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JMazetis

JUL 7 1975

Docket No.: 50-346

Toledo Edison Company
ATTN: Mr. Lowell E. Roe
Vice President
Edison Plaza
300 Madison Avenue
Toledo, Ohio 43652

Gentlemen:

B&W Topical Report No. 10105 is presently under review in support of your application to construct and operate the Davis-Besse, Unit 1 facility. To complete the review of your application with regard to compliance with 10 CFR 50.46, certain material in addition to that submitted in the referenced topical report is needed.

Attachment 1 to this latter is an overall requirements statement delineating all information necessary for the staff to complete its review of ECCS capability on each and every application docket. Each NSSS vendor (including B&W) has already been provided with all the attached information except the first two pages.

We urge you to evaluate these requirements and be assured that your submittals on the Davis-Besse, Unit 1 docket include all the required information outlined in Attachment 1. Please advise this office within 10 days of your schedule for submitting additional information as required.

Sincerely,

Original Signed by

A. Schwencer, Chief
Light Water Reactors Branch 2-3
Division of Reactor Licensing

Attachment 1
Required Information

ccs: See next page

OFFICE ▶	x7886/LWR2-3	C-LWR2-3:RL				
SURNAME ▶	LEngle:rm	ASchwencer				
DATE ▶	7/7/75	7/7/75				

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

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w/o enclosure
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bccs: J. R. Buchanan, ORNL w/o encl.
T. B. Abernathy, DTIE w/o encl.

OFFICE ▶						
SURNAME ▶						
DATE ▶						

Attachment 1

REQUIRED INFORMATION

1. Break Spectrum and Partial Loop Operation

The information provided for each plant shall comply with the provisions of the attached memorandum entitled, "Minimum Requirements for ECCS Break Spectrum Submittals."

2. Potential Boron Precipitation (PWR's Only)

The ECCS system in each plant should be evaluated by the applicant (or licensee) to show that significant changes in chemical concentrations will not occur during the long term after a loss-of-coolant accident (LOCA) and these potential changes have been specifically addressed by appropriate operating procedures. Accordingly, the applicant should review the system capabilities and operating procedures to assure that boron precipitation would not compromise long-term core cooling capability following a LOCA. This review should consider all aspects of the specific plant design, including component qualification in the LOCA environment in addition to a detailed review of operating procedures. The applicant should examine the vulnerability of the specific plant design to single failures that would result in any significant boron precipitation.

3. Single Failure Analysis

A single failure evaluation of the ECCS should be provided by the applicant (or licensee) for his specific plant design, as required by Appendix K to 10 CFR 50, Section I.D.1. In performing this evaluation, the effects of a single failure or operator error that causes any manually controlled, electrically-operated valve to move to a position that could adversely affect the ECCS must be considered. Therefore, if this consideration has not been specifically reported in the past, the applicants upcoming submittal must address this consideration. Include a list of all of the ECCS valves that are currently required by the plant Technical Specifications to have power disconnected, and any proposed plant modifications and changes to the Technical Specifications that might be required in order to protect against any loss of safety function caused by this type of failure. A copy of Branch Technical Position EICSB 18 from the U.S. Nuclear Regulatory Commission's Standard Review Plan is attached to provide you with guidance.

The single failure evaluation should include the potential for passive failures of fluid systems during long term cooling following a LOCA as well as single failures of active components. For PWR plants, the single failure analysis is to consider the potential boron concentra- problem as an integral part of long term cooling.

4. Submerged Valves

The applicant should review the specific equipment arrangement within his plant to determine if any valve motors within containment will become submerged following a LOCA. The review should include all valve motors that may become submerged, not only those in the safety injection system. Valves in other systems may be needed to limit boric acid concentration in the reactor vessel during long term cooling or may be required for containment isolation.

The applicant (or licensee) is to provide the following information, fo- each plant:

(1) Whether or not any valve motors will be submerged following a LOCA in the plant being reviewed.

(2) If any valve motors will be flooded in their plant, the applicant (or licensee) is to:

(a) Identify the valves that will be submerged.

(b) Evaluate the potential consequences of flooding of the valves for both the short term and long term ECCS functions and containment isolation. The long term should consider the potential problem of excessive concentrations of boric acid in PWR's.

(c) Propose a interim solution while necessary modifications are being designed and implemented. (currently operating plants only).

(d) Propose design changes to solve the potential flooding problem.

5. Containment Pressure (PWR's Only)

The containment pressure used to evaluate the performance capability of the ECCS shall be calculated in accordance with the provisions of Branch Technical Position CSB 6-1, which is enclosed.

6. Low ECCS Reflood Rate (Westinghouse NSSS Only)

Plants that have a Westinghouse nuclear steam supply shall perform their ECCS analyses utilizing the proper version of the evaluation model, as defined below:

(1) The December 25, 1974 version of the Westinghouse evaluation model, i.e., the version without the modifications described in WCAP-8471 is acceptable for previously analyzed plants for which the peak clad temperature turnaround was identified prior to the reflood rate decreasing below 1.1 inches per second or for which the reflood rate was identified to remain above 1.0 inch per second; conditions for which the December 25, 1974 and March 15, 1975 versions would be equivalent.

(2) The March 15, 1975 version of the Westinghouse evaluation model is an acceptable model to be used for all previously analyzed plants for which the peak clad temperature turnaround was identified to occur after the reflood rate decreased below 1.1 inches per second, and for which steam cooling conditions (reflood rate less than 1 inch per second) exist prior to the time of peak clad temperature turnaround. The March 15, 1975 version will be used for all future plant analyses.

MINIMUM REQUIREMENTS FOR ECCS BREAK SPECTRUM SUBMITTALS

I. INTRODUCTION

The following outline shall be used as a guideline in the evaluation of LOCA break spectrum submittals. These guidelines have been formulated for contemporary reactor designs only and must be re-assessed when new reactor concepts are submitted.

The current ECCS Acceptance Criteria requires that ECCS cooling performance be calculated in accordance with an acceptable evaluation model and for a number of postulated loss-of-coolant accidents of different sizes, locations and other properties sufficient to provide assurance that the entire spectrum of postulated loss-of-coolant accidents is covered. In addition, the calculation is to be conducted with at least three values of a discharge coefficient (C_D) applied to the postulated break area, these values spanning the range from 0.6 to 1.0.

Sections IIA and IIIA define the acceptable break spectrum for most operating plants which have received Safety Orders. Sections IIB and IIIB define the break spectrum requirements for most CP and OL case work (exceptions noted later). Sections IIC and IIIC provide an outline of the minimum requirements for an acceptable complete break spectrum. Such a complete break spectrum could be appropriately referenced by some plants. Sections IIID and IIIE provide the exceptions to certain plant types noted above.

A plant due to reload a portion of its core will have previously submitted all or part of a break spectrum analysis (either by reference or by specific calculations). If it is the intention of the Licensee to replace expended fuel with new fuel of the same design (no mechanical design differences which could affect thermal and hydraulic performance), and if the Licensee intends to operate the reloaded core in compliance with previously approved Technical Specifications, no additional calculations are required. If the reload core design has changed, the Licensee shall adopt either of Sections IIA or IIC, or of Sections IIIA or IIIC of this document, as appropriate to the plant type (BWR or PWR). The criterion for establishing whether paragraph A or C shall be satisfied will be determined on the basis of whether the Licensee can demonstrate that the shape of the PCT versus break size curve has not been modified as a consequence of changes to the reload core design. When the reload is supplied by a source other than the NSSS supplier, the break spectrum analyses specified by Sections IIC or IIIC shall be submitted as a minimum (as appropriate to the plant type, BWR or PWR). Additional sensitivity studies may be required to assess the sensitivity of fuel changes in such areas as single failures and reactor coolant pump performance.

II. PRESSURIZED WATER REACTORS

A. Operating Reactor Reanalyses (Plants for which Safety Orders were issued)

If calculational changes* were made to the LBM** to make it wholly in

* Calculational changes/Model changes--those revisions made to calculational techniques or fixed parameters used for the referenced complete spectrum.

** LBM--Large Break Model; SBM--Small Break Model conformance with 10CFR50, Appendix K, the following minimum number of break sizes should be reanalyzed. Each sensitivity study performed during the development of the ECCS evaluation model shall be individually verified as remaining applicable, or shall be repeated. A plant may reference a break spectrum analysis conducted on another plant if it is the same configuration and core design.

1. If the largest break size results in the highest PCT:

   a. Reanalyze the limiting break.

   b. Reanalyze two smaller breaks in the large break region.

2. If the largest break size does not result in the highest PCT:

   a. Reanalyze the limiting break.

   b. Reanalyze a break larger and a break smaller than the limiting break. If the limiting break is outside the range of Moody multipliers of 0.6 to 1.0 (i.e., less than 0.6), then the limiting break plus two larger breaks must be analyzed.

If calculational changes have been made to the SBM to make it wholly in conformance with 10CFR50, Appendix K, the analysis of the worst small break (SBM) as previously determined from paragraph C below should be repeated.

B. New CP and OL Case Work

A complete break spectrum should be provided in accordance with paragraph C below, except for the following:

1. If a new plant is of the same general design as the plant used as a basis for a referenced complete spectrum analysis, but operating parameters have changed which would increase PCT or metal-water reaction, or approved calculational changes resulting in more than 20°F change in PCT have been made to the ECCS model used for the referenced complete spectrum, the analyses of paragraph A above should be provided plus a minimum of three small breaks (SBM), one of which is the transition break.* The shape of the break spectrum in the referenced analysis should be justified as remaining applicable, including the sensitivity studies used as a basis for the ECCS evaluation model.

2. If a new plant (configuration and core design) is applicable to all generic studies because it is the same with respect to the generic plant design and parameters used as a basis for a referenced complete spectrum defined in paragraph C, and no calculational changes resulting in more than 20°F change in PCT were made to the ECCS model used for the referenced complete spectrum, then no new spectrum analyses are required. The new plant may instead reference the applicable analysis.

---

* Transition Break (TB)--that break size which is analyzed with both the LBM and SBM.

C. Minimum Requirements for a Complete Break Spectrum

Since it is expected that applicants will prefer to reference an applicable complete break spectrum previously conducted on another plant, this paragraph defines the minimum number of breaks required for an acceptable complete break spectrum analysis, assuming the cold leg pump discharge is established as the worst break location. The worst single failure and worst-case reactor coolant pump status (running or tripped) shall be established utilizing appropriate sensitivity studies. These studies should show that the worst single failure has been justified as a function of break size. Each sensitivity study published during the development of the ECCS evaluation model shall be individually justified as remaining applicable, or shall be repeated. Also, a proposal for partial loop operation shall be supported by identifying and analyzing the worst break size and location (i.e., idle loop versus operating loop). In addition, sufficient justification shall be provided to conclude that the shape of the PCT versus Break Size curve would not be significantly altered by the partial loop configuration. Unless this information is provided, plant Technical Specifications shall not permit operation with one or more idle reactor coolant pumps.

It must be demonstrated that the containment design used for the break spectrum analysis is appropriate for the specific plant analyzed. It should be noted that this analysis is to be performed with an approved evaluation model wholly in conformance with the current ECCS Acceptance Criteria.

1. LBM--Cold Leg-Reactor Coolant Pump Discharge

   a. Three guillotine type breaks spanning at least the range of Moody multipliers between 0.6 and 1.0.

   b. One split type break equivalent in size to twice the pipe cross-sectional area.

   c. Two intermediate split type breaks.

   d. The large-break/small-break transition split.

2. LBM--Cold Leg-Reactor Coolant Pump Suction

   Analyze the largest break size from part 1 above. If the analyses in part 1 above should indicate that the worst cold leg break is an intermediate break size, then the largest break in the pump suction should be analyzed with an explanation of why the same trend would not apply.

3. LBM--Hot Leg Piping

   Analyze the largest rupture in the hot leg piping.

4. SBM--Splits

Analyze five different small break sizes. One of these breaks must include the transition split break. The CFT line break must be analyzed for B&W plants. This break may also be one of the five small breaks.

III. BOILING WATER REACTORS

The generic model developed by General Electric for BWRs proposed that split and guillotine type breaks are equivalent in determining blowdown phenomena. The staff concluded this was acceptable and that the break area may be considered at the vessel nozzle with a zero loss coefficient using a two phase critical flow model. Changes in the break area are equivalent to changes in the Moody multiplier.

The minimum number of breaks required for a complete break spectrum analysis, assuming a suction side recirculation line break is the design basis accident (DBA) and the worst single failure has been established utilizing appropriate sensitivity studies, are shown in paragraph C below. Also, a proposal for partial loop operation shall be supported by identifying and analyzing the worst break size and location (i.e., idle loop versus operating loop). In addition, sufficient justification shall be provided to conclude that the shape of the PCT versus Break Size curve would not be significantly altered by the partial loop configuration. Unless this information is provided, plant Technical Specifications shall not permit operation with one or more idle reactor coolant pumps.

A. BWR2, BWR3. and BWR4 Reanalysis (Plants for which Safety Orders were issued)

If the referenced lead plant analysis is in accordance with Section III, paragraph C below, the following minimum number of break sizes should be reanalyzed. It is to be noted that the lead plant analysis is to be performed with an approved evaluation model wholly in conformance with the current ECCS Acceptance Criteria. A plant may reference a break spectrum analysis conducted on another plant if it is the same configuration and core design.

Each sensitivity study published during the development of the ECCS evaluation model shall be individually justified as remaining applicable, or shall be repeated.

1. If the largest break results in the highest PCT:

   a. Reanalyze the limiting break with the appropriate referenced single failure.

   b. Reanalyze the worst small break with the appropriate referenced single failure.

   c. Reanalyze the transition break with the single failure and model that predicts the highest PCT.

2. If the largest break does not result in the highest PCT:

   a. Reanalyze the limiting break, the largest break, and a smaller break.

If calculational changes have been made to the SBM to make it wholly in conformance with 10CFR50, Appendix K, reanalyze the small break (SBM) in accordance with Section IIIC.

B. New CP and OL Case Work

A complete break spectrum should be provided in accordance with Section III, paragraph C below, except for the following:

1. If a new plant is of the same general design as the plant used as a basis for the lead plant analysis, but operating parameters have changed which would increase PCT or metal-water reaction, or approved calculational changes have been made to the ECCS model resulting in more than 20°F change in PCT, the analyses of Section III, paragraph A above should be provided plus a minimum of three small breaks (SBM), one of which is the transition break. The shape of the break spectrum in the lead plant analysis should be justified as remaining applicable, including the sensitivity studies used as a basis for the ECCS evaluation model.

2. If a new plant (configuration or core design) is applicable to all generic studies because it is the same with respect to the generic plant design and parameters used as a basis for a referenced complete spectrum defined in paragraph C, and no calculational changes resulting in more than 20°F change in PCT were made to the ECCS model used for the referenced complete spectrum, then no new spectrum analyses are required. The new plant may instead reference the applicable analysis.

C. Minimum Requirements for a Complete Break Spectrum

This paragraph defines the minimum number of breaks required for an acceptable complete spectrum analysis. This complete spectrum analysis is required for each of the lead plants of a given class (BWR2, BWR3, BWR4, BWR5, and BWR6). Each sensitivity study published during the development of the ECCS evaluation model shall be individually justified as remaining applicable, or shall be repeated.

1. Four recirculation line breaks at the worst location (pump suction or discharge), using the LBM, covering the range from the transition break (TB) to the DBA, including C_D coefficients of from 0.6 to 1.0 times the DBA.

2. Five recirculation line breaks, using the SBM, covering the range from the smallest line break to the TB.

3. The following break locations assuming the worst single failure:

   a. largest steamline break

   b. largest feedwater line break

c. largest core spray line break

d. largest recirculation pump discharge or suction break (opposite side of worst location)

D. BWR4 with "Modified" ECCS

Same as Section IIIC.

E. BWR5

Same as Section IIIC.

F. BWR6

Same as Section IIIC.

IV. LOCA PARAMETERS OF INTEREST

A. On each plant and for each break analyzed, the following parameters (versus time unless otherwise noted) should be provided on engineering graph paper of a quality to facilitate calculations.

--Peak clad temperature (ruptured and unruptured node)

--Reactor vessel pressure

--Vessel and downcomer water level (PWR only)

--Water level inside the shroud (BWR only)

--Thermal power

--Containment pressure (PWR only)

B. For the worst break analyzed, the following additional parameters (versus time unless otherwise noted) should be provided on engineering graph paper of a quality to facilitate calculations. The worst single failure and worst-case reactor coolant pump status will have been established utilizing appropriate sensitivity studies.

--Flooding rate (PWR only)

--Core flow (inlet and outlet)

--Core inlet enthalpy (BWR only)

--Heat transfer coefficients

--MAPLHGR versus Exposure (BWR only)

--Reactor coolant temperature (PWR only)

--Mass released to containment (PWR only)

--Energy released to containment (PWR only)

—PCT versus Exposure (BWR only)

—Containment condensing heat transfer coefficient (PWR only)

—Hot spot flow (PWR only)

—Quality (hottest assembly) (PWR only)

—Hot pin internal pressure

—Hot spot pellet average temperature

—Fluid temperature (hottest assembly) (PWR only)

C. A tabulation of peak clad temperature and metal-water reaction (local and core-wide) shall be provided across the break spectrum.

D. Safety Analysis Reports (SARs) filed with the NRC shall identify on each plot the run date, version number, and version date of the computer model utilized for the LOCA analysis. Should differences exist in version number or version date from the most current code listings made available to the NRC staff, then each modification shall be identified with an assessment of impact upon PCT and metal-water reaction (local and core-wide).

E. A tabulation of times at which significant events occur shall be provided on each plant and for each break analyzed. The following events shall be included as a minimum:

—End-of-bypass (PWR only)

—Beginning of core recovery (PWR only)

—Time of rupture

—Jet pumps uncovered (BWR only)

—MCPR (BWR only)

—Time of rated spray (BWR only)

—Can quench (BWR only)

—End-of-blowdown

—Plane of interest uncovery (BWR only)

Possible grouping of plants for the purpose of performing generic as well as individual plant break spectrum analyses.

CURRENT DOCKETED

APPLICATIONS

BABCOCK AND WILCOX

CATEGORY I: 177 FA w/Lowered Loops Arrangement

Re-analysis (Safety Order Plants):

Oconee 1, 2, 3 2568	-- IIA	These plants must resubmit at least 3 breaks. (They will do so by reference to a complete break spectrum reanalysis submitted generically by B&W.)
Three Mile Island 1 2535	-- IIA	
Arkansas Power 1 2563	-- IIA	
Rancho Seco 2772	-- IIA	

New OLs:

Three Mile Island 2 2772	--IIB(2)	Since these plants are the same design as the above plant, they may reference the same reanalysis of the complete spectrum above.
Crystal River 3 2452	--IIB(2)	
Midland 1, 2	--IIB(2)	

New CPs:

None

CATEGORY II: 177 FA w/Raised Loop Arrangement

New OLs:

Davis Besse 1	--IIB	Complete spectrum required.

New CPs

Davis Besse 2, 3	--IIB	Complete spectrum required.

CATEGORY III: 205-FA Plants

New OLs:

None

New CPs:

Bellefonte 1, 2	-- IIB	Complete spectrum required. (Plans are for all to reference a complete spectrum submitted probably on WPPSS.)
Greenwood 2, 3	-- IIB	
WPPSS 1, 4	-- IIB	
Pebble Springs 1, 2	-- IIB	

CATEGORY IV: 145-FA Plants

New OLs:

None

New CPs:

North Anna 3, 4	-- IIB	Complete spectrum required. (One will probably reference the other.)
Surry 3, 4	-- IIB	

GENERAL ELECTRIC

BWR-2 Oyster Creek -- LP* Complete spectrum required. (IIIA)**

Nine Mile Point -- Reference only required. (IIIA)

BWR-3 Quad Cities 2 -- LP* Complete spectrum required. (IIIA)**
2511

Millstone -- IIIA - 3 breaks required
2011
Monticello -- IIIA - 3 breaks required
1670
Dresden 2, 3 -- IIIA
2527
Quad Cities 1 -- IIIA
2511
May reference LP
Pilgrim -- IIIA - 3 breaks required
1998

BWR-4 Without fix Hatch 1 -- LP* Complete spectrum required. (IIIA)**
2436

Peach Bottom 2, 3 -- IIIA
3293
Browns Ferry 1, 2, 3 -- IIIA
3293
Complete spectrum required. One may reference the other.

Cooper -- IIIA
2381
Fitzpatrick -- IIIA
2436
Duane Arnold -- IIIA - 3 breaks required
1658
Hatch 2 -- IIIA
2436
Brunswick 1 -- IIIA
2436
3 breaks required. Hatch 1 may serve as a reference for the others.

Shoreham -- IIIB

Fermi -- IIIB

Newbold -- IIIB

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* Lead Plant

** Original break spectrum not wholly in conformance with 10CFR50, Appendix K.

BWR-4 Wit^ fix Brunswick 2 (Lead Plant) IIIA - Complete spectrum required.**
2436

Vermont Yankee -- IIIA - 3 breaks required (Lead Plant can be referenced, if appropriate)
1593

Browns Ferry* 1, 2, & 3

Peach Bottom* 2, 3

Fitzpatrick*

See preceding page

BWR-5 Lead Plant -- IIIE - Complete spectrum required.

Nine Mile Point 2 -- IIIB

LaSalle 1, 2 -- IIIB

Bailly -- IIIB

Zimmer -- IIIB

Susquehanna 1, 2 -- IIIB

Complete spectrum required. (Lead Plant can be referenced by other BWR-5 plants, if appropriate.)

BWR-6 Lead Plant -- IIIF - Complete spectrum required.

Grand Gulf -- IIIB

Black Fox -- IIIB

Barton 1, 2, 3, 4 -- IIIB

Perry 1, 2 -- IIIB

Clinton 1, 2 -- IIIB

Douglas Point -- IIIB

Hanford 2 -- IIIB

Skagit 1, 2 -- IIIB

Hartsville -- IIIB

Somerset -- IIIB

River Bend Station -- IIIB

Allens Creek -- IIIB

Complete spectrum required. (Lead Plant can be referenced by other BWR-6 plants, if appropriate.)

---

* May or may not have the LPCI fix

** Original break spectrum not wholly in conformance with 10CFR50, Appendix K.

PLANT SPECIF

Oyster Creek	-- IIIA	Complete spectrum required.
Nine Mile Point	-- IIIA	" " "
Limmerick 1, 2	-- IIIB	" " "
Hope Creek	-- IIIB	" " "
Humboldt Bay	-- IIIA	" " "
Dresden 1	-- IIIA	" " "
Big Rock	-- IIIA	" " "

COMBUSTION ENGINEERING

The following list is grouped according to similarities in design. Some of the older, operating plants are fairly unique, as indicated, and don't fall conveniently into any other groups. The list is in approx. chronological order.

1. Palisades (Unique) -- IIA } 3 breaks required
2. Ft. Calhoun (Unique) -- IIA }
*3. Maine Yankee (Unique) -- IIA }
4. 2560 MWt Series
   a. Calvert Cliffs Units 1 & 2 -- IIA - 3 breaks required
   b. Millstone Unit 2 -- IIB } Complete spectrum required. (One may reference the other.)
   c. St. Lucie 1 -- IIB }
   **d. St. Lucie 2 -- IIB · Complete spectrum required
5. 3400 MWt Series (3410 MWt 217 Fuel Assemblies)
   a. Pilgrim 2 (3470 Mwt) -- IIB } Complete spectrum required. (One may reference the other.)
   b. Forked River 1 -- IIB }
   c. San Onofre 2 & 3 -- IIB }
   d. Waterford 3 -- IIB }
6. Arkansas Class (2900 MWt 177 Fuel Assemblies)
   a. Russelville 1 -- IIB } Complete spectrum required. (One may reference the other.)
   b. Blue Hills 1 -- IIB }

*Maine Yankee is unique in that it has 3 steam generators, 3 hot legs and 3 cold legs. All other CE plants have 2 steam generators, 2 hot legs and 4 cold legs.

**All plants shown above listed before St. Lucie 2 are of the 14x14 fuel design. All plants after, and including, St. Lucie 2 are 16x16.

7. System 80 Class-(CESSAR) -- IIB Complete spectrum required

These plants have not all been named yet. The utility and approx. number of plants expected are as follows:

a. Duke (6) b. WUPPS (1) c. Arizona Power and Light (2) d. TVA (2)	May reference complete spectrum, if applicable.

Westinghouse

Operating Reactors (Safety Order Plants)*

2-loop	3-loop	4-loop
Ginna	Surry 1/2	Yankee Rowe
Kewaunee	Turkey Pt. 3/4	IP2
Pt. Beach 1/2	H. B. Robinson 2	D. C. Cook 1
Prairie Island 1/2		Zion 1/2

Operating License**

2-loop	3-loop	4-loop
— — — —	Beaver Valley 1	- Trojan*
	Farley 1/2	- Salem 1/2*
	North Anna 1/2	- Diablo Canyon 1/2*
		IP-3
		D. C. Cook 2
		McGuire 1/2
		Sequoyah 1/2

* 3 breaks required (IIA). One plant may reference another if applicable.

** Complete spectrum required. One plant may reference another if applicable (see paragraph IIB).

Construction Permit **

2-loop	3-loop	4-loop
North Coast	Sharon Harris 1/4	Byron/Braidwood 1/2
	Koshkonong 1/2	Catawba 1/2
	Summer 1	Floating Nuclear 1/8
	Beaver Valley 2	Jamesport 1/2
	Wisconsin Utilities	Seabrook 1/2
		SNUPPS 1-5
		South Texas 1/2
		Comanche Peak 1/2
		Watts Bar 1/2
		Millstone 3
		Vogtle 1/2

** Complete spectrum required. One plant may reference another if applicable (see paragraph IIB).

BRANCH TECHNICAL POSITION EICSB 18

APPLICATION OF THE SINGLE FAILURE CRITERION TO MANUALLY-CONTROLLED ELECTRICALLY-OPERATED VALVES

A. BACKGROUND

Where a single failure in an electrical system can result in loss of capability to perform a safety function, the effect on plant safety must be evaluated. This is necessary regardless of whether the loss of safety function is caused by a component failing to perform a requisite mechanical motion, or by a component performing an undesirable mechanical motion.

This position establishes the acceptability of disconnecting power to electrical components of a fluid system as one means of designing against a single failure that might cause an undesirable component action. These provisions are based on the assumption that the component is then equivalent to a similar component that is not designed for electrical operation, e.g., a valve that can be opened or closed only by direct manual operation of the valve. They are also based on the assumption that no single failure can both restore power to the electrical system and cause mechanical motion of the components served by the electrical system. The validity of these assumptions should be verified when applying this position

B. BRANCH TECHNICAL POSITION

1. Failures in both the "fail to function" sense and the "undesirable function" sense of components in electrical systems of valves and other fluid system components should be considered in designing against a single failure, even though the valve or other fluid system component may not be called upon to function in a given safety operational sequence.

2. Where it is determined that failure of an electrical system component can cause undesired mechanical motion of a valve or other fluid system component and this motion results in loss of the system safety function, it is acceptable, in lieu of design changes that also may be acceptable, to disconnect power to the electric systems of the valve or other fluid system component. The plant technical specifications should inolude a list of all electrically-operated valves, and the required positions of these valves, to which the requirement for removal of electric power is applied in order to satisfy the single failure criterion.

3. Electrically-operated valves that are classified as "active" valves, i.e., are required to open or close in various safety system operational sequences, but are manually-controlled, should be operated from the main control room. Such valves may not be included among those valves from which power is removed in order to meet the single failure criterion unless: (a) electrical power can be restored to the valves from the main control room, (b) valve operation is not necessary for at least [illegible] minutes following occurrence of the event requiring such operation, and (c) it is demonstrated

that there is reasonable assurance that all necessary operator actions will be performed within the time shown to be adequate by the analysis. The plant technical specifications should include a list of the required positions of manually-controlled, electrically-operated valves and should identify those valves to which the requirement for removal of electric power is applied in order to satisfy the single failure criterion.

4. When the single failure criterion is satisfied by removal of electrical power from valves described in (2) and (3), above, these valves should have redundant position indication in the main control room and the position indication system should, itself, meet the single failure criterion.

5. The phrase "electrically-operated valves" includes both valves operated directly by an electrical device (e.g., a motor-operated valve or a solenoid-operated valve) and those valves operated indirectly by an electrical device (e.g., an air-operated valve whose air supply is controlled by an electrical solenoid valve).

C. REFERENCES

1. Memorandum to R. C. DeYoung and V. A. Moore from V. Stello, October 1, 1973.

BRANCH TECHNICAL POSITION CSB 6-1

MINIMUM CONTAINMENT PRESSURE MODEL
FOR PWR ECCS PERFORMANCE EVALUATION

A. BACKGROUND

Paragraph I.D.2 of Appendix K to 10 CFR Part 50 (Ref. 1) requires that the containment pressure used to evaluate the performance capability of a pressurized water reactor (PWR) emergency core cooling system (ECCS) not exceed a pressure calculated conservatively for that purpose. It further requires that the calculation include the effects of operation of all installed pressure - ducing systems and processes. Therefore, the following branch technical position has been developed to provide guidance in the performance of minimum containment pressure analysis. The approach described below applies only to the ECCS-related containment pressure evaluation and not to the containment functional capability evaluation for postulated design basis accidents.

B. BRANCH TECHNICAL POSITION

1. Input Information for Model

a. Initial Containment Internal Conditions

The minimum containment gas temperature, minimum containment pressure, and maximum humidity that may be encountered under limiting normal operating conditions should be used.

b. Initial Outside Containment Ambient Conditions

A reasonably low ambient temperature external to the containment should be used.

c. Containment Volume

The maximum net free containment volume should be used. This maximum free volume should be determined from the gross containment volume minus the volumes of internal structures such as walls and floors, structural steel, major equipment, and piping. The individual volume calculations should reflect the uncertainty in the component volumes.

2. Active Heat Sinks

a. Spray and Fan Cooling Systems

The operation of all engineered safety feature containment heat removal systems operating at maximum heat removal capacity; i.e., with all containment spray trains operating at maximum flow conditions and all emergency fan cooler units operating, should be assumed. In addition, the minimum temperature of the stored water for the spray cooling system and the cooling water supplied to the fan coolers, based on technical specification limits, should be assumed.

Deviations from the foregoing will be accepted if it can be shown that the worst conditi regarding a single active failure, stored water temperature, and cooling water temperature have been selected from the standpoint of the overall ECCS model.

b. Containment Steam Mixing With Spilled ECCS Water

The spillage of subcooled ECCS water into the containment provides an additional heat sink as the subcooled ECCS water mixes with the steam in the containment. The effect of the steam-water mixing should be considered in the containment pressure calculations.

c. Containment Steam Mixing With Water from Ice Melt

The water resulting from ice melting in an ice condenser containment provides an additional heat sink as the subcooled water mixes with the steam while draining from the ice condenser into the lower containment volume. The effect of the steam-water mixing should be considered in the containment pressure calculations.

3. Passive Heat Sinks

a. Identification

The passive heat sinks that should be included in the containment evaluation model should be established by identifying those structures and components within the containment that could influence the pressure response. The kinds of structures and components that should be included are listed in Table 1.

Data on passive heat sinks have been compiled from previous reviews and have been used as a basis for the simplified model outlined below. This model is acceptable for minimum containment pressure analyses for construction permit applications, and until such time (i.e., at the operating license review) that a complete identification of available heat sinks can be made. This simplified approach has also been followed for operating plants by licensees complying with Section 50.46 (a)(2) of 10 CFR Part 50. For such cases, and for construction permit reviews, where a detailed listing of heat sinks within the containment often cannot be provided, the following procedure may be used to model the passive heat sinks within the containment:

(1) Use the surface area and thickness of the primary containment steel shell or steel liner and associated anchors and concrete, as appropriate.

(2) Estimate the exposed surface area of other steel heat sinks in accordance with Figure 1 and assume an average thickness of 3/8 inch.

(3) Model the internal concrete structures as a slab with a thickness of 1 foot and exposed surface of 160,000 ft^2.

The heat sink thermophysical properties that would be acceptable are shown in Table 2.

At the operating license stage, applicants should provide a detailed list of pass' heat sinks, with appropriate dimensio nd properties.

b. Heat Transfer Coefficients

The following conservative condensing heat transfer coefficients for heat transfer to the exposed passive heat sinks during the blowdown and post-blowdown phases of the loss-of-coolant accident should be used (See Figure 2):

(1) During the blowdown phase, assume a linear increase in the condensing heat transfer coefficient from $h_{initial} = 8$ Btu/hr-ft^2-°F, at t = 0, to a peak value four times greater than the maximum calculated condensing heat transfer coefficient at the end of blowdown, using the Tagami correlation (Ref. 2),

$$h_{max} = 72.5 \left[\frac{Q}{Vt_p}\right]^{0.62}$$

where h_{max} = maximum heat transfer coefficient, Btu/hr-ft^2-°F
Q = primary coolant energy, Btu
V = net free containment volume, ft^3
t_p = time interval to end of blowdown, sec.

(2) During the long-term post-blowdown phase of the accident, characterized by low turbulence in the containment atmosphere, assume condensing heat transfer coefficients 1.2 times greater than those predicted by the Uchida data (Ref. 3) and given in Table 3.

(3) During the transition phase of the accident, between the end of blowdown and the long-term post-blowdown phase, a reasonably conservative exponential transition in the condensing heat transfer coefficient should be assumed (see Figure 2).

The calculated condensing heat transfer coefficients based on the above method should be applied to all exposed passive heat sinks, both metal and concrete, and for both painted and unpainted surfaces.

Heat transfer between adjoining materials in passive heat sinks should be based on the assumption of no resistance to heat flow at the material interfaces. An example of this is the containment liner to concrete interface.

C. REFERENCES

1. 10 CFR §50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors," and 10 CFR Part 50, Appendix K, "ECCS Evaluation Models."

2. T. Tagami, "Interim Report on Safety Assessment and Facilities Establishment Project in Japan for Period Ending June 1965 (No. 1)," prepared for the National Reactor Testing Station, February 28, 1966 (unpublished work).

3. H. Uchida, A. Oyama, and Y. Toga, "Evaluation of Post-Incident Cooling Systems of Light-Water Power [illegible]tors," Proc. Third International Conf[illegible]nce on the Peaceful Uses of Atomic Energy, Volume 13, Session 3.9, United Nations, Geneva (1964).

POOR ORIGINAL

TABLE 1

IDENTIFICATION OF CONTAINMENT HEAT SINKS

1. Containment Building (e.g., liner plate and external concrete walls, floor, and sump, and liner anchors).

2. Containment Internal Structures (e.g., internal separation walls and floors, refueling pool and fuel transfer pit walls, and shielding walls).

3. Supports (e.g., reactor vessel, steam generator, pumps, tanks, major components, pipe supports, and storage racks).

4. Uninsulated Systems and Components (e.g., cold water systems, heating, ventilation, and air conditioning systems, pumps, motors, fan coolers, recombiners, and tanks).

5. Miscellaneous Equipment (e.g., ladders, gratings, electrical cable trays, and cranes).

TABLE 2

HEAT SINK THERMOPHYSICAL PROPERTIES

Material	Density lb/ft^3	Specific Heat Btu/lb-°F	Thermal Conductivity Btu/hr-ft-°F
Concrete	145	0.156	0.92
Steel	490	0.12	27.0

TABLE 3

UCHIDA HEAT TRANSFER COEFFICIENTS

Mass Ratio (lb air/lb steam)	Heat Transfer Coefficient (Btu/hr-ft^2-°F)	Mass Ratio (lb air/lb steam)	Heat Transfer Coefficient (Btu/hr-ft^2-°F)
50	2	3	29
20	8	2.3	37
18	9	1.8	46
14	10	1.3	63
10	14	0.8	98
7	17	0.5	140
5	21	0.1	280
4	24		

Figure 1
Area of Steel Heat Sinks Inside Containment

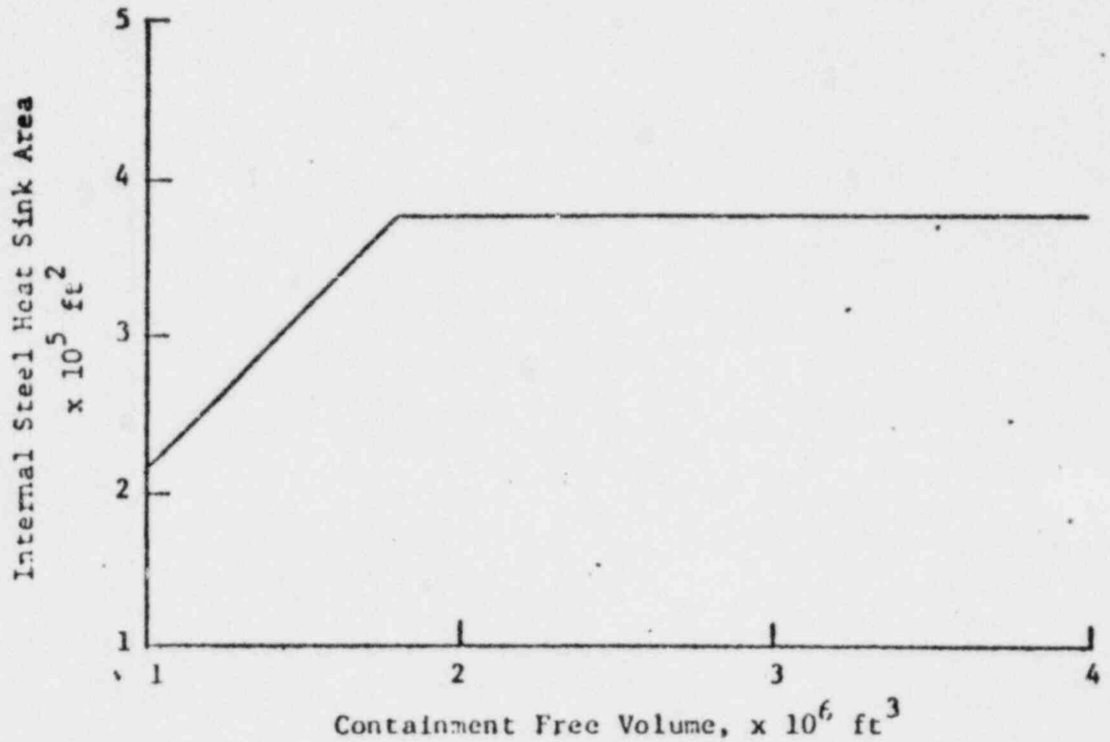

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Figure 2

Condensing Heat Transfer Coefficients for Static Heat Sinks