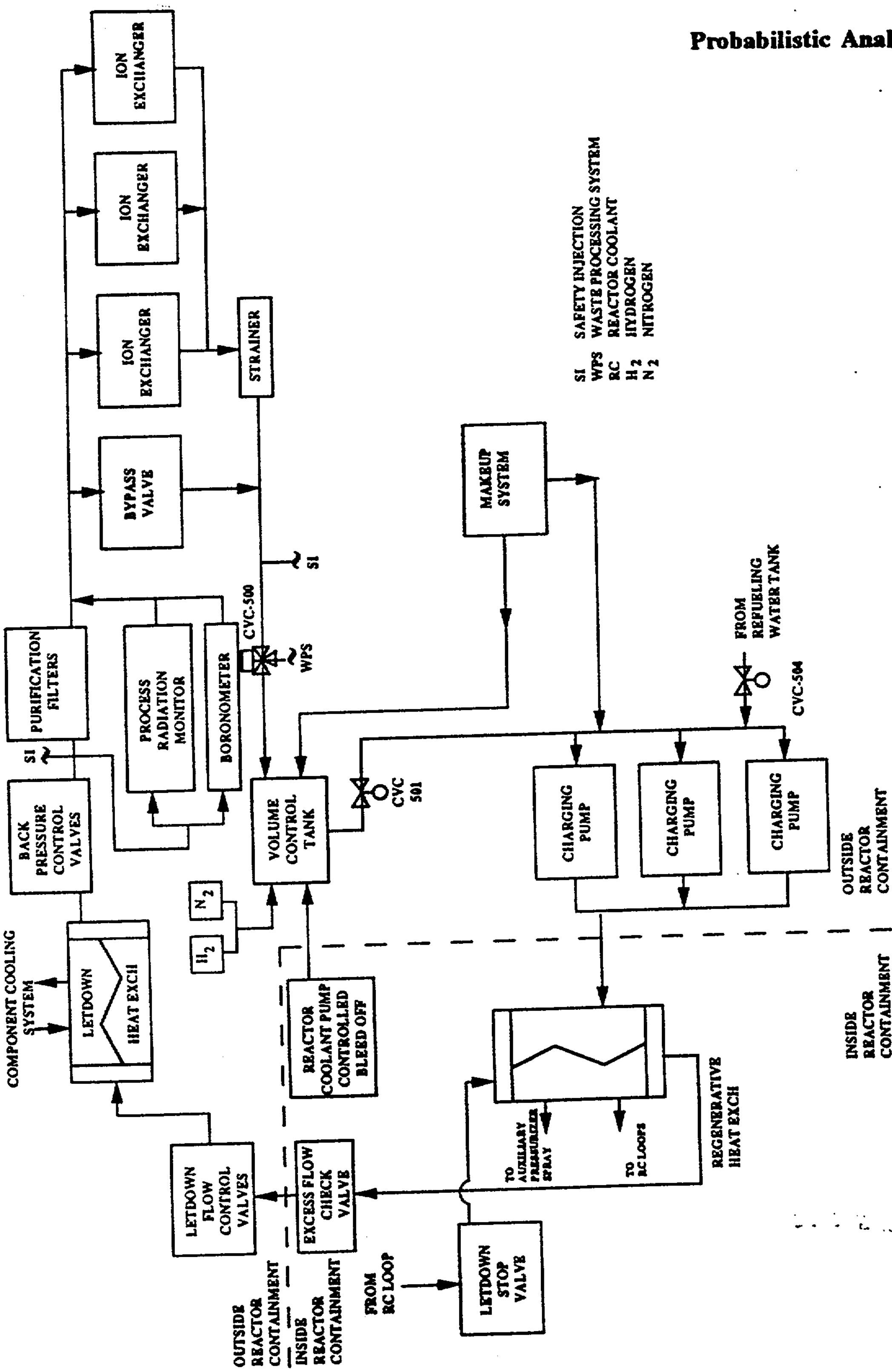

Figure 3.8 Charging and Letdown System - Calvert Cliffs

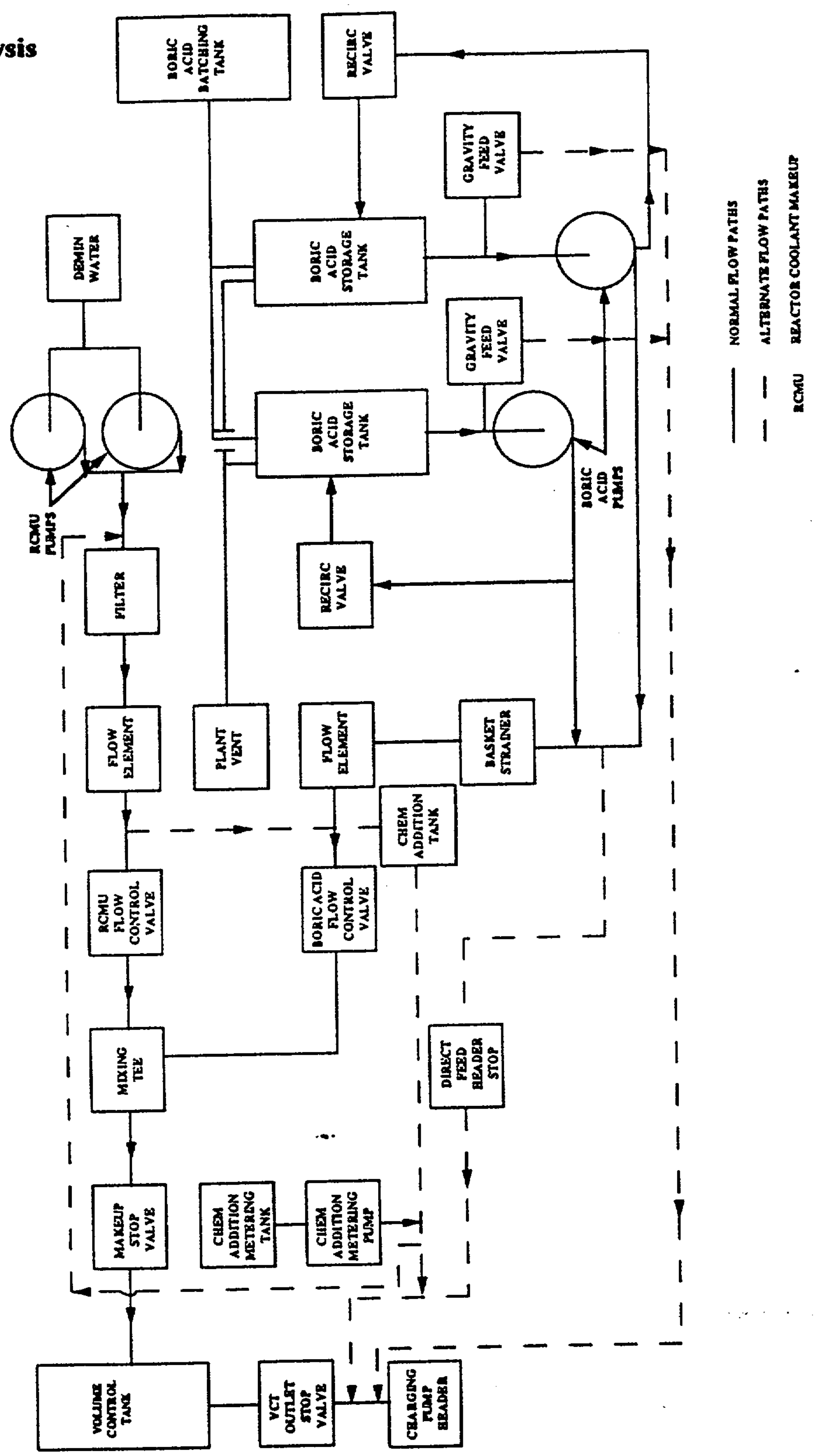

Figure 3.9 Makeup System - Calvert Cliffs

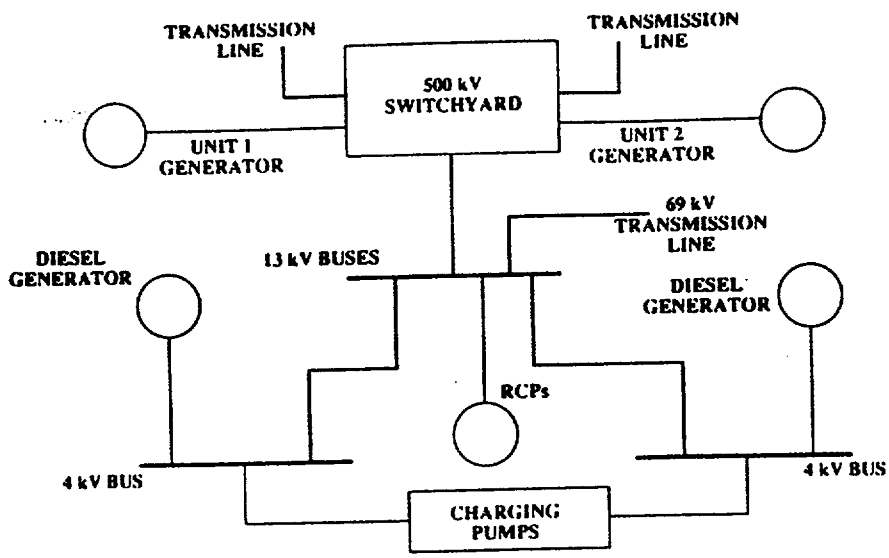

Figure 3.10 Electric System - Calvert Cliffs

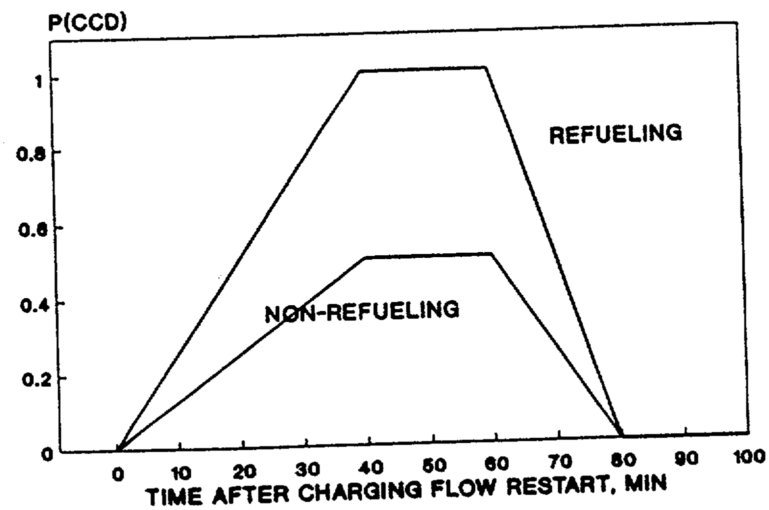

Figure 3.11 Conditional Core Damage Probability Calvert Cliffs Plant - Option A

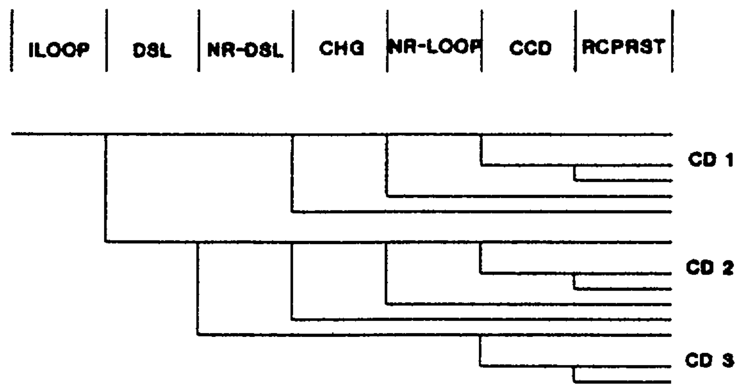

Figure 3.12 Boron Dilution Event Tree - Calvert Cliffs

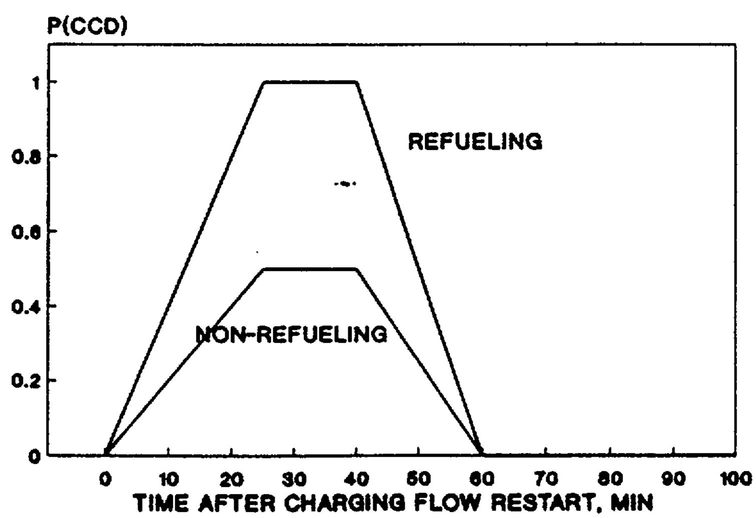

Figure 3.13 Conditional Core Damage Probability Calvert Cliffs Plant - Option B

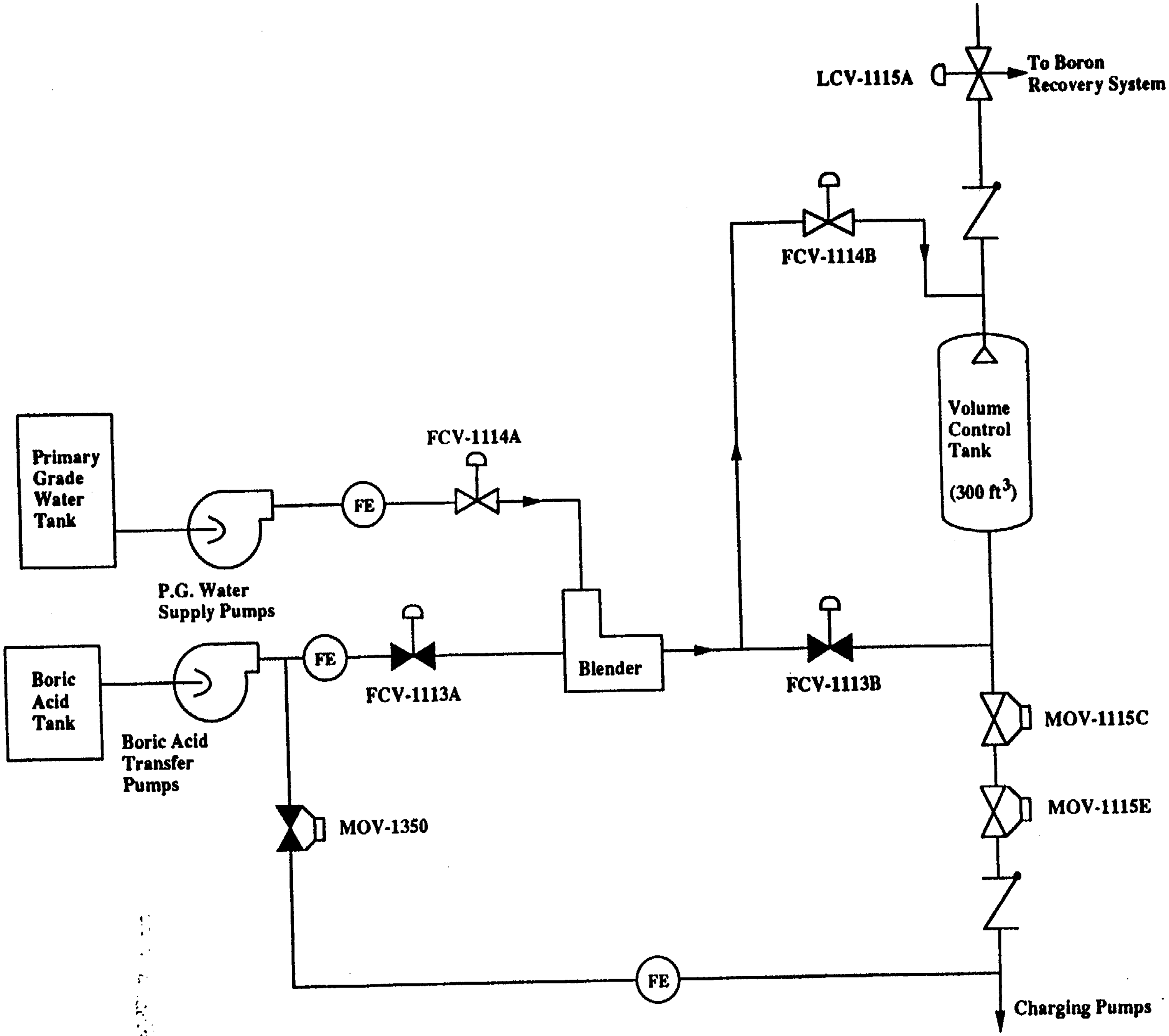

Figure 3.14 Boron Dilution Surry

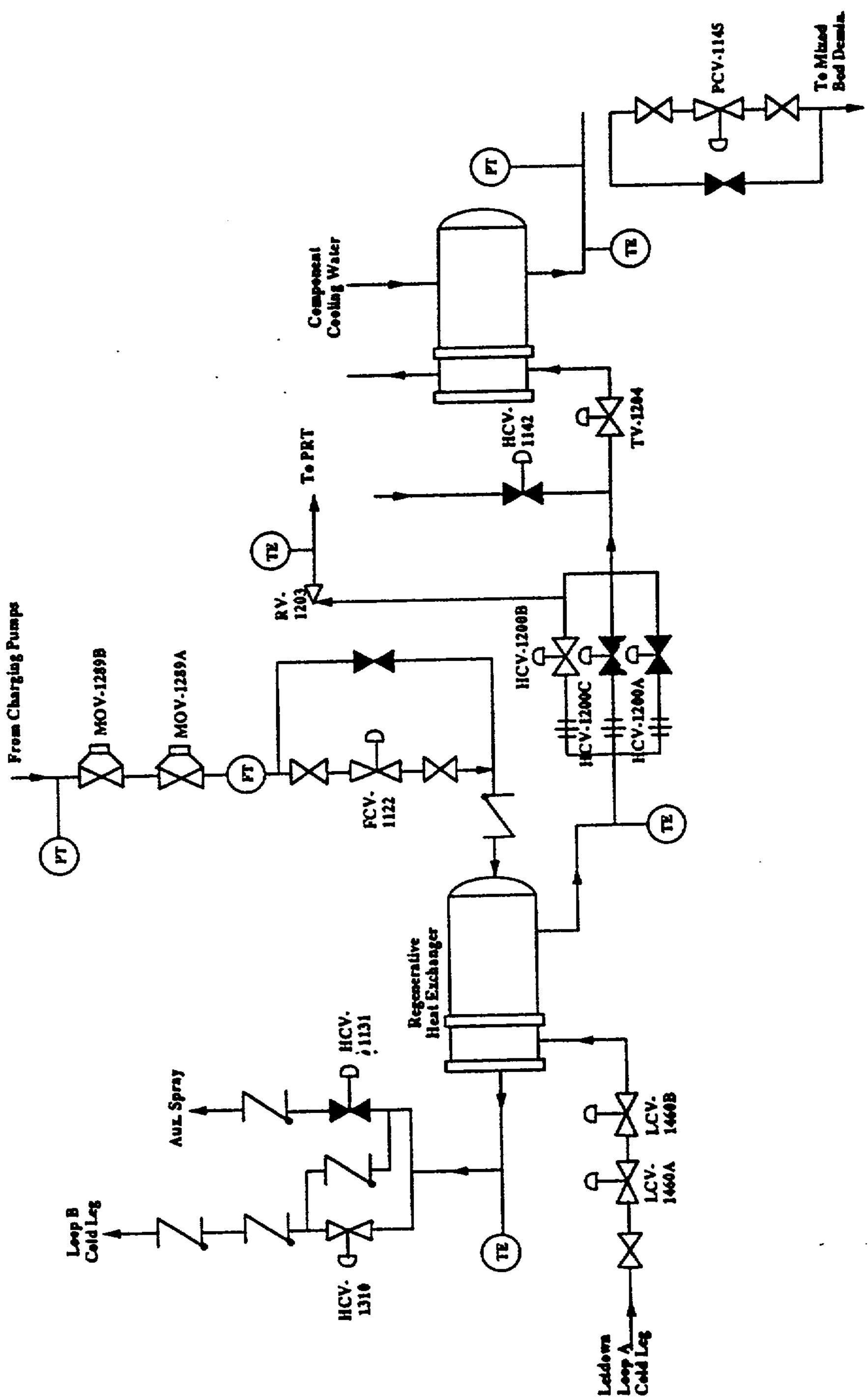

Figure 3.15 Charging and Letdown Subsystem - Surry

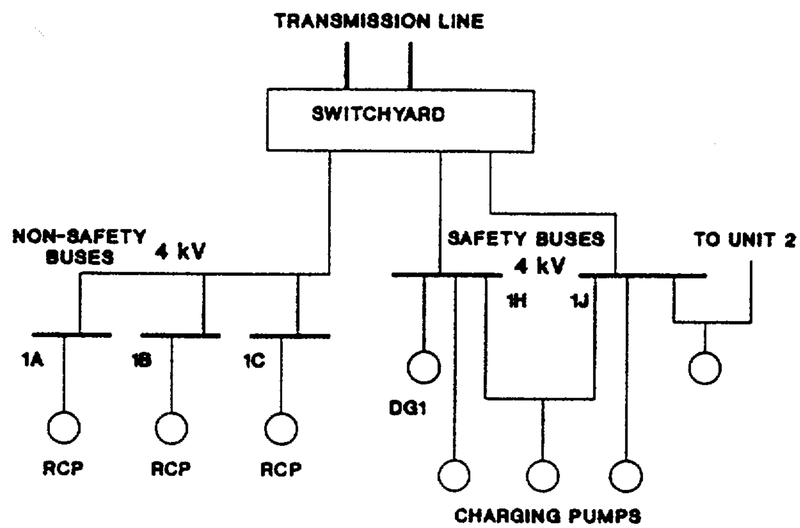

Figure 3.16 Electrical System - Surry

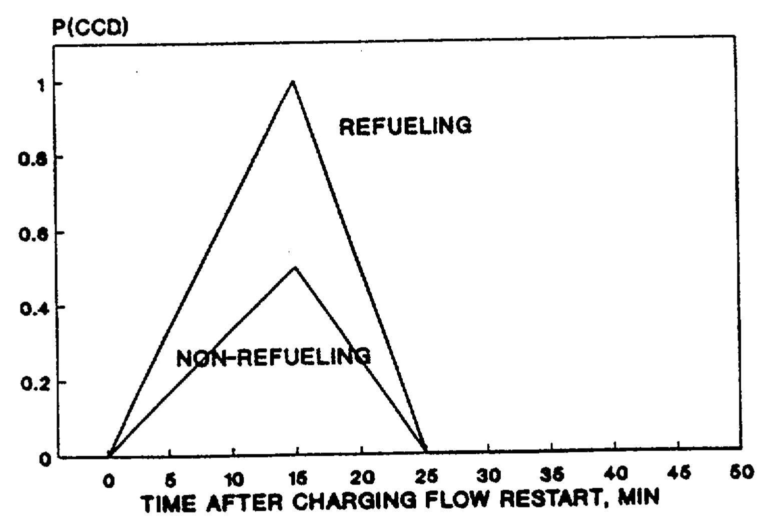

Figure 3.17 Conditional Core Damage Probability - Surry Plant

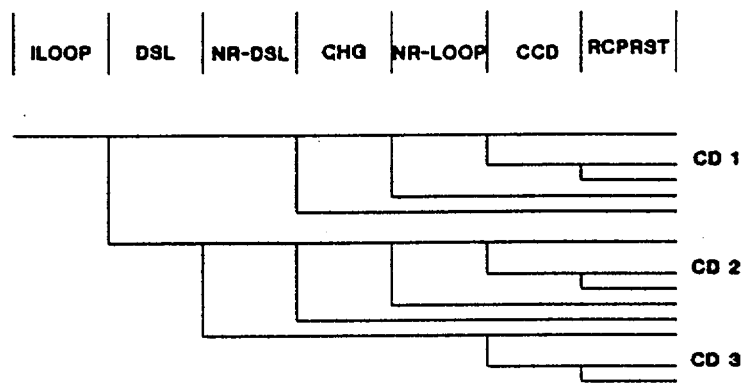

Figure 3.18 Boron Dilution Event Tree - Surry

4 Thermal-Hydraulic Analysis

4.1 Introduction

In Section 3 the conservative assumption was made that the charging flow, consisting of unborated water, does not mix sufficiently with the borated water in the RCS so that a diluted region accumulates in the lower plenum with the potential to cause a power excursion. It is known that there will be some mixing and in this chapter the extent of this mixing is quantified. The analysis assumes that the unborated charging flow is colder than the water in the RCS and it is injected into the cold leg which is otherwise stagnant or at a low natural circulation flow rate.

The modelling approach is similar to that used in the regional mixing model developed by Nourbakhsh and Theofanous [4.1, 4.2]. That work was in support of the NRC Pressurized Thermal Shock (PTS) study to predict the overcooling transients due to high pressure safety injection into a stagnant loop of a PWR. The analysis includes quantification of mixing (entrainment) at locations where the mixing is expected to be intense such as at the connection of the charging line to the reactor coolant system and in the downcomer. These mixing models are then used to determine the dilution boundary as a function of time.

4.2 Thermal Mixing Considerations

Qualitatively, the physical condition may be described with the help of Figure 4.1. In the absence of loop flow, the relevant parts of the system include the loop seal, pump, cold leg, downcomer, and the lower plenum. Initially, this portion of the primary system is filled with borated water with a boron concentration of ~1500 ppm and at a temperature near that of normal operation (~550°F). The dilution transient occurs with charging pump(s) injecting unborated water into the cold leg at a rate of ~45 to 96 gpm. Typical temperatures of the makeup (charging) flow are ~400°F to 500°F, although, depending on the plant and the stoppage of letdown flow during LOOP, lower temperatures, on the order of ~160°F, are also possible.

The ensuing flow regime is schematically illustrated in Figure 4.1. A "cold diluted stream" originates with the charging buoyant jet at the point of injection, continues toward both ends of the cold leg, and decays away as the resulting buoyant jets fall into the downcomer and pump/loop-seal regions. A "hot stream" flows counter to this "cold diluted stream" supplying the flow necessary for mixing (entrainment) at each location. This mixing is most intensive in certain locations identified as mixing regions (MRs). MR1 indicates that mixing associated with the highly buoyant charging jet. MR3 and MR5 are regions where mixing occurs because of the transitions (jumps) from horizontal layers into falling jets. MR4 is the region where the downcomer (planar) buoyant jet finally decays. The cold streams have special significance since they induce a global recirculating flow pattern with flow rates significantly higher than the charging flow. The whole process may be viewed as the quasi-static decay of the cold diluted stream within a slowly varying "ambient" temperature and boron concentration.

4.3 Regional Mixing Model

The quantitative aspects of this physical behavior were incorporated in the Regional Mixing Model [4.1-4.3]. This model accounts for countercurrent flow limitations between the cold and hot streams at the cold leg/downcomer junction and incorporates plume mixing rates which are consistent with data from idealized plume geometries.

The computation proceeds at two levels. The one is global and seeks to establish a "mean system response", referred to as "ambient" in the discussion of the flow pattern above. The other is local and seeks to partition mass and energy into the cold and hot stream consistent with mixing rates and countercurrent flow requirements. The global computation depends on the dilution sequence conditions and is discussed in Section 4.4.

At the local level of the computation the flows, energies, boron concentration and the volumes of the cold and hot streams must be established. The mass, energy, and boron balances for the control volume around MR1 (see Figure 4.1) yield:

$$\rho_{ch}Q_{ch} + \rho_h Q_h = \rho_c Q_c \tag{4.1}$$

$$\rho_{ch}Q_{ch}h_{ch} + \rho_h Q_h h_h = \rho_c Q_c h_c \tag{4.2}$$

$$\rho_{ch}Q_{ch}C_{ch} + \rho_h Q_h C_h = \rho_c Q_c C_c \tag{4.3}$$

where ρ, Q, h, and C represent density, flow rate, enthalpy, and boron concentration, respectively, and the subscripts ch, h, and c refer to the charging, hot, and cold streams, respectively. The hot stream flow rate is equal to that entrained into the jet and depends on the injection Froude number, Fr_{ch}, location of injection (side, top or bottom) into the cold leg, and the path length of the jet before reaching the cold streams, i.e.,

$$Q_h = Q_e(Fr_{ch}, D_{cl}, H_c) \tag{4.4}$$

where D_{cl} and H_c represent the cold leg diameter and the height of cold stream, respectively.

This entrainment function can be obtained by using the analytical or experimental results from the idealized buoyant jet geometries. Note that the Froude number is the ratio of inertial to buoyancy forces. Energy and boron concentration can be partitioned into the hot and cold stream volumes such that the total energy and boron mass remain equal to their corresponding mean values obtained from the global calculations.

The essential control of the overall process is provided by the countercurrent flow requirement as expressed by the condition of stationarity of long, neutrally stable waves at the interface between the cold and hot streams, i.e.,

$$Fr_h^2 + Fr_c^2 = 1 \tag{4.5}$$

The Froude numbers in Equation 4.5 must be based on the actual cold stream and hot stream hydraulic diameters (stream cross-sectional area divided by the width of contact between the two streams, W), and respective flow rates exiting or entering the cold leg. A parameter ß is used to express the fraction of jet entrainment, Q_e, coming from the direction of the vessel, i.e., the hot stream flow for use in Frh is $ßQ_e$. Therefore, the portion arriving for entrainment from the loop seal side would be $(1-ß)Q_e$. Since there is no outflow from the horizontal part of the loop seal, an equal volumetric rate of cold stream must flow in that direction. As a consequence, the net flow to be used in Fr_c of Equation 4.5 should be $Q_{ch} + ßQ_e$.

A symmetric behavior, i.e., ß = 0.5, is appropriate if charging flow is injected into a horizontal cold leg. A ß = 1 is used when the charging flow is injected into an inclined portion of the cold leg (e.g., as in the Oconee injection configuration).

Equation 4.5 can be put in dimensionless form [4.4, 4.5] as

$$Q^{*3} + aQ^{*2} + bQ^* + c = 0 \tag{4.6}$$

where

$$a = \frac{1}{\beta\rho^*\sigma}\left\{\frac{\rho^*}{(1-A^*)^3} + \frac{1+2\rho^*}{A^{*3}}\right\}, \quad b = \frac{1}{\beta^2\rho^*\sigma}\left\{\frac{\rho^* + 2}{A^{*3}}\right\} \tag{4.7}$$

$$c = \frac{1}{\beta^3\rho^*\sigma}\left\{\frac{1}{A^{*3}} - \frac{1}{W^* Fr_{ch,d}^2}\right\} \tag{4.8}$$

$$\sigma = \frac{1}{(1-A^*)^3} + \frac{1}{A^{*3}} \tag{4.9}$$

$$Q^* = \frac{Q_e}{Q_{ch}}, \quad \rho^* = \frac{\rho_h}{\rho_{ch}} \tag{4.10}$$

$$W^* = \frac{WD_d}{A_d}, A^* = \frac{A_c}{A_d}, H^* = \frac{H_c}{D_d} \tag{4.11}$$

and

$$Fr_{ch,d} = \frac{(Q_{ch}/A_d)}{\left\{gD_d \frac{\rho_{ch} - \rho_h}{\rho_{ch}}\right\}^{1/2}} \tag{4.12}$$

Since W^*, A^*, and H_c^* are all geometrically related, Equation 4.6 provides a simple relationship of the form:

$$Q^* = f_1(H^*_c, \rho^*, Fr_{ch,d}, \beta) \tag{4.13}$$

Similarly, Equation 4.4 can be put in a dimensionless form:

$$Q^* = f_2(Fr_{ch,d}, H^*_c, D^*) \tag{4.14}$$

where

$$D^* = \frac{D_d}{D_{ch}} \tag{4.15}$$

In reactor applications the variation of ρ^* during a dilution transient is small and the effect of this variation on the results of Equation 4.13 is negligible. The $Fr_{ch,d}$ increases gradually during a dilution scenario.

The countercurrent flow limiting condition (Equation 4.13) and jet entrainment (Equation 4.14) can be reduced to two sets of plots such that the stratification (H_c^*) and entrainment Q^* can be determined by a simple superposition procedure (see Section 4.4).

The temperature and boron concentration in the downcomer may be estimated on the basis of mixing of the cold stream spilling out of the cold leg. A highly complicated three dimensional mixing pattern occurs in MR3. In the original formulation of the Regional Mixing Model [4.1], the approach was to conservatively neglect this contribution to the mixing in the downcomer. Rather, the cold stream exiting the cold leg was assumed to form smoothly into the planar plume within the downcomer and to decay according to the K-ϵ-θ turbulence model prediction. A refinement was possible on the basis of Purdue's 1/2 scale data [4.3]. The planar plume is taken to form within a distance of $2D_d$ below the cold leg centerline and to be fed in equal volumetric flow rates by the cold stream and surrounding hot volume fluid. Below this point the decay is approximated by that of a planar plume of initial width equal to D_d and Fr = 1.0 as show in Figure 4.2. The plot shows the temperature function vs distance down the plume. The centerline temperature of the plume, T, temperature of the mixed mean region outside the plume, T_m, and temperature at the jump, T_j, are related to concentrations in the present problem.

It should be noted that such thermal stratification is obtained at low (and zero) loop flow, and it cannot be represented with typical system thermal-hydraulic codes (e.g., TRAC and RELAP5) to simulate rapid boron dilution transients. For a well-mixed condition, when system codes are applicable, there must be sufficient loop flow not only to break up the charging plume (jet) but also to produce stable flow into the downcomer. Nourbakhsh and Theofanous [4.6] used the boundary of stability (Fr_d = 1) and developed a criterion for the existence of perfect mixing in the presence of loop flow. Their stratification/mixing boundary, shown in Figure 4.3, can be expressed by:

$$Fr_d \approx \left[1 + \frac{Q_L}{Q_{ch}}\right]^{-7/5} \tag{4.16}$$

Although Equation 4.16 has been developed for the conditions of high pressure safety injection, it is also valid for the Fr_d range of interest for charging injection. Loop flows of 20 (for Surry) to 45 (for Calvert Cliffs) times the charging flow are required to have perfect mixing in the cold leg and, therefore, to be able to apply the typical system thermal-hydraulic codes.

4.4 Boron Mixing Calculations

The regional mixing model was used to assess the extent of boron mixing during a rapid dilution scenario for the Surry and Calvert Cliffs stations. At Surry, charging pumps can deliver 96 gpm of demineralized water from the volume control tank (VCT). The available volume for injection into the RCS is about 1500 gal. This flow is directed to one of the three cold legs via 3-inch ID piping. The charging injection line connects with a 26-inch ID cold leg at the top.

Two cases with charging flow temperature of 160°F and 450°F were considered. No loop circulation was assumed for the duration of the dilution transient.

The "mean system response" was calculated from the global energy balance. Neglecting the heat released from the walls and assuming ρ_m to be a constant we have:

$$\frac{h_m - h_{ch}}{h_{mo} - h_{ch}} = \exp\left\{\frac{-tQ_{ch}\rho_{ch}}{V_{mix}\rho_m}\right\} \tag{4.17}$$

The mixing volume, V_{mix} = 1156 ft³, representing the volume of one cold leg, one pump, one loop seal (excluding the upstream vertical leg), lower plenum (up to the lower edge of core barrel) and the portion of downcomer below the cold leg was used in the calculations. Assuming a charging flow temperature of 160°F, h_m is decreasing from its initial value (h_{mo}) of 547 Btu/lb$_m$ to 463 Btu/lb$_m$ at the end of the dilution transient (940 s based on the capacity of the VCT and the flow rate).

This corresponds to a cooldown from 548°F to 478°F. The variation of $Fr_{ch,d}$ is very small during this dilution transient (0.014-0.016).

The entrainment at the injection point (Q^*=3.5) and stratification (H^*_c=0.22) was obtained by superposition of plots of counter-current flow limited entrainment and jet entrainment as shown in Figure 4.4 The jet entrainment correlation based on the results of turbulence model calculations [4.1, 4.2] was used for the analysis for Surry.

After the mixing patterns were calculated, the results were converted to boron dilution using the equivalence between the dimensionless boron concentration and energy distribution. The boron concentration transients at several important locations in the system are shown in Figure 4.5 The mixed mean boron concentration, C_m, exponentially decreases from its initial value of 1500 ppm to 1193 ppm during the dilution transient. The boron concentration at the cold stream, C_c, was obtained from a boron balance for the control volume around MR1 (Equation 4.3) and by assuming the hot stream boron concentration to be equal to the mixed mean concentration. The boron concentration at the junction of the cold leg and downcomer (~2D_d below the cold leg), C_j, was obtained by assuming the mixing of equal volumetric flow rates by the cold stream and surrounding hot volume fluid as discussed in Section 4.3. Finally the minimum boron concentration reaching the lower plenum, C_{min}, was obtained from the centerline planar plume decay (Figure 4.2) obtained from the results of the K-ϵ-θ turbulence model [4.1, 4.2]. The actual boron concentration in the lower plenum is higher than C_{min} because of mixing in the lower plenum. The result from Figure 4.5 shows that the lowest boron concentration in the lower plenum will be no less than 1080 ppm. This occurs at the time when the VCT would be emptied of diluted water and would start to be replenished with highly borated water from the refueling water storage tank.

Similar calculations were performed assuming the charging flow temperature to be 450°F. The results of the entrainment solution and boron concentration transients at different locations are presented in Figures 4.6 and 4.7. Higher charging flow temperature (higher Froude numbers) slightly decreases the entrainment at the injection point. However, due to a lower mass flow of the makeup water (due to the lower density) the boron concentration transients are slightly higher than for the case with lower temperature. The result is that the minimum boron concentration in the lower plenum is only 1100 ppm in this case.

Boron mixing calculations were also performed for Calvert Cliffs. Under normal boron dilution conditions at Calvert Cliffs three charging pumps are used, each delivering 44 gpm of demineralized water from the volume control tank (VCT) to two of the four cold legs via 2-inch ID piping. The charging injection lines connect with the 30-inch ID cold leg pipes at the side. The available volume for injection into the RCS is about 2900 gal. During a LOOP only two of the three charging pumps are transferred to the emergency electrical bus. Therefore, each cold leg with a charging line will receive 44 gpm. As was done with Surry, two cases with charging temperature of 160°F and 450°F were considered.

The mixing analysis for Calvert Cliffs was similar to that for Surry. However, the jet entrainment correlation used for top injection (as found in Surry) is not applicable to Calvert Cliffs because the charging injection is from the side. Using the correlation obtained by Riester et al., [4.7] for the prediction of the trajectory of horizontal buoyant submerged jets and using a simple entrainment coefficient for the horizontal part of the jet, the entrainment correlation was modified to estimate the mixing due to side injection. The results for the entrainment solution and boron concentration transients are presented in Figures 4.8-4.11. Significant mixing is predicted during the boron dilution transients and the minimum boron concentration in the lower plenum is 900 ppm with the cold water injection (Figure 4.9) and 960 ppm with the hotter water (Figure 4.11). It should be noted that the Froude number of injection (based on the injection nozzle diameter) for both Surry and Calvert Cliffs is approximately 3-7 and thus there is additional mixing due to forceful jet impingement and splashing off the opposite wall in the cold leg which is neglected here.

4.5 Summary and Conclusions

The Regional Mixing Model, which has been developed to study the thermal mixing of interest to pressurized thermal shock, was utilized to assess the extent of boron mixing in the absence of loop flow during a reactor restart scenario. Illustrative reactor predictions for Surry and Calvert Cliffs indicate significant mixing during the boron dilution transients. Indeed, for the cases considered the boron concentration in the lower plenum does not fall below 900 ppm. However, these cases do not encompass all possible physical situations for these plants. It would also be desirable to assess the applicability of the model when the temperature of the charging flow is higher, to improve the understanding of mixing when the injectant enters at the side or bottom of the cold leg piping, and to quantify the additional mixing due to jet impingement for the range of injection Froude numbers of interest to boron dilution.

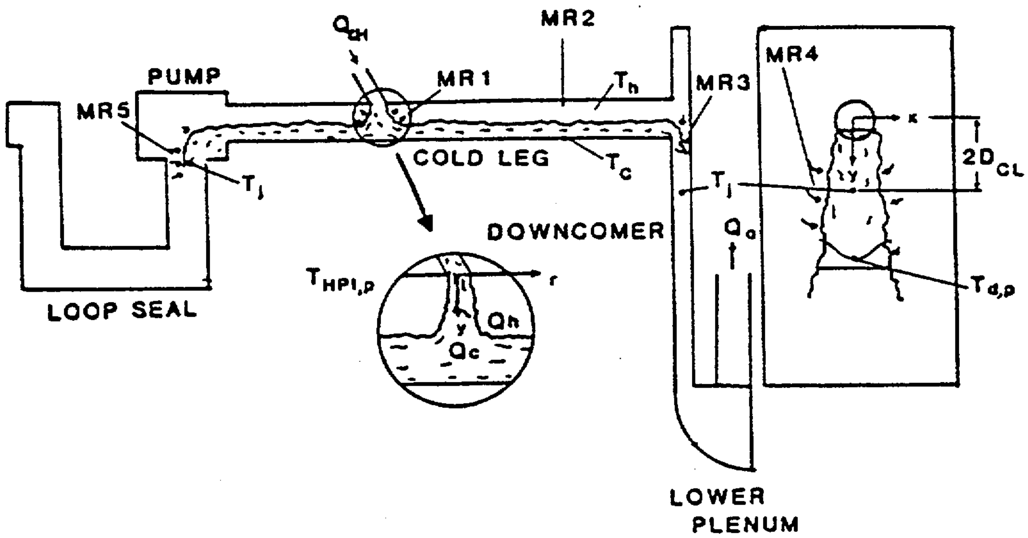

Figure 4.1 Schematic of the Flow Regime and Regional Mixing Model [4.2]

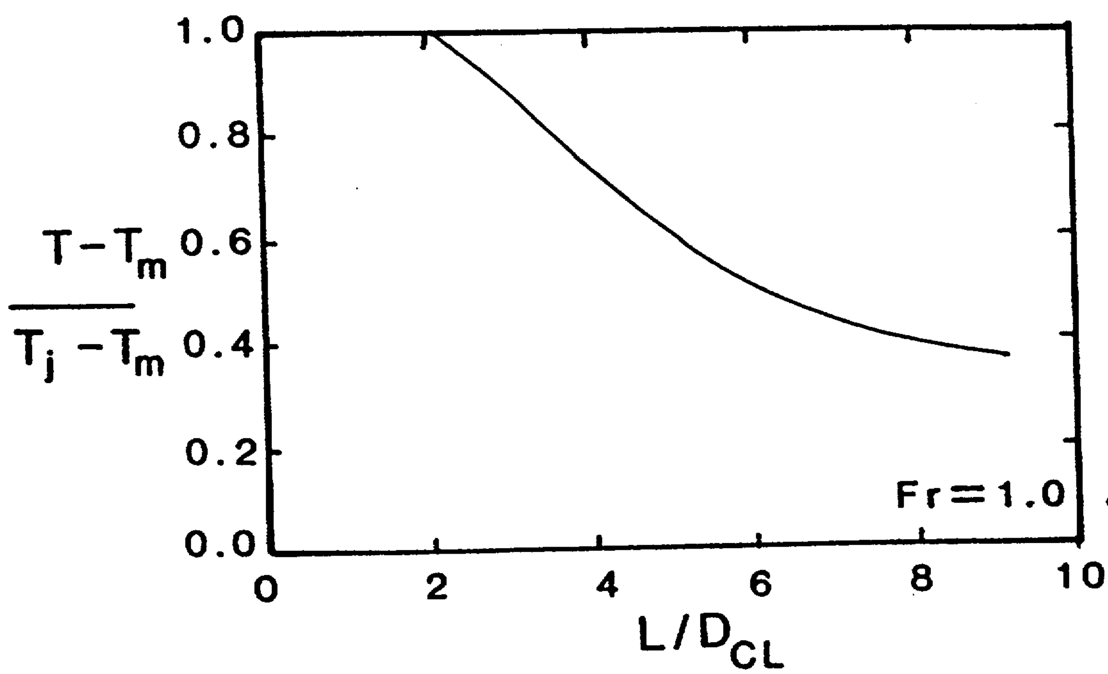

Figure 4.2 Prediction of Centerline Temperature in the Downcomer Planar Plume [4.2, 4.3]

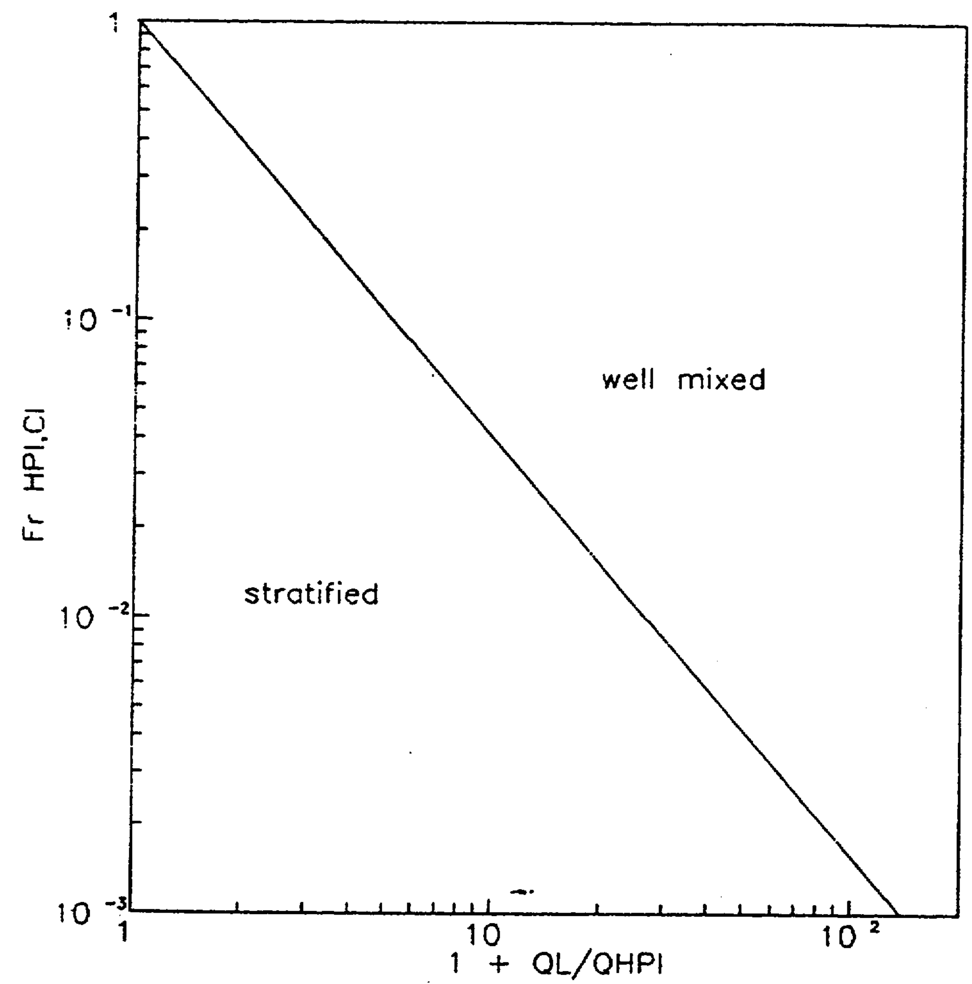

Figure 4.3 Theoretical Stratification Criterion (Equation 4.16)

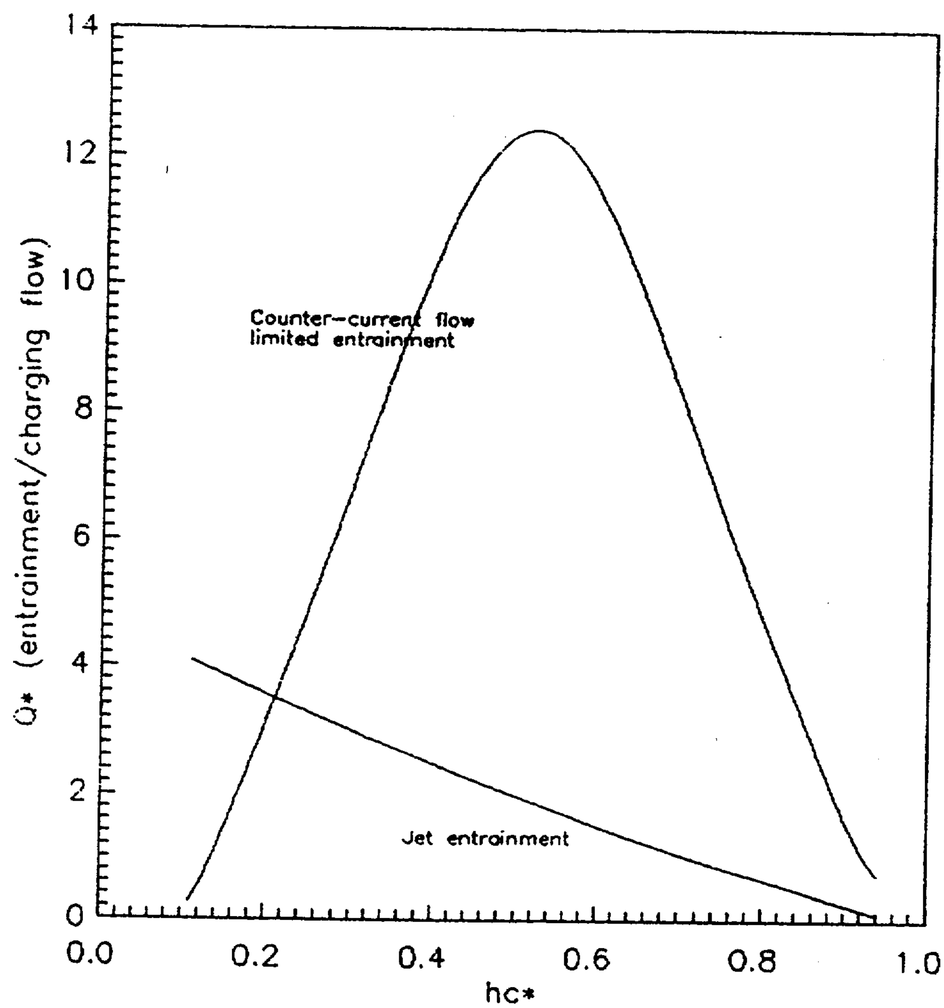

Figure 4.4 Entrainment Solution for Surry with Charging Temperature of 160°F ($Fr_{ch,d}$=0.014,ρ^*=0.76)

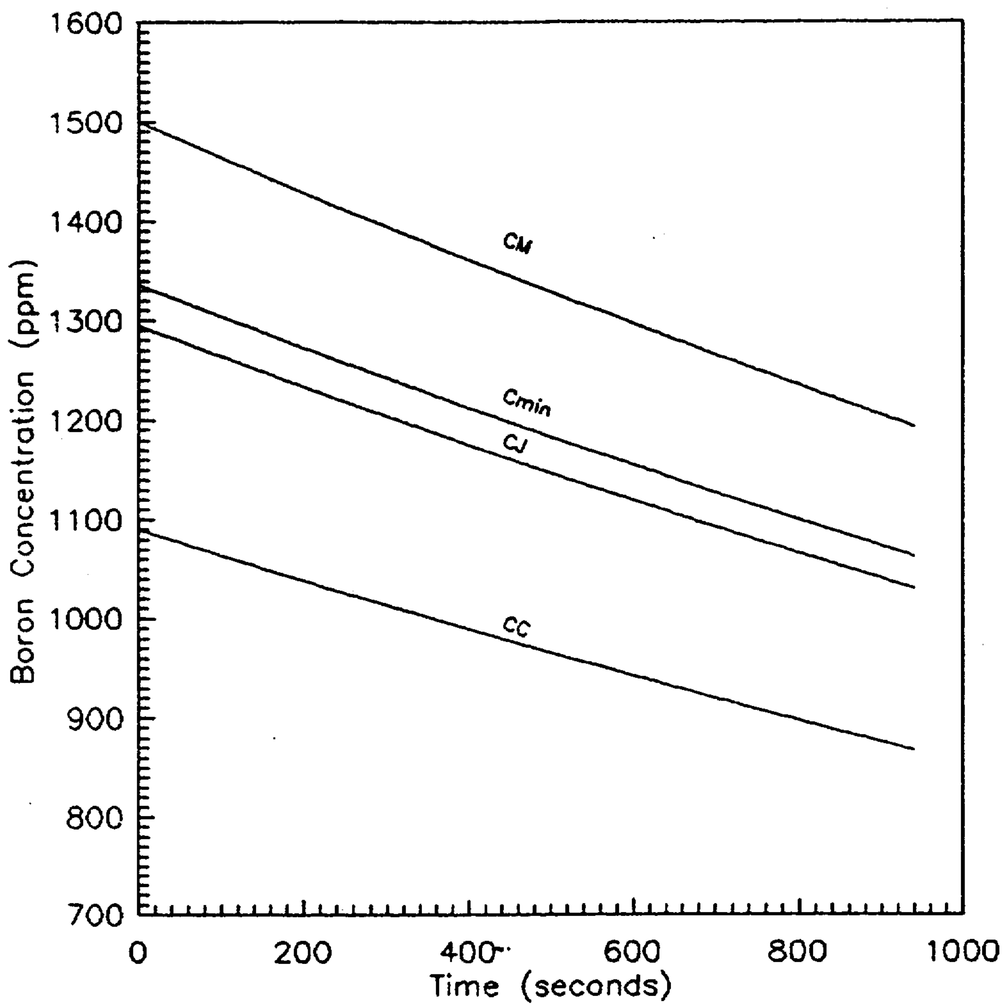

Figure 4.5 Predicted Boron Concentration Transients for Surry with Charging Temperature of 160°F

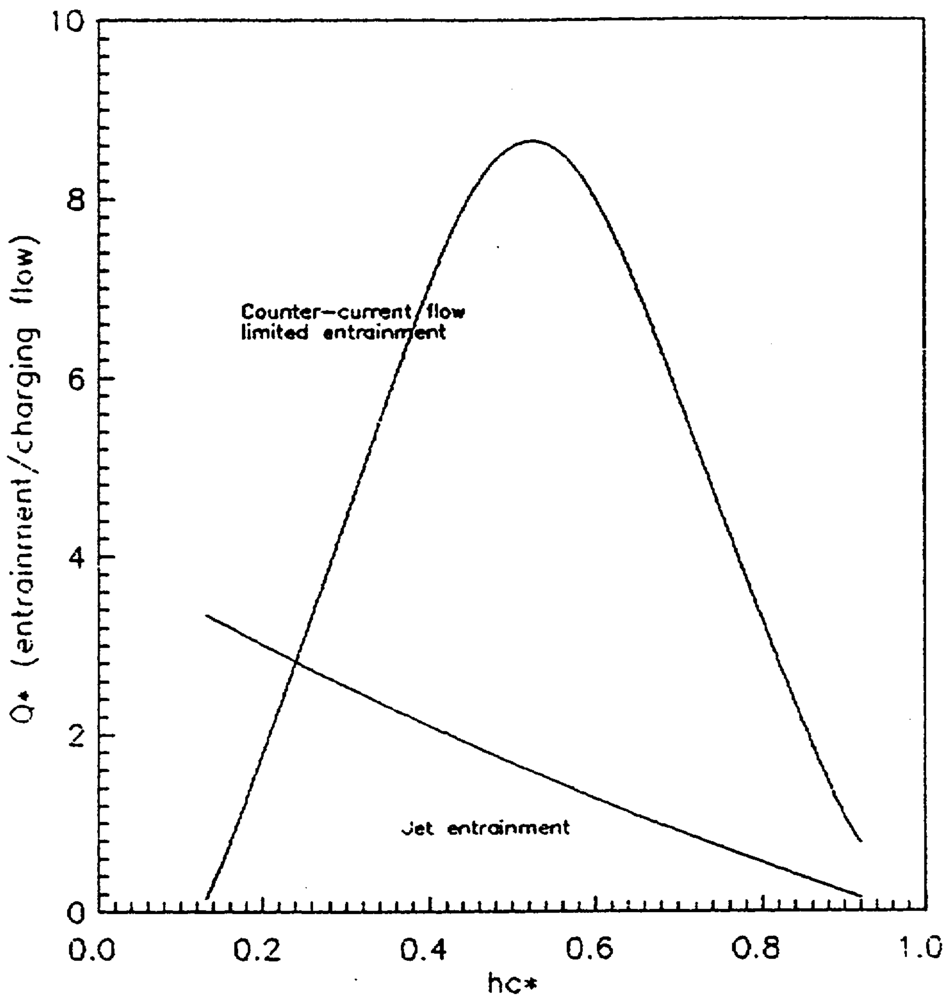

Figure 4.6 Entrainment Solution for Surry with Charging Temperature of 450°F ($Fr_{ch,d}$=0.021,ρ^*=0.9)

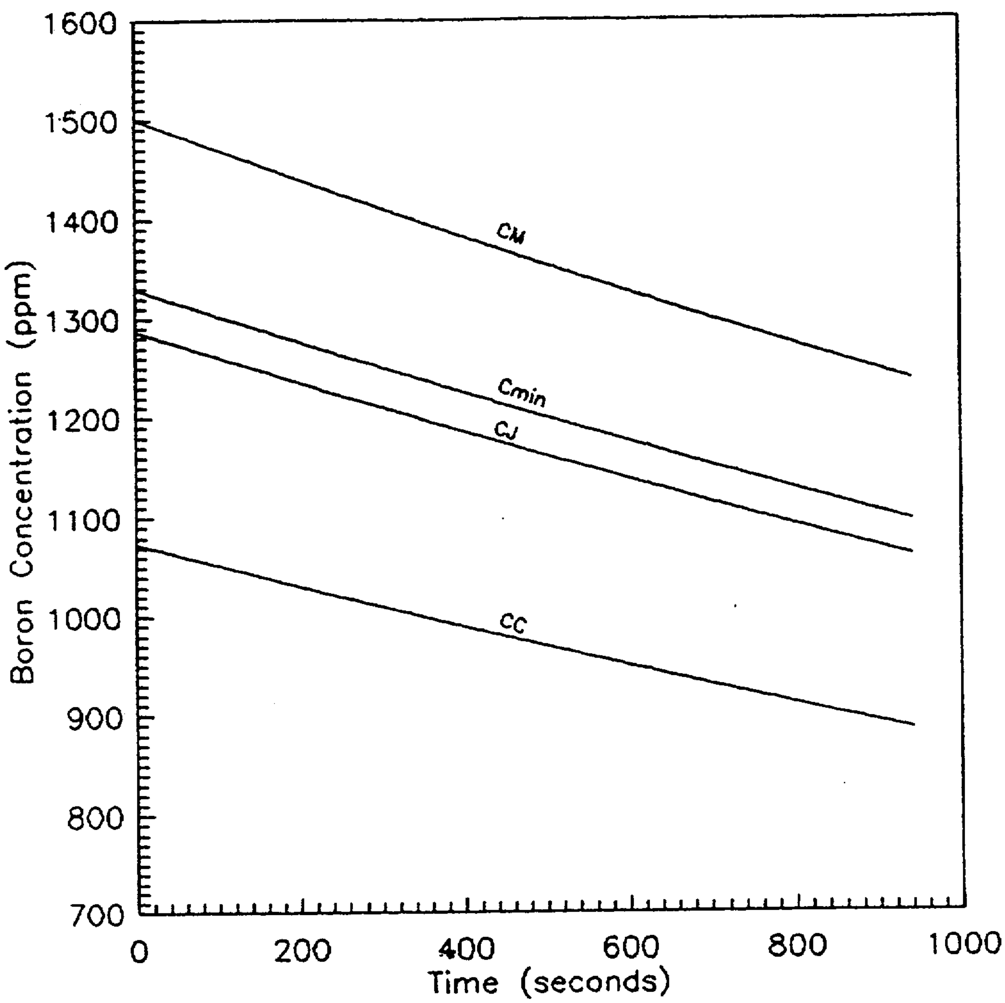

Figure 4.7 Predicted Boron Concentration Transients for Surry with Charging Temperature of 450°F

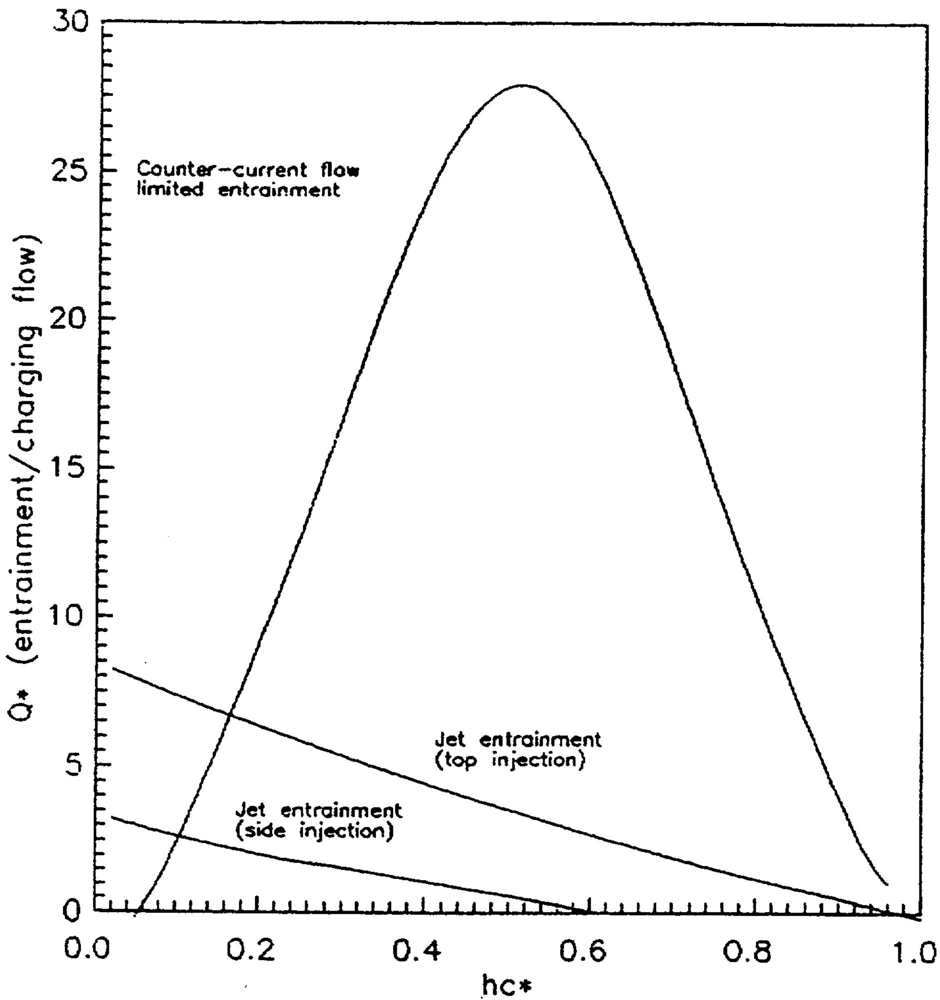

Figure 4.8 Entrainment Solution for Calvert Cliffs with Charging Temperature of 160°F($Fr_{ch,d}$=0.0045, ρ^*=0.76)

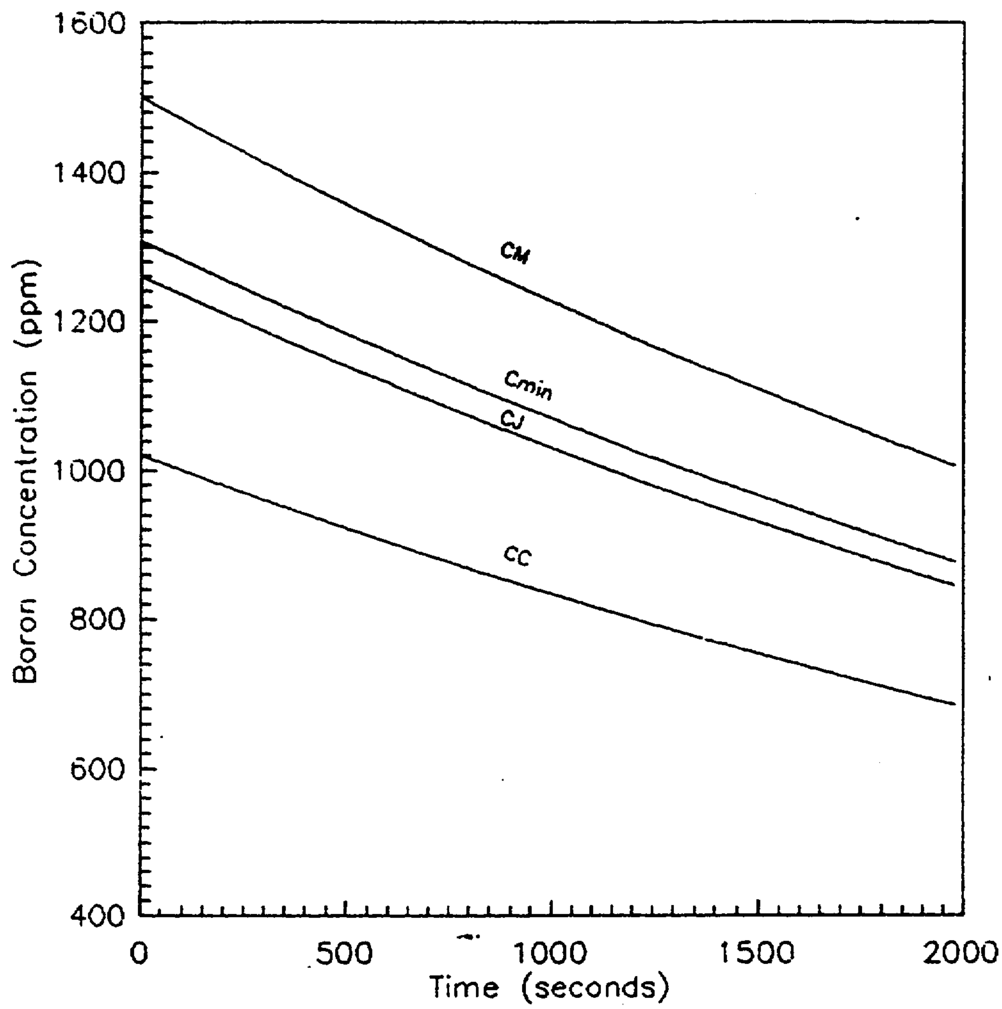

Figure 4.9 Predicted Boron Concentration Transients for Calvert Cliffs with Charging Temperature of 160°F

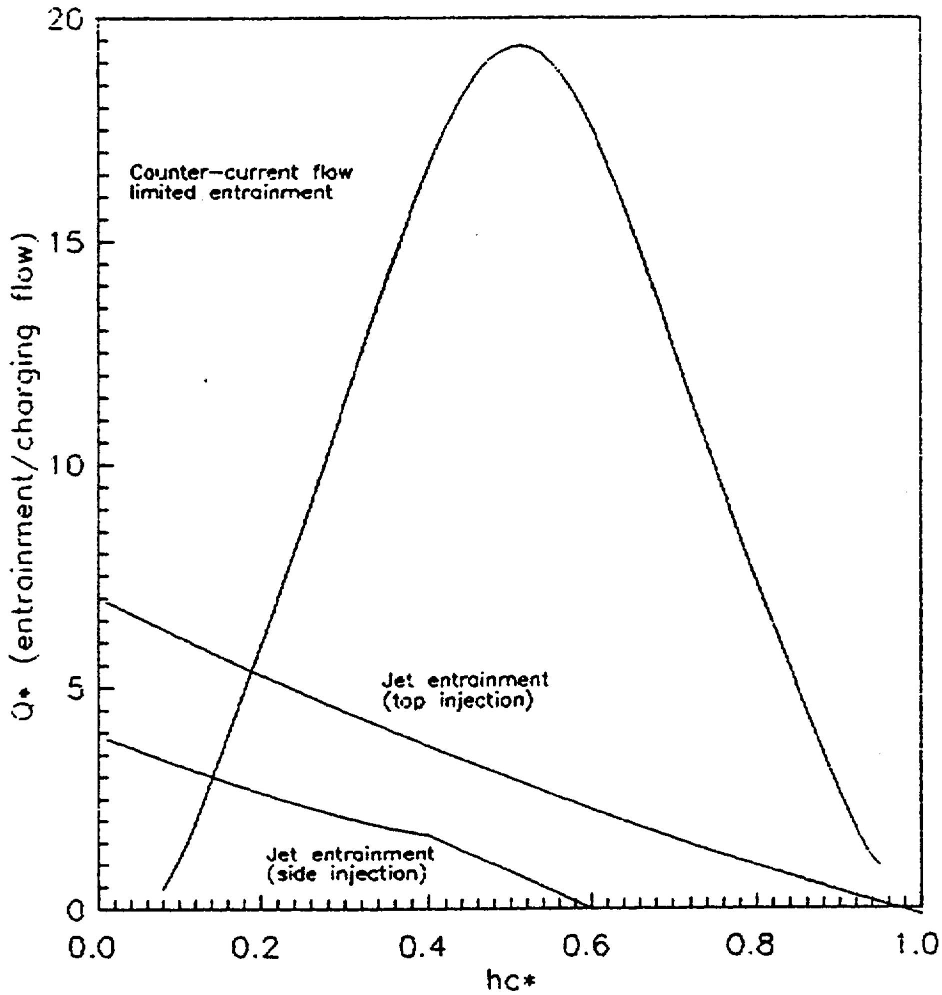

Figure 4.10 Entrainment Solution for Calvert Cliffs with Charging Temperature of 450°F ($Fr_{ch,d}$=0.007,ρ^*=0.9)

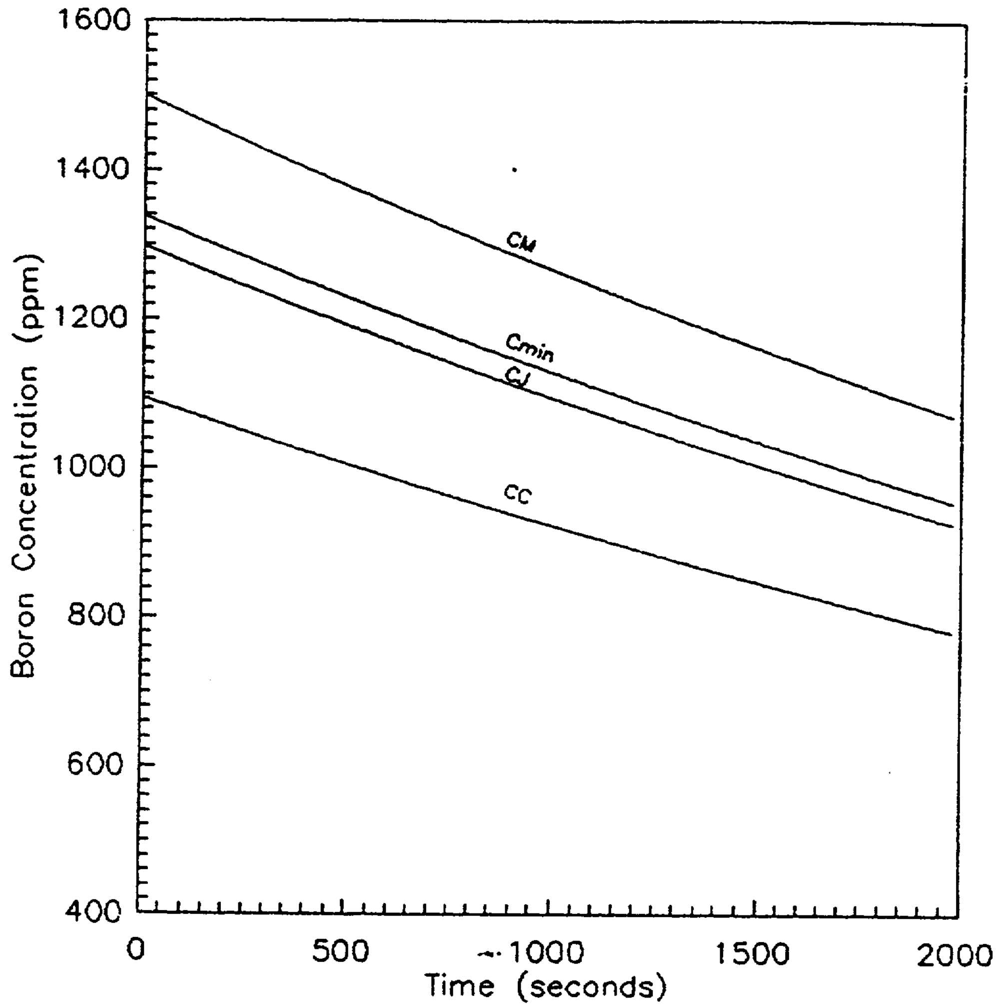

Figure 4.11 Predicted Boron Concentration Transients for Calvert Cliffs with Charging Temperature of 160°F

5 Analysis of Consequences

5.1 General Methodology

In order to rigorously calculate the response of the core to the injection of a slug of diluted water, a three-dimensional dynamic core model is needed. That model should have three-dimensional neutron kinetics and, if not three-dimensional thermal-hydraulics, then at least one-dimensional thermal-hydraulics in multiple channels. It would calculate the transport of boron throughout the core for a given distribution at the core inlet. Since this capability was not available for this project, and since resources were not available to develop a rigorous model from those partial models that do exist, the approach taken was to do a study based on an approximate synthesis method.

In this synthesis method static three-dimensional core calculations are combined with point neutron kinetics calculations to determine the power excursion. The static calculations determine the slug reactivity which is input to the power calculation. In lieu of doing detailed boron transport calculations different slug geometries and boron concentrations are assumed. The slug is assumed to enter the core uniformly across either the entire inlet area, or across only a section of the inlet. The slug reactivity is calculated as a function of the position of the slug front and a constant speed is assumed in order to translate the space dependence into a time dependence.

The neutron kinetics model, with the standard six groups of delayed neutron precursors, is combined with a heat conduction model in order to improve the accuracy of the fuel temperature calculation relative to an adiabatic model. This model also calculates the core average fuel enthalpy. At the time at which the fuel enthalpy is at a maximum the position of the slug front is noted and the corresponding static calculation is used to determine the power peaking factor. The local peak fuel enthalpy is then calculated by adding to the initial core average fuel enthalpy the increase in core average fuel enthalpy multiplied by the power peaking factor. The local peak fuel enthalpy can then be compared to the criterion for catastrophic fuel damage. This is taken to be 280 cal/g (1.2 MJ/kg) because it is equal to the approximate threshold for mechanical energy release, as determined from experiments [5.1], and because fuel fragmentation has been observed [5.2] at this fuel enthalpy. Hence, we equate catastrophic fuel damage with a change in geometry which could in turn lead to other fuel damage mechanisms.

Note that this approach does not take into account other consequences of rapid dilution events. If there is no catastrophic fuel damage there is still the possibility of release of fission products due to cladding damage either because of stresses caused at lower enthalpies or because of dryout on the surface of the clad. There is also the possibility of a pressure increase that could be excessive under shutdown conditions.

5.2 Static Core Model

The three-dimensional core calculations were carried out using NODE-P2 [5.3] This code models the neutronics with one energy group and a nodal method where k_∞ and M^2 are the basic neutronic data for each fuel assembly. NODE-P2 has successfully been applied to core performance problems by many PWR licensees. Cycle 9 of the Calvert Cliffs 2 plant [5.4] was chosen to be modeled with the code because modeling information was available from another recent Brookhaven National Laboratory study at shutdown conditions [5.5].

Analysis of Consequences

The basic neutronics data needed for each fuel assembly found in Cycle 9 were generated using CASMO [5.6]. CASMO is a multigroup, two-dimensional, transport theory code for burnup calculations of light water reactor fuel assemblies. The code has been extensively validated. The data were generated for fresh fuel containing either 0, 4, 8, or 12 burnable poison rods (containing B_4C) and for four types of burned fuel from Cycle 8, each with a different enrichment and/or number of burnable poison rods. Each burned fuel assembly was assumed to be burned to the average exposure for that fuel type. To simplify the data generation two fuel types representing only five assemblies in Cycle 9 were not explicitly represented. A single burned assembly from Cycle 7 was represented as one of the bundles from Cycle 8 and 4 erbia bearing demonstration assemblies were represented as fresh fuel with four burnable poison rods.

Table 5.1 gives the enrichment, number of burnable poison rods, and burnup of each assembly type actually used. The table also indicates whether data was generated with and/or without control element assemblies (CEAs) present. There are two types of CEAs, each axially zoned differently with rods containing B_4C, Ag-In-Cd, Al_2O_3, or stainless steel. The assumption was made that all CEAs were identical and contained B_4C rods since more than 85% of the rods are of this composition. Part length assemblies were neglected. For each assembly the data were generated at boron concentrations of 1500, 750, and 0 ppm. At 750 ppm the data were generated at the base fuel temperature of 548°F (560 K) and at an elevated temperature of 1200°F (922 K). The moderator temperature was 548°F in all cases.

The core layout for Cycle 9 as modeled is shown in Figure 5.1. There is octant symmetry. In addition to the fuel type, the location of CEAs are noted, with those that are in the shutdown banks and regulating banks identified separately.

Table 5.1
Fuel Assemblies for NODE-P2 Model

Fuel Type	Enrichment w/o	Burnable Poison Rods	Burnup MWd/t	Control Rods
J	4.05	0	30	Yes/No
K	4.08	0	15	No
K/	4.08	8	21	Yes/No
K*	4.08	12	21	No
L	4.30	0	0	Yes/No
LX	4.30	4	0	Yes
L/	4.30	8	0	Yes
L*	4.30	12	0	Yes/No

5.3 Static Calculations

5.3.1 Base Calculations

The NODE-P2 core model with the shutdown banks removed and the boron concentration at 1500 ppm is meant to represent the core during the period of deboration when the final boron concentration has been reached. (The hot, full power (HFP) critical boron concentration with all rods out and equilibrium xenon is 1460 ppm [5.4]; without xenon it would be higher.) At this point the CEAs belonging to the regulating banks would be removed to achieve criticality and then to increase the power to operating conditions. The calculated k_{eff} using NODE-P2 is 1.0083 which is less than 1% too high and within the expected uncertainty. Changes in reactivity are relative to this value.

The worth of the shutdown bank at these conditions is calculated by NODE-P2 to be 4.6%. This is less than the value of 5.9% calculated independently at beginning-of-life, but at full power conditions [5.7], and greater than the worth expected at zero power conditions based on measurements for some of the shutdown banks [5.8].

The worth of a core-wide boron dilution over the range from 1500 to 750 ppm is calculated by NODE-P2 to be 7.8 pcm/ppm. The boron reactivity coefficient quoted for full power conditions is 8.2 pcm/ppm [5.4]. The difference between these boron worths has the correct trend as the removal of control rods at the full power condition is expected to result in a higher value for the coefficient.

At a boron concentration of 1500 ppm a core-average increase in fuel temperature (with the spatial distribution determined by power) from 548°F to 1200°F was calculated by NODE-P2 to give a Doppler coefficient of -2.6 pcm/°F. Again the only comparison that can easily be made is with the coefficient calculated for full power operation. At full power the coefficient is expected to range from -1.0 to -2.4 pcm/°F [5.4]. The magnitude of the Doppler coefficient is expected to be higher at zero power.

5.3.2 Pseudo Time-Dependent Calculations

The worth of a slug of diluted water was calculated by assuming a particular geometry and then doing a sequence of calculations representing the slug as it moved through the core. In these calculations all CEAs were inserted. The 9 cases considered are given in Table 5.2 which lists the change in boron concentration (relative to 1500 ppm) and the slug geometry.

Table 5.2
Diluted Slug Conditions

Case	Change in Boron Concentration, ppm	Slug Geometry
1	600	Semi-infinite step
2	750	Semi-infinite step
3	1000	Semi-infinite step
4	1500	Semi-infinite step
5	750	535 ft^3 rectangular step
6	750	535 ft^3 trapazoidal step
7	750	Semi-infinite step - core center
8	1000	Semi-infinite step - core center
9	1000	Semi-infinite step - core edge

In Cases 1-4 the dilution is assumed to occur uniformly across the core and the change in boron concentration is represented as a sharp wave front (step). The reactivity effect of the dilution in these cases is shown in Figure 5.2 as a function of the position of the front of the dilution boundary. Each node corresponds to almost one foot so that when Node 12 is reached the entire core (136.7 in length) is diluted to the new boron concentration. The curves follow an 'ess' shape with the most rapid changes in reactivity occurring when the slug front is moving through the bottom of the core. (Note that this is the complement to the problem of control rods being worth relatively little until they move *past* the midway point.) An additional non-linearity is the effect of the degree of boron dilution with the reactivity of a 1500 ppm change being more than twice that for a 750 ppm change. As expected, Figure 5.2 shows that the magnitude of the slug reactivity can be very large if the dilution is large.

In Cases 5 and 6 the slug is not semi-infinite but rather corresponds to a fixed volume of 535 ft^3. This corresponds to 2000 gal of water available from the volume control tank (VCT) multiplied by two to account for an amount of VCT water (with an assumed concentration of 0 ppm) mixing with an equal amount of water from the reactor coolant system (RCS) (with an assumed concentration of 1500 ppm) to create the slug of diluted water at a concentration of 750 ppm. The reactivity change for these two cases is shown in Figure 5.3. The rectangular case starts off identical to the semi-infinite step shown in Figure 5.2 and reaches approximately the same peak reactivity since the length of the slug is almost equal to (actually 10/12) the core height. As the wave front approaches the top of the core the boron concentration in the bottom of the core changes back to 1500 ppm and the total reactivity begins to decrease. The volume of the slug is the same in the trapazoidal case but rather than a jump change in concentration it changes over a distance of 5 ft. The total length of the slug in this case is almost 15 ft. As expected, the effect of changing the geometry is a slower rise and a delay in returning to the initial condition.

In Figure 5.4 the effect of different radial geometries is shown by plotting Cases 2, 7, 8, and 9. In Case 7 the dilution is at the center and affects a 7x7 array of fuel assemblies. Since this is only 49/217 (=0.23) of the core the effect of a 750 ppm dilution is less than that obtained when the dilution is uniform (Case 2). This can be seen on Figure 5.4. However, since the slug is in the center of the core, where the neutron importance is relatively high, the effect is greater than 0.23 of that obtained in Case 2. Also shown on the figure is the effect of increasing the dilution in the center from 750 ppm to 1000 ppm (i.e., reducing the boron concentration to 500 ppm). With the larger dilution the figure shows that the reactivity effect is almost equal to that achieved with a uniform dilution.

For the same 1000 ppm dilution Figure 5.4 also shows the effect if only assemblies on the core edge are affected. For Case 9 there are 25 assemblies diluted at either end of one of the core axes. This is shown on Figure 5.5 (which also shows the pattern when the dilution was at the core center). The total of 50 assemblies affected is almost equal to the 49 affected in the center of the core but since the neutron importance at the core periphery is less than at the center, the reactivity effect is much smaller.

5.4 Dynamic Core Model

The dynamic core model consists of the point neutron kinetics equations, including six groups of delayed neutron precursors, and a simple thermal-hydraulic model to obtain the core average fuel enthalpy. The thermal-hydraulic model consists of equations for the average pellet, clad, and coolant temperature. A gap heat conductance (as a function of fuel temperature) is used and a fixed heat transfer coefficient for the clad is determined to yield the proper initial conditions. The model was solved with the DESIRE software package [5.9] on a personal computer.

Boron dilution reactivity as a function of time was taken from the static calculations described in Section 5.3.2 by assuming a constant speed for the slug front. This was taken to be 2.0 ft/s which corresponds to 13% of rated flow. This is an approximation to the flow which would increase from close to zero (assuming little natural circulation) to 20% of full flow in about 20 seconds.

A constant negative reactivity representing the initial shutdown margin was used. This is meant to account for the worth of the shutdown banks in the reactor startup scenario and in general for any other contribution to shutdown margin that might be present before the dilution begins. In the present calculation -4.0% shutdown is assumed to be the base initial condition. In Calvert cliffs this is approximately equal to the shutdown bank worth but in other plants the shutdown worth might be smaller.

Fuel temperature reactivity was calculated during the dynamic simulation using the core average fuel temperature calculated by the model and a Doppler feedback coefficient expressed per unit change in square root of absolute temperature. The Doppler feedback is strongest in the region where the fuel temperature is highest and since the power, and hence the neutron importance, is also highest in this region, it is a gross inaccuracy to represent the Doppler feedback using a core-average fuel temperature. To improve upon the accuracy of the fuel temperature feedback a Doppler weighting factor (DWF) was applied to the reactivity calculated using the core average fuel temperature.

The DWF is defined as the actual change in fuel temperature reactivity during a dilution divided by the change which would occur with the same average temperature change uniformly distributed over the core (or as assumed when generating a core average Doppler coefficient). In practice it is approximated as the change in reactivity if the fuel temperature is increased according to the power distribution expected during the dilution divided by the change in reactivity for the same average temperature change with the power distribution at the initial condition. Using this definition, the DWF is calculated by doing additional static calculations at elevated power (and thus temperature), at each of the slug positions used to determine the boron dilution reactivity. The change in reactivity for this change in temperature at this slug position is then divided by the change in reactivity when the temperature change is made prior to the slug moving into the core. It is then input to the dynamic calculation as a function of time just as the slug reactivity is input.

Some of the nominal initial conditions used for the dynamic calculations are given in Table 5.3. The delayed neutron fraction represents a midpoint in the range of 0.0044 to 0.0070 expected for the cycle [5.4].

The dynamic core model calculates the core average enthalpy. At the time at which this is a maximum the power peaking factor from the static calculation corresponding to this time is extracted and used to correct the enthalpy so that the final result is the local peak enthalpy.

Table 5.3
Nominal Initial Core Conditions

Power (Decay Heat)	10 MW
Fuel Temperature	548°F
Coolant Temperature	548°F
Inlet Flowrate	13% of Rated, 2.0 ft/s
Delayed Neutron Fraction	0.0056
Shutdown Margin	4.0%
Doppler Coefficient	-2.6 pcm/°F

5.5 Dynamic Calculations

The dynamic calculations were carried out to determine the peak fuel enthalpy during a rapid dilution event in order to know if catastrophic fuel damage could occur. No attempt was made to calculate other effects such as the extent of fuel melting in the center of the pellet, clad temperatures if boiling transition is observed, or the pressure rise in the system. A more rigorous calculational model would be necessary to observe this behavior and to understand how the core would respond other than through catastrophic fuel damage.

The power response for the base case described in Section 5.4 is shown in Figure 5.6. In this case a 535 ft^3 slug of diluted water at a boron concentration of 750 ppm passes through the core at a speed of 2.0 ft/s. The power trace is typical for a prompt-critical reactivity excursion in a PWR. When the reactivity addition due to the diluted water exceeds the assumed shutdown margin by the delayed neutron fraction the power rises rapidly. In this case, with an assumed shutdown margin of 4% it can be seen from Figure 5.3 that when the slug-front is between Node 5 and 6 this condition is satisfied. Hence, at approximately 3 seconds into the transient the power rises rapidly until the almost instantaneous fuel temperature response (typical of a PWR) causes sufficient negative feedback to terminate the initial power rise. Although the peak power at 75 GW is very high, what is important in determining the fuel response is the integral of power, i.e., the energy that is deposited in the fuel.

The energy deposition causes an increase in fuel temperature and enthalpy. The radial pellet average enthalpy is the quantity used to determine whether or not catastrophic fuel damage has occurred (cf Section 5.1). The core average fuel enthalpy for this case is shown in Figure 5.7. In order to determine the peak enthalpy in the core a power peaking factor obtained from the steady state calculation is applied as explained in Section 5.1. The power peaking factor is 6.3 and this means that the initial power rise corresponds to a peak fuel enthalpy of 69 cal/g which is much less than the criterion for fuel damage. The power peaking factor is large because the slug is only half-way into the core at this time.

After the initial power rise the power decreases but then, as can be seen on Figure 5.6, the power rises again before it slowly starts to decrease after 5-6 seconds. The increase comes from the fact that reactivity is continuing to be added until that time period. The decrease in power comes from the decrease in slug reactivity but the decrease in core average fuel enthalpy comes primarily from the fact that there is heat transfer from the fuel into the coolant and this becomes appreciable in the period after 6 seconds. Note that when there is a large amount of energy transferred into the coolant the model is no longer applicable. Two-phase flow would be expected and significant negative feedback from the coolant would affect the response of the fuel rods. Although this was not modeled in this calculation the peak fuel enthalpy was calculated during the period of 5-6 seconds when the core average fuel enthalpy reaches a peak. The value was 102 cal/g which again is an indication that no catastrophic fuel damage would occur.

These results will be most sensitive to the initial shutdown margin, the Doppler feedback, and the properties of the slug. Other factors which will have a secondary impact are the delayed neutron fraction, the neutron lifetime, and the speed at which the slug moves through the core.

In order to quantify the effect of initial shutdown margin, several additional calculations were completed. The initial shutdown margin represents what the condition of the core might be by virtue of the operating mode combined with the effect of any control rods that might insert prior to the slug passing through the core. In the reactor restart scenario even if the core was initially at its final boron concentration there would still be some negative reactivity as the reactor is usually brought to critical on the movement of the regulating banks. Assuming that the reactivity hold-down of the regulating banks is small, then the initial shutdown margin is just the worth of the shutdown banks which would scram when there was a loss of off-site power. Although this was assumed to be 4% in the base case, for some plants this might only be 2%. If there were no rods initially withdrawn then the smallest shutdown margin that could be expected would be the 1% requirement when at hot shutdown conditions. Hence, the calculations were done with an initial shutdown margin down to 1%. At the other end of the scale is the fact that if the shutdown margin were equal to or greater than the amount of reactivity which could be inserted by the diluted slug then no power excursion can take place. In the case being considered with a 750 ppm slug, this is 5.9%.

The results of these calculations are shown in Figure 5.8. Peak fuel enthalpy is plotted for the time corresponding to immediately after the initial power spike and for the time at which the core average enthalpy exhibits a broad maximum (cf Figure 5.7). The times at which these peaks occur become later with an increase in initial shutdown margin as it takes longer to overcome that barrier and become supercritical. As can be seen, when the initial shutdown margin is between 1% and 2% the peak fuel enthalpy can exceed the 280 cal/g criterion for catastrophic fuel damage. This is not the result of the initial power burst but rather the fact that the power remains high due to the continued presence of the diluted slug.

Figure 5.8 also shows the effect of reducing the Doppler feedback by a factor of 0.5. As explained in Section 5.3.1 the reduction in the Doppler coefficient from -2.6/°F to -1.3/°F is consistent with the range of coefficients expected during operation of this cycle of Calvert Cliffs and similar to how other PWRs operate.

An estimate of the boron dilution necessary to cause the fuel enthalpy to exceed 280 cal/g when the initial shutdown margin is 4% can be made by using the results shown in Figure 5.8. Since that figure shows that a reduction of the shutdown margin to approximately 1.5% would cause the peak fuel enthalpy to exceed 280 cal/g it can be estimated that at an initial shutdown margin of 4% one would need additional reactivity worth 2.5%, or approximately an additional dilution of 320 ppm. Hence, if the boron concentration of the slug was approximately 430 ppm and the initial shutdown margin was 4%, catastrophic fuel damage might be likely. Note that with this boron concentration the volume of the diluted slug would be 1.4 times the volume of the unborated water assumed to be available from the VCT or 375 ft^3. The length of the slug would then be approximately 7 ft rather than the 10 ft for the nominal case. This is not expected to alter the behavior of the power excursion during the period where the fuel enthalpy reaches a maximum, however, if the slug was even smaller and more dilute it is not clear that the situation would lead to a higher fuel enthalpy. The competition between a higher positive reactivity insertion and a shorter insertion time in the limit of zero volume leads to a smaller effect.

If the slug were to be located in the center of the core as shown for Cases 7 and 8 in Section 5.3.2 then there are several factors which change the response of the core. It takes a larger dilution to cause the same power excursion with a uniform radial distribution as can be inferred from the curves in Figure 5.4. However, for a given power excursion the peak fuel enthalpy would be higher with the slug localized in the core center because the power peaking factor might be higher. For the cases considered herein the increase in power peaking is approximately 40%. If the situation of concern is a highly dilute slug (with perhaps a high initial shutdown margin) then the power excursion with the slug in the core center will be different from the case with a uniform radial distribution because the smaller inlet area for the slug means that it will be in the core longer. Hence, the severity of the excursion will be a complicated function of slug dilution, geometry, and volume as well as initial shutdown margin and Doppler feedback.

In the above calculations the reactivity feedback from the reactor coolant was neglected. After power rises, the increase in coolant temperature and the possibility of steam voids would tend to reduce the power after sufficient time has elapsed for significant heat transfer. Another, potentially more important effect of coolant temperature occurs at the outset of the event if the diluted slug is significantly cooler than the initial water in the core. This may be the case as one of the important mechanisms for not having mixing of any diluted water entering the system is the relatively high density of the injectant if it is colder than the water in the RCS. The reactivity addition due to the temperature of the slug could be significant, especially at the end of a fuel cycle when the moderator temperature coefficient (MTC) has its largest magnitude. With the same conditions used to generate Figure 5.8 (with shutdown margin of 4%) the peak fuel enthalpy would exceed 280 cal/g if the MTC was greater in magnitude than -20 pcm/°F. This is based on a slug temperature obtained by mixing equal amounts of unborated water at 298°F with 1500 ppm RCS water at 548°F.

The interpretation of the effect of cooler slug temperatures is complicated by the fact that the above neutronics calculations are for a fixed boron density (g/cm^3) and the boron concentrations (ppm) quoted above assume a slug temperature equal to the initial temperature (548°F) in the core. If the coolant density is lower then the boron concentration quoted must be lower. For example, the boron concentration would have to be reported as 675 ppm at 298°F rather than 750 ppm in order to have an equivalent density.

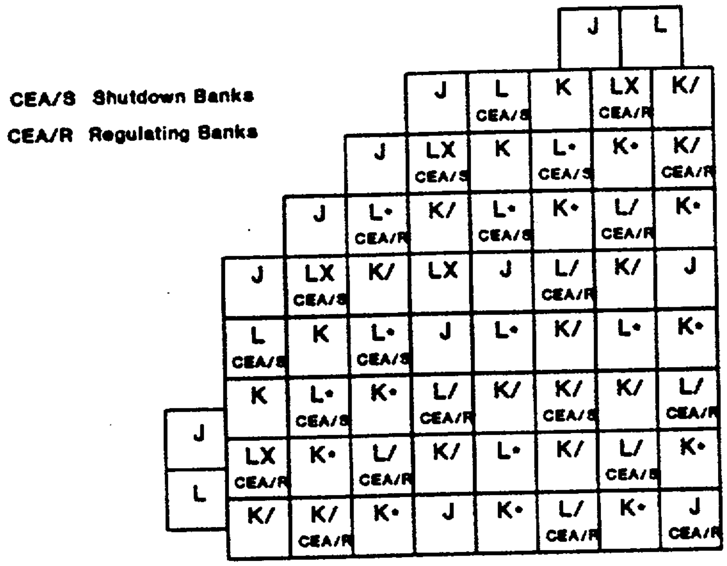

Figure 5.1 Distribution of Fuel Types and CEAs

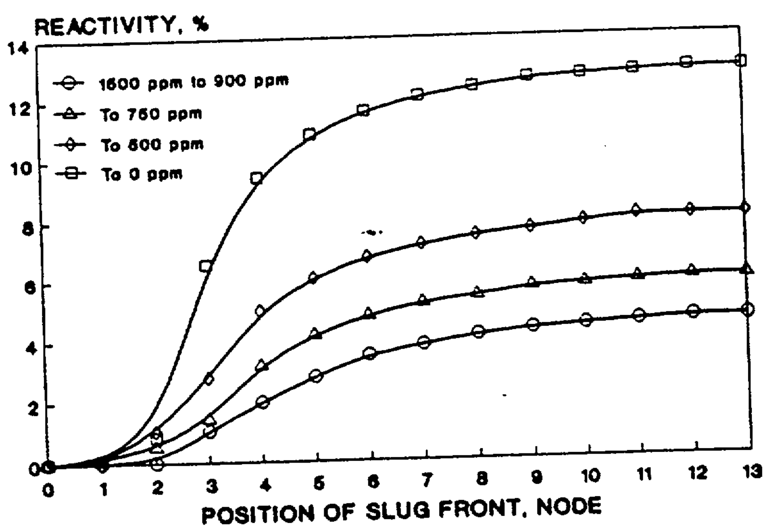

Figure 5.2 Reactivity Effect of Diluted Slug

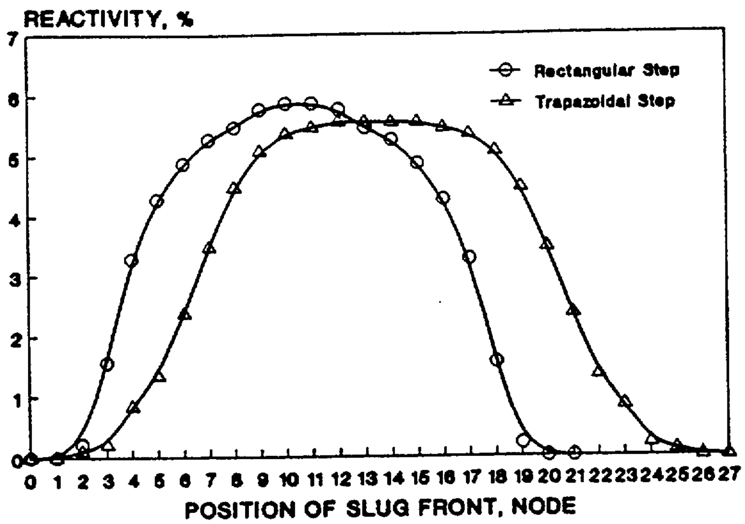

Figure 5.3 Reactivity Effect of Diluted Slug

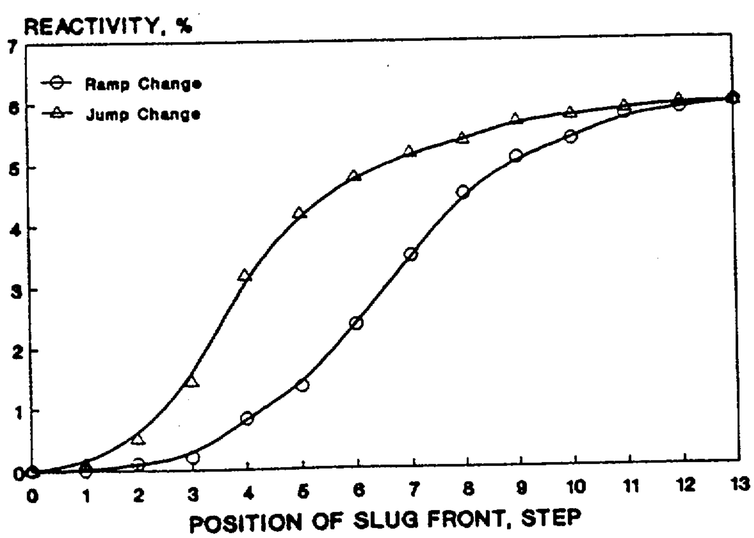

Figure 5.4 Reactivity Effect of Diluted Slug

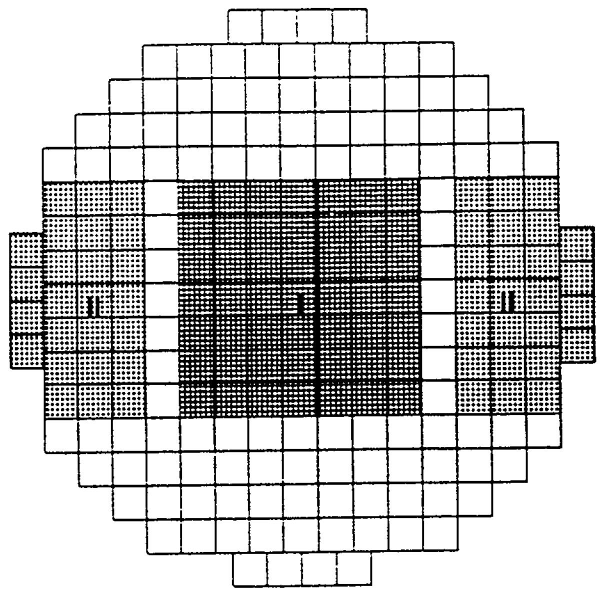

Figure 5.5 Core Center (I) and Core Edge (II) Patterns

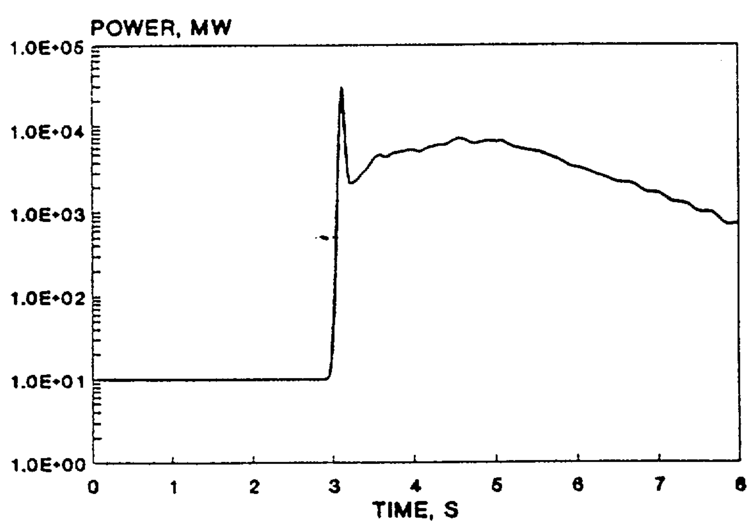

Figure 5.6 Core Power with Finite Slug

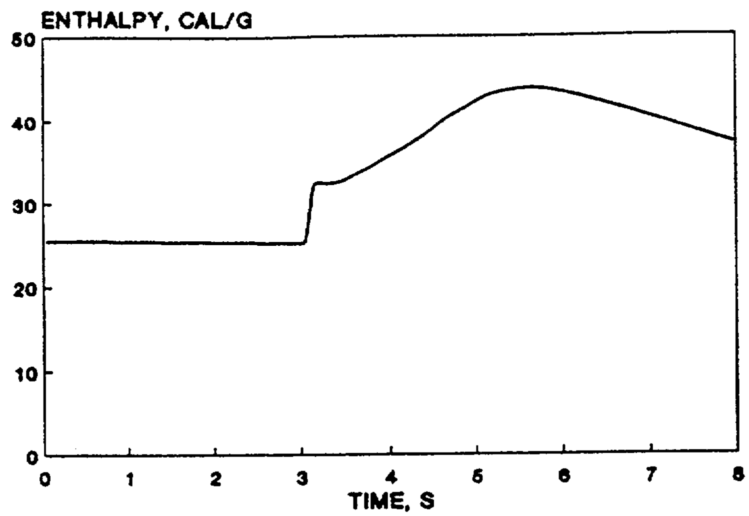

Figure 5.7 Core Average Fuel Enthalpy

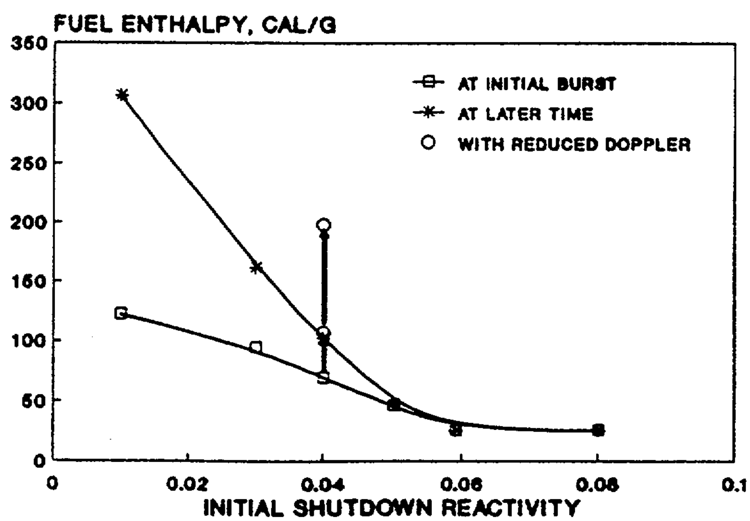

Figure 5.8 Peak Fuel Enthalpy vs Shutdown Margin

6 Summary of Results and Conclusions

The analysis of a rapid boron dilution event has been carried out in three different ways: 1) a probabilistic analysis for the core damage frequency, 2) a mixing analysis to determine the extent of dilution before injection into the core, and 3) neutronics calculations to determine core behavior and the consequences in terms of fuel damage.

The probabilistic analysis was done for reactors from each of the three U.S. reactor vendors. The CDF varies from 9.7E-6/yr to 2.8E-5/yr for the three plants which is what is calculated for other internal events that are considered to be important. It is, however, less than that calculated for the same event in a European reactor. One of the reasons for this is that the initiating frequency is lower in the present analysis. No other conclusions can be reached on the differences because the details of the European analysis were not available for this study.

The analysis shows the importance of the primary grade water pump. For all three plants the potential for an accident is limited by the amount of diluted water in the VCT as the supply of primary grade water is stopped by the trip of the PG makeup pump. However, there are plants where this pump is connected to the emergency bus and the probability of an accident will be increased if primary grade water continues to be pumped into the VCT. This appears to be the case for some plants in France and Sweden and is another reason why the problem may be more serious there. The question of whether the makeup pump trips or continues to run has to be evaluated on a unit by unit basis.

For all three plants the dilution is done with flow from the VCT. It should be noted that in some plants the suction for the charging flow comes directly from the primary grade makeup water source and once the PG water pump is tripped there is no longer the potential for adding unborated water to the RCS.

The results are dependent on assumptions used in the analysis which are summarized below. Note that the assumptions result in overestimating the core damage frequency. More detailed analysis would have to be carried out to quantify the effect of these assumptions.

1. The dilution time during startup is 8 hrs. The consequences of the event are assumed independent of when during this period the loss of off-site power occurs. In reality, an event occurring early during this period will have more shutdown margin to overcome and is, therefore, expected to have less of an effect than an event occurring near the end of the normal dilution procedure.

2. No credit is given for the operator to take action and stop the charging flow from the VCT after the LOOP. Although dilution while the shutdown banks are inserted or the RCPs are stopped, (as would be the case after a LOOP) is not a normal procedure, it is assumed that since the operator knows that the flow from the primary water makeup pump has ceased, that no other action is deemed warranted. An action that could be taken by the operator would be to switch the charging pump suction to the RWST.

3. The analysis does not account for any mixing at the point of injection or mixing of the diluted water due to its flow into the downcomer and down to the lower plenum. It does, however, account for the effect of natural circulation on mixing. The assumption is made that if it is a startup without refueling then there is sufficient decay heat and natural circulation to mix the injectant to the extent that the probability of causing core damage is reduced by a factor of 0.5. Mixing analysis performed for two of the plants shows that this may be very conservative under certain conditions.
4. Another important assumption is that if off-site power, or another adequate power source, is available, the reactor coolant pumps (RCPs) will be started during a 30-minute interval.

The mixing analysis was done with the Regional Mixing Model developed to study the thermal mixing of interest to pressurized thermal shock. Illustrative reactor predictions for a Westinghouse and a Babcock & Wilcox plant show significant mixing during the boron dilution transients. For the cases considered, the boron concentration in the lower plenum does not fall below 900 ppm when the initial boron concentration in the vessel is 1500 ppm. However, these cases do not encompass all possible physical situations for these plants. It would be desirable to assess the extent of mixing when the injectant enters at the side or bottom of the cold leg piping, as is the case in some plants, and to quantify the additional mixing due to jet impingement for the range of injection conditions of interest.

The neutronic results show that there is the possibility of catastrophic fuel damage depending on a) the initial shutdown margin, b) the Doppler feedback, and c) the properties of the slug, especially the boron concentration. The initial shutdown margin depends on the reactor state at the time of initiation of the event and the reactivity worth of the shutdown bank which will scram before the slug enters the core. The Doppler feedback is responsible for initially terminating the power excursion. This number can vary by a factor of two during a fuel cycle and therefore results will be sensitive to where in the fuel cycle the event takes place. After the initial power excursion the power remains high until the slug has passed sufficiently through the core so that power decreases. Since there may not be much shutdown margin to begin with, after the slug passes through the core the decline in power may not be as rapid as occurs with reactor trip. This also impacts the consequences in terms of fuel damage.

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11. ABSTRACT (200 words or less)

This report documents the results of a scoping study of rapid dilution events in pressurized water reactors. It reviews the subject in broad terms and focuses on one event of most interest. This event could occur during a restart if there is a loss-of-offsite power when the reactor is being deborated. If the volume control tank is filled with water at a low boron concentration then a slug of this water could accumulate in the lower plenum. This would be the result of the trip of the reactor coolant power. The concern is that this diluted slug will rapidly enter the core after a reactor coolant pump is restarted and this could cause a power excursion leading to fuel damage. This problem was studied probabilistically for three plants and the important design features that affect the core damage frequency were identified. This analysis was augmented by an analysis of the mixing of the diluted water with the borated water already present in the vessel. The mixing was found to be significant so that neglect of this mechanism in the probabilistic analysis leads to very conservative results. Neutronic calculations for one plant were carried out to understand the effect of nuclear design on the consequences of the event.

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