NRC PDR

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

February 10, 1983

Docket No. 50-29

Mr. Jeremiah J. Ray, Chairman
Advisory Committee on Reactor Safeguards
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Ray:

Enclosed are three sets of the Safety Evaluation Reports referenced in Appendix E of NUREG-0825, the Integrated Plant Safety Assessment Report for Yankee Nuclear Power Station. These references are presented for the use of the ACRS and its staff in the review of NUREG-0825. Complete sets of the Safety Evaluation Reports are also being placed in the NRC Public Document Room and the Local Public Document Room for ease of reference by the general public.

Sincerely,

Dennis M. Crutchfield

Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing

Enclosure:
As stated

XA Copy Has Been Sent to PDR

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DESIGNATED ORIGINAL

Certified By [signature] 2/17/83

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

September 1, 1982

Docket No. 50-29
LS05-82-09-006

Mr. James A. Kay
Senior Engineer - Licensing
Yankee Atomic Electric Company
1671 Worcester Road
Framingham, Massachusetts 01701

Dear Mr. Kay:

SUBJECT: SEP TOPIC III-7.B, DESIGN CODES, DESIGN CRITERIA AND LOAD COMBINATIONS - YANKEE NUCLEAR POWER STATION

Enclosed is a copy of our draft evaluation of SEP Topic III-7.B. The evaluation identifies areas of codes where changes have occurred to decrease safety margins. It also identifies loads applicable to some or all of the structures at Yankee which have increased in magnitude. After reviewing structural drawings of your facility, we concluded that some code changes of concern were not applicable to your facility because the structural elements to which these code changes are referring were not found in the structural drawings of Yankee which we reviewed. These changes are identified in Appendix A of the enclosure. You are to review how these areas of the codes were applied in the design of Yankee and the ability of structures to resist increased loads and assess the current safety margins.

You are requested to examine the facts upon which the staff has based its evaluation and respond by confirming that the facts are correct or by identifing errors and supplying the corrected information. We encourage you to supply any other material that might affect the staff's evaluation of this topic or be significant in the Integrated Assessment of your facility.

Sincerely,

Ralph Caruso

Ralph Caruso, Project Manager
Operating Reactors Branch No. 5
Division of Licensing

Enclosure:
As stated

cc w/enclosure:
See next page

29 p 820901

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Mr. James A. Kay

Yankee
Docket No. 50-29
Revised 3/30/82

cc
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SYSTEMATIC EVALUATION PROGRAM

TOPIC III-7.B

YANKEE NUCLEAR POWER STATION

Topic: III-7.B, Design Codes, Design Criteria and Loading Combinations

I. INTRODUCTION

SEP plants were generally designed and constructed during the time span from the late 1950's to late 1960's. They were designed according to criteria and codes which differ from those accepted by the NRC for new plants.

The purpose of this topic is to assess the safety margins existing in Category I structures as a result of changes in design codes and criteria.

II. REVIEW GUIDELINES

The current licensing criteria which governs the safety issue in this topic is 10 CFR 50, Appendix A, GDC 1, 2, and 4 as interpreted by Standard Review Plan 3.8.

III. RELATED SAFETY TOPICS

The following SEP topics are related to III-7.B:

1. III-2, Wind and Tornado Loadings
2. III-3.A, Effects of High Water and Level on Structures
3. III-4.A, Tornado Missiles
4. III-5.A, Effects of High Energy Pipe Breaks Inside Containment
5. III-5.B, Effects of High Energy Pipe Breaks Outside Containment
6. III-6, Seismic Design Considerations
7. VI-2.D, Mass and Energy Release for Postulated Pipe Break Inside Containment
8. VI-3, Containment Pressure and Heat Removal Capability

IV. EVALUATION

The evaluation is based on a Technical Evaluation Report (TER) prepared by the Franklin Research Center (FRC) in conjunction with the NRC staff through contract. The report is entitled, "Design Codes, Design Criteria and Loading Combinations" and is attached to this Safety Evaluation Report as Enclosure (1).

We have compared structural design codes employed in the design of Category I structures at Yankee to present codes. This was done through generic code versus code comparison without investigating specifically how the original code was applied to the Yankee design; however, after reviewing drawings of structures at Yankee, we concluded that certain portions of the codes were not applicable to Yankee because the types of structures to which the codes are referring were non-existent at Yankee. We have compared the loads and loading combinations employed in the design of Yankee as described in the Yankee FSAR to those required today.

A result of these comparisons is that a number of code changes could potentially impact significantly margins of safety (denoted by scale A and Ax in Enclosure 1). This can be attributed to several factors such as:

1) New codes have imposed stricter limitations than old,

2) New codes have included sections governing design of certain types of structures which were not included in the older codes,

3) Design loads required today were not included in the plant design; and

4) Certain load combinations judged to be significant were not included in plant design.

In Enclosure (1), some items have been judged to potentially impact margins of safety regarding the containment as a result of comparing ASME BPV Code Section VIII, 1956 to ASME BPV Code Section III, Subsection NE, 1980. These items are discussed in Section 11 of the report.

The code changes of concern from Enclosure (1) are: (See next page)

Structural Elements to be Examined	Code Change Affecting These Elements New Code	Old Code
Beams/Columns	AISC 1980	AISC 1953
Hollow circular sections subject to bending	1.5.1.4.1 Subpara 7	--*
Composite Construction	AISC 1980	AISC 1953
1. Shear connectors in composite beams	1.11.4	13
2. Composite beams or girders with formed steel deck	1.11.5	--
3. Width of concrete flange - limitations	1.11.1	13(a)
Compression Elements	AISC 1980	AISC 1953
1. With width-to-thickness ratio higher than specified in 1.9.1.2	1.9.1.2 and Appendix C	18(b)
2. Hollow circular sections subject to axial compression	1.9.2.3 and Appendix C	--
3. Members where sideway is not prevented	1.8.3	16
Tension Members	AISC 1980	AISC 1953
1. When load is transmitted by bolts or rivets	1.14.2.2	--
2. Built up members	1.18.3	28(b)
Connections	AISC 1980	AISC 1953
1. Beam ends with top flange coped, if subject to shear	1.5.1.2.2	--

*Double dash (--) indicates that older code had no provisions.

Structural Elements to be Examined	Code Change Affecting These Elements New Code	Old Code
2. Connections carrying moment or restrained member connection	1.15.5.2 1.15.5.3 1.15.5.4	--
Members Designed to Operate in an Inelastic Regime	AISC 1980	AISC 1953
Spacing of lateral bracing	2.9	--
Rolled Sections and Built up Members	AISC 1980 --	AISC 1953 15(a)(3)
Partial length cover plates	1.10.4	26(d)
Members Subject to Axial and Bending Stresses	AISC 1980 1.6	AISC 1953 12(a)
Web Plate Girders	AISC 1980	AISC 1953
1. Subject to shear and tension stresses	1.10.7	--
2. Stiffeners	1.10.10.2	26
Partial Penetration Weld Effective throat thickness	1.14.6.1	15(f)
Short Brackets and Corbels having a shear span-to-depth ratio of unity or less	ACI 349-76 11.13	ACI 318-56 --
Shear Walls used as a primary load-carrying member	ACI 349-76 11.16	ACI 318-56 --
Precast Concrete Structural Elements, where shear is not a measure of diagonal tension	ACI 349-76 11.15	ACI 318-56 --
Concrete Regions Subject to High Temperatures	ACI 349-76	ACI 318-56
Time-dependent and position-dependent temperature variations	Appendix A	--

Structural Elements to be Examined	Code Change Affecting These Elements New Code	Old Code
All Structural Elements	ACI 349-76	ACI 318-56
1. Ultimate bond strength	Chapter 12	--
2. Allowable bond stress	--	Table 305(a)
Columns with Spliced Reinforcement	ACI 349-76	ACI 318-56
subject to stress reversals; f_y in compression to 1/2 f_y in tension	7.10.3	--
Steel Embedments used to transmit load to concrete	ACI 349-76 Appendix B	ACI 318-56 --
Element Subject to Impulsive and Impactive Loads whose failure must be precluded	ACI 349-76 Appendix C	ACI 318-56 --
Composite Construction	ACI 349-76 Chapter 17	ACI 318-56 --
Containment Vessels		
1. Plates, if understrength	ASME Sec. III, 1980 NE-3112.4	ASME Sec. VIII, 1956 UG-5(b)
2. Containment vessels of materials no longer listed as code acceptable	ASME Sec. III, 1980 NE-3112.4	ASME Sec. VIII, 1956 UG-23
3. Containment vessels designed by formula and subject to substantial thermal or mechanical loads	ASME Sec. III, 1980 NE-3131	ASME Sec. VIII, 1956 Various paragraphs
4. Stiffening rings for cylindrical shells subject to buckling loads	ASME Sec. III, 1980 NE-3133.5(a)	ASME Sec. VIII, 1956 UG-29

Structural Elements to be Examined	Code Change Affecting These Elements: New Code	Old Code
5. Stiffening rings of material different than shell material	ASME Sec. III, 1980 NE-3133.5(b)	ASME Sec. VIII, 1956 --
6. Vessels with Quick Actuating Closures	ASME Sec. III, 1980 NE-3327.1	ASME Sec. VIII, 1956 Footnote to UG-35
Shell Openings and Attachments		
1. Unstayed flat heads and covers	ASME Sec. III, 1980 NE-3325 Figs. (c) and (m)	ASME Sec. VIII, 1956 UG-34(d) Figs. (b) and (a)
2. Openings and reinforcements; subject to cyclic loads	ASME Sec. III, 1980 NE-3331(b)	ASME Sec. VIII, 1956 --
3. Reinforcement for openings	ASME Sec. III, 1980 NE-3334.1, NE-3334.2	ASME Sec. VIII, 1956 UG-40
4. Bellows and bellows expansion joints	ASME Sec. III, 1980 NE-3365	ASME Sec. VIII, 1956 --
Roofs	---	--

Extreme environmental snow loads are provided by SEP Topic II-2.A. NRC Regulatory Guide 1.102 (Position 3) provides guidance to preclude adverse consequences from ponding or parapet roofs. Failure of roofs not designed for such circumstances could generate impulsive loadings and water damage, possibly extending to Seismic Category I components of all floor levels.

1. Not shown in tabular summary of code change impacts.

Section 10 of Enclosure (1) addresses load and load combination changes which occurred as a result of code changes and identifies specific plant structures for which various load combinations may be significant. Based upon a lack of detailed information on the stress results for loads and load combinations used during design of structures at Yankee, these loads and load combinations may be potentially significant.

After reviewing your submittals for SEP Topic III-6, the licensee should identify the maximum stresses at critical locations in the vapor container and supports considering appropriate load combinations.

V. CONCLUSION

We conclude that after comparing design codes, criteria, loads and load combinations, a number of changes have occurred which could potentially impact margins of safety. These changes are identified above. These differences between plant design and current licensing criteria should be resolved as follows:

1) Review Seismic Category I Structures at Yankee to determine if any of the structural elements for which a concern exists are a part of the facility design of Yankee. For those that are, assess the impact of the code changes on margins of safety on a plant specific basis, and

2) Examine on a sampling basis the margins of safety of Seismic Category I Structures for loads and load combinations not covered by another SEP topic and denoted by Ax in Enclosure (1). (The load tables should be reviewed to assure their technical accuracy concerning applicability of the loads for each of the structures and their significance. The Category I Structures considered should be reviewed to insure completeness.)

The licensee should review the vapor container's ability to withstand accident and seismic loads when analyzed to current criteria loading combinations.

TECHNICAL EVALUATION REPORT

DESIGN CODES, DESIGN CRITERIA, AND LOADING COMBINATIONS (SEP, III-7B)

YANKEE ATOMIC ELECTRIC COMPANY
YANKEE ROWE NUCLEAR POWER STATION

NRC DOCKET NO. 50-029

NRC TAC NO. 41494

NRC CONTRACT NO. NRC-03-79-118

FRC PROJECT C5257

FRC ASSIGNMENT 11

FRC TASK 316

Prepared by

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Lead NRC Engineer: D. Persinko

August 30, 1982

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TECHNICAL EVALUATION REPORT

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August 30, 1982

Prepared by: Thomas C. Stilwell — Principal Author — Date: 8-30-82

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CONTENTS (Cont.)

FOREWORD

This Technical Evaluation Report was prepared by Franklin Research Center under a contract with the U.S. Nuclear Regulatory Commission (Office of Nuclear Reactor Regulation, Division of Operating Reactors) for technical assistance in support of NRC operating reactor licensing actions. The technical evaluation was conducted in accordance with criteria established by the NRC.

Principal contributors to the technical preparation of this report were T. Stilwell, M. Darwish, W. A. Segraves, and S. J. Triolo of the Franklin Research Center.

Dr. E. W. Wallo, Chairman of the Civil Engineering Department, Villanova University, and Dr. R. Koliner, Professor of Civil Engineering, Villanova University, provided assistance both as contributing authors and in an advisory capacity as consultants under subcontract with the Franklin Research Center.

The report also incorporates the suggestions, guidance, and supportive efforts provided by Mr. D. Persinko, the NRC Lead Engineer for this task.

1. INTRODUCTION

For the Seismic Category I buildings and structures at the Yankee Rowe Nuclear Power Station, this report provides a comparison of the structural design codes and loading criteria used in the actual plant design against the corresponding codes and criteria currently used for licensing of new plants.

The objective of the code comparison review is to identify deviations in design criteria from current criteria, and to assess the effect of these deviations on margins of safety, as they were originally perceived and as they would be perceived today.

The work was conducted as part of the Nuclear Regulatory Commission's (NRC) Systematic Evaluation Program (SEP) and provides technical assistance for Topic III-7.B, "Design Codes, Design Criteria, and Load Combinations." The report was prepared at the Franklin Research Center under NRC Contract No. NRC-03-79-118.

2. BACKGROUND

With the development of nuclear power, provisions addressing facilities for nuclear applications were progressively introduced into the codes and standards to which plant building and structures are designed. Because of this evolutionary development, older nuclear power plants conform to a number of different versions of these codes, some of which have since undergone considerable revision.

There has likewise been a corresponding development of other licensing criteria, resulting in similar non-uniformity in many of the requirements to which plants have been licensed. With this in mind, the NRC undertook an extensive program to evaluate the safety of 11 older plants (and eventually all plants) to a common set of criteria. The program, entitled the Systematic Evaluation Program (SEP), employs current licensing criteria (as defined by NRC's Standard Review Plan) as the common basis for these evaluations.

To make the necessary determinations, the NRC is investigating, under the SEP, 137 topics spanning a broad spectrum of safety-related issues. The work reported herein constitutes the results of part* of the investigation of one of these topics, Topic III-7.B, "Design Codes, Design Criteria, and Load Combinations."

This topic is charged with the comparison of structural design criteria in effect in the late 1950's to the late 1960's (when the SEP plants were constructed) with those in effect today. Other SEP topics also address other aspects of the integrity of plant structures. All these structurally oriented tasks, taken together, will be used to assess the structural adequacy of the SEP plants with regard to current requirements. The determinations with respect to structural safety will then be integrated into an overall SEP evaluation encompassing the entire spectrum of safety-related topics.

*The report addresses only the Yankee Rowe plant.

3. REVIEW OBJECTIVES

The broad objective of the NRC's Systematic Evaluation Program (SEP) is to reassess the safety of 11 older nuclear power plants in accordance with the intent of the requirements governing the licensing of current plants, and to provide assurance, possibly involving backfitting, that operation of these plants conforms to the general level of safety required of modern plants.

Task III-7.B of the SEP effort seeks to compare actual and current structural design criteria for the major civil engineering structures at each SEP plant site, i.e., those important to shutdown, containment, or both, and therefore designated Seismic Category I structures. The broad safety objective of SEP Task III-7.B is (when integrated with several other interfacing SEP topics) to assess the capability of all Seismic Category I structures to withstand all design conditions stipulated by the NRC, at least to a degree sufficient to assure that the nuclear power plant can be safely shut down under all circumstances.

The objective of the present effort under Task III-7.B is to provide, through code comparisons, a rational basis for making the required technical assessments, and a tool which will assist in the structural review.

Finally, the objective of this report is to present the results of Task III-7.B as they relate to the Yankee Rowe Nuclear Power Station.

4. SCOPE

In general, the scope of work requires comparison of the provisions of the structural codes and standards used for the design of SEP plant Seismic Category I civil engineering structures* against the corresponding provisions governing current licensing practice. The review includes the containment and all Category I structures within and exterior to it. Explicit among the criteria to be reviewed are loads and loading combinations postulated for these structures.

The review scope consists of the following specific tasks:

1. Identify current design requirements, based on a review of NRC Regulations; 10CFR50.55a, "Codes and Standards"; and the NRC Standard Review Plan (SRP).

2. Review the structural design codes, design criteria, design and analysis procedures, and load combinations (including combinations involving seismic loads) used in the design of all Seismic Category I structures as defined in the Final Safety Analysis Report (FSAR) for each SEP plant.

3. Based upon the plant-specific design codes and standards identified in Task 2 and current licensing codes and standards from Task 1, identify plant-specific deviations from current licensing criteria for design codes and criteria.

4. Assess the significance of the identified deviations, performing (where necessary) comparative analyses to quantify significant deviations. Such analyses may be made on typical elements (beams, columns, frames, and the like) and should be explored over a range of parameters representative of plant structures.

5. Prepare a Technical Evaluation Report for each SEP plant including:

   a. comparisons of plant design codes and criteria to those currently accepted for licensing

   b. assessment of the significance of the deviations

*In general, these are the structures normally examined in licensing reviews under Section 3.8 of the SRP (but note the list at the end of this section of structures specifically excluded from the scope of this review).

c. results of any comparative stress analyses performed in order to assess the significance of the code changes on safety margins

d. overall evaluation of the acceptability of structural codes used at each SEP plant.

A number of SEP topics examine aspects of the integrity of the structures composing SEP facilities. Several of these interface with the Task III-7.B effort as shown below:

Topic	Designation
III-1	Classification of Structures, Components, Equipment, and Systems (Seismic and Quality)
III-2	Wind and Tornado Loading
III-3.A	Effects of High Water Level on Structures
III-4	Missile Generation and Protection
III-5	Evaluation of Pipe Breaks
III-6	Seismic Design Considerations
III-7.D	Structural Integrity Tests
VI-2	Mass and Energy Release for Postulated Pipe Break.

Because they are covered either elsewhere within the SEP review or within other NRC programs, the following matters are explicitly excluded from the scope of this review:

Mark I torus shell, supports, vents, local region of drywell at vent penetrations	Reviewed in Generic Task A-7.
Reactor pressure vessel supports, steam generator supports, pump supports	Reviewed in Generic Task A-2, A-12.
Equipment supports in SRP 3.8.3	Reviewed generically in Topic III-6, Generic Task A-12.

Other component supports (steel and concrete)	Specific supports have been analyzed in detail in Topic III-6. (Component supports may be included later if items of concern applicable to component supports are found as a result of reviewing the structural codes.)
Testing of containment	Reviewed in Topic III-7.D.
Inservice inspection; quality control/assurance	Should be considered in the review only to the extent that it affects design criteria and design allowables. Aspects of inservice inspection are being reviewed in Topics III-7.A and III-3.C
Determination of structures that should be classified Seismic Category I	Not within scope.
Shield walls and subcompartments inside containment	Reviewed in Generic Task A-2.
Masonry walls	Reviewed generically in IE Bulletin 80-11.
Seismic analysis	Being reviewed as an independent SEP Topic.

5. MARGINS OF SAFETY

There are several bases upon which margins of safety* may be defined and discussed.

The most often used is the margin of safety based on yield strength. This is a particularly useful concept when discussing the behavior of steels, and became ingrained into the engineering vocabulary at the time when steel was the principal metal of engineering structures. In this usage, the margin of safety reflects the reserve capacity of a structure to withstand extra loading without experiencing an incipient permanent change of shape anywhere throughout the structure. Simultaneously, it reflects the reserve load carrying capacity existing before the structure is brought to the limit for which an engineer could be certain the computations (based on elastic behavior of the metal) applied.

This is the conventional use of the term and the meaning which engineers take as intended, unless the term is further qualified to show something else is meant. Thus, if a structure is stated to have a margin of safety of 1.0 under a given set of loads, then it will be generally understood that every load on the structure may be simultaneously doubled without encountering (anywhere) inelastic stresses or deflections. On the other hand, if (under load) a structure has no margin of safety, any increment to any load will cause the structure to experience, in a least one (and possibly more than one) location, some permanent distortion (however small) of its original shape.

Because the yield strengths of common structural steels are generally well below their ultimate strengths, the engineer knows that in most (but not all) cases, the structure possesses substantial reserve capacity--beyond his computed margin--to carry additional load.

There are other useful ways, however, to speak of safety margins and these (not the conventional one) are particularly relevant to the aims of the systematic evaluation program.

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*Factors of safety (FS) are related to margins of safety (MS) through the relation, MS = FS - 1.

One may speak of margins of safety with respect to code allowable limits. This margin reflects the reserve capacity of a structure to withstand extra loading while still conforming to all criteria governing its design.

One may also speak (if it is made clear in advance that this is the intended meaning) of margins of safety against actual failure. Both steel and concrete structures exhibit much higher "margins of safety" on this second basis than is shown by computation of margins of safety based on code allowables.

These latter concepts of "margin of safety" are very significant to the SEP review. Indeed the basic review concept, at least as it relates to structural integrity, cannot be easily defined in any quantitative manner without considering both. The SEP review concept is predicated on the assumption that it is unrealistic to expect that plants which were built to, and were in compliance with, older codes will still conform to current criteria in all respects. The SEP review seeks to assess whether or not plants meet the "intent" of current licensing criteria as defined by the Standard Review Plan (SRP). The objective is not to require that older plants be brought into conformance with all SRP requirements to the letter, but rather to assess whether or not their design is sufficient to provide the general level of safety that current licensing requirements assure.

With respect to aspects of the SEP program that involve the integrity of structures, the SEP review concept can be rephrased in a somewhat more quantitative fashion in terms of these two "margins of safety." Thus, it is not expected or demanded that all structures show positive margins of safety based upon code allowables in meeting all current SRP requirements; but it is demanded that margins of safety based upon ultimate strength are noc only positive, but ample. In fact, the critical judgments to be made (for SEP plants) are:

1. to what extent may current code margins be infringed upon.

2. what minimum margin of safety based on ultimate strength must be assured.

The choice of method for :opic III-7.B review can be discussed in terms of these two key considerations.

6. CHOICE OF REVIEW APPROACH

The approach taken in the review process depends on which key questions (of Section 5) one chooses to emphasize and address first.

One could give primary consideration to the second. If this approach is chosen, one first sets up a minimum margin of safety (based on failure) that will be acceptable for SEP plants. This margin is to be computed in accordance with current criteria. Then one investigates structures designed in accordance with earlier code provisions, and to different loading combinations, to see if they meet the chosen SEP margin when challenged by current loading combinations and evaluated to current criteria. This approach gives the appearance of being efficient. The review proceeds from the general (the chosen minimum margin of safety) to the particular (the ability of a previously designed structure to meet the chosen margin). Moreover, issues are immediately resolved on a "go; no-go" basis. The initial step in this approach is not easy, nor are the necessary evaluations. One is dealing with highly loaded structures in regions where materials behave inelastically. Rulemaking in such areas is sure to be difficult, and likely to be highly controversial.

The alternative approach is taken in this review. It proceeds from the particular to the general, and places initial emphasis upon seeking to answer (for SEP plants) questions as to what, how many, and of what magnitude are the infringements on current criteria. No new rulemaking is involved (at least at the outset). All initial assessments are based on existing criteria.

Current and older codes are compared paragraph-by-paragraph to see the effects that code changes may have on the load carrying ability of individual elements (beams, columns, frames, and the like). It should be noted that this process, although involving judgments, is basically fact-finding -- not decisionmaking.

This kind of review is painstaking, and there is no assurance in advance that it in itself will be decisive. It may turn out, after examination of the facts, that designs predicated upon the older criteria infringe upon current design allowables in many cases and to extensive depths. If so, such information will certainly be of value to the final safety assessment, but many unresolved questions will remain.

On the other hand, it may turn out that infringements upon current criteria are infrequent and not of great magnitude. If this is the case, many issues will have been resolved, and questions of structural integrity will be sharply focused upon a few remaining key issues.

7. METHOD

A brief description of the approach used to carry out SEP Topic III-7.B follows. For discussion of the work, it is convenient to divide the approach into six areas:

1. information retrieval and assembly
2. appraisal of information content
3. code comparison reviews
4. code change impact assessment
5. plant-specific review of the relevancy of code change impacts
6. summarizing plant status vis-a-vis design criteria changes.

7.1 INFORMATION RETRIEVAL

The initial step (and to a lesser extent an ongoing task of the review) was to collect and organize necessary information. At the outset, NRC forwarded files relevant to the work. These submittals included pertinent sections of plant FSARs, Standard Review Plan (SRP) 3.8, responses to questions on Topic III-7.B previously requested of licensees by the NRC, and other relevant data and reports.

These submittals were organized into Topic III-7.B files on a plant-by-plant basis. The files also contain subsequently received information, as well as other documents developed for the plant review.

A number of channels were used to gather additional information. These included information requests to NRC; letter requests for additional information sent to licensees; plant site visits and retrieval of representative structural drawings, design calculations, and design specifications.

In addition, a separate file was set up to maintain past and present structural codes, NRC Regulatory Guides, Staff Position Papers, and other relevant documents (including, where available, reports from SEP tasks interfacing with the III-7.B effort).

7.2 APPRAISAL OF INFORMATION CONTENT

Most of the information sources were originally written for purposes other than those of the Task III-7.B review. Consequently, much of the information sought was embedded piecemeal in the documents furnished. These sources were searched for the relevant information that they did contain. Generally, it was found that information gaps remained (i.e., some items were not referenced at all or were not specific enough for Task III-7.B purposes). The information found was assembled and the gaps were filled through the information retrieval efforts mentioned earlier.

7.3 CODE COMPARISON REVIEWS

The codes and standards used to represent current licensing practice were selected as described in Appendix I of this report. Briefly summarized, the criteria selection corresponds to NUREG-800 (NRC's Standard Review Plan), the operative document providing guidance to NRC reviewers on licensing matters (see Reference 1).

Next, the Seismic Category I structures at the Yankee Rowe Nuclear Power Station were identified (see Section 8). For these, the codes and standards which were used for actual design were likewise identified on a structure-by-structure basis (see Section 9). Each code was then paired with its counterpart which would govern design were the structure to be licensed today.

Workbooks were prepared for each code pair. The workbook format consisted of paragraph-by-corresponding-paragraph photocopies of the older and the current versions laid out side-by-side on 11-by-17-inch pages. A central column between the codes was left open to provide space for reviewer comments.

The code versions were initially screened to discover areas where the text either remained identical in both versions or had been reedited without changing technical content. Code paragraphs which were found to be essentially the same in both versions were so marked in the comments column.

The review then focused on the remaining portions of the codes where textual disparities existed. Pertinent comments were entered. Typical comments address either the reason the change had been introduced, the intent of the change, its impact upon safety margins, or a combination of such considerations.

As can be readily appreciated, many different circumstances arise in such evaluations--some simple, some complex. A few examples are cited and briefly discussed below.

Provisions were found where code changes liberalized requirements, i.e., less stringent criteria are in force today than were formerly required. Such changes are introduced from time to time as new information becomes available regarding the provision in question. Not infrequently, code committees are called upon to protect against failure modes where the effects are well known; but too little is yet clear concerning the actual failure mechanism and the relative importance of the contributing factors. The committee often cannot defer action until a full investigation has been completed, but must act on behalf of safety. Issues such as these are usually resolved with prudence and caution--sometimes by the adoption of a rule (based upon experience and judgment) known to be conservative enough to assure safety. Subsequent investigation may produce evidence showing the adopted rule to be overly cautious, and provide grounds for its relaxation.

On the other hand, some changes which on first view may appear to reflect a relaxation of code requirements do not in fact actually do so. Structural codes tend to be documents with interactive provisions. Sometimes apparent liberalization of a code paragraph may really reflect a general tightening of criteria, because the change is associated with stiffening of requirements elsewhere.

To cite a simple example, a newly introduced code provision may be found making it unnecessary to check thin flanged, box section beams of relatively small depth-to-width ratio for buckling. This might appear to be a relaxation of requirements; however, elsewhere the code has also introduced a requirement that the designer must space end supports closely enough to preclude buckling. Thus, code requirements have been tightened, not relaxed.

Whenever it was found that code requirements had truly been relaxed, this was noted in the reviewer's comments in the code comparison review. Because liberalization of code criteria clearly cannot give rise to safety issues concerning structures built to more stringent requirements, such matters were not considered further.

On the other hand, whenever it was clear that a code change introduced more stringent criteria, the potential impact of the change on margins of safety shown for the structure was assessed. When it was felt that the change (although more restrictive) would not significantly affect safety margins, this judgment was entered as a reviewer comment. When it was clear that the code change had the potential to significantly affect the perceived margin of safety, this was noted in the comments and the paragraph flagged for further consideration.

Sometimes the effects of a code change are not apparent. Indeed, depending upon a number of factors,* the change may reflect a tightening of requirements for some structures and a liberalization for others. When doubtful or ambiguous situations were encountered in the review, the effect of the code change was explored analytically using simple models.

A variety of analytical techniques were used, depending on the situation at hand. One general approach was to select a basic structural element (a beam, a column, a frame, a slab, or the like) and analytically test it, under both the older and the current criteria. For example, a typical structural element and a simple loading were selected; the element was then designed to the older code requirements. Next, the load carrying capacity of this structure was reexamined using current code criteria. Finally, the load carrying capacities of the element, as shown by the older criteria and as determined by the current criteria, were compared. Examples of investigations performed to assess code change impacts are found in Appendix C.

*Geometry, material properties, magnitude or type of loading, type of supports--to name a few.

In making these studies, an attempt was made to use structural elements, model dimensions, and load magnitudes that were representative of actual structures. For studies that were parametized, an attempt was made to span the parametric range encountered in nuclear structures.

Although one must be cautious about claiming that results from simplified models may be totally applicable to the more complex situations occurring in real structures, it was felt that such examples provided reasonable guidance for making rational judgments concerning the impact of changed code provisions on perceived margins of safety.

7.4 ASSESSMENT OF THE POTENTIAL IMPACT OF CODE CHANGES

As the scope of the Task III-7.B assignment indicates, a limited objective is sought in assessing the effects of code changes on Seismic Category I structures.

The scope of this review is not set at the level of appraisal of individual, as-built structures on plant sites. Consequently, the review does not attempt to make quantitative assessments as to the structural adequacy under current NRC criteria of specific structures at particular SEP plants.

To the contrary, the scope is confined to the comparison of former structural codes and criteria with counterpart current requirements. Correspondingly, the assessment of the impact of changes in codes and criteria is confined to what can be deduced solely from the provisions of the codes and criteria.

Although the review is therefore carried out with minimal reference to actual structures in the field, the assessments of code change impacts that can be made at the code comparison level hold considerable significance for actual structures.

In this respect, two important points should be noted:

1. The review brings sharply into focus the changes in code provisions that may give rise to concern with respect to structural margins of

safety as perceived from the standpoint of the requirements that NRC now imposes upon plants currently being licensed.

The review simultaneously culls away a number of code changes that do not give rise to such concerns, but which (because they are there) would otherwise have to be addressed, on a structure-by-structure basis.

2. The effects of code changes that can be determined from the level of code review are confined to potential or possible impacts on actual structures.

   A review conducted at the code comparison level cannot determine whether or not potentially adverse impacts are actually realized in a given structure. The review may only warn that this may be the case.

   For example, current criteria may require demonstration of structural integrity under a loading combination that includes an additional load not specified in the corresponding loading combination to which the structure was designed. If the non-considered load is large (i.e., in the order of or larger than other major loads that were included), then it is quite possible that some members in the structure would appear overloaded as viewed by current criteria. Thus a potential concern exists.

   However, no determination as to actual overstress in any member can be made by code review alone. Actual margins of safety in the controlling member (and several others*) must certainly be examined before even a tentative judgment of this kind may be attempted.

In order to carry out the code review objective of identifying criteria changes that could potentially impair perceived margins of safety, the following scheme classifying code change impacts was adopted.

7.4.1 Classification of Code Changes

Where code changes involve technical content (as opposed to those which are editorial, organizational, administrative, and the like), the changes are classified according to the following scheme.

*The addition of a new load can change the location of the point of highest stress.

Each such code change is classified according to its potential to alter perceived margins of safety* in structural elements to which it applies. Four categories are established:

Scale A Change - The new criteria have the potential to substantially impair margins of safety as perceived under the former criteria.

Scale A_x Change - The impact of the code change on margins of safety is not immediately apparent. Scale A_x code changes require analytical studies of model structures to assess the potential magnitude of their effect upon margins of safety.

Scale B Change - The new criteria operate to impair margins of safety but not enough to cause engineering concern about the adequacy of any structural element.

Scale C Change - The new criteria will give rise to larger margins of safety than were exhibited under the former criteria.

7.4.1.1 General and Conditional Classifications of Code Change Impacts

Scale ratings of code changes are found in two different forms in this report. For example, some are designated as "Scale A," and others as "Scale C." Others have dual designation, such as "Scale A if --- [a condition statement] or Scale C if --- [a second condition statement]."

In assigning scale classifications, an efficient design to original criteria is assumed. That is, it is postulated that (a) the provision in question controls design, and (b) the structural member to which the code provision applies was proportioned to be at (or close to) the allowable limit. The impact scale rating is assigned accordingly.

If the code change is Scale A, and it applies (in a particular structure) to a member which is not highly stressed, then this may afford excellent grounds for asserting that this particular member is adequate; but it does not thereby downgrade the ranking to, say, a Scale B change for that member. The

*That is, if (all other considerations remaining the same) safety margins as computed by the older code rules were to be recomputed for an as-built structure in accordance with current code provisions, would there be a difference due only to the code change under consideration?

scale ranking is neither a function of member stress* nor a ranking of member adequacy. The scale system ranks code change impact, not individual members.

However, a number of code provisions are framed so that the allowable limit is made a function of member proportion. When this kind of a code provision is changed, the change may affect members of certain proportions one way and members of other proportions differently.

For example, assume a change in column design requirements is introduced into the code and is framed in terms of the ratio of the effective column length to its radius of gyration. The new rule acts to tighten design requirements for slender columns, but liberalizes former requirements for columns that are not slender. This change may be rated Scale A for slender columns, and simultaneously, Scale C for non-slender ones. Although some columns now appear to be Scale A columns while others appear to be Scale C columns, the distinction between them resides in the code, and is not a reflection of member adequacy. Clearly, it is still the code changes that are ranked; but, in this case, the code change does not happen to affect all columns in a unilateral way.

7.4.1.2 Code Impact or Structural Margins

This classification of code changes identifies both (a) changes that have the potential to significantly impair perceived margins of safety (Scale A) and (b) changes that have the potential to enhance perceived margins of safety (Scale C).

Emphasis is subsequently placed on Scale A changes, not on Scale C changes. The purpose of the code comparison review is to narrow down and bring into sharper focus the areas where structures shown adequate under former criteria may not fully comply with current criteria. Once such criteria changes have been identified, actual structures may be checked to see if the potential concern is applicable to the structure. Depending upon a number of structure-specific circumstances, it may or may not pertain.

*There are exceptions, but these are code-related, not adequacy-related.

The same thing is true of Scale C changes, i.e., those that may enhance perceived structural margins. Specific structures must be examined to see if the potential benefit is actually applicable to the structure. If it is applicable, credit may be taken for it. However, this step can only be taken at the structural level, not at the code level.

A simple example may help clarify this point. Assume a steel beam exists in a structure designed by AISC 1963 rules for the then-specified loading combination. Current criteria require inclusion of an additional load in the loading combination (Scale A change), but the current structural code permits a higher allowable load if the beam design conforms to certain stipulated proportions (Scale C change). Several circumstances are possible for beams in actual structures, as shown below.

New Load	Higher Stress Limit	Results
Maximum stress in beam under original loading conditions was low with ample margin for additional load	Applicability immaterial	Beam adequate under current criteria
Maximum stress in beam under original loading condition was near former allowable limit	Beam qualifies for higher stress limit	Beam may be adequate under current criteria
Maximum stress in beam under original loading condition was near former allowable limit	Beam does not qualify for increased stress limit	Beam unlikely to be adequate under current criteria

It is clear from this example that the function of the code review is to point out code changes which might impair perceived margins of safety, and that assessment of their pertinence is best accomplished at the structure-specific level.

7.5 PLANT-SPECIFIC CODE CHANGES

There is substantial overlap among the SEP plants in the codes and standards used for structural design. Several plants, for example, followed the provisions of ACI-318, 1963 edition, in designing major concrete structures.

Thus, the initial work of comparing older and current criteria is not plant-specific. However, when the reviewed codes are packaged in sets containing only those code comparisons relevant to design of Seismic Category I structures in a particular SEP plant, the results begin to take on plant-specific character.

The code changes potentially applicable to particular structures at a particular SEP plant have then been identified. However, this list is almost surely overly long because the list has been prepared without reference to actual plant structures. For example, the code change list might include an item relating to recently introduced provisions for the design of slender columns, while none actually exist in any structures in that particular plant.

In-depth examination of design drawings, audit of structural analyses, and review of plant specifications were beyond the scope of the III-7.B task; accordingly, such activities were not attempted. However, occasional reference to such documents was necessary to the review work. Consequently, it was possible to cull from the list some items that were obviously inappropriate to the Yankee Rowe plant structures. Wherever this was done, the reason for removal was documented, but no attempt was made to remove every such item.

Code changes that may be significant for structures in general but did not appear applicable to any of the Seismic Category I structures at the Yankee Rowe plant were relegated to Appendix A. The Scale A or Scale A_x changes that remained are listed on a code-by-code basis in Section 11.

8. YANKEE ROWE SEISMIC CATEGORY I STRUCTURES

SEP Topic III-1 has for its objectives the classification of components, structures, and systems with respect to both quality group and seismic designation. Based upon a review of the Yankee Rowe FSAR [5] and Reference 6, the present report considers the following to be Seismic Category I structures:

- containment structure (vapor container)
- fuel transfer vault
- diesel generator building
- control room (part of turbine building)
- battery rooms (located in auxiliary bay of turbine building)
- primary auxiliary building
- screen well house
- spent fuel chute
- spent fuel pit
- primary vent stack (steel).

Although the primary vent stack is not mentioned in Reference 6, which gives the seismic classification of Yankee Rowe structures, it is appropriate to include the primary vent stack in the above list as a Seismic Category I structure because of its proximity to other Seismic Category I equipment and structures.

9. STRUCTURAL DESIGN CRITERIA

The structural codes governing design of the major Seismic Category I structures for the Yankee Rowe Nuclear Power Station are detailed in the following table.

	Structure	Design Criteria	Current Criteria
1.	Containment Structure (vapor container)	ASME B&PV Code, Sect. VIII, 1956, and Code Cases 1177-2, 1224-1, 1226-3, and 1235	ASME B&PV Code, Sect. III, Div. I, Subsect. NE, Class MC components, 1980
2.	Fuel Transfer Vault	ACI 318-56 AISC, 1956	ACI 349-76 AISC, 1980
3.	Safety Injection Building (diesel generator rooms)	ACI 318-56 AISC, 1956	ACI 349-76 AISC, 1980
4.	Control Room (part of turbine building)	ACI 318-56 AISC, 1956	ACI 349-76 AISC, 1980
5.	Battery Rooms (located in auxiliary bay of turbine building)	ACI 318-56 AISC, 1956	ACI 349-76 AISC, 1980
6.	Primary Auxiliary Building	ACI 318-56 AISC, 1956	ACI 349-76 AISC, 1980
7.	Screen Well House	ACI 318-56 AISC, 1956	ACI 349-76 AISC, 1980
8.	Spent Fuel Chute	ACI 318-56 AISC, 1956	ACI 349-76 AISC, 1980
9.	Spent Fuel Pool	ACI 318-56	ACI 349-76
10.	Primary Vent Stack (steel)*	Design criteria not stated	*

* Comparisons of the previous design code with current versions for the primary vent stack are not carried out in this report since a complete reanalysis of the stack to current criteria will be carried out within the SEP program.

References identifying major codes used for the original design are:

1. Yankee Atomic Electric Company letters to the NRC dated 9-6-79 and 1-31-79

2. Yankee Rowe Final Safety Analysis Report, Volume I, January 8, 1974.

3. Stone and Webster's "Specification for Structural Steel for Yankee Atomic Electric Plant" is dated August 7, 1957; and invokes "The Latest Revision of the AISC Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings." The fifth edition (in effect from 1946 to 1963) was then current, and the 1953 printing has been used as the reference in this report.

10. LOADS AND LOAD COMBINATION CRITERIA

10.1 DESCRIPTION OF TABLES OF LOADS AND LOAD COMBINATIONS

The requirements governing loads and load combinations to be considered in the design of civil engineering structures for nuclear service have been revised since the older nuclear power plants were constructed and licensed. Such changes constitute a major aspect of the general pattern of evolving design requirements; consequently, they are singled out for special consideration in this section of this report.

The NRC Regulatory Guides and Standard Review Plans provide guidance as to what loads and load combinations must be considered. In some cases, the required loads and load combinations are also specified within the governing structural design code; other structural codes have no such provisions and take loads and load combinations as given a priori. In this report, loads and load combinations are treated within the present section whether or not the structural design codes also include them.

Later sections of this report address, paragraph by paragraph, changes in text between design codes current at the time the plant was constructed and those governing design today; however, to avoid repetition, code changes related to loads and load combinations will not be evaluated again although they may appear as provisions of the structural design codes.

To provide a compact and systematic comparison of previous and present requirements, two sets of tables are used:

1. load tables
2. load combination tables.

Both sets of tables are constructed in accordance with current requirements for Seismic Category I structures, i.e., the load tables list all loads that must be considered in today's design of these structures (as enumerated in NRC's Standard Review Plan), and the load combination tables list all combinations of these loadings for which current licensing procedures require demonstration of structural integrity.

In general, the loads and load combinations to be considered are determined by the structure under discussion. The design loads for the structure housing the emergency power diesel generator, for example, are quite different than those for the design of the containment vessel. Consequently, structures must be considered individually. Each structure usually requires a load table and load combination table appropriate to its specific design requirements.

The design requirements for the various civil engineering structures within a nuclear power plant are echoed in applicable sections of NRC's Standard Review Plan (SRP) 3.8. The tables in the present report correspond to, and summarize, these requirements for each structure. A note at the bottom of each table provides the reference to the applicable section of the Standard Review Plan. Section 10.2 of this report lists, for reference, the load symbols used in the charts together with their definitions.

The loads actually used for design are considered, structure by structure, and the load tables are filled in according to the following scheme:

1. The list of potentially applicable loads (according to current requirements) is examined to eliminate loads which either do not occur on, or are not significant for, the structure under consideration.

2. The loads included in the actual design basis are then checked against the reduced list to see if all applicable loads (according to current requirements) were actually considered during design.

3. Each load that was considered during design is next screened to see if it appears to correspond to current requirements. Questions such as the following are addressed: Were all the individual loads encompassed by the load category definition represented in the applied loading? Do all loads appear to match present requirements (1) in magnitude? (2) in method of application?

4. An annotation is made as to whether deviations from present requirements exist, either because of load omissions or because the loads do not correspond in magnitude or in other particulars.

5. If a deviation is found, a judgment (in the form of a scale ranking) is made as to the potential impact of the deviation on perceived margins of safety.

6. Relevant notes or comments are recorded.

Of particular importance to the Topic III-7.B review are comments indicating that the effects of certain loadings (tornado and seismic loads, in particular) are being examined under other SEP topics. In all such cases, the findings of these special SEP topics (where review in depth of the indicated loading conditions will be undertaken) will be definitive for the overall SEP effort. Consequently, no licensee investigation of such issues is required under Topic III-7.B nor is such effort within the scope of Topic III-7.B (see Section 4). Licensee participation in the resolution of such issues may, however, be requested under the scope of other SEP topics devoted to such issues.

After the load tables have been filled out, the load combination tables are compiled. Like the load tables, the load combination tables are drawn up to current requirements and the load combinations actually used in the design basis are matched against these requirements.

Current criteria require consideration during plant design of 13 load combinations for most structures, as shown in the load combination tables. These specific requirements were not in effect at the time when SEP plants were designed. Consequently, other sets of load combinations were used. In comparing actual and current criteria, an attempt was made to match each of the load combinations actually considered to its nearest counterpart under present requirements. For example, consider a plant where the safe shutdown earthquake was addressed in combination with other loads, but not in combination with the effects of a LOCA (load combination 13). The load combination tables would reflect this by showing that load case 9 was addressed, but that load case 13 was not. If six load cases were considered, only six (nearest counterpart) load cases are indicated in the table--not partial fulfillment of all 13.

For ease of comparison, the load combinations actually used are superimposed on the load combinations currently required. This is accomplished in two steps:

1. Currently specified load combinations include loads sufficient for the most general cases. In particular applications, some of these are either inappropriate or insignificant. Therefore, the first step

is to strike all loads that are not applicable to the structure under consideration from all load combinations in which they appear.

2. Next, loads actually combined are indicated by encircling (in the appropriate load combinations) each load contributing to the summation considered for design.

Thus, the comparison between what was actually done and what is required today is readily apparent. If the load combinations used are in complete accord with current requirements, each load symbol on the sheet appears as either struck or encircled. Load combinations not considered, and loads omitted from the load combinations stand out as unencircled items.

A scale ranking is next assigned to the load combinations; however (unlike the corresponding ranking of loads), a scale ranking is not necessarily assigned to each one. When the load combinations used for design correspond closely to current requirements, scale ratings may be assigned to all combinations. However, when the number of load combinations considered in design was substantially fewer than current criteria prescribe, it did not appear to serve any engineering purpose to rank the structure for each currently required load combination. Instead, a limited number of loading cases (usually two) were ranked.

The following considerations guided the selection of these cases:

1. For purposes of the SEP review, it was not believed necessary to require an extensive reanalysis of structures under all load combinations currently specified.

2. SEP plants have been in full power operation for a number of years. During this time, they have experienced a wide spectrum of operating and upset conditions. There is no evidence that major Seismic Category I structures lack integrity under these operating conditions.

3. The most severe load combinations occur under emergency and accident conditions. These are also the conditions associated with the greatest consequences to public health and safety.

4. If demonstration of structural adequacy under the most severe load combinations currently specified for emergency and accident conditions is provided, a reasonable inference can be drawn that the structure is also adequate to sustain the less severe loadings associated with less severe consequences.

The scale rankings assigned to loads and load combinations in tables are intended as an appraisal of plant status, with respect to demonstration of compliance with current design criteria, based on information available to the NRC prior to the inception of the SEP review. A number of structurally related SEP topics review some loads and load combinations in detail based upon current calculational methods. In order that a consistent basis for the tables be maintained, they are based upon load combinations considered in the original design of the facility or, in the case of facility modifications, they are based upon the combinations used in the design of the modification. Loads that were not included in the original design or that have increased in magnitude and have not been specifically addressed in another SEP topic should be addressed by the Licensee.

10.2 LOAD DEFINITIONS

D Dead loads or their related internal moments and forces (such as permanent equipment loads).

E or E_o Loads generated by the operating basis earthquake.

E' or E_{ss} Loads generated by the safe shutdown earthquake.

F Loads resulting from the application of pre-stress.

H Hydrostatic loads under operating conditions.

H_a Hydrostatic loads generated under accident conditions, such as post-accident internal flooding. (F_L is sometimes used by others* to designate post-LOCA internal flooding.)

L Live loads or their related internal moments and forces (such as movable equipment loads).

P_a Pressure load generated by accident conditions (such as those generated by the postulated pipe break accident).

P_o or P_v Loads resulting from pressure due to normal operating conditions.

*See, for example, SRP 3.8.2.

P_s All pressure loads which are caused by the actuation of safety relief valve discharge including pool swell and subsequent hydrodynamic loads.

R_a or R_r Pipe reactions under accident conditions (such as those generated by thermal transients associated with an accident).

R_o Pipe reactions during startup, normal operating, or shutdown conditions, based on the critical transient or steady-state condition.

R_s All pipe reaction loads which are generated by the discharge of safety relief valves.

T_a Thermal loads under accident conditions (such as those generated by a postulated pipe break accident).

T_o Thermal effects and loads during startup, normal operating, or shutdown conditions, based on the most critical transient or steady-state condition.

T_s All thermal loads which are generated by the discharge of safety relief valves.

W Loads generated by the design wind specified for the plant.

W' or W_t Loads generated by the design tornado specified for the plant. Tornado loads include loads due to tornado wind pressure, tornado-created differential pressure, and tornado-generated missiles.

Y_j Equivalent static load on the structure generated by the impingement of the fluid jet from the broken pipe during the design basis accident.

Y_m Missile impact equivalent static load on the structure generated by or during the design basis accident, such as pipe whipping.

Y_r Equivalent static load on the structure generated by the reaction on the broken pipe during the design basis accident.

The load combination charts correspond to loading cases and load definitions as specified in the appropriate SRP. Each chart is associated with a specific SRP as identified in the notes accompanying the chart. Guidance with respect to the specific loads which must be considered in forming each load combination is provided by the referenced SRP. All SRPs are prepared to a standard format; consequently, subsection 3 of each plan always contains the appropriate load definitions and load combination guidance.

10.3 DESIGN LOAD TABLES

"COMPARISON OF DESIGN BASIS LOADS"

COMPARISON OF DESIGN BASIS LOADS

STRUCTURE: CONTAINMENT STRUCTURE (VAPOR CONTAINER)

PLANT: YANKEE ROWE

	Current Design Basis Loads	Is Load Applicable To This Structure?	Is Load Included In Plant Design Basis?	SEP Topic Reviewing This Load	Does Load Magnitude Correspond To Present Criteria?	Does Deviation Exist In Load Basis?	Code Impact Scale Ranking	Comments
Gravity	D	Yes	Yes	-	Yes	No	-	
	L	Yes	Yes	-	Yes	No	-	1.
Pressure	F	No	-	-	-	-	-	
	H	Yes	Yes	III-5.A	*	*	*	
	P_o	Yes	Yes	-	-	No	-	
	P_a	Yes	Yes	VI-2.D, III-7.B	*	*	*	2.
	P_s	No	-	-	-	-	-	
Thermal	T_o	Yes	Yes	-	-	No	-	
	T_a	Yes	Yes	VI-2.D, III-7.B	*	*	*	
	T_S	No	-	-	-	-	-	
Pipe & Mech.	R_o	Yes	No	-	-	Yes	-	
	R_a	Yes	No	-	-	Yes	A_x	
	R_S	No	-	-	-	No	-	
Environmental	E'	Yes	No	III-6	*	*	A_x	3. ,5.
	E	Yes	No	III-6	*	*	*	3.
	W'	Yes	No	III-2, III-4.A	*	*	A_x	
	W	Yes	Yes	III-2, III-4.A	*	*	*	4.
Impulse	Y_r	Yes	No	III-5.A	*	*	*	
	Y_j	Yes	No	III-5.A	*	*	*	
	Y_m	Yes	No	III-5.A	*	*	*	

Ref.; SRP(1981) Section 3.8.1 or 3.8.2

Comments

* To be determined per results of SEP topics. Scale ranking shown for SEP topic items are independent judgments, based on information in the FSAR or other original design documents.

1. Snow loads have increased per Topic II-2.A.
2. FSAR Sect. 6.2.1.2.1 states that design pressure is 31.5 psig.
3. Attachment B, specification for vapor container, P.5 states "Design of vapor container for earthquake conditions is not required."
4. Containment designed for 100 MPH wind.
5. In "Seismic Analysis and Stress Report for steel vapor container structure Yankee Nuclear Power Station" by Earthquake Engineering Systems, Inc., dated March 2, 1979, analysis of the containment was performed for safe shutdown earthquake.

STRUCTURE: FUEL TRANSFER VAULT (OUTSIDE CONTAINMENT)

COMPARISON OF DESIGN BASIS LOADS

PLANT: YANKEE ROWE

	Current Design Basis Loads	Is Load Applicable To This Structure?	Is Load Included In Plant Design Basis? (1)	SEP Topic Reviewing This Load	Does Load Magnitude Correspond To Present Criteria?	Does Deviation Exist In Load Basis?	Code Impact Scale Ranking	Comments
Gravity	D	Yes	Yes	-	Yes	No	-	
	L	Yes	Yes	-	No	Yes	A_x	2.
Pressure	F	No	-	-	-	-	-	
	H	Yes	Yes	III-3.A	*	*	*	
	P_a	No	-	III-5.B	*	*	-	
Thermal	T_o	Yes	-	-	-	-	-	
	T_a	No	-	III-5.B	*	*	*	
Pipe & Mech.	R_o	No	-	-	-	-	-	
	R_a	No	-	-	-	-	-	
Environmental	E'	Yes	Yes	III-6	*	*	*	
	E	Yes	Yes	III-6	*	*	*	
	W'	Yes	No	III-2, III-4.A	*	*	*	
	W	Yes	Yes	III-2, III-4.A	*	*	*	
Impulse	Y_r	No	-	III-5.B	*	*	-	
	Y_j	Yes	No	III-5.B	*	*	*	3.
	Y_m	Yes	No	III-5.B	*	*	*	3.

Ref.; SRP(1981) Section 3.[illegible]

Comments

* To be determined per [illegible] topics. Scale ranking shown for SEP topic items are independent judgments, based on information in the FSAR or other original design documents.

1. According to letter from Yankee Rowe to NRC with Attachments A to F, dated 9/6/79, all design loads (except for containment) comply with American Standard Building Code A58.1-1955.
2. Snow loads have increased per Topic II-2A and may increase per Topic II-2B for parapet roofs.
3. Main steam lines appear to be close enough for main steam line break to possibly cause jet impingement and impact loads.

STRUCTURE: SAFETY INJECTION BUILDING (DIESEL GENERATOR ROOMS)

COMPARISON OF DESIGN BASIS LOADS

PLANT: YANKEE ROWE

	Current Design Basis Loads	Is Load Applicable To This Structure?	Is Load Included In Plant Design Basis? (1)	SEP Topic Reviewing This Load	Does Load Magnitude Correspond To Present Criteria?	Does Deviation Exist In Load Basis?	Code Impact Scale Ranking	Comments
Gravity	D	Yes	Yes	-	Yes	No	-	
	L	Yes	Yes	-	No	Yes	A_x	2.
Pressure	F	No	-	-	-	-	-	
	H	*	Yes	III-3.A	*	*	*	3.
	P_a	*	No	III-5.B	*	*	*	
Thermal	T_o	Yes	-	-	-	-	-	
	T_a	*	No	III-5.B	*	*	*	
Pipe & Mech.	R_o	No	No	-	-	-	-	
	R_a	No	No	-	-	-	-	
Environmental	E'	Yes	Yes	III-6	*	*	*	
	E	Yes	Yes	III-6	*	*	*	
	W'	Yes	No	III-2, III-4.A	*	*	*	
	W	Yes	Yes	III-2, III-4.A	*	*	*	
Impulse	Y_r	*	No	III-5.B	*	*	*	
	Y_j	*	No	III-5.B	*	*	*	
	Y_m	*	No	III-5.B	*	*	*	

Ref.; SRP(1981) Section 3.8.4

Comments

* To be determined per results of SEP topics. Scale ranking shown for SEP topic items are independent judgments, based on information in the FSAR or other original design documents.

1. According to letter from Yankee Rowe to NRC with Attachments A to F, dated 9/6/79, all design loads (except for containment) comply with American Standard Building Code A58.1-1955.

2. Snow loads have increased per Topic II-2A and may increase per Topic II-3B for parapet roofs.

3. According to letter referred to in Comment 1., no record of flood conditions used for initial design input is available. However, a flood analysis was recently undertaken and included in YAEC Report 1139, July 1977.

STRUCTURE: CONTROL ROOM
(PART OF TURBINE BUILDING)

COMPARISON OF DESIGN BASIS LOADS

PLANT: YANKEE ROWE

	Current Design Basis Loads	Is Load Applicable To This Structure?	Is Load Included In Plant Design Basis? (1)	SEP Topic Reviewing This Load	Does Load Magnitude Correspond To Present Criteria?	Does Deviation Exist In Load Basis?	Code Impact Scale Ranking	Comments
Gravity	D	Yes	Yes	-	Yes	No	-	
	L	Yes	Yes	-	No	Yes	A_x	2.,4.
Pressure	F	No	-	-	-	-	-	
	H	No	-	III-3.A	*	*	-	
	P_a	Yes	No	III-5.B	*	*	*	
Thermal	T_o	Yes	-	-	-	-	-	
	T_a	Yes	No	III-5.B	*	*	*	
Pipe & Mech.	R_o	No	No	-	-	-	-	
	R_a	No	No	-	-	-	-	
Environmental	E'	Yes	Yes	III-6	*	*	*	
	E	Yes	Yes	III-6	*	*	*	
	W'	Yes	No	III-2, III-4.A	*	*	*	
	W	Yes	Yes	III-2, III-4.A	*	*	*	
Impulse	Y_r	Yes	No	III-5.B	*	*	*	3.,5.
	Y_j	Yes	No	III-5.B	*	*	*	3.,5.
	Y_m	Yes	No	III-5.B	*	*	*	3.,5.

Ref.; SRP(1981) Section 3.8.4

Comments

* To be determined per results of SEP topics. Scale ranking shown for SEP topic items are independent judgments, based on information in the FSAR or other original design documents.

1. According to letter from Yankee Rowe to NRC with Attachments A to F, dated 9/6/79, all design loads (except for containment) comply with American Standard Building Code A58.1-1955.

2. Snow loads have increased per Topic II-2A and may increase per Topic II-3B for parapet roofs.

3. Some of main steam line support framing is anchored to shield wall outside control room.

4. According to Stone and Webster Drawing 9699-FC-13A, control room designed for floor live load of 50 psf.

5. Letter from YAEC to US-AEC dated July 2, 1973 states that effects of piping system break was investigated in 1973.

STRUCTURE: BATTERY ROOMS
(Located in Auxiliary bay of Turbine Building.)

COMPARISON OF DESIGN BASIS LOADS

PLANT: YANKEE ROWE

	Current Design Basis Loads	Is Load Applicable To This Structure?	Is Load Included In Plant Design Basis? (1)	SEP Topic Reviewing This Load	Does Load Magnitude Correspond To Present Criteria?	Does Deviation Exist In Load Basis?	Code Impact Scale Ranking	Comments
Gravity	D	Yes	Yes	-	Yes	No		
	L	Yes	Yes	-	Yes	No	-	
Pressure	F	No	-	-	-	-	-	
	H	No	-	III-3.A	*	*	-	
	P_a	*	No	III-5.B	*	*	*	
Thermal	T_o	Negl.	-	-	-	-	-	
	T_a	*	No	III-5.B	*	*	*	
Pipe & Mech.	R_o	No	-	-	-	-	-	
	R_a	No	-	-	-	-	-	
Environmental	E'	Yes	Yes	III-6	*	*	*	
	E	Yes	Yes	III-6	*	*	*	
	W'	*	No	III-2, III-4.A	*	*	*	
	W	-	-	III-2, III-4.A	*	*	*	
Impulse	Y_r	Yes	No	III-5.B	*	*	*	2.
	Y_j	Yes	No	III-5.B	*	*	*	2.
	Y_m	Yes	No	III-5.B	*	*	*	2.

Ref.; SRP(1981) Section 3.8.4

Comments

* To be determined per results of SEP topics. Scale ranking shown for SEP topic items are independent judgments, based on information in the FSAR or other original design documents.

1. According to letter from Yankee Rowe to NRC with Attachments A to F, dated 9/6/79, all design loads (except for containment) comply with American Standard Building Code A58.1-1955.

2. Letter from YAEC to US-AEC dated July 2, 1973 states that effects of piping system break was investigated in 1973.

STRUCTURE: PRIMARY AUXILIARY BUILDING

COMPARISON OF DESIGN BASIS LOADS

PLANT: YANKEE ROWE

	Current Design Basis Loads	Is Load Applicable To This Structure?	Is Load Included In Plant Design Basis? (1)	SEP Topic Reviewing This Load	Does Load Magnitude Correspond To Present Criteria?	Does Deviation Exist In Load Basis?	Code Impact Scale Ranking	Comments
Gravity	D	Yes	Yes	-	Yes	No	-	
	L	Yes	Yes	-	No	Yes	A_x	2.
Pressure	F	No	-	-	-	-	-	
	H	*	Yes	III-3.A	*	*	*	3.
	P_a	*	No	III-5.B	*	*	*	
Thermal	T_o	Negl.	-	-	-	-	-	
	T_a	*	No	III-5.B	*	*	*	
Pipe & Mech.	R_o	Yes	No	-	-	Yes	-	
	R_a	Yes	No	-	-	Yes	-	
Environmental	E'	Yes	Yes	III-6	*	*	*	
	E	Yes	Yes	III-6	*	*	*	
	W'	Yes	No	III-2, III-4.A	*	*	*	
	W	Yes	Yes	III-2, III-4.A	*	*	*	
Impulse	Y_r	Yes	No	III-5.B	*	*	*	4.
	Y_j	Yes	No	III-5.B	*	*	*	4.
	Y_m	Yes	No	III-5.B	*	*	*	4.

Ref.; SRP(1981) Section 3.8.4

Comments

* To be determined per results of SEP topics. Scale ranking shown for SEP topic items are independent judgments, based on information in the FSAR or other original design documents.

1. According to letter from Yankee Rowe to NRC with Attachments A to F, dated 9/6/79, all design loads (except for containment) comply with American Standard Building Code A58.1-1955.

2. Snow loads have increased per Topic II-2A and may increase per Topic II-3B for parapet roofs.

3. According to letter referred to in Comment 1., no record of flood conditions used for initial design input is available. However, a flood analysis was recently undertaken and included in YAEC Report 1139, July 1977.

4. Letter from YAEC to US-AEC dated July 2, 1973 states that effects of piping system break was investigated in 1973.

STRUCTURE: SCREEN WELL HOUSE

COMPARISON OF DESIGN BASIS LOADS

PLANT: YANKEE ROWE

	Current Design Basis Loads	Is Load Applicable To This Structure?	Is Load Included In Plant Design Basis? (1)	SEP Topic Reviewing This Load	Does Load Magnitude Correspond To Present Criteria?	Does Deviation Exist In Load Basis?	Code Impact Scale Ranking	Comments
Gravity	D	Yes	Yes	-	Yes	No	-	
	L	Yes	Yes	-	No	Yer	A_x	2.
Pressure	F	No	-	-	-	-	-	
	H	*	Yes	III-3.A	*	*	*	3.
	P_a	*	No	III-5.B	*	*	*	
Thermal	T_o	Yes	-	-	-	-	-	
	T_a	*	No	III-5.B	*	*	*	
Pipe & Mech.	R_o	Negl.	-	-	-	-	-	
	R_a	No	-	-	-	-	-	
Environmental	E'	Yes	Yes	III-6	*	*	*	
	E	Yes	Yes	III-6	*	*	*	
	W'	Yes	No	III-2, III-4.A	*	*	*	
	W	Yes	Yes	III-2, III-4.A	*	*	*	
Impulse	Y_r	No	No	III-5.B	*	*	*	
	Y_j	No	No	III-5.B	*	*	*	
	Y_m	No	No	III-5.B	*	*	*	

Ref.; SRP(1981) Section 3.8.4

Comments

* To be determined per results of SEP topics. Scale ranking shown for SEP topic items are independent judgments, based on information in the FSAR or other original design documents.

1. According to letter from Yankee Rowe to NRC with Attachments A to F, dated 9/6/79, all design loads (except for containment) comply with American Standard Building Code A58.1-1955.

2. Snow loads have increased per Topic II-2A and may increase per Topic II-3B for parapet roofs.

3. According to letter referred to in Comment 1., no record of flood conditions used for intial design input is available. However, a flood analysis was recently undertaken and included in YAEC Report 1139, July 1977.

STRUCTURE: SPENT FUEL CHUTE

COMPARISON OF DESIGN BASIS LOADS

PLANT: YANKEE ROWE

	Current Design Basis Loads	Is Load Applicable To This Structure?	Is Load Included In Plant Design Basis? (1)	SEP Topic Reviewing This Load	Does Load Magnitude Correspond To Present Criteria?	Does Deviation Exist In Load Basis?	Code Impact Scale Ranking	Comments
Gravity	D	Yes	Yes	-	Yes	No	-	
	L	Yes	Yes	-	Yes	No	A_x	2.
Pressure	F	No	-	-	-	-	-	
	H	No	-	III-3.A	*	*	-	
	P_a	*	No	III-5.B	*	*	*	
Thermal	T_o	Yes	-	-	-	-	-	
	T_a	Yes	No	III-5.B	*	*	*	
Pipe & Mech.	R_o	No	-	-	-	-	-	
	R_a	No	-	-	-	-	-	
Environmental	E'	Yes	Yes	III-6	*	*	A_x	3.
	E	Yes	Yes	III-6	*	*	*	
	W'	Yes	No	III-2, III-4.A	*	*	A_x	4.
	W	Yes	Yes	III-2, III-4.A	*	*	*	
Impulse	Y_r	*	No	III-5.B	*	*	*	
	Y_j	*	No	III-5.B	*	*	*	4.
	Y_m	*	No	III-5.B	*	*	*	4.

Ref.; SRP(1981) Section 3.8.4

Comments

* To be determined per results of SEP topics. Scale ranking shown for SEP topic items are independent judgments, based on information in the FSAR or other original design documents.

1. According to letter from Yankee Rowe to NRC with Attachments A to F, dated 9/6/79, all design loads (except those for containment) comply with American Standard Building Code A58.1-1955.

2. Snow loads have increased per Topic II-2A.

3. Fuel transfer chute may be vulnerable to damage from differential seismic movement between vapor container and fuel transfer vault.

4. Fuel transfer chute may be vulnerable to damage from tornado missiles, primary vent stack collapse, or main steam line break (jet impingement or whipping pipe).

STRUCTURE:

SPENT FUEL POOL (Concrete)

COMPARISON OF DESIGN BASIS LOADS

PLANT: YANKEE ROWE

	Current Design Basis Loads	Is Load Applicable To This Structure?	Is Load Included In Plant Design Basis?	SEP Topic Reviewing This Load	Does Load Magnitude Correspond To Present Criteria?	Does Deviation Exist In Load Basis?	Code Impact Scale Ranking	Comments
Gravity	D	Yes	Yes	——	Yes	No		
	L	Yes	Yes	——	Yes	No		
Pressure	F	No	—	——	—	—		
	H	Yes	Yes	III-3.A	*	*		
	P_a	No	—	III-5.B	*	*		
Thermal	T_o	Negl.	—	——	—	—		
	T_a	Yes	*	III-5.B	*	*		
Pipe & Mech.	R_o	No	—	——	—	—		
	R_a	No	—	——	—	—		
Environmental	E'	Yes	No 1	III-6	*	*		
	E	Yes	No	III-6	*	*	*	
	W'	Yes 2	No	III-2, III-4.A	*	*	*	4.
	W	No	—	III-2, III-4.A	*	*		
Impulse	Y_r	—	—	III-5.B	*	*	*	3.
	Y_j	—	—	III-5.B	*	*	*	3.
	Y_m	—	—	III-5.B	*	*	*	3.

Ref.; SRP(1981) Section 3.8.4

Comments

* To be determined per results of SEP topics. Scale ranking shown for SEP topic items are independent judgments, based on information in the FSAR or other original design documents.

1. The seismic capacity of the pool was evaluated in 1980 and found to provide adequate seismic resistance to qualify the plant for continued operation during the time period of the SEP Program.
2. Applicable only if roof over spent pool is not tornado resistant.
3. Pipe break external to containment is evaluated in SEP Topic III-5.B.
4. SEP Topic III-2 will determine whether or not pool exposure to possible tornado effects is an allowable spent fuel pool load.

STRUCTURE: PRIMARY VENT STACK (STEEL)

COMPARISON OF DESIGN BASIS LOADS

PLANT: YANKEE ROWE

	Current Design Basis Loads	Is Load Applicable To This Structure?	Is Load Included In Plant Design Basis?	SEP Topic Reviewing This Load	Does Load Magnitude Correspond To Present Criteria?	Does Deviation Exist In Load Basis?	Code Impact Scale Ranking	Comments
Gravity	D	Yes	Yes	-	-	No	-	
	L	Yes	Yes	-	-	No	-	
Pressure	F	No	-	-	-	-	-	
	H	No	-	III-3.A	*	*	*	
	P_a	No	-	III-5.B	*	*	*	
Thermal	T_o	Negl.	-	-	-	-	-	
	T_a	Negl.	-	III-5.B	*	*	*	
Pipe & Mech.	R_o	Yes	-	-	-	-	-	(Duct)
	R_a	No	-	-	-	-	-	
Environmental	E'	Yes		III-6	*	*	*	
	E	Yes		III-6	*	*	*	
	W'	Yes		III-2, III-4.A	*	*	*	
	W	Yes		III-2, III-4.A	*	*	*	
Impulse	Y_r	No	-	III-5.B	*	*	*	
	Y_j	No	-	III-5.B	*	*	*	
	Y_m	No	-	III-5.B	*	*	*	

Ref.; SRP(1981) Section 3.8.4

Comments

* To be determined per results of SEP topics. Scale ranking shown for SEP topic items are independent judgments, based on information in the FSAR or other original design documents.

10.4 LOAD COMBINATION TABLES

"COMPARISON OF LOADING COMBINATION CRITERIA"

COMPARISON OF LOADING COMBINATION CRITERIA

PLANT: YANKEE ROWE

STRUCTURE: CONTAINMENT STRUCTURE (VAPOR CONTAINER)

	Combined Loading Cases	Gravity Dead, Live	Thermal	Pressure	Mechanical	Natural Phenomena	Impulsive Loading	Scale Ranking
Service Level A	1	D + L	T_o	P_o	R_o			
	2	D + L	T_s	P_s	R_s			
	3	D + L	T_a	P_a	R_a			
	4	D + L	$T_a + T_s$	$P_a + P_s$	$R_a + R_s$			
Service Level B	1	D + L	T_a	P_a	R_a	E		
	2	D + L	T_o	P_o	R_o	E		
	3	D + L	T_s	P_s	R_s	E		
	4	D + L	$T_a + T_s$	$P_a + P_s$	$R_a + R_s$			
Service Level C	1	D + L	T_a	P_a	R_a	E'	(2)	
	2	D + L	T_o	P_o	R_o	E'		
	3	D + L	$T_a + T_s$	$P_a + P_s$	$R_a + R_s$	E'		
Service Level D	1	D + L	T_a	P_a	R_a	E'	$Y_r+Y_j+Y_m$	
	2	D + L	$T_a + T_s$	$P_a + P_s$	$R_a + R_s$	E'	$Y_r+Y_j+Y_m$	A_x 3.
Post - LOCA Flooding	1	D + L		P_L		E		A_x 3.

Ref.: SRP Section 3.8.2 Steel Containment

Notes

1. Encircled loads are those actually considered in the design per FSAR. When load factors different from those currently required were used, the factor used is also encircled.

2. This combination is stated in "Seismic Analysis & Stress Report for the Steel Vapor Container Structure Yankee Nuclear Power Station" by Earthquake Engineering Systems, Inc., dated March 2, 1979.

3. For purposes of the SEP Review, demonstration that structural integrity is maintained for load cases indicated above (per current criteria) may be considered as providing reasonable assurance that this structure meets the intent of current design criteria.

COMPARISON OF STRESS LIMITS

FOR

STEEL CONTAINMENT STRUCTURES

PLANT: YANKEE ROWE

SERVICE LEVEL	CURRENT CRITERIA (REF.: TABLE NE - 3221-1, ASME SECTION III, 1980) CRITERIA		VALUE, psi	DESIGN CRITERIA (REF.: ASME B&PV CODE SECTION VIII, 1956) CRITERIA		VALUE, psi
A	P_m	$1.0\ S_{mc}$	16,500			
	P_L	$1.5\ S_{mc}$	24,750			
	$P_L + P_b$	$1.5\ S_{mc}$	24,750			
	$P_L + P_b + Q$ (See note 6)	$3.0\ S_{mi}$	57,618			
B	P_m	$1.0\ S_{mc}$	16,500			
	P_L	$1.5\ S_{mc}$	24,750			
	$P_L + P_b$	$1.5\ S_{mc}$	24,750			
	$P_L + P_b + Q$ (See note 6)	$3.0\ S_{mi}$	57,618			
C	P_m	$1.2\ S_{mc}$ or $1.0\ S_y$	32,000			
	P_L	$1.8\ S_{mc}$ or $1.5\ S_y$	48,000			
	$P_L + P_b$ (See notes 3, 4 & 6)	$1.8\ S_{mc}$ or $1.5\ S_y$	48,000			
D	P_m	$1.0\ S_f$	33,660			
	P_L	$1.5\ S_f$	50,490			
	$P_L + P_b$ (See notes 2, 5 & 6)	$1.5\ S_f$	50,490	$(P_L$ or $P_m) + P_b + Q$ (see Note 8)	2.4S	36,000
POST-FLOODING CONDITION	P_m	$1.2\ S_{mc}$ or $1.0\ S_y$	32,000			
	P_L	$1.8\ S_{mc}$ or $1.5\ S_y$	48,000			
	$P_L + P_b$	$1.8\ S_{mc}$ or $1.5\ S_y$	48,000			
	$P_L + P_b + Q$ (See notes 4 & 6)	$3.0\ S_{mi}$	57,618			

SHELL MATERIAL (See Notes 7 & 9)	
SPEC. NO. SA201	GRADE: B to SA300
YIELD STRESS (S_y) - 32,000 psi	
ULT. STRENGTH (S_u) - 60,000 psi	
CURRENT PRIMARY STRESS INTENSITY LIMIT	S_{mc} = 16,500 psi; S_{mi} = 19,206 psi @ 249 °F (See note 1)
DESIGN PRIMARY MEMBRANE STRESS LIMIT	S = 15,000 psi @ 249 °F

NOTES:

1. NOTE THAT CURRENT PRIMARY STRESS INTENSITY LIMITS PRESUME (AMONG OTHER CODE QUALITY CONTROLS) MODERN COMPUTERIZED METHODS OF ANALYSIS. CONSEQUENTLY, CAUTION SHOULD BE OBSERVED IN MAKING DIRECT COMPARISONS WITH DESIGN STRESS LIMITS APPROPRIATE FOR LESS MODERN ANALYTICAL PROCEDURES.
2. THE COMPARABLE CURRENT CRITERIA ASSUMING ELASTIC METHODS WERE USED FOR THE ORIGINAL DESIGN ANALYSIS.
3. VALUES SHOWN PERTAIN TO INTEGRAL AND CONTINUOUS STRUCTURES ONLY.
4. THE LARGER OF THE TWO LIMITS IS APPLICABLE.
5. S_f IS 85% OF THE GENERAL PRIMARY MEMBRANE ALLOWABLE PERMITTED IN APPENDIX F OF SECTION III, ASME CODE.
6. IN ALL INSTANCES FATIGUE AND BUCKLING CRITERIA MUST ALSO BE SATISFIED.
7. IN ACCORDANCE WITH ASME III, DIV. 1, SUBSECT. NE, SUBPARA. NE 2121, THIS MATERIAL IS NOT LISTED AMONG THOSE CURRENTLY PERMITTED. REF: APPENDICES TABLE I-10.1 "CURRENT" STRESS VALUES LISTED ARE DERIVED USING $S_{mc} = 1.1 \times 1/4 \times S_u$, and S_{mi} @ 249°F FROM TABLE N-421 ASME B&PV CODE SECT. III, CLASS A, (1965).
8. REF.: "SEISMIC ANALYSIS & STRESS REPORT FOR THE STEEL VAPOR CONTAINER STRUCTURE YANKEE NUCLEAR POWER STATION" BY EARTHQUAKE ENGINEERING SYSTEMS, INC., DATED MARCH 2, 1979.
9. FSAR SECT. 6.2.1.2.1 STATES THAT "THE PLATE MATERIAL IS ASTM SPECIFICATION A-300, CLASS A-201, GRADE B..."

COMPARISON OF LOADING COMBINATION CRITERIA

CONCRETE STRUCTURES

PLANT: YANKEE ROWE

STRUCTURE: FUEL TRANSFER VAULT (concrete portion) (outside containment)

Combined Loading Cases	Gravity Dead, Live	Thermal	Pressure	Mechanical	Natural Phenomena	Impulsive Loading	Scale Ranking
1	1.4D + 1.7L						
2	1.4D + 1.7L				1.9E		
3	1.4D + 1.7L				1.7W		
4	.75 (1.4D + 1.7L)	.75 x 1.7 T_o		.75 x 1.7 R_o			
5	.75 (1.4D + 1.7L)	.75 x 1.7 T_o		.75 x 1.7 R_o	.75 x 1.9E		
6	.75 (1.4D + 1.7L)	.75 x 1.7 T_o		.75 x 1.7 R_o	.75 x 1.7W		
7	1.2D				1.9E		
8	1.2D				1.7W		
9	D + L	T_o		R_o	E'		
10	D + L	T_o		R_o	W_t		A_x
11	D + L	T_a	1.5 P_a	R_a			
12	D + L	T_a	1.25 P_a	R_a	1.25E	$Y_r + Y_j + Y_m$	
13	D + L	T_a	P_a	R_a	E'	$Y_r + Y_j + Y_m$	A_x

Ref.: SRP (1981) Sect. 3.8.4 Other Category I structures (concrete)

Notes

1. Ultimate strength method required by ACI-349 (1977).
2. Methods used in design { working stress ✓ ~~ultimate strength~~ consequently no load factors were used.
3. Loads deemed inapplicable or negligible struck from loading combinations.
4. Encircled loads are those actually considered in the design. When load factors different from those currently required were used, the factor used is also encircled.
5. For purposes of the SEP Review, demonstration that structural integrity is maintained for load cases 10 and 13 (per current criteria) may be considered as providing reasonable assurance that this structure meets the intent of current design criteria.

COMPARISON OF LOADING COMBINATION CRITERIA

STEEL STRUCTURES (Elastic Analysis)

PLANT: YANKEE ROWE

STRUCTURE: FUEL TRANSFER VAULT (steel portion) (outside containment)

Combined Loading Cases	Gravity Dead, Live	Thermal	Pressure	Mechanical	Natural Phenomena	Impulsive Loading	Scale
1	D + L						
2	D + L				E		
3	D + L				W		
4	D + L	T_o		R_o			
5	D + L	T_o		R_o	E		
6	D + L	T_o		R_o	W		
7	D + L	T_o		R_o	E'		
8	D + L	T_o		R_o	W_t		A_x
9	D + L	T_a	P_a	R_a			
10	D + L	T_a	P_a	R_a	E	$Y_j + Y_r + Y_m$	
11	D + L	T_a	P_a	R_a	E'	$Y_j + Y_r + Y_m$	A_x

Ref; SRP (1981) SECT. 3.8.4 Other Category I structures (steel)

Notes

1. Encircled loads are those actually considered in the design. When load factors are different from those currently required were used, the factor used is also encircled.

2. Loads deemed inapplicable or negligible struck from loading combinations.

3. For purposes of the SEP Review, demonstration that structural integrity is maintained for load cases 8 and 11 (per current criteria) may be considered as providing reasonable assurance that this structure meets the intent of current design criteria.

COMPARISON OF LOADING COMBINATION CRITERIA

CONCRETE STRUCTURES

PLANT: YANKEE ROWE

STRUCTURE: SAFETY INJECTION BUILDING (diesel/generator rooms) (concrete portion)

Combined Loading Cases	Gravity Dead, Live	Thermal	Pressure	Mechanical	Natural Phenomena	Impulsive Loading	Scale Ranking
1	1.4D + 1.7L						
2	1.4D + 1.7L				1.9E		
3	1.4D + 1.7L				1.7W		
4	.75 (1.4D + 1.7L)	.75 x 1.7 T_o		.75 x 1.7 R_o			
5	.75 (1.4D + 1.7L)	.75 x 1.7 T_o		.75 x 1.7 R_o	.75 x 1.9E		
6	.75 (1.4D + 1.7L)	.75 x 1.7 T_o		.75 x 1.7 R_o	.75 x 1.7W		
7	1.2D				1.9E		
8	1.2D				1.7W		
9	D + L	T_o		R_o	E'		
10	D + L	T_o		R_o	W_t		A_x
11	D + L	T_a	1.5 P_a	R_a			
12	D + L	T_a	1.25 P_a	R_a	1.25E	$Y_r + Y_j + Y_m$	
13	D + L	T_a	P_a	R_a	E'	$Y_r + Y_j + Y_m$	A_x

Ref.: SRP (1981) Sect. 3.8.4 Other Category I structures (concrete)

Notes

1. Ultimate strength method required by ACI-349 (1977).
2. Methods used in design { working stress, ~~ultimate strength~~ } consequently no load factors were used.
3. Loads deemed inapplicable or negligible struck from loading combinations.
4. Encircled loads are those actually considered in the design. When load factors different from those currently required were used, the factor used is also encircled.
5. For purposes of the SEP Review, demonstration that structural integrity is maintained for load cases 10 and 13(per current criteria) may be considered as providing reasonable assurance that this structure meets the intent of current design criteria.

COMPARISON OF LOADING COMBINATION CRITERIA

STEEL STRUCTURES (Elastic Analysis)

PLANT: YANKEE ROWE

STRUCTURE: SAFETY INJECTION BUILDING (diesel generator rooms) (steel framing)

Combined Loading Cases	Gravity Dead, Live	Thermal	Pressure	Mechanical	Natural Phenomena	Impulsive Loading	Scale
1	D + L						
2	D + L				E		
3	D + L				W		
4	D + L	T_o		R_o			
5	D + L	T_o		R_o	E		
6	D + L	T_o		R_o	W		
7	D + L	T_o		R_o	E'		
8	D + L	T_o		R_o	W_t		A_x
9	D + L	T_a	P_a	R_a			
10	D + L	T_a	P_a	R_a	E	$Y_j + Y_r + Y_m$	
11	D + L	T_a	P_a	R_a	E'	$Y_j + Y_r + Y_m$	A_x

Ref; SRP (1981) SECT. 3.8.4 Other Category I structures (steel)

Notes

1. Encircled loads are those actually considered in the design. When load factors are different from those currently required were used, the factor used is also encircled.

2. Loads deemed inapplicable or negligible struck from loading combinations.

3. For purposes of the SEP Review, demonstration that structural integrity is maintained for load cases 8 and 11 (per current criteria) may be considered as providing reasonable assurance that this structure meets the intent of current design criteria.

COMPARISON OF LOADING COMBINATION CRITERIA

CONCRETE STRUCTURES

PLANT: YANKEE ROWE

STRUCTURE: CONTROL ROOM
(Part of turbine building)

Combined Loading Cases	Gravity Dead, Live	Thermal	Pressure	Mechanical	Natural Phenomena	Impulsive Loading	Scale Ranking
1	1.4D + 1.7L						
2	1.4D + 1.7L				1.9E		
3	1.4D + 1.7L				1.7W		
4	.75 (1.4D + 1.7L)	.75 x 1.7 T_o		.75 x 1.7 R_o			
5	.75 (1.4D + 1.7L)	.75 x 1.7 T_o		.75 x 1.7 R_o	.75 x 1.9E		
6	.75 (1.4D + 1.7L)	.75 x 1.7 T_o		.75 x 1.7 R_o	.75 x 1.7W		
7	1.2D				1.9E		
8	1.2D				1.7W		
9	D + L	T_o		R_o	E'		
10	D + L	T_o		R_o	W_t		A_x
11	D + L	T_a	1.5 P_a	R_a			
12	D + L	T_a	1.25 P_a	R_a	1.25E	$Y_r + Y_j + Y_m$	
13	D + L	T_a	P_a	R_a	E'	$Y_r + Y_j + Y_m$	A_x

Ref.: SRP (1981) Sect. 3.8.4 Other Category I structures (concrete)

Notes

1. Ultimate strength method required by ACI-349 (1977).
2. Methods used in design { working stress ✓ consequently no load factors were used. ~~ultimate strength~~
3. Loads deemed inapplicable or negligible struck from loading combinations.
4. Encircled loads are those actually considered in the design. When load factors different from those currently required were used, the factor used is also encircled.
5. For purposes of the SEP Review, demonstration that structural integrity is maintained for load cases 10 and 13 (per current criteria) may be considered as providing reasonable assurance that this structure meets the intent of current design criteria.

COMPARISON OF LOADING COMBINATION CRITERIA

CONCRETE STRUCTURES

PLANT: YANKEE ROWE

STRUCTURE: BATTERY ROOMS (located in Auxiliary Bay of Turbine Building)

Combined Loading Cases	Gravity Dead, Live	Thermal	Pressure	Mechanical	Natural Phenomena	Impulsive Loading	Scale Ranking
1	1.4D + 1.7L						
2	1.4D + 1.7L				1.9E		
3	1.4D + 1.7L				1.7W		
4	.75 (1.4D + 1.7L)	.75 x 1.7 T_o		.75 x 1.7 R_o			
5	.75 (1.4D + 1.7L)	.75 x 1.7 T_o		.75 x 1.7 R_o	.75 x 1.9E		
6	.75 (1.4D + 1.7L)	.75 x 1.7 T_o		.75 x 1.7 R_o	.75 x 1.7W		
7	1.2D				1.9E		
8	1.2D				1.7W		
9	D + L	T_o		R_o	E'		
10	D + L	T_o		R_o	W_t		A_x
11	D + L	T_a	1.5 P_a	R_a			
12	D + L	T_a	1.25 P_a	R_a	1.25E	$Y_r + Y_j + Y_m$	
13	D + L	T_a	P_a	R_a	E'	$Y_r + Y_j + Y_m$	A_x

Ref.: SRP (1981) Sect. 3.8.4 Other Category I structures (concrete)

Notes

1. Ultimate strength method required by ACI-349 (1977).
2. Methods used in design { working stress ✓ consequently no load factors were used. ~~ultimate strength~~
3. Loads deemed inapplicable or negligible struck from loading combinations.
4. Encircled loads are those actually considered in the design. When load factors different from those currently required were used, the factor used is also encircled.
5. For purposes of the SEP Review, demonstration that structural integrity is maintained for load cases 10 and 13 (per current criteria) may be considered as providing reasonable assurance that this structure meets current design criteria.

COMPARISON OF LOADING COMBINATION CRITERIA

CONCRETE STRUCTURES

PLANT: YANKEE ROWE

STRUCTURE: PRIMARY AUXILIARY BUILDING

Combined Loading Cases	Gravity Dead, Live	Thermal	Pressure	Mechanical	Natural Phenomena	Impulsive Loading	Scale Ranking
1	1.4D + 1.7L						
2	1.4D + 1.7L				1.9E		
3	1.4D + 1.7L				1.7W		
4	.75 (1.4D + 1.7L)	.75 x 1.7 T_o		.75 x 1.7 R_o			
5	.75 (1.4D + 1.7L)	.75 x 1.7 T_o		.75 x 1.7 R_o	.75 x 1.9E		
6	.75 (1.4D + 1.7L)	.75 x 1.7 T_o		.75 x 1.7 R_o	.75 x 1.7W		
7	1.2D				1.9E		
8	1.[illegible]D				1.7W		
9	D + L	T_o		R_o	E'		
10	D + L	T_o		R_o	W_t		A_x
11	D + L	T_a	1.5 P_a	R_a			
12	D + L	T_a	1.25 P_a	R_a	1.25E	$Y_r + Y_j + Y_m$	
13	D + L	T_a	P_a	R_a	E'	$Y_r + Y_j + Y_m$	A_x

Ref.: SRP (1981) Sect. 3.8.4 Other Category I structures (concrete)

Notes

1. Ultimate strength method required by ACI-349 (1977).
2. Methods used in design { working stress ✓ consequently no load factors were used. ~~ultimate strength~~
3. Loads deemed inapplicable or negligible struck from loading combinations.
4. Encircled loads are those actually considered in the design. When load factors different from those currently required were used, the factor used is also encircled.
5. For purposes of the SEP Review, demonstration that structural integrity is maintained for load cases 10 and 13 (per current criteria) may be considered as providing reasonable assurance that this structure meets current design criteria.

COMPARISON OF LOADING COMBINATION CRITERIA

CONCRETE STRUCTURES

PLANT: YANKEE ROWE

STRUCTURE: SCREEN WELL HOUSE

Combined Loading Cases	Gravity Dead, Live	Thermal	Pressure	Mechanical	Natural Phenomena	Impulsive Loading	Scale Ranking
1	1.4D + 1.7L						
2	1.4D + 1.7L				1.9E		
3	1.4D + 1.7L				1.7W		
4	.75 (1.4D + 1.7L)	.75 x 1.7 T_o		.75 x 1.7 R_o			
5	.75 (1.4D + 1.7L)	.75 x 1.7 T_o		.75 x 1.7 R_o	.75 x 1.9E		
6	.75 (1.4D + 1.7L)	.75 x 1.7 T_o		.75 x 1.7 R_o	.75 x 1.7W		
7	1.2D				1.9E		
8	1.2D				1.7W		
9	D + L	T_o		R_o	E'		
10	D + L	T_o		R_o	W_t		A_x
11	D + L	T_a	1.5 P_a	R_a			
12	D + L	T_a	1.25 P_a	R_a	1.25E	$Y_r + Y_j + Y_m$	
13	D + L	T_a	P_a	R_a	E'	$Y_r + Y_j + Y_m$	A_x

Ref.: SRP (1981) Sect. 3.8.4 Other Category I structures (concrete)

Notes

1. Ultimate strength method required by ACI-349 (1977).
2. Methods used in design { working stress ✓ consequently no load factors were used. ~~ultimate strength~~
3. Loads deemed inapplicable or negligible struck from loading combinations.
4. Encircled loads are those actually considered in the design. When load factors different from those currently required were used, the factor used is also encircled.
5. For purposes of the SEP Review, demonstration that structural integrity is maintained for load cases 10 and 13 (per current criteria) may be considered as providing reasonable assurance that this structure meets current design criteria.

COMPARISON OF LOADING COMBINATION CRITERIA

CONCRETE STRUCTURES

PLANT: YANKEE ROWE

STRUCTURE: SPENT FUEL CHUTE

Combined Loading Cases	Gravity Dead, Live	Thermal	Pressure	Mechanical	Natural Phenomena	Impulsive Loading	Scale Ranking
1	1.4D + 1.7L						
2	1.4D + 1.7L				1.9E		
3	1.4D + 1.7L				1.7W		
4	.75 (1.4D + 1.7L)	.75 x 1.7 T_o		.75 x 1.7 R_o			
5	.75 (1.4D + 1.7L)	.75 x 1.7 T_o		.75 x 1.7 R_o	.75 x 1.9E		
6	.75 (1.4D + 1.7L)	.75 x 1.7 T_o		.75 x 1.7 R_o	.75 x 1.7W		
7	1.2D				1.9E		
8	1.2D				1.7W		
9	D + L	T_o		R_o	E'		
10	D + L	T_o		R_o	W_t		A_x
11	D + L	T_a	1.5 P_a	R_a			
12	D + L	T_a	1.25 P_a	R_a	1.25E	$Y_r + Y_j + Y_m$	
13	D + L	T_a	P_a	R_a	E'	$Y_r + Y_j + Y_m$	A_x

Ref.: SRP (1981) Sect. 3.8.4 Other Category I structures (concrete)

Notes

1. Ultimate strength method required by ACI-349 (1977).
2. Methods used in design { working stress ✓ consequently no load factors were used, ~~ultimate strength~~ }
3. Loads deemed inapplicable or negligible struck from loading comb' ations.
4. Encircled loads are those actually considered in the design. When load factors different from those currently required were used, the factor used is also encircled.
5. For purposes of the SEP Review, demonstration that structural integrity is maintained for load cases 10 and 13 (per current criteria) may be considered as providing reasonable assurance that this structure meets current design criteria.

COMPARISON OF LOADING COMBINATION CRITERIA

CONCRETE STRUCTURES

PLANT: YANKEE ROWE

STRUCTURE:
SPENT FUEL POOL CONCRETE

Combined Loading Cases	Gravity Dead, Live	Thermal	Pressure	Mechanical	Natural Phenomena	Impulsive Loading	Scale Ranking
1	1.4D + 1.7L						
2	1.4D + 1.7L				1.9E		
3	1.4D + 1.7L				1.7W		
4	.75 (1.4D + 1.7L)	.75 x 1.7 T_o		.75 x 1.7 R_o			
5	.75 (1.4D + 1.7L)	.75 x 1.7 T_o		.75 x 1.7 R_o	.75 x 1.9E		
6	.75 (1.4D + 1.7L)	.75 x 1.7 T_o		.75 x 1.7 R_c	.75 x 1.7W		
7	1.2D				1.9E		
8	1.2D				1.7W		
9	D + L	T_o		R_o	E'		
10	D + L	T_o		R_o	W_t		A_x
11	D + L	T_a	1.5 P_a	R_a			
12	D + L	T_a	1.25 P_a	R_a	1.25E	$Y_r + Y_j + Y_m$	
13	D + L	T_a	P_a	R_a	E'	$Y_r + Y_j + Y_m$	A_x

Ref.: SRP (1981) Sect. 3.8.4 Other Category I structures (concrete)

Notes

1. Ultimate strength method required by ACI-349 (1977).
2. Methods used in design { working stress ✓ consequently no load factors were used. ~~ultimate strength~~
3. Loads deemed inapplicable or negligible struck from loading combinations.
4. Encircled loads are those actually considered in the design. When load factors different from those currently required were used, the factor used is also encircled.
5. For purposes of the SEP Review, demonstration that structural integrity is maintained for load case 10, 13 (per current criteria) may be considered as providing reasonable assurance that this structure meets the intent of current design criteria.

COMPARISON OF LOADING COMBINATION CRITERIA

STEEL STRUCTURES (Elastic Analysis)

PLANT: YANKEE ROWE

STRUCTURE: PRIMARY VENT STACK

Combined Loading Cases	Gravity Dead, Live	Thermal	Pressure	Mechanical	Natural Phenomena	Impulsive Loading	Scale
1	D + L						
2	D + L				E		
3	D + L				W		
4	D + L	T_o		R_o			
5	D + L	T_o		R_o	E		3.
6	D + L	T_o		R_o	W		
7	D + L	T_o		R_o	E'		
8	D + L	T_o		R_o	W_t		A_x
9	D + L	T_a	P_a	R_a			
10	D + L	T_a	P_a	R_a	E	$Y_j + Y_r + Y_m$	
11	D + L	T_a	P_a	R_a	E'	$Y_j + Y_r + Y_m$	A_x

Ref; SRP (1981) SECT. 3.8.4 Other Category I structures (steel)

Notes

1. Encircled loads are those actually considered in the design. When load factors are different from those currently required were used, the factor used is also encircled.
2. Loads deemed inapplicable or negligible struck from loading combinations.
3. Licensee report that this load combination mav have also been considered.
4. For purposes of the SEP Review, demonstration that structural integrity is maintained for load cases 8 and 11 (per current criteria) may be considered as providing reasonable assurance that this structure meets current design criteria.

11. REVIEW FINDINGS

The most important findings of the review are summarized in this section in tabular form.

The major structural codes used for design of Seismic Category I buildings and structures for the Yankee Rowe Nuclear Power Station were:

1. AISC, "Specification for Design, Fabrication, and Erection of Structural Steel for Buildings," 1956

2. ACI 318-56, "Building Code Requirements for Reinforced Concrete," 1956

3. ASME Boiler and Pressure Vessel Code, Section VIII, 1956.

Each of these design codes has been compared with the corresponding structural code governing current licensing criteria. Tables follow, in the order listed above, summarizing important results of these comparisons for each code.

These tables provide:

1. identification by paragraph number (both of the original code and of its current counterpart) of code provisions where Scale A or Scale A_x deviations exist.

2. identification of structural elements to which each such provision may apply.

Some listed provisions may apply only to elements that do not exist in the Yankee Rowe structures. When it could be determined that this was the case, such provisions were struck from the list. Any provisions that appeared to be inapplicable for other reasons also were eliminated. Items so removed are listed in Appendix A to this report.

Access to further information concerning code provision changes is provided by additional appendixes. Each pair of codes (the design and the current ones) has a tabular summary within the report (Appendix B) which lists all code changes by scale ranking.

In addition, a separately bound appendix exists for each code pair. The appendix provides:

1. full texts of each revised provision in both the former and current versions

2. comments or conclusions, or both, relevant to the code change

3. the scale ranking of the change.

11.1 MAJOR FINDINGS OF AISC-1953 VS. AISC-1980 CODE COMPARISON

Franklin Research Center
A Division of The Franklin Institute

MAJOR FINDINGS OF AISC 1953 VS. AISC 1980 CODE COMPARISON

(Summary of Code Changes with the Potential to Significantly Degrade Perceived Margin of Safety)

Scale A

Referenced Subsection				
AISC 1980	AISC 1963	AISC 1953	Structural Elements Potentially Affected	Comments
1.5.1.2.2	--	--	Beam end connection where the top flange is coped and subject to shear, or failure by shear along a plane through fasteners or by a combination of shear along a plane through fasteners plus tension along a perpendicular plane	See case study 1 for details.
1.5.1.4.1	1.5.1.4.1	15(a)(3)	Rolled sections, plate girders and built up members.	New requirements added in the 1963 Code limiting the allowable stresses for tension due to bending.
1.5.1.4.1 Subpara. 7	1.5.1.4.1	--	Hollow circular sections subject to bending	New requirement in the 1980 Code
1.6	1.6	12(a)	Members subject to axial and bending stresses	New requirement for combined stresses added in the 1963 Code
1.8.3	1.8.3	16	Axially loaded compression members where sideway is not prevented	New requirements for slenderness ratio added in the 1963 Code

MAJOR FINDINGS OF AISC 1953 VS. AISC 1980 CODE COMPARISON

(Summary of Code Changes with the Potential to Significantly Degrade Perceived Margin of Safety)

Scale A (Cont.)

Referenced Subsection				
AISC 1980	AISC 1963	AISC 1953	Structural Elements Potentially Affected	Comments
1.9.1.2 and Appendix C	1.9.1	18(b)	Slender compression unstiffened elements subject to axial compression or compression due to bending when actual width-to-thickness ratio exceeds the values specified in subsection 1.9.1.2	New provisions added in the 1963 and the 1980 Code, Appendix C.
1.9.2.3 and Appendix C	--	--	Circular tubular elements subject to axial compression	New requirements added in the 1980 Code
1.10.4	1.10.4	26(d)	Partial length cover plates in plate girders and rolled beams	New requirements added in the 1963 Code
1.10.7	1.10.7	--	Plate girder web	New requirements for combined shear and tension stress added to the 1963 Code
1.10.10.2	1.10.10.2	26	Stiffeners for web plate girders	Change in the requirements of the 1953 Code

MAJOR FINDINGS OF AISC 1953 VS. AISC 1980 CODE COMPARISON

(Summary of Code Changes with the Potential to Significantly Degrade Perceived Margin of Safety)

Scale A (Cont.)

Referenced Subsection				
AISC 1980	AISC 1963	AISC 1953	Structural Elements Potentially Affected	Comments
1.11.1	1.11.1	13(a)	Composite construction	Limitation on effective width of concrete flange is introduced in the 1953 Code
1.11.4	1.11.4	13	Shear connectors in composite beams	New requirements added in the 1963 Code and the 1980 Code
1.11.5	--	--	Composite beams or girders with formed steel deck	New requirements added in the 1980 Code
1.14.2.2	--	--	Axially loaded tension members where the load is transmitted by bolts or rivets through some but not all of the cross-sectional elements of the members	New requirement added in the 1980 Code
1.14.6.1	1.14.7	15(f)	Effective throat thickness for partial penetration weld	
1.15.5.2 1.15.5.3 1.15.5.4	--	--	Restrained members when flange or moment connection plates for end connections of beams and girders are welded to the flange of I or H shaped columns	New requirement added in the 1980 Code

MAJOR FINDINGS OF AISC 1953 VS. AISC 1980 CODE COMPARISON

(Summary of Code Changes with the Potential to Significantly Degrade Perceived Margin of Safety)

Scale A (Cont.)

Referenced Subsection					
AISC 1980	AISC 1963	AISC 1953	Structural Elements Potentially Affected	Comments	Scale
1.18.3	1.18.3	28(b)	Built up members under tension	New requirement added in the 1963 Code	
2.9	2.8	--	Lateral bracing of members to resist lateral and torsional displacement	$0.0 < M/Mp < 1.0$ $0.0 > M/Mp > -1.0$ See case study 7 for details.	A C

11.2 MAJOR FINDINGS OF ACI 318-56 VS. ACI 349-76 CODE COMPARISON

MAJOR FINDINGS OF ACI 318-56 VS. ACI 349-76 CODE COMPARISON

(Summary of Code Changes with the Potential to Significantly Degrade Perceived Margin of Safety)

Scale A

Referenced Subsection				
ACI 349-76	ACI 318-63	ACI 318-56	Structural Elements Potentially Affected	Comments
7.10.3	805	--	Columns designed for stress reversals with variation of stress from f_y in compression to 1/2 f_y in tension	Splices of the main reinforcement in such columns must be reasonably limited to provide for adequate ductility under all loading conditions.
11.13	--	--	Short brackets and corbels which are primary load-carrying members	As this provision is new, any existing corbels or brackets may not meet these criteria and failure of such elements could be non-ductile type failure. Structural integrity may be seriously endangered if the design fails to fulfill these requirements.
11.15	--	--	Applies to any elements loaded in shear where it is inappropriate to consider shear as a measure of diagonal tension and the loading could induce direct shear-type cracks	Structural integrity may be seriously endangered if the design fails to fulfill these requirements.

MAJOR FINDINGS OF ACI 318-56 VS. ACI 349-76 CODE COMPARISON

(Summary of Code Changes with the Potential to Significantly Degrade Perceived Margin of Safety)

Scale A (Cont.)

Referenced Subsection				
ACI 349-76	ACI 318-63	ACI 318-56	Structural Elements Potentially Affected	Comments
11.16	--	--	All structural walls - those which are primary load carrying, e.g., shear walls and those which serve to provide protection from impacts of missile-type objects	Guidelines for these kinds of wall loads were not provided by older codes; therefore, structural integrity may be seriously endangered if the design fails to fulfill these requirements.
Chap. 12	Chap. 18	--	All	New chapter; old code did not have ultimate strength criteria for bond. This chapter presents some changes in bond stresses allowed and a change in philosophy. Allowable bond values are higher on small bars, but lower on large bars because of this shift in philosophy introduced by ultimate strength logic here.

MAJOR FINDINGS OF ACI 318-56 VS. ACI 349-76 CODE COMPARISON

(Summary of Code Changes with the Potential to Significantly Degrade Perceived Margin of Safety)

Scale A (Cont.)

Referenced Subsection ACI 349-76	ACI 318-63	ACI 318-56	Structural Elements Potentially Affected	Comments
Chapter 12 (cont.)				Splice lengths in column steel are the same as the 56 code and permissible bond stress for compression bars was set to match when reduced to working stress.
--	1301(c)	Table 305(a)	All	Allowable bond stresses are presented in the new code as a function of concrete strength and bar diameter. Values in the new code are higher for small diameter bars and lower for large diameter bars as compared to the old code. See case study (14).
Chap. 17	Chapter 25	--	Composite construction	New chapter; ACI 318-56 did not contain specific sections on composite concrete flexural members and composite construction.

MAJOR FINDINGS OF ACI 318-56 VS. ACI 349-76 CODE COMPARISON

(Summary of Code Changes with the Potential to Significantly Degrade Perceived Margin of Safety)

Scale A (Cont.)

Referenced Subsection				
ACI 349-76	ACI 318-63	ACI 318-56	Structural Elements Potentially Affected	Comments
Appendix A	--	--	All elements subject to time-dependent and position-dependent temperature variations and which are restrained such that thermal strains will result in thermal stresses	For structures subject to effects of pipe break, especially jet impingement, thermal stresses may be significant. Scale A for areas of jet impingement or where the conditions could develop causing concrete temperature to exceed limitations of A.4.2. For structures not subject to effects of pipe break accident, thermal stresses are unlikely to be significant (Scale B).
Appendix B	--	--	All steel embedments used to transmit loads from attachments into the reinforced concrete structures	New appendix; therefore, considerable review of older designs is warranted.**

**Since stress analysis associated with these conditions is highly dependent on definition of failure planes and allowable stress for these special conditions, past practice varied with designers' opinions. Stresses may vary significantly from those thought to exist under previous design procedures.

MAJOR FINDINGS OF ACI 318-56 VS. ACI 349-76 CODE COMPARISON

(Summary of Code Changes with the Potential to Significantly Degrade Perceived Margin of Safety)

Scale A (Cont.)

Referenced Subsection				
ACI 349-76	ACI 318-63	ACI 318-56	Structural Elements Potentially Affected	Comments
Appendix C	--	--	All elements whose failure under impulsive and impactive loads must be precluded	New appendix; therefore, considerations and review of older designs is considered important.**

**Since stress analysis associated with these conditions is highly dependent on definition of failure planes and allowable stress for these special conditions, past practice varied with designers' opinions. Stresses may vary significantly from those thought to exist under previous design procedures.

11.3 MAJOR FINDINGS OF ASME B&PV CODE COMPARISON, SECTION VIII, 1956 VS. SECTION III, SUBSECTION NE, 1980

ASME B&PV CODE COMPARISON
SECTION VIII, 1956 VS. SECTION III, SUBSECTION NE, 1980

(Summary of Code Changes with the Potential to Significantly Degrade Perceived Margin of Safety)

Scale A

Referenced Subsection				
Section III 1980	Section VIII 1962	Section VIII 1956	Structural Elements Potentially Affected	Comments
NE-3112.4	---	UG-5(b)	Plates, if under-strength	The 1956 Code permits conditional use of understrength plate, if: 1. The local allowable stress is correspondingly reduced; and 2. The UTS range is maintained. This practice has been terminated and current codes are blind to such situations in older structures. Scale A - if additional loads, not originally designed for, are required by current criteria. No scale ranking applicable otherwise.
NE-3112.4	UG-23	UG-23	Vessels of materials no longer listed as Code acceptable	Section III, 1980 Code references some materials which are identical to those referenced in Section VIII, 1956 Code. However, several materials which were referenced in Section VIII, are no longer listed in Section III, 1980. Justification of such use would be necessary to show equivalence to current requirements.

ASME B&PV CODE COMPARISON
SECTION VIII, 1956 VS. SECTION III, SUBSECTION NE, 1980

(Summary of Code Changes with the Potential to Significantly
Degrade Perceived Margin of Safety)

Scale A (Cont.)

Referenced Subsection				
Section III 1980	Section VIII 1962	Section VIII 1956	Structural Elements Potentially Affected	Comments
NE-3131	---	Various Paragraphs	Containment shells designed by formula	Section VIII, 1956 Code calls for the design of vessels by formula, while Section III, 1980 Code requires that the rules of Subsection NE-3200 (Design by Analysis) be satisfied. In the absence of substantial thermal or mechnical loads other tha[illegible]re, the rules of "De[illegible]ormula" may still be use[illegible] The [illegible]ing for containment shell[illegible]otential thermal or mec[illegible] other than pressure [illegible]s Scale B; otherwise [illegible] A.
NE-3133.5(a)	UG-29	UG-29	Stiffening rings for cylindrical shells subject to buckling loads.	The requirements [illegible] 1980 Code defining the minim[illegible]ent of inertia for stiffen[illegible]s as compared to the requir[illegible] of the 1956 Code may result [illegible] lower margin of safety. Scale $I_s' > 1.28\ I_s$ — C $I_s' > 1.22\ I_s$ — B $I_s' < 1.22\ I_s$ — A

ASME B&PV CODE COMPARISON
SECTION VIII, 1956 VS. SECTION III, SUBSECTION NE, 1980

(Summary of Code Changes with the Potential to Significantly Degrade Perceived Margin of Safety)

Scale A (Cont.)

Referenced Subsection				
Section III 1980	Section VIII 1962	Section VIII 1956	Structural Elements Potentially Affected	Comments
NE-3133.5(b)	---	---	Shell and stiffening rings of different materials.	This new insert in Section III of the 1980 Code requires using the material chart which gives the larger value of the factor A. This may result in a larger stiffening ring section needed to meet the requirements of the Code. Scale A for ring-stiffened shells where (1) the ring and the shell are of different materials and, in addition, (2) the "factor A" (as computed by the procedures of NE-3133.5) for the two materials differs by more than 6%; otherwise Scale B.
NE-3221.5	---	---	Containment components subject to cyclic loadings.	Requirements for fatigue analysis of vessels or parts which experience cyclic loadings are provided in Section III, Subsection NE, of the 1980 Code. No specific guidance was provided by Section VIII, 1956.

ASME B&PV CODE COMPARISON
SECTION VIII, 1956 VS. SECTION III, SUBSECTION NE, 1980

(Summary of Code Changes with the Potential to Significantly Degrade Perceived Margin of Safety)

Scale A (Cont.)

Referenced Subsection				
Section III 1980	Section VIII 1962	Section VIII 1956	Structural Elements Potentially Affected	Comments
NE-3325 Figs. (c) and (m)	UG34(d) Figs. (d) and (p)	UG-34(d) Figs. (b) and (a)	Unstayed flat heads and covers of the designs in the referenced figures	Present Code requires thicker plates.
NE-3327	UG-35	Footnote to UG-35	Quick-acting closures	Subsection NE, 1980 has expanded requirements for safety devices including: o positive interlocks on remotely operated doors o warning devices on manually operated doors o visibility of pressure indicators from operating floor.
NE-3334.1 NE-3334.2	UG-40(b) UG-40(c)	UG-40	Reinforcement for vessel openings	New requirements in the 1980 Code impose additional restrictions on metal that may be counted as reinforcement.
NE-3365	---	--	Bellows and bellows expansion joints	The 1980 Code imposes new design requirements.

12. SUMMARY

The table that follows provides a summary of the status of the findings from the Task III-7.B criteria comparison review of structural codes and loading requirements for Seismic Category I structures at the Yankee Rowe Nuclear Power Station.

The first and second columns of the table show the extent to which all Seismic Category I structures external to containment comply with current design criteria codes. The first column applies to the concrete portion of these structures; the second column applies to the portions which are of steel frame construction and the upper container supports. The third column applies only to the vapor container.

The salient feature of this table is the limited number of code change impacts requiring a Scale A ranking. Consequently, resolution, at the structural level, of potential concerns with respect to changes in structural code requirements appears, at least for the Yankee Rowe plant, to be an effort of tractable size.

SUMMARY

NUMBER OF CODE CHANGE IMPACTS FOR YANKEE ROWE CATEGORY I STRUCTURES

SCALE RANKING		ACI 318-56 VS. ACI 349-76	AISC 1953 VS. AISC 1980	ASME B&PV SEC. VIII, 1956 VS. SEC. III SUBSECT. NE, CLASS MC,1980
TOTAL CHANGES FOUND		113	50	30
Do Not Require Further Investigation	A or A_x not Applicable to Yankee Rowe	3 + 4*	11	4 + 3*
	B	84	13	10
	C	12	8	3
To Be Further Investigated	A	10	18	10
	A_x	0	0	0

SCALE RATINGS:

Scale A Change - The new criteria have the potential to substantially impair margins of safety as perceived under the former criteria.

Scale A_x Change - The impact of the code change on margins of safety is not immediately apparent. Scale A_x code changes require analytical studies of model structures to assess the potential magnitude of their effect upon margins of safety.

Scale B Change - The new criteria operate to impair margins of safety but not enough to cause engineering concern about the adequacy of any structural element.

Scale C Change - The new criteria will give rise to larger margins of safety than were exhibited under the former criteria.

---

*These changes are related to loads and load combinations. Loading criteria are addressed in Section 10. Consequently, to avoid duplication, such items are not counted in the above tabulation of code changes to be addressed under Section 11.

13. RECOMMENDATIONS

Potential concerns with respect to the ability of Seismic Category I buildings and structures in SEP plants to conform to current structural criteria are raised by the review at the code comparison level. These must ultimately be resolved by examination of individual as-built structures.

It is recommended that Yankee Atomic Electric Comapny be requested to take three actions:

1. Review individually all Seismic Category I structures at the Yankee Rowe plant to see if any of the structural elements listed in the following table occur in their designs. These are the structural elements for which a potential exists for margins of safety to be less than originally computed, due to criteria changes since plant design and construction. For structures which do incorporate these features, assess the actual impact of the associated code changes on margins of safety.

2. Reexamine the margins of safety of Seismic Category I structures under loads and load combinations which correspond to current criteria. Only those load combinations assigned a Scale A or Scale A_x rating in Section 10 of this report need be considered in this review. If the load combination includes individual loads which have themselves been ranked A or A_x, indicating that they do not conform to current criteria, update such loads.

   Full reanalysis of these structures is not necessarily required. Simple hand computations or appropriate modifications of existing results can qualify as acceptable means of demonstrating structural adequacy.

3. Review Appendix A of this report to confirm that all items listed there have no impact on safety margins at the Yankee Rowe plant.

LIST OF STRUCTURAL ELEMENTS TO BE EXAMINED

Structural Elements to be Examined	Code Change Affecting These Elements: New Code	Old Code	Scale
Beams/Columns	AISC 1980	AISC 1953	
Hollow circular sections subject to bending	1.5.1.4.1 Subpara 7	--*	A
Composite Construction	AISC 1980	AISC 1953	
1. Shear connectors in composite beams	1.11.4	13	A
2. Composite beams or girders with formed steel deck	1.11.5	--	A
3. Width of concrete flange - limitations	1.11.1	13(a)	A
Compression Elements	AISC 1980	AISC 1953	
1. With width-to-thickness ratio higher than specified in 1.9.1.2	1.9.1.2 and Appendix C	13(b)	A
2. Hollow circular sections subject to axial compression	1.9.2.3 and Appendix C	--	A
3. Members where sideway is not prevented	1.8.3	16	A
Tension Members	AISC 1980	AISC 1953	
1. When load is transmitted by bolts or rivets	1.14.2.2	--	A
2. Built up members	1.18.3	28(b)	A
Connections	AISC 1980	AISC 1953	
1. Beam ends with top flange coped, if subject to shear	1.5.1.2.2	--	A

*Double dash (--) indicates that older code had no provisions.

LIST OF STRUCTURAL ELEMENTS TO BE EXAMINED (Cont.)

Structural Elements to be Examined	Code Change Affecting These Elements: New Code	Old Code	Scale
2. Connections carrying moment or restrained member connection	1.15.5.2 1.15.5.3 1.15.5.4	--	A
Members Designed to Operate in an Inelastic Regime	AISC 1980	AISC 1953	
Spacing of lateral bracing	2.9	--	A
Rolled Sections and Built up Members	AISC 1980 --	AISC 1953 15(a)(3)	A
Partial length cover plates	1.10.4	26(d)	A
Members Subject to Axial and Bending Stresses	AISC 1980 1.6	AISC 1953 12(a)	A
Web Plate Girders	AISC 1980	AISC 1953	
1. Subject to shear and tension stresses	1.10.7	--	A
2. Stiffeners	1.10.10.2	26	A
Partial Penetration Weld Effective throat thickness	1.14.6.1	15(f)	A
Short Brackets and Corbels having a shear span-to-depth ratio of unity or less	ACI 349-76 11.13	ACI 318-56 --	A
Shear Walls used as a primary load-carrying member	ACI 349-76 11.16	ACI 318-56 --	A
Precast Concrete Structural Elements, where shear is not a measure of diagonal tension	ACI 349-76 11.15	ACI 318-56 --	A
Concrete Regions Subject to High Temperatures	ACI 349-76	ACI 318-56	
Time-dependent and position-dependent temperature variations	Appendix A	--	A

LIST OF STRUCTURAL ELEMENTS TO BE EXAMINED (Cont.)

Structural Elements to be Examined	Code Change Affecting These Element New Code	Old Code	Scale
All Structural Elements	ACI 349-76	ACI 318-56	
1. Ultimate bond strength	Chapter 12	--	A
2. Allowable bond stress	--	Table 305(a)	A
Columns with Spliced Reinforcement	ACI 349-76	ACI 318-56	
subject to stress reversals; f_y in compression to 1/2 f_y in tension	7.10.3	--	A
Steel Embedments used to transmit load to concrete	ACI 349-76 Appendix B	ACI 318-56 --	A
Element Subject to Impulsive and Impactive Loads whose failure must be precluded	ACI 349-76 Appendix C	ACI 318-56 --	A
Composite Construction	ACI 349-76 Chapter 17	ACI 318-56 --	A
Containment Vessels			
1. Plates, if understrength	ASME Sec. III, 1980 NE-3112.4	ASME Sec. VIII, 1956 UG-5(b)	A
2. Containment vessels of materials no longer listed as code acceptable	ASME Sec. III, 1980 NE-3112.4	ASME Sec. VIII, 1956 UG-23	A
3. Containment vessels designed by formula and subject to substantial thermal or mechanical loads	ASME Sec. III, 1980 NE-3131	ASME Sec. VIII, 1956 Various paragraphs	A
4. Stiffening rings for cylindrical shells subject to buckling loads	ASME Sec. III, 1980 NE-3133.5(a)	ASME Sec. VIII, 1956 UG-29	A

LIST OF STRUCTURAL ELEMENTS TO BE EXAMINED (Cont.)

Structural Elements to be Examined	Code Change Affecting These Elements: New Code	Old Code	Scale
5. Stiffening rings of material different than shell material	ASME Sec. III, 1980 NE-3133.5(b)	ASME Sec. VIII, 1956 --	A
6. Vessels with Quick Actuating Closures	ASME Sec. III, 1980 NE-3327.1	ASME Sec. VIII, 1956 Footnote to UG-35	A
Shell Openings and Attachments			
1. Unstayed flat heads and covers	ASME Sec. III, 1980 NE-3325 Figs. (c) and (m)	ASME Sec. VIII, 1956 UG-34(d) Figs. (b) and (a)	A
2. Openings and reinforcements; subject to cyclic loads	ASME Sec. III, 1980 NE-3331(b)	ASME Sec. VIII, 1956 --	A
3. Reinforcement for openings	ASME Sec. III, 1980 NE-3334.1, NE-3334.2	ASME Sec. VIII, 1956 UG-40	A
4. Bellows and bellows expansion joints	ASME Sec. III, 1980 NE-3365	ASME Sec. VIII, 1956 --	A
Roofs	---	--	A[1]

Extreme environmental snow loads are provided by SEP Topic II-2.A. NRC Regulatory Guide 1.102 (Position 3) provides guidance to preclude adverse consequences from ponding or parapet roofs. Failure of roofs not designed for such circumstances could generate impulsive loadings and water damage, possibly extending to Seismic Category I components of all floor levels.

1. Not shown in tabular summary of code change impacts.

14. REFERENCES

1. Standard Review Plan
NRC, Rev. 1, July 1981
NUREG-0800 (Formerly NUREG-75/087)

2. American Institute of Steel Construction, Inc.
Specification for Design, Fabrication, and Erection of Structural Steel for Buildings
New York, NY: 1963

3. American Concrete Institute
Building Code Requirements for Reinforced Concrete
Detroit MI: 1963
ACI 318-63

4. American Society of Mechanical Engineers
Boiler and Pressure Vessel Code, Section III, Subsection B
New York, NY: 1962

5. Yankee Rowe Nuclear Power Station
Final Safety Analysis Report
Volumes I, II, & III
8-Jan-74

6. Yankee Atomic Electric Company
Letter to NRC
Subject: Yankee Rowe Classification of Structures, Components, and Systems (Seismic and Quality)
31-Jan-79

7. Yankee Atomic Electric Company
Letter to NRC, with Attachments A to F
Subject: Request for Additional Information, Systematic Evaluation Program Structural Topics, Yankee Rowe Nuclear Power Plant
6-Sep-79

8. Appendix I to Technical Evaluation Report, "Design Codes, Design Criteria, and Loading Combinations"
Contains List of Basic Documents Defining Current Licensing Criteria for SEP Topic III-7.B
Franklin Research Center, 1981
TER-C5257-327

APPENDIX A

SCALE A AND SCALES A_x CHANGES
DEEMED INAPPROPRIATE TO YANKEE ROWE PLANT

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APPENDIX A-1

AISC 1953 VS. AISC 1980 CODE COMPARISON

(SCALE A AND SCALE A_x CHANGES DEEMED INAPPROPRIATE TO YANKEE ROWE PLANT
OR CODE CHANGES RELATED TO LOADS OR LOAD COMBINATIONS
AND THEREFORE TREATED ELSEWHERE)

AISC 1953 VS. AISC 1980 CODE COMPARISON

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Scale A

Referenced Subsection				
AISC 1980	AISC 1963	AISC 1953	Structural Elements Potentially Affected	Comments
1.5.1.1	1.5.1.1	--	Structural members under tension, except for pin connected members	Structural steel used in Yankee Rowe Cat. I structures is A-7. Thus, $F_y < 0.83\ F_u$ Therefore, Scale C for Yankee Rowe.

Limitations	Scale
$F_y \leq 0.833\ F_u$	C
$0.833\ F_u < F_y < 0.875\ F_u$	B
$F_y \geq 0.875\ F_u$	A

1.5.1.4.1 Subpara. 6	1.5.1.4.1	--	Box-shaped members (subject to bending) of rectangular cross section whose depth is not more than 6 times its width and whose flange thickness is not more than 2 times the web thickness New requirement in the 1980 Code	Box-shaped members not found to be used in Yankee Rowe Cat. I structures; therefore, not applicable
1.5.1.4.4	--	--	Lateral support requirements for box sections whose depth is larger than 6 times their width New requirement in the 1980 Code	Box section members not found to be used in Yankee Rowe Cat. I structures; therefore; not applicable

AISC 1953 VS. AISC 1980 CODE COMPARISON

Scale A (Cont.)

Referenced Subsection				
AISC 1980	AISC 1963	AISC 1953	Structural Elements Potentially Affected	Comments
1.5.2.2	1.7	11(b)	Rivets, bolts, and threaded parts subject to 20,000 cycles or more	Cat. I structures are not subject to such cyclic loading; therefore, not applicable
1.7 and Appendix	1.7	11	Members and connections subject to 20,000 cycles or more	Cat. I structures are not subject to such cyclic loading; therefore, not applicable
1.9.2.1 and Appendix C	1.9.2	18(c)	Stiffened Compression members	All structural steel is A-7, Fy < 40 ksi; therefore, Scale C
1.10.6	1.10.6	26	Hybrid girder - reduction in flange stress	All structural steel is A-7. No hybrid girders found in Yankee Rowe, therefore, not applicable.
1.13.3	--	--	Roof surface not provided with sufficient slope towards points of free drainage or adequate individual drains to prevent the accumulation of rain water (ponding)	

Scale A (Cont.)

Referenced Subsection				
AISC 1980	AISC 1963	AISC 1953	Structural Elements Potentially Affected	Comments
2.4 1st Para.	2.3 1st Para.	--	Slenderness ratio for columns. Must satisfy: $\frac{l}{r} \leq \frac{2\pi^2 E}{F_y}$ Scale: $F_y \leq 40$ ksi — C $40 \leq F_y < 44$ ksi — B $F_y \geq 44$ ksi — A	Scale C for Yankee Rowe. See case study 4 for details.
2.7	2.6	--	Flanges of rolled W, M, or S shapes and similar built-up single-web shapes subject to compression Scale: $F_y \leq 36$ ksi — C $36 \leq F_y < 38$ ksi — B $F_y \geq 38$ ksi — A	Scale C for Yankee Rowe. See case study 6 for details.
Appendix D	--	--	Web tapered members	New requirements added in the 1980 Code Web tapered member are not found to be used in Yankee Rowe Cat. I structures, therefore, not applicable

APPENDIX A-2

ACI 318-56 VS. ACI 349-76 CODE COMPARISON

(SCALE A AND SCALE A_x CHANGES DEEMED INAPPROPRIATE TO YANKEE ROWE PLANT
OR CODE CHANGES RELATED TO LOADS OR LOAD COMBINATIONS
AND THEREFORE TREATED ELESEWHERE)

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ACI 318-56 VS. ACI 349-76 SUMMARY OF CODE COMPARISON

Scale A

Referenced Subsection				
ACI 349-76	ACI 318-63	ACI 318-56	Structural Elements Potentially Affected	Comments
10.1 and 10.10	--	--	All primary load-carrying members.	Design loads here refer to Chapter 9 load combinations.*
11.1	--	--	All primary load-carrying members.	Design loads here refer to Chapter 9 load combinations.*
Chap. 18	Chapter 26	--	Prestressed concrete.	New chapter; ACI 318-56 did not contain specific sections or criteria for prestressed concrete. No prestressed elements outside primary containment; therefore, not applicable.
18.1.4 and 18.4.2	--	--	Prestressed concrete elements.	New load combinations here refer to Chapter 9 load combinations.* No prestressed elements outside containment; therefore, not applicable.
Chap. 19	--	--	Shell structures with thickness equal to or greater than 12 inches.	No concrete shell structure; therefore, not applicable.

*Special treatment of loads and loading combinations is addressed in other sections of the report.

ACI 318-56 VS. ACI 349-76 SUMMARY OF CODE COMPARISON

Scale A (Cont.)

Referenced Subsection ACI 349-76	ACI 318-63	ACI 318-56	Structural Elements Potentially Affected	Comments
3.5	405 (e),(f)	--	Prestressed elements.	New insert lists ASTM specifications for prestressing wire and strands. 318-56 did not have sections dealing with prestressed concrete. Controls other than ACI Codes or recommended practice would apply to this type of construction prior to 1963. No prestressed elements outside containment; therefore, not applicable.
Chap. 9 9.1, 9.2, & 9.3 most specifically	Chap. 15	A604	All primary load-carrying members or elements of the structural system are potentially affected.	Definition of new loads not normally used in design of traditional buildings and redefinition of load factors and capacity reduction factors has altered the traditional analysis requirements.*

*Special treatment of load and loading combinations is addressed in other sections of the report.

ACI 318-56 VS. ACI 349-76 SUMMARY OF CODE COMPARISON

Scale A (Cont.)

Referenced Subsection				
ACI 349-76	ACI 318-63	ACI 318-56	Structural Elements Potentially Affected	Comments
Chap. 19 (Cont.)	--	--	Shell Structures	This chapter is completely new; therefore, shell structures designed by the general criteria of older codes may not satisfy all aspects of this chapter. In addition, this chapter refers to Chapter 9 provisions.

APPENDIX A-3

ASME B&PV CODE COMPARISON

SECTION VIII, 1956, VS. SECTION III, SUBSECTION NE, 1980

(SCALE A OR SCALE A_x CHANGES DEEMED NOT APPLICABLE TO YANKEE ROWE PLANT
OR CODE CHANGES RELATED TO LOAD COMBINATIONS
AND THEREFORE TREATED ELSEWHERE)

ASME B&PV CODE COMPARISON
SECTION VIII, 1956 VS. SECTION III, SUBSECTION NE, 1980

(Scale A Changes Deemed Not Applicable or Changes Related to Loadings)

Scale A

Referenced Subsection				
Section III 1980	Section VIII 1962	Section VIII 1956	Structural Elements Potentially Affected	Comments
---	UG-25(d)	UG-25(d) UW-15(b)	Vessels containing telltale holes	The 1956 Code required telltale holes at reinforcing plates and saddles at nozzles to be left open. The 1962 Code permitted plugging. The removal of these provisions from Section III, 1962 Code, bans the use of telltale holes. Moreover, the more recent version of Section VIII specifically excludes using telltale holes for lethal substances.
NE-3324.3	UG-27(c)	UG-27(c)	Vessel components where welding efficiency of circumferential joints is less than half the longitudinal joint efficiency	The 1956 Code did not require computation of axial stress in cylindrical shells. The wide disparity in welding efficiencies is deemed improbable.
NE-3325	UG-34(c)3	UG-34(c)3	Heads, covers, or blind flanges on non-circular shape	The 1956 Code did not distinguish between circular and non-circular plates when specifying plate thickness. These thicknesses are now regarded as inadequate for most non-circular plates.

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ASME B&PV CODE COMPARISON
SECTION VIII, 1956 VS. SECTION III, SUBSECTION NE, 1980

(Scale A Changes Deemed Not Applicable or Changes Related to Loadings)

Scale A

Referenced Subsection				
Section III 1980	Section VIII 1962	Section VIII 1956	Structural Elements Potentially Affected	Comments
NE-2124(b)	UG-16(c)	UG-16(c)	Pressure retaining plates less than 0.167 in thick	The 1956 Code granted a blanket 0.010 in mill undertolerance on all plate. The present Code allows 6% or 0.010 in under-tolerance (whichever is least).
NE-3111	UG-22	UG-22	Loading as applied to load-carrying components*	Section III, 1980 Code, specifies additional loads to be considered in designing the vessel. These include: o dynamic head of liquids o snow loads and vibration loads o reaction to steam and water jet impingement
NE-3112.2	---	UG-20	Design temperature as applied to the vessel and its components*	The effects of internal heat generation due to radiation (in addition to all external sources) must be included in establishing design temperature.
NE-3112.3	---	---	Design mechanical loads as applied to the vessel and its components*	Currently, the design load combination includes mechanical loads. In 1956, the Code considered pressure at temperature only.

*Treatment of loads and load combinations is addressed in Section 10.

APPENDIX B

SUMMARIES OF CODE COMPARISON FINDINGS

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APPENDIX B-1

AISC 1953 VS. AISC 1980

SUMMARY OF CODE COMPARISON

(SYNTHESIS OF AISC 1953 VS. AISC 1963 VS. AISC 1980 CODE COMPARISONS)

AISC 1953 VS. AISC 1980
SUMMARY OF CODE COMPARISON

Scale A

Referenced Subsection: AISC 1980	AISC 1963	AISC 1953	Structural Elements Potentially Affected	Comments: Limitations	Scale
1.5.1.1	1.5.1.1	--	Structural members under tension, except for pin connected members	$F_y \le 0.833\ F_u$	C
				$0.833\ F_u < F_y < 0.875\ F_u$	B
				$F_y \ge 0.875\ F_u$	A
1.5.1.2.2	--	--	Beam end connection where the top flange is coped and subject to shear, failure by shear along a plane through fasteners, or shear and tension along and perpendicular to a plane through fasteners	See case study 1 for details.	
1.5.1.4.1	1.5.1.4.1	15(a)(3)	Rolled sections, plate girders and built up members.	New requirements added in the 1963 Code limiting the allowable stresses for tension due to bending.	
1.5.1.4.1 Subpara. 6	1.5.1.4.1	--	Box-shaped members (subject to bending) of rectangular cross section whose depth is not more than 6 times their width and whose flange thickness is not more than 2 times the web thickness	New requirement in the 1980 Code	

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AISC 1953 VS. AISC 1980
SUMMARY OF CODE COMPARISON

Scale A (Cont.)

Referenced Subsection				
AISC 1980	AISC 1963	AISC 1953	Structural Elements Potentially Affected	Comments
1.5.1.4.1 Subpara. 7	1.5.1.4.1	--	Hollow circular sections subject to bending	New requirement in the 1980 Code
1.5.1.4.4	--	--	Lateral support requirements for box sections whose depth is larger than 6 times their width	New requirement in the 1980 Code
1.5.2.2	1.7	11(b)	Rivets, bolts, and threaded parts subject to 20,000 cycles or more	Change in the requirements
1.6	1.6	12(a)	Members subject to axial and bending stresses	New requirement for combined stresses added in the 1963 Code
1.7 and Appendix B	1.7	11	Members and connections subject to 20,000 cycles or more	Change in the requirements
1.8.3	1.8.3	16	Axially loaded compression members where sideway is not prevented	New requirements for slenderness ratio added in the 1963 Code

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AISC 1953 VS. AISC 1980
SUMMARY OF CODE COMPARISON

Scale A (Cont.)

Referenced Subsection				
AISC 1980	AISC 1963	AISC 1953	Structural Elements Potentially Affected	Comments
1.9.1.2 and Appendix C	1.9.1	18(b)	Slender compression unstiffened elements subject to axial compression or compression due to bending when actual width-to-thickness ratio exceeds the values specified in subsection 1.9.1.2	New provisions added in the 1963 and the 1980 Code, Appendix C.
1.9.2.1 and Appendix C	1.9.2	18(c)	Stiffened compression members	New requirements added in the 1963 Code and the 1980 Code
1.9.2.3 and Appendix C	--	--	Circular tubular elements subject to axial compression	New requirements added in the 1980 Code
1.10.4	1.10.4	26(d)	Partial length cover plates in plate girders and rolled beams	New requirements added in the 1963 Code
1.10.6	1.10.6	26	Hybrid girder - reduction in flange stress	New requirement added in the 1980 Code. Hybrid girders were not covered in the 1963 Code. See case study 9 for details.

AISC 1953 VS. AISC 1980
SUMMARY OF CODE COMPARISON

Scale A (Cont.)

Referenced Subsection				
AISC 1980	AISC 1963	AISC 1953	Structural Elements Potentially Affected	Comments
1.10.7	1.10.7	--	Plate girder web	New requirements for combined shear and tension stress added to the 1963 Code
1.10.10.2	1.10.10.2	26	Stiffeners for web plate girders	Change in the requirements of the 1953 Code
1.11.1	1.11.1	13(a)	Composite construction	Limitation on effective width of concrete flange is introduced in the 1953 Code
1.11.4	1.11.4	13	Shear connectors in composite beams	New requirements added in the 1963 Code and the 1980 Code
1.11.5	--	--	Composite beams or girders with formed steel deck	New requirements added in the 1980 Code
1.13.3	--	--	Roof surface not provided with sufficient slope towards points of free drainage or adequate individual drains to prevent the accumulation of rain water (ponding)	

AISC 1953 VS. AISC 1980
SUMMARY OF CODE COMPARISON

Scale A (Cont.)

Referenced Subsection				
AISC 1980	AISC 1963	AISC 1953	Structural Elements Potentially Affected	Comments
1.14.2.2	--	--	Axially loaded tension members where the load is transmitted by bolts or rivets through some but not all of the cross-sectional elements of the members	New requirement added in the 1980 Code
1.14.6.1	1.14.7	15(f)	Effective throat thickness for partial penetration weld	
1.15.5.2 1.15.5.3 1.15.5.4	--	--	Restrained members when flange or moment connection plates for end connections of beams and girders are welded to the flange of I or H shaped columns	New requirement added in the 1980 Code
1.15.7	1.15.7	21(g)	Connections of tension and compression members in trusses	
1.18.3	1.18.3	28(b)	Built-up members under tension	New requirement added in the 1963 Code

AISC 1953 VS. AISC 1980
SUMMARY OF CODE COMPARISON

Scale A (Cont.)

Referenced Subsection					
AISC 1980	AISC 1963	AISC 1953	Structural Elements Potentially Affected	Comments	
2.4 1st Para.	2.3 1st Para.	--	Columns, Slenderness ratio for columns. Must satisfy: $\frac{l}{r} \leq \frac{2\pi^2 E}{F_y}$	See case study 4 for details.	Scale
				$F_y \leq 40$ ksi	C
				$40 \leq F_y < 44$ ksi	B
				$F_y \geq 44$ ksi	A
2.7	2.6	--	Flanges of rolled W, M, or S shapes and similar built-up single-web shapes subject to compression	See case study 6 for details.	Scale
				$F_y \leq 36$ ksi	C
				$36 \leq F_y < 38$ ksi	B
				$F_y \geq 38$ ksi	A
2.9	2.8	--	Lateral bracing of members to resist lateral and torsional displacement	See case study 7 for details.	
Appendix D	--	--	Web tapered members	New requirements added in the 1980 Code	

AISC 1953 VS. AISC 1980
SUMMARY OF CODE COMPARISON

Scale B

Referenced Subsection				
AISC 1980	AISC 1963	AISC 1953	Structural Elements Potentially Affected	Comments
1.9.2.2	1.9.2	--	Flanges of square and rectangular box sections of uniform thickness, of stiffened elements, when subject to axial compression or to uniform compression due to bending	The 1980 Code limit on width-to-thickness ratio of flanges is slightly more stringent than that of the 1963 Code.
1.10.1	--	--	Hybrid girders	Hybrid girders were not covered in the 1963 Code. Application of the new requirement could not be much different from other rational method.
1.10.5	1.10.5	26(e)	Intermediate stiffeners for plate girders and rolled beams	Change of in the requirements of the 1953 Code
1.11.4	1.11.4	--	Flat soffit concrete slabs, using rotary kiln produced aggregates conforming to ASTM C330	Lightweight concrete is not permitted in nuclear plants as structural members (Ref. ACI-349).
1.13.2	--	--	Beams and girders supporting large floor areas free of partitions or other source of damping, where transient vibration due to pedestrian traffic might not be acceptable	Lightweight construction not applicable to nuclear structures which are designed for greater loads

AISC 1953 VS. AISC 1980
SUMMARY OF CODE COMPARISON

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Scale B (Cont.)

Referenced Subsection				
AISC 1980	AISC 1963	AISC 1953	Structural Elements Potentially Affected	Comments
1.14.2	1.14.3	19(g)	Member with through hole	The 1963 Code specifies slightly more stringent requirements
1.14.6.1.3	--	--	Flare type groove welds when flush to the surface of the solid section of the bar	
1.15.5.5	--	--	Connections having high shear in the column web	New insert in the 1980 Code
1.15.11	1.15.11	--	Friction type joints	
1.16.4.2	1.16.4	--	Fasteners, minimum spacing, requirements between fasteners	
1.16.5	1.16.5	--	Structural joints, edge distances of holes for bolts and rivets	
2.3.1 2.3.2	--	--	Braced and unbraced multi-story frame - instability effect	Instability effect on short buildings will have negligible effect.
2.4	2.3	--	Members subject to combined axial and bending moments	Procedure used in the 1963 Code for the interaction analysis is replaced by a different procedure. See case study 8 for details.

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AISC 1953 VS. AISC 1980
SUMMARY OF CODE COMPARISON

Scale C

Referenced Subsection				
AISC 1980	AISC 1963	AISC 1953	Structural Elements Potentially Affected	Comments
1.3.3	1.3.3	--	Support girders and their connections - pendant operated traveling cranes	
			The 1963 Code requires 25% increase in live loads to allow for impact as applied to traveling cranes, while the 1980 Code requires 10% increase.	The 1963 Code requirement is more stringent, and, therefore, conservative.
1.5.1.3.1	1.5.1.3.1	15(a)(2)	Axially loaded members under compression	New requirements added the 1963 Code - See Case Study 15 for details
1.5.1.5.3	1.5.2.2	--	Bolts and rivets - bearing stress on projected area - in bearing type connections $F_p = 1.5\ F_u$ (1980 Code) $F_p = 1.35\ F_y$ (1963 Code)	New provisions added in the 1963 Code.
1.10.2	1.10.2	26(b)	Web girders and rolled beams	The requirements of the 1963 Code are more liberal

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AISC 1953 VS. AISC 1980
SUMMARY OF CODE COMPARISON

Scale C (Cont.)

Referenced Subsection				
AISC 1980	AISC 1963	AISC 1953	Structural Elements Potentially Affected	Comments
1.10.5.3	1.10.5.3	--	Stiffeners in girders - added spacing between stiffeners at end panels, at panels containing large holes, and at panels adjacent to panels containing large holes	New design concept in 1980 Code giving less stringent requirements. See case study 5 for details.
1.11.4	1.11.4	--	Continuous composite beams, where longitudinal reinforcing steel is considered to act compositely with the steel beam in the negative moment regions	New requirement added in the 1980 Code
1.14.5	1.14.6	19(g)	Pin Connected Members	
1.15.1	1.15.1	21(a)	Connections	More stringent requirements were specified in the 1953 Code.

APPENDIX B-2

ACI 318-56 VS. ACI 349-76

SUMMARY OF CODE COMPARISON

(SYNTHESIS OF ACI 318-56 VS. ACI 318-63 VS. ACI 349-76 CODE COMPARISONS)

ACI 318-56 VS. ACI 349-76
SUMMARY OF CODE COMPARISON

Scale A

Referenced Subsection ACI 349-76	ACI 318-63	ACI 318-56	Structural Elements Potentially Affected	Comments
3.5	405 (e),(f)	--	Prestressed elements	New insert lists ASTM specifications for prestressing wire and strands. 318-56 did not have sections dealing with prestressed concrete. Controls other than ACI Codes or recommended practice would apply to this type of construction prior to 1963.
7.10.3	805	--	Columns designed for stress reversals with variation of stress from f_y in compression to 1/2 f_y in tension	Splices of the main reinforcement in such columns must be reasonably limited to provide for adequate ductility under all loading conditions.
Chap. 9 9.1, 9.2, & 9.3 most specifically	Chap. 15	A604	All primary load-carrying members or elements of the structural system are potentially affected	Definition of new loads not normally used in design of traditional buildings and redefinition of load factors and capacity reduction factors has altered the traditional analysis requirements.*

*Special treatment of load and loading combinations is addressed in other sections of the report.

ACI 318-56 VS. ACI 349-76
SUMMARY OF CODE COMPARISON

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Scale A (Cont.)

Referenced Subsection				
ACI 349-76	ACI 318-63	ACI 318-56	Structural Elements Potentially Affected	Comments
10.1 and 10.10	--	--	All primary load-carrying members	Design loads here refer to Chapter 9 load combinations.*
11.1	--	--	All primary load-carrying members	Design loads here refer to Chapter 9 load combinations.*
11.13	--	-	Short brackets and corbels which are primary load-carrying members	As this provision is new, any existing corbels or brackets may not meet these criteria and failure of such elements could be non-ductile type failure. Structural integrity may be seriously endangered if the design fails to fulfill these requirements.
11.15	--	--	Applies to any elements loaded in shear where it is inappropriate to consider shear as a measure of diagonal tension and the loading could induce direct shear-type cracks	Structural integrity may be seriously endangered if the design fails to fulfill these requirements.

*Special treatment of load and loading combinations is addressed in other sections of the report.

ACI 318-56 VS. ACI 349-76
SUMMARY OF CODE COMPARISON

Scale A (Cont.)

Referenced Subsection				
ACI 349-76	ACI 318-63	ACI 318-56	Structural Elements Potentially Affected	Comments
11.16	--	--	All structural walls - those which are primary load carrying, e.g., shear walls and those which serve to provide protection from impacts of missile-type objects	Guidelines for these kinds of wall loads were not provided by older codes; therefore, structural integrity may be seriously endangered if the design fails to fulfill these requirements.
Chapter 12	Chapter 18	--	All	New chapter; old code did not have ultimate strength criteria for bond. This chapter presents some changes in bond stresses allowed and a change in philosophy. Allowable bond values are higher on small bars, but lower on large bars because of this shift in philosophy introduced by ultimate strength logic here. Splice lengths in column steel are the same as the 56 code and permissible bond stress for compression bars was set to match when reduced to working stress.

ACI 318-56 VS. ACI 349-76
SUMMARY OF CODE COMPARISON

Scale A (Cont.)

Referenced Subsection ACI 349-76	ACI 318-63	ACI 318-56	Structural Elements Potentially Affected	Comments
--	1301(c)	Table 305(a)	All	Allowable bond stresses are presented in the new code as a function of concrete strength and bar diameter. Values in the new code are higher for small diameter bars and lower for large diameter bars as compared to the old code. See case study (14).
Chap. 17	Chapter 25	--	Composite construction	New chapter; ACI 318-56 did not contain specific sections on composite concrete flexural members and composite consruction.
Chap. 18	Chapter 26	--	Prestressed concrete	New chapter; ACI 318-56 did not contain specific sections or criteria for prestressed concrete.
18.1.4 and 18.4.2	--	--	Prestressed concrete elements	New load combinations here refer to Chapter 9 load combinations.*
Chap. 19	--	--	Shell structures with thickness equal to or greater than 12 inches	This chapter is completely new; therefore, shell structures designed by the general

*Special treatment of loads and loading combinations is addressed in other sections of the report.

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Scale A (Cont.)

Referenced Subsection				
ACI 349-76	ACI 318-63	ACI 318-56	Structural Elements Potentially Affected	Comments
Chap. 19 (Cont.)				criteria of older codes may not satisfy all aspects of this chapter. Additionally, this chapter refers to Chapter 9 provisions.
Appendix A	--	--	All elements subject to time-dependent and position-dependent temperature variations and which are restrained such that thermal strains will result in thermal stresses	New appendix; older did not give specific guidelines on short-term temperature limits for concrete. The possible effects of strength loss in concrete at high temperatures should be assessed. Scale A for any accident temperature or other thermal condition exceeding limits of paragraph A.4.2.
Appendix B	--	--	All steel embedments used to transmit loads from attachments into the reinforced concrete structures	New appendix; therefore, considerable review of older designs is warranted.**

**Since stress analysis associated with these conditions is highly dependent on definition of failure planes and allowable stress for these special conditions, past practice varied with designers' opinions. Stresses may vary significantly from those thought to exist under previous design procedures.

ACI 318-56 VS. ACI 349-76
SUMMARY OF CODE COMPARISON

Scale A (Cont.)

Referenced Subsection				
ACI 349-76	ACI 318-63	ACI 318-56	Structural Elements Potentially Affected	Comments
Appendix C	--	--	All elements whose failure under impulsive and impactive loads must be precluded	New appendix; therefore, considerations and review of older designs is considered important.**

**Since stress analysis associated with these conditions is highly dependent on definition of failure planes and allowable stress for these special conditions, past practice varied with designers' opinions. Stresses may vary significantly from those thought to exist under previous design procedures.

ACI 318-56 VS. ACI 349-76
SUMMARY OF CODE COMPARISON

Scale B

Referenced Subsection ACI 349-76	ACI 318-63	ACI 318-56	Structural Elements Potentially Affected	Comments
1.3.2	103(b)	--	Ambient temperature control for concrete inspection - upper limit reduced 5° (from 100°F to 95°F) applies to all structural concrete	Tighter control to ensure adequate control of curing environment for cast-in-place concrete.
1.5	--	--	Requirement of a "Quality Assurance Program" is new. Applies to all structural concrete	Previous codes required inspection but not the establishment of a quality assurance program.
Chap. 3	Chap. 4	Chap. 2	Any elements containing steel with $f_y > 60,000$ psi or lightweight concrete	Use of lightweight concrete in a nuclear plant not likely. Elements containing steel with $f_y > 60,000$ psi may have inadequate ductility or excessive deflections at service loads.
--	1208	--	Elements where lightweight concrete was used.	Probably does not apply to nuclear structures.

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Scale B (Cont.)

Referenced Subsection				
ACI 349-76	ACI 318-63	ACI 318-56	Structural Elements Potentially Affected	Comments
3.2	402	205	Cement	This serves to clarify intent of previous code.
3.3	403	206	Aggregate	Eliminated reference to lightweight aggregate.
3.3.1	403	206	Any structural concrete covered by ACI 349-76 and expected to provide for radiation shielding in addition to structural capacity	Controls of ASTM C567, "Standard Specifications for Aggregates for Radiation Shielding Concrete," closely parallel those for ASTM C33, "Standard Specification for Concrete Aggregates."
3.3.3	403	206	Aggregate	To ensure adequate control.
3.4.2	404	207	Water for concrete	Improve quality control measures.
3.5	405	208	Metal reinforcement	Removed all reference to steel with f_y > 60,000 psi.
3.5.1	405(a)	--	Reinforcing bar welds	Older code did not reference A.W.S. literature, but specific jobs that allowed welding of reinforcing bars normally listed requirements in the job specifications.

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ACI 318-56 VS. ACI 349-76
SUMMARY OF CODE COMPARISON

Scale B (Cont.)

Referenced Subsection				
ACI 349-76	ACI 318-63	ACI 318-56	Structural Elements Potentially Affected	Comments
3.6	406, 407, & 408	--	Concrete admixtures	Added requirements to improve quality control.
3.6.3 & 3.6.4	407 & 408	--	Concrete where admixtures were used	Extensive use of these admixtures before 1963 was not common.
4.1 & 4.2	501 & 502	302 & 303	Concrete proportioning	Proportioning logic improved to account for statistical variation and statistical quality control.
4.2.5 & 4.2.7	501(c) & 501(d)	--	Concrete exposed to freezing or chemically aggressive environments	Past practice used other sources to guide designs in chemically aggressive environments.
4.3	504	304	Evaluation and acceptance of concrete	Added provision to allow for design specified strength at age > 28 days to be used. Not considered to be a problem, since large cross sections will allow concrete in place to continue to hydrate.

ACI 318-56 VS. ACI 349-76
SUMMARY OF CODE COMPARISON

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Scale B (Cont.)

Referenced Subsection ACI 349-76	ACI 318-63	ACI 318-56	Structural Elements Potentially Affected	Comments
4.3.3	504(c)	304(c)	Concrete quality control	Changed to separate quality control on strength for working stress and ultimate strength. Control for working stress in new code made somewhat more conservative.
--	505	--	Lightweight concrete	New section added for lightweight aggregate concrete diagonal tension control. Old code did not specify this parameter.
5.7	607	--	Curing of very large concrete elements and control of hydration temperature	Attention to this is required because of the thicker elements encountered in nuclear-related structures.
6.3.3	--	--	All structural elements with embedded piping containing high tempera-ture materials in excess of 150°F, or 200°F in localized areas not insulated from the concrete	Previous codes did not address the problem of long periods of exposure to high temperature and did not provide for reduction in design allowables to account for strength reduction at high (>150°F) temperatures.

Scale B (Cont.)

Referenced Subsection				
ACI 349-76	ACI 318-63	ACI 318-56	Structural Elements Potentially Affected	Comments
7.5.5.1	805(d)	1103 (c)(3)	Welded splices	Welded splice requirement is more conservative as the 56 Code only required splices in compression to develop 100% of yield. Design allowables were reasonably below yield. This is not considered critical.
7.5, 7.6, & 7.8	805	506, 1002(d), 1103(c)	Members with spliced reinforcing steel	Sections on splicing and tie requirements amplified to better control strength at splice locations and provide ductility.
7.8.1 & 7.8.2	805(f)	--	Elements which used welded wire fabric as main reinforcement	This type of reinforcement not generally used in large structures and main structural elements; therefore, not considered a problem.
7.9	805	--	Members containing deformed wire fabric	New sections to define requirements for this new material.
7.10 & 7.11	--	--	Connection of primary load-carrying members and at splices in column steel	To ensure adequate ductility.

ACI 318-56 VS. ACI 349-76
SUMMARY OF CODE COMPARISON

Scale B (Cont.)

Referenced Subsection ACI 349-76	ACI 318-63	ACI 318-56	Structural Elements Potentially Affected	Comments
7.12.3 & 7.12.4	--	--	Lateral ties in columns	To provide for adequate ductility.
7.13.1 through 7.13.3	--	--	Reinforcement in exposed concrete	New requirements to conform with the expected large thicknesses in nuclear-related structures.
8.6	--	--	Continuous nonprestressed flexural members.	Allowance for redistribution of negative moments has been redefined as a function of the steel percentage.
9.2	1504(b)	--	All	Concept of a capacity reduction factor ϕ applied to the ultimate strength equations is new. This in a way replaces the old code use of different load factors for different structural elements.
9.3.1 & 9.3.2	1506	A604	All	Load factors have changed - also the use of different load factors for different structural elements was dropped. These changes have been offset by the introduction of the capacity reduction factor; therefore, overall effect not critical.

ACI 318-56 VS. ACI 349-76
SUMMARY OF CODE COMPARISON

Scale B (Cont.)

Referenced Subsection				
ACI 349-76	ACI 318-63	ACI 318-56	Structural Elements Potentially Affected	Comments
9.4	1505	A603(c)	Reinforcing steel - design strength limitation	See comments in Chapter 3 summary.
9.5.1.1	--	--	Reinforced concrete members subject to bending - deflection limits	Allows for more stringent controls on deflection in special cases.
9.5.1.2 through 9.5.1.4	--	--	Slab and beams - minimum thickness requirements	Minimum thickness generally would not control this type of structure.
9.5.2.4	909	--	Beams and one-way slabs	New section on control of deflections needed because of use of new high strength steels and concrete. Will, generally, not be a problem in structures carrying heavy loads as minimum. thickness would not control.
9.5.3	--	--	Nonprestressed two-way construction	Immediate and long time deflections generally not critical in structures designed for very large live loadings; however, design by ultimate strength requires more attention to deflection controls.

Scale B (Cont.)

Referenced Subsection				
ACI 349-76	ACI 318-63	ACI 318-56	Structural Elements Potentially Affected	Comments
9.5.4 & 9.5.5	--	--	Prestressed concrete members	Control of camber, both initial and long time in addition to service load deflection, requires more attention for designs by ultimate strength.
10.2.7	--	--	Flexural members - new limit on B factor	Lower limit on B of 0.65 would correspond to an f'_c of 8,000 psi. No concrete of this strength likely to be found in a nuclear structure.
10.3.6	--	--	Compression members, with spiral reinforcement or tied reinforcement, non-prestressed and prestressed	Limits on axial design load for these members given in terms of design equations. See case study 2
10.3.6	Chapter 19	A600	Columns	The introduction of the capacity reduction factor ϕ viewed alone would significantly effect the ultimate design code results; however, the introduction of lower load factors at the same time minimizes the effect. Sample calculations show reasonable parity between safety margins with the older code being generally more conservative.

ACI 318-56 VS. ACI 349-76
SUMMARY OF CODE COMPARISON

Scale B (Cont.)

Referenced Subsection ACI 349-76	ACI 318-63	ACI 318-56	Structural Elements Potentially Affected	Comments
10.6.1 through 10.6.4	1508	A604(a)	Beams and one-way slabs	Changes in distribution of reinforcement for crack control.
10.6.5	--	--	Beams	New insert
10.7	910	--	Deep beams	Older code did not address "deep beams" as a specific case.
10.11	916	1107	Long columns	For long columns, h/t limit removed and a new strength reduction logic, which includes factors such as resistance to lateral displacement of the ends and mode of curvature in the formulation, replaces load reduction based on h/t. The old code designs were generally conservative and long slender columns were not allowed.
10.11.1 through 10.11.7 & 10.12	915 & 916	1107	Compression members, slenderness effects	For slender columns, moment magnification concept replaces the so-called strength reduction concept, but for the limits stated in ACI 318-63 both methods yield equal accuracy and both are acceptable methods.

ACI 318-56 VS. ACI 349-76
SUMMARY OF CODE COMPARISON

Scale B (Cont.)

Referenced Subsection				
ACI 349-76	ACI 318-63	ACI 318-56	Structural Elements Potentially Affected	Comments
--	1102(c)	--	Flexural elements which contain compression steel	New requirements defined for computing the compression steel contribution to the transformed area. This was to account for stress increase which results from creep. Will not be significant where design dead load is not a large part of the design load.
10.15.1 through 10.15.6	1404 through 1406	--	Composite compression members	New items - no way to compare; ACI 318-63 contained only working stress method of design for these members.
10.17	--	--	Massive concrete members, more than 48 in thick	New item - no comparison.
--	1407	1109	Columns	Both codes use interaction logic; however, new code working stress interaction diagram is derived from the ultimate strength diagram. The definition of the tension controlled region changes since balanced eccentricity is the new limit as opposed to the old "Kern" definition.

ACI 318-56 VS. ACI 349-76
SUMMARY OF CODE COMPARISON

Scale B (Cont.)

Referenced Subsection				
ACI 349-76	ACI 318-63	ACI 318-56	Structural Elements Potentially Affected	Comments
-- (Cont.)				Comparison is complex but in general it is probable that the old code is more conservative.
11.2.1 & 11.2.2	--	--	Concrete flexural members	For nonprestressed members, concept of minimum area of shear reinforcement is new. For prestressed members, Eqn. 11-2 is the same as in ACI 318-63. Requirement of minimum shear reinforcement provides for ductility and restrains inclined crack growth in the event of unexpected loading.
11.3	Chapter 17	--	All	This chapter is completely new; previous codes did not contain ultimate strength design criteria for shear and diagonal tension.
11.7 through 11.8.6	--	--	Nonprestressed members	Detailed provisions for this load combination were not part of ACI 318-63. These new sections provide a conservative logic which

ACI 318-56 VS. ACI 349-76
SUMMARY OF CODE COMPARISON

Scale B (Cont.)

Referenced Subsection ACI 349-76	ACI 318-63	ACI 318-56	Structural Elements Potentially Affected	Comments
11.7 through 11.8.6 (Cont.)				requires that the steel needed for torsion be added to that required for transverse shear, which is consistent with the logic of ACI 318-63. This is not considered to be critical, as ACI 318-63 required the designer to consider torsional stresses; assuming that some rational method was used to account for torsion, no problem is expected to arise.
11.9 through 11.9.6	--	--	Deep beams	Special provisions for shear stresses in deep beams are new. The minimum steel requirements are similar to the ACI 318-63 requirements of using the wall steel limits. Deep beams designed under previous ACI 318-63 criterion were reinforced as walls at the minimum and therefore no unreinforced section would have resulted.

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Scale B (Cont.)

Referenced Subsection				
ACI 349-76	ACI 318-63	ACI 318-56	Structural Elements Potentially Affected	Comments
11.10 through 11.10.7	--	--	Slabs and footings	New provision for shear reinforcement in slabs or footings for the two-way action condition and new controls where shearhead reinforcement is used. Logic consistent with ACI 318-63 for these conditions and change is not considered major.
--	1207	808-809	Slabs and footings	Shear stress logic for working stress design in ACI 318-63 was developed by applying a factor of 2 to the ultimate strength logic. In slabs and footings, the critical section for shear was defined at a distance d/2 (not d) from the face of the support or column. Allowable stresses in the new code are larger; however, overall differences are not great in the final design.
--	2101(e)(2)	--	Slabs	New section added to give a specific method of defining the effect of a slab opening on the critical section around a column.

ACI 318-56 VS. ACI 349-76
SUMMARY OF CODE COMPARISON

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Scale B (Cont.)

Referenced Subsection ACI 349-76	ACI 318-63	ACI 318-56	Structural Elements Potentially Affected	Comments
--	1604	--	Members with nonsymmetrical cross sections	Old code did not address this problem. Old designs generally done by very conservative assumptions.
11.11.1	1707	--	Slabs and footings	The change which deletes the old requirement that steel be considered as only 50% effective and allows concrete to carry 1/2 the allowable for two-way action is new. Also deleted was the requirement that shear reinforcement not be considered effective in slabs less than 10 in thick. Change is based on recent research which indicates that such reinforcement works even in thin slabs.
11.11.2 through 11.11.2.5	--	--	Slabs	Details for the design of shearhead is new. ACI 318-63 had no provisions for shearhead design. The requirements in this section for slabs and footings are not likely to have been used in older plant designs. If such devices were used, it is assumed a rational design method was used.

Scale B (Cont.)

Referenced Subsection ACI 349-76	ACI 318-63	ACI 318-56	Structural Elements Potentially Affected	Comments
11.12	--	--	Openings in slabs and footings	Modification for inclusion of shearhead design. See above conclusion.
11.13.1 & 11.13.2	--	--	Columns	No problem anticipated since previous code required design consideration by some analysis.
Chap. 12	--	--	Reinforcement	Development length concept replaces bond stress concept in ACI 318-63. The various l_d lengths in this chapter are based entirely on ACI 318-63 permissible bond stresses. There is essentially no difference in the final design results in a design under the new code compared to ACI 318-63.
12.1.6 through 12.1.6.3	918(C)	--	Reinforcement	Modified with minimum added to ACI 318-63, 918(C).
12.2.2 & 12.2.3	--	--	Reinforcement	New insert in ACI 349-76.

ACI 318-56 VS. ACI 349-76
SUMMARY OF CODE COMPARISON

Scale B (Cont.)

Referenced Subsection ACI 349-76	ACI 318-63	ACI 318-56	Structural Elements Potentially Affected	Comments
12.4	--	--	Reinforcement of special members	New insert. Gives emphasis to special member consideration.
12.8.1 & 12.8.2	--	--	Standard hooks	Based on ACI 318-63 bond stress allowables in general; therefore, no major change.
12.10.1 & 12.10.2(b)	--	--	Wire fabric	New insert. Use of such reinforcement not likely in Category I structures for nuclear plants.
12.11.2	--	--	Wire fabric	New insert. Mainly applies to precast prestressed members.
12.11	918	--	Beams	Tensile steel cut off conditions are new. Older design practice did not terminate bars in high tension zones and generally bent up bars where not needed.
12.13.1.4	--	--	Wire fabric	New insert. Use of this material

Scale B (Cont.)

Referenced Subsection ACI 349-76	ACI 318-63	ACI 318-56	Structural Elements Potentially Affected	Comments
12.13.1.4 (Cont.)				for stirrups not likely in heavy members of a nuclear plant.
13.2.4	2102(g)	--	Slabs	New section added to ensure moment transfer between supports and the slab.
13.5	--	--	Slab reinforcement	New details on slab reinforcement intended to produce better crack control and maintain ductility. Past practice was not inconsistent with this in general.
14.2	--	--	Walls with loads in the Kern area of the thickness	Change of the order of the empirical equation (14-1) makes the solution compatible with Chapter 10 for walls with loads in the Kern area of the thickness.
15.5	--	--	Footings - shear and development of reinforcement	Changes here are intended to be compatible with change in concept of checking bar development instead of nominal bond stress consistent with Chapter 12.

ACI 318-56 VS. ACI 349-76
SUMMARY OF CODE COMPARISON

Scale B (Cont.)

Referenced Subsection ACI 349-76	ACI 318-63	ACI 318-56	Structural Elements Potentially Affected	Comments
15.5	2305(d)	1205(e)	Footings	Removal of the 85% shear used to compute tensile reinforcement bond in two-way reinforced footings; now 100% shear is required.
15.9	--	--	Minimum thickness of plain footing on piles	Reference to minimum thickness of plain footing on piles which was in ACI 318-63 was removed entirely.
16.2	--	--	Design considerations for a structure behaving monolithically or not, as well as for joints and bearings.	New but consistent with the intent of previous code.
17.5.3	2505	--	Horizontal shear stress in any segment	Use of Nominal Average Shear Stress equation (17-1) replaces the theoretical elastic equation (25-1) of ACI 318-63. It makes design computations easier.
18.4.1	--	--	Concrete immediately after prestress transfer	Change allows more tension, thus is less conservative but not considered a problem.

ACI 318-56 VS. ACI 349-76
SUMMARY OF CODE COMPARISON

Scale B (Cont.)

Referenced Subsection				
ACI 349-76	ACI 318-63	ACI 318-56	Structural Elements Potentially Affected	Comments
18.5	2606	--	Tendons (steel)	Augmented to include yield and ultimate in the jacking force requirement.
18.7.1	--	--	Bonded and unbonded members	Eqn. 18-4 is based on more recent test data.
18.9.1 18.9.2 18.9.3	--	--	Two-way flat plates (solid slabs) having minimum bonded reinforcement	Intended primarily for control of cracking.
18.11.3 18.11.4	--	--	Bonded reinforcement at supports	New to allow for consideration of the redistribution of negative moments in the design.
18.13 18.14 18.15 18.16.1	--	--	Prestressed compression members under combined axial load and bending. Unbonded tendons. Post tensioning ducts. Grout for bonded tendons.	New to emphasize details particular to prestressed members not previously addressed in the codes in detail.
18.16.2	--	--	Proportions of grouting materials	Expanded definition of how grout properties may be determined.
18.16.4	--	--	Grouting temperature	Expanded definition of temperature controls when grouting.

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ACI 318-56 VS. ACI 349-76
SUMMARY OF CODE COMPARISON

Scale C

Referenced Subsection				
ACI 349-76	ACI 318-63	ACI 318-56	Structural Elements Potentially Affected	Comments
7.13.4	--	--	Reinforcement in flexural slabs	
Chapter 7	2408, 2409 and 2410	--	Precast elements	New sections identify special conditions allowed by new code as exceptions to the general code provisions. Old code required precase elements to meet all Code provisions.
10.3.6	1403(a)	1104(a)	Tied columns	New code allows more load to be carried on tied columns, i.e., 85% as compared to 80% factor in old code. Also new code allows a higher % of steel to be used in tied columns. This is less conservative than the old code.
10.8.1 10.8.2 10.8.3	912	1101	Compression members, limiting dimensions	Minimum size limitations are deleted in newer code giving the designer more freedom in cross-sectional dimensioning.

ACI 318-56 VS. ACI 349-76
SUMMARY OF CODE COMPARISON

Scale C

Referenced Subsection				
ACI 349-76	ACI 318-63	ACI 318-56	Structural Elements Potentially Affected	Comments
--	1502(d)	--	Continuous beams	New Code allows for moment redistribution where sufficient ductility exists. Old designs produce steel % on the order of 0.4 p_b; therefore, ductility was there.
10.14	2306	1206	Bearing - sections controlled by design bearing stresses	ACI 318-63 is more conservative, allowing a stress of $1.9(0.25\ f'_c) = 0.475\ f'_c < 0.6\ f'_c$
11.2.5	1706	805 & 806	Reinforcement concrete members without prestressing	Allowance of spirals as shear reinforcement is new. Requirement of 2 lines of web reinforcment, where shear stress exceeds $6\phi\sqrt{f'_c}$, was removed.
13.0 to end	--	--	Two-way slabs with multiple square or rectangular panels	Slabs designed by the previous criteria of ACI 318-63 are generally the same or more conservative.
13.4.1.5	--	--	Equivalent column flexibility stiffness and attached torsional members	Previous code did not consider the effect of stiffness of members normal to the plane of the equivalent frame.

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ACI 318-56 VS. ACI 349-76
SUMMARY OF CODE COMPARISON

Scale C (Cont.)

Referenced Subsection				
ACI 349-76	ACI 318-63	ACI 318-56	Structural Elements Potentially Affected	Comments
15.6	2306(b)	1206(b)	Columns	New code requires only transfer of actual stress carried by the column longitudinal bars. Old code required transfer of full working value. Older code more conservative.
17.5.4 17.5.5	--	--	Permissible horizontal shear stress for any surface, ties provided or not provided	Nominal increase in allowable shear stress under new code.

APPENDIX B-3

SECTION VIII, 1956 VS. SECTION III, SUBSECTION NE, 1980

SUMMARY OF CODE COMPARISONS

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ASME B&PV CODE COMPARISON
SECTION VIII, 1956 VS. SECTION III, SUBSECTION NE, 1980

Scale A

Referenced Subsection				
Section III 1980	Section VIII 1962	Section VIII 1956	Structural Elements Potentially Affected	Comments
NE-2124(b)	UG-16(c)	UG-16(c)	Pressure retaining plates less than 0.367 in thick	The 1956 Code granted a blanket 0.010 in mill undertolerance on all plate. The present code allows 6% or 0.010 inch undertolerance (whichever is least).
NE-3111	UG-22	UG-22	Load-carrying components	Section III, 1980 Code, specifies additional loads to be considered in designing the vessel. These include: o dynamic head of liquids o snow loads and vibration loads o reaction to steam and water jet impingement
NE-3112.2	---	UG-20	Vessel and components	The effects of internal heat generation due to radiation (in addition to all external sources) must be included in establishing design temperature.
NE-3112.3	---	---	Vessel and components	Currently, the design load combination includes mechanical loads. In 1956, the code considered pressure at temperature only.

ASME B&PV CODE COMPARISON
SECTION VIII, 1956 VS. SECTION III, SUBSECTION NE, 1980

Scale A (Cont.)

Referenced Subsection				
Section III 1980	Section VIII 1962	Section VIII 1956	Structural Elements Potentially Affected	Comments
NE-3112.4	---	UG-5(b)	Plates, if under-strength	The 1956 Code permits conditional use of understrength plate, if: 1. The local allowable stress is correspondingly reduced; and 2. The UTS range is maintained. This practice has been terminated and current codes are blind to such situations in older structures. Scale A - if additional loads, not originally designed for, are required by current criteria. No scale ranking applicable otherwise.
NE-3112.4	UG-23	UG-23	Vessels of materials no longer listed as Code acceptable	Section III, 1980 Code, references some materials which are identical to those referenced in Section VIII, 1956 Code. However, several materials which were referenced in Section VIII, are no longer listed in Section III, 1980. Justification of such use would be necessary to show equivalence to current requirements.

ASME B&PV CODE COMPARISON
SECTION VIII, 1956 VS. SECTION III, SUBSECTION NE, 1980

Scale A (Cont.)

Referenced Subsection				
Section III 1980	Section VIII 1962	Section VIII 1956	Structural Elements Potentially Affected	Comments
NE-3131	---	Various Paragraphs	Containment shells designed by formula	Section VIII, 1962 Code, calls for the design of vessels by formula, while Section III, 1980 Code requires that the rules of Subsection NE-3200 (Design by Analysis) be satisfied. In the absence of substantial thermal or mechnical loads other than pressure, the rules of "Design by Formula" may still be used. The scale rating for containment shells where substantial thermal or mechnical loads other than pressure are absent is Scale B; otherwise it is Scale A.
---	UG-25(d)	UG-25(d) UW-15(b)	Vessels containing telltale holes	The 1956 Code required telltale holes at reinforcing plates and saddles at nozzles to be left open. The 1962 Code permitted plugging. The removal of these provisions from Section III, 1962 Code, bans the use of telltale holes. Moreover, the more recent version of Section VIII specifically excludes using telltale holes for lethal substances.

ASME B&PV CODE COMPARISON
SECTION VIII, 1956 VS. SECTION III, SUBSECTION NE, 1980

Scale A (Cont.)

Referenced Subsection				
Section III 1980	Section VIII 1962	Section VIII 1956	Structural Elements Potentially Affected	Comments
NE-3133.5(a)	UG-29	UG-29	Stiffening rings for cylindrical shells subject to buckling loads.	The requirements of the 1980 Code defining the minimum moment of inertia for stiffening rings as compared to the requirements of the 1956 Code may result in a lower margin of safety. Scale $I_s' > 1.28\ I_s$ — C $I_s' > 1.22\ I_s$ — B $I_s' < 1.22\ I_s$ — A
NE-3133.5(b)	---	---	Shell and stiffening rings of different materials.	This new insert in Section III of the 1980 Code requires using the material chart which gives the larger value of the factor A. This may result in a larger stiffening ring section needed to meet the requirements of the Code. Scale A for ring-stiffened shells where (1) the ring and the shell are of different materials and, in addition, (2) the "factor A" (as computed by the procedures of NE-3133.5) for the two materials differs by more than 6%; otherwise Scale B.

ASME B&PV CODE COMPARISON
SECTION VIII, 1956 VS. SECTION III, SUBSECTION NE, 1980

Scale A (Cont.)

Referenced Subsection				
Section III 1980	Section VIII 1962	Section VIII 1956	Structural Elements Potentially Affected	Comments
NE-3221.5	---	---	Containment components subject to cyclic loadings.	Requirements for fatigue analysis of vessels or parts which experience cyclic loadings are provided in Section III, Subsection NE, of the 1980 Code. No specific guidance was provided by Section VIII, 1956.
NE-3324.3	UG-27(c)	UG-27(c)	Vessel components where welding efficiency of circumferential joints is less than half the longitudinal joint efficiency	The 1956 Code did not require computation of axial stress in cylindrical shells. The wide disparity in welding efficiencies is deemed improbable.
NE-3325	UG-34(c)3	UG-34(c)3	Heads, covers, or blind flanges on non-circular shape	The 1956 Code did not distinguish between circular and non-circular plates when specifying plate thickness. These thicknesses are now regarded as inadequate for most non-circular plates.
NE-3325 Figs. (c) and (m)	UG34(d) Figs. (d) and (p)	UG-34(d) Figs. (b) and (a)	Unstayed flat heads and covers of the designs in the referenced figures	Present code requires thicker plates.

ASME B&PV CODE COMPARISON
SECTION VIII, 1956 VS. SECTION III, SUBSECTION NE, 1980

Scale A (Cont.)

Referenced Subsection				
Section III 1980	Section VIII 1962	Section VIII 1956	Structural Elements Potentially Affected	Comments
NE-3327	UG-35	Footnote to UG-35	Quick-acting closures	Subsection NE, 1980 has expanded requirements for safety devices including: o positive interlocks on remotely operated doors o warning devices on manually operated doors o visibility of pressure indicators from operating floor.
NE-3334.1 NE-3334.2	UG-40(b) UG-40(c)	UG-40	Reinforcement for vessel openings	New requirements in the 1980 Code impose additional restrictions on metal that may be counted as reinforcement.
NE-3365	---	--	Bellows and bellows expansion joints	The 1980 Code imposes new design requirements.

ASME B&PV CODE COMPARISON
SECTION VIII, 1956 VS. SECTION III, SUBSECTION NE, 1980

(Summary of Code Changes with the Potential to Significantly Degrade Perceived Margin of Safety)

Scale B

Referenced Subsection: Section III 1980	Section VIII 1962	Section VIII 1956	Structural Elements Potentially Affected	Comments
NE-3133.1 & NE-3133.6	UG-28	UG-28 & UG-29	Components under external pressure and axial compression	The curves associated with the buckling of short cylinders appear to have been replotted to slightly different values.
NE-3324.8(c)	---	---	Torisphereical heads made of materials having minimum tensile strength exceeding 80 ksi	The allowable stress is restricted to values less than 22 ksi at room temperature by the the 1980 Code. Allowable stresses for some plate materials specified in the 1956 Code are slightly higher.
NE-3325 No figure	UG-34(d) Fig. (s)	UG-34(d) Fig. (m)	Unstayed flat heads and covers secured by spinning	Not a code-recommended practice for Section III vessels.
NE-3328	---	---	Combinations units	This new insert gives the design requirements for pressure vessels consisting of more than one independent pressure

ASME B&PV CODE COMPARISON
SECTION VIII, 1956 VS. SECTION III, SUBSECTION NE, 1980

(Summary of Code Changes with the Potential to Significantly Degrade Perceived Margin of Safety)

Scale B (Cont.)

Referenced Subsection				
Section III 1980	Section VIII 1962	Section VIII 1956	Structural Elements Potentially Affected	Comments
NE-3328 (cont.)				chamber. These requirements are standard practice for designing such vessels.
NE-3335	UG-40	UG-45	Reinforcement in nozzles and vessel walls	These new provisions of Section III, 1980 Code, detail specific requirements which are usually considered in good design practice.
NE-3336	UG-41(a)	---	Reinforcement for openings where welding is counted as reinforcement	The 1962 Code has provision that weld strength be taken as that of the weaker of the metal joined.
NE-3700	---	---	Electrical and mechanical penetration assemblies	Provisions usually adopted in standard engineering design of such assemblies.
NE-4120	UG-11(c)	---	Welded pressure parts other than the vessel shell	Documentation of code acceptability of welding practices as presently required by 1980 Code was not code enforced in 1956 and may not be available.

ASME B&PV CODE COMPARISON
SECTION VIII, 1956 VS. SECTION III, SUBSECTION NE, 1980

(Summary of Code Changes with the Potential to Significantly Degrade Perceived Margin of Safety)

Scale B (Cont.)

Referenced Subsection				
Section III 1980	Section VIII 1962	Section VIII 1956	Structural Elements Potentially Affected	Comments
NE-4232.1	UG-36(d)5	---	Reducers	Restriction on alignment of joints introduced. Local bending moments could be induced if offset joints were controlled.
NA-3767.4(a)2	UG-85	---	Heat-treated components	Requirements for written documentation of heat-treatment process were not provided by the 1956 Code.
NE-6000	UG-101	---	Vessel shell and other pressure retaining parts	The code has expanded the methods for, and exerts greater control over, the acceptable methods of proof testing.

ASME B&PV CODE COMPARISON
SECTION VIII, 1956 VS. SECTION III, SUBSECTION NE, 1980

Scale C

Referenced Subsection				
Section III 1980	**Section VIII 1962**	**Section VIII 1956**	**Structural Elements Potentially Affected**	**Comments**
NE-3332.2	UG-37(b)	UG-37(b)	Area of reinforcement - vessels under internal pressure	The 1980 Code includes a correction factor, F, in the equation for required area for reinforcement. This area is the same or less than the uncorrected equation required.
NE-3325 Figs. (a), (b), & (f)	UG-34(d) Figs. (a), (c), & (g)	UG-34(d) Figs. (d), (c), & (f)	Well-proportioned flat heads or covers of circular shape of the configurations shown in the referenced figures.	Thinner heads for these designs are now code acceptable.
NE-3362(b)	UG-43	UG-43	Studded connections	These paragraphs (although addressing different issues) provide rules for minimum depth of studs in general. The minimum engagement length can be less under the 1980 Code.

APPENDIX C

COMPARATIVE EVALUATIONS AND MODEL STUDIES

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CASE STUDY -1-

The allowable stress for structural steel subject to shear is specified in section 1.5.1.2 of the AISC code both in the 1963 and 1980 editions as

$$F_v = 0.40\ F_y \text{ ——— (1)}$$ based on the sectional area effective in resisting shear

However, in the 1980 Code a new section 1.5.1.2.2 is introduced stating that;

"AT beam end connections where the top flange is coped, and in similar situations where failure might occur by shear along a plane through the fasteners, or by a combination of shear along a plane through the fasteners plus tension along a perpendicular plane, on the area effective in resisting tearing failure: $F_v = 0.30\ F_u$ where the effective area is the minimum net failure surface, bounded by the bolt holes."

Referring to the 1980 Commentary and Fig. C.1.5.1.2 The connection allowable capacity in the tearing failure mode can be taken as

$$0.30\ A_v\ F_u + 0.50\ A_t\ F_u \text{ ——— (2)}$$

where A_v and A_t are the net shear and net tension areas respectively.

In order to evaluate the effect of the code change, 3 sets of each; Material, beam size & coefficients for web tear out (Table 1-G page 4-11 of the AISC Steel Manual) were used.

The results obtained by using equations (1) & (2) above indicate that the 1980 Code gives less conservative results as shown on the following tabulation.

Therefore, Scale -A-

BEAM END CONNECTION WHERE TOP FLANGE IS COPED, CASE STUDY -1-

FY,PSI	FU,PSI	H,IN	C1	C2	ALLOWABLE LOAD,LB 1963 CODE	1980 CODE	PCT.
36000.	60000.	12.00	1.00	0.74	172800.	104400.	40.
36000.	60000.	12.00	1.50	0.74	172800.	134400.	22.
36000.	60000.	24.00	1.00	0.74	345600.	104400.	70.
36000.	60000.	24.00	1.00	2.48	345600.	206800.	40.
36000.	60000.	24.00	1.50	0.74	345600.	134400.	61.
36000.	60000.	24.00	1.50	2.48	345600.	238800.	31.
36000.	60000.	24.00	2.25	0.74	345600.	179400.	48.
36000.	60000.	24.00	2.25	2.48	345600.	283800.	18.
36000.	60000.	36.00	1.00	2.48	518400.	208800.	60.
36000.	60000.	36.00	1.00	4.81	518400.	348600.	33.
36000.	60000.	36.00	1.50	2.48	518400.	238800.	54.
36000.	60000.	36.00	1.50	4.81	518400.	378600.	27.
36000.	60000.	36.00	2.25	2.48	518400.	283800.	45.
36000.	60000.	36.00	2.25	4.81	518400.	423600.	18.
50000.	70000.	12.00	1.00	0.74	240000.	121800.	49.
50000.	70000.	12.00	1.50	0.74	240000.	156800.	35.
50000.	70000.	12.00	2.25	0.74	240000.	209300.	13.
50000.	70000.	24.00	1.00	0.74	480000.	121800.	75.
50000.	70000.	24.00	1.00	2.48	480000.	243600.	49.
50000.	70000.	24.00	1.50	0.74	480000.	156800.	67.
50000.	70000.	24.00	1.50	2.48	480000.	278600.	42.
50000.	70000.	24.00	2.25	0.74	480000.	209300.	56.
50000.	70000.	24.00	2.25	2.48	480000.	331100.	31.
50000.	70000.	36.00	1.00	2.48	720000.	243600.	66.
50000.	70000.	36.00	1.00	4.81	720000.	406700.	44.
50000.	70000.	36.00	1.50	2.48	720000.	278600.	61.
50000.	70000.	36.00	1.50	4.81	720000.	441700.	39.
50000.	70000.	36.00	2.25	2.48	720000.	331100.	54.
50000.	70000.	36.00	2.25	4.81	720000.	494200.	31.
65000.	80000.	12.00	1.00	0.74	312000.	139200.	55.
65000.	80000.	12.00	1.50	0.74	312000.	179200.	43.
65000.	80000.	12.00	2.25	0.74	312000.	239200.	23.
65000.	80000.	24.00	1.00	0.74	624000.	139200.	78.
65000.	80000.	24.00	1.00	2.48	624000.	278400.	55.
65000.	80000.	24.00	1.50	0.74	624000.	179200.	71.
65000.	80000.	24.00	1.50	2.48	624000.	318400.	49.
65000.	80000.	24.00	2.25	0.74	624000.	239200.	62.
65000.	80000.	24.00	2.25	2.48	624000.	378400.	39.
65000.	80000.	36.00	1.00	2.48	936000.	278400.	70.
65000.	80000.	36.00	1.00	4.81	936000.	464800.	50.
65000.	80000.	36.00	1.50	2.48	936000.	318400.	66.
65000.	80000.	36.00	1.50	4.81	936000.	504800.	46.
65000.	80000.	36.00	2.25	2.48	936000.	378400.	60.
65000.	80000.	36.00	2.25	4.81	936000.	564800.	40.

NOTES:

1- ALLOWABLE LOADS ARE GIVEN PER INCH OF WEB THICKNESS
2- PCT= PERCENT OF THE REDUCTION OF PERCEIVED MARGIN OF SAFETY

CASE STUDY 2

AXIALLY LOADED COLUMNS

Maximum allowable axial load on tied columns by working stress design criteria is defined by

$$P = 0.85\ [A_g\ (0.25\ f'_c + f_s\ p_g)]$$

where $p_g = \dfrac{A_{st}}{A_g}$ and allowable $f_s = 0.4 f_y \leq$ 30,000 psi

that is, max $f_y \leq$ 75,000 psi

therefore, the maximum load could be expressed as:

$$P_{allow} = (0.21\ A_g\ f'_c + 0.34\ f_y\ A_{st})$$

Maximum allowable axial load on tied columns by strength design criteria is defined by

$$P_{allow} = \phi P_o = \phi 0.8\ [0.85\ f'_c\ (A_g - A_{st}) + A_{st}\ f_y]$$

for a tied column in axial compression $\phi = 0.7$ and $P_u = 1.4\ D + 1.7\ L$.

Reducing these equations to be comparable to working stress limits and considering all extremes of steel % and D. to L. load ratios, we get

if $A_{st} = 0.01\ A_g$ $\quad P_u = \phi P_o = \phi\ (0.673\ f'_c\ A_g + 0.8\ A_{st}\ f_y)$

if $A_{st} = 0.08\ A_g$ $\quad P_u = \phi P_o = \phi\ (0.626\ f'_c\ A_g + 0.8\ A_{st}\ f_y)$

and to bracket extremes, consider the following three cases.

(a) $D = 0$

(b) $L = D$ and

(c) $L = 0$ with $P_{allow} = \dfrac{P_u}{L.F.}$

(a) for L.F. = 1.7

$$P_{allow} = 0.28\ f_c'\ A_g + 0.33\ f_y\ A_{st} \quad \text{or}$$

$$P_{allow} = 0.26\ f_c'\ A_g + 0.33\ f_y\ A_{st}$$

(b) for L.F. = 1.55

$$P_{allow} = 0.30\ f_c'\ A_g + 0.36\ f_y\ A_{st} \quad \text{or}$$

$$P_{allow} = 0.28\ f_c'\ A_g + 0.36\ f_y\ A_{st}$$

(c) for L.F. = 1.4

$$P_{allow} = 0.34\ f_c'\ A_g + 0.40\ f_y\ A_{st} \quad \text{or}$$

$$P_{allow} = 0.31\ f_c'\ A_g + 0.40\ f_y\ A_{st}$$

Comparison of these resulting equations to the P_{allow} by working stress design criteria shows that the new code allows from 1.24 to 1.62 times more load on the concrete in a tied column and from 0.97 to 1.18 times more load on the longitudinal steel in a tied column.

Therefore, Scale C

CASE STUDY 3

FLEXURAL MEMBERS

Sections with Tension Reinforcing Only:

For purposes of code comparison, with emphasis on comparing safety margins of designs conforming to older codes and practices with corresponding margins provided by current criteria, the following case studies were prepared.

For designs prepared by working stress criteria, a comparison with strength design was made by reducing the strength equation to an allowable moment by the following definition.

$$M_{allow} = \frac{\phi M_u}{L.F.}$$

To bracket extremes of load ratios, the following three cases were considered in each working stress comparison.

(a) when L = 0   L.F. = 1.4
(b) when L = D   L.F. = 1.55
(c) when D = 0   L.F. = 1.7

For designs prepared by yield-strength criteria, a comparison with strength design was made directly with a load factor equal to 1.0. The yield-strength definition used here was not a code endorsed practice; but was the method widely adopted by architect engineers, at the time, to design for the extreme loadings postulated for accident and faulted conditions. It possesses the practical advantage of permitting an extended use of linearly elastic computer codes to provide design guidance for extreme loading cases and is documented in Ref. 1*

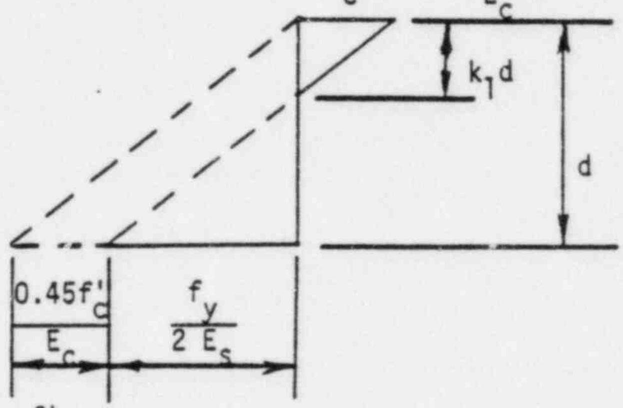
Since older codes did not contain any strict limitation on the percent of reinforcement, the comparisons presented here used the defined balanced steel percentage and additionally steel percentages 60 percent lower and 50 percent higher than balanced in order to show the effect of this parameter on the comparisons.

*Ref. 1
A Study of the Design and Construction Practices of Prestressed Concrete and Reinforced Concrete Containment Vessels by C. P. Tan prepared by FIRL for the U. S. Atomic Energy Commission, Aug. 1969 under contract to the ORNL (TID 25176).

For Working Stress Design

The definition of balanced design is that both concrete and steel reach their theoretical working stress allowable limit simultaneously.

The strain diagram and neutral axis location for this condition are:



$$\frac{k_1 d}{d} = \frac{0.45\ \frac{f'_c}{E_c}}{\frac{0.45\ f'_c}{E_c} + \frac{f_y}{2\ E_s}} \qquad k_1 = \frac{1}{1 + \left(\frac{1}{0.9}\right)\left(\frac{f_y}{f'_c}\right)\left(\frac{E_c}{E_s}\right)}$$

$$\text{let } r = \frac{f_y}{f'_c} \quad \text{and} \quad n = \frac{E_s}{E_c}$$

then for elastic balanced design:

$$k_1 = \frac{1}{1 + 1.11\left(\frac{r}{n}\right)}$$

and from equilibrium:

$$\frac{F_s}{F_c} = \frac{A_s\ f_y}{(0.45\ f'_c)\ b\ k\ d} = \left(\frac{A_s}{bd}\right)\left(\frac{f_y}{f'_c}\right)\frac{1}{0.45k} = 1$$

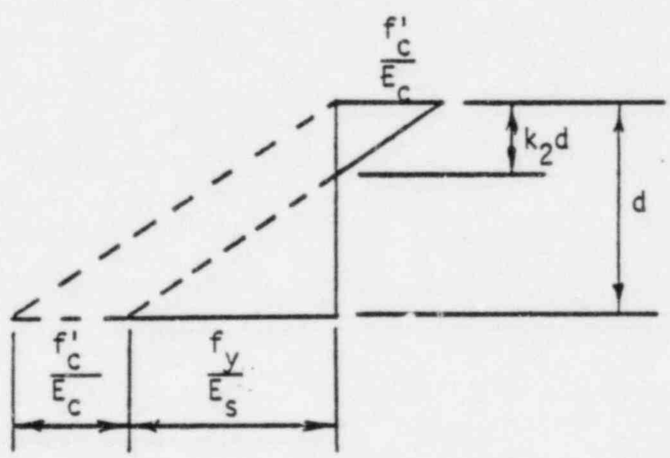
$$\rho_1 = \frac{A_s}{bd} = 0.45 \ \frac{k_1}{r}$$

$$M_t = f_s A_s jd \quad \text{or } M_c = 1/2\ f_c bd^2 j\ k$$

For Yield-Limit Design

The Yield-Limit concept assumes that the system behaves in a linear fashion up to the yield of the steel or to the ultimate strength of the concrete. For the balanced condition again $f_s = f_y$ and $f_c = f'_c$ simultaneously.

The strain diagram and neutral axis location for this condition are:



$$\frac{k_2 d}{d} = \frac{\frac{f'_c}{E_c}}{\frac{f'_c}{E_c} + \frac{f_y}{E_s}} \qquad k_2 = \frac{1}{1 + \frac{f_y}{f'_c}\ \frac{E_c}{E_s}}$$

then for balanced conditions

$$k_2 = \frac{1}{1 + \left(\frac{r}{n}\right)}$$

and from equilibrium

$$\frac{F_s}{F_c} = \frac{A_s f_y}{1/2(f'_c)\ bkd} = 2\ \rho_2\ \frac{r}{k_2} = 1$$

$$\rho_2 = \frac{k_2}{2r}$$

$$M_t = f_y A_s jd \quad \text{or } M_c = 1/2\ f'_c\ bd^2 jk$$

For Strength Design

Ultimate strength capacity is defined as:

$$M_u = A_s f_y d \left[1-0.59\ \rho\ \frac{f_y}{f'_c}\right]$$

Example 1.

for Yield-Limit design at balanced design

$$M_t = f_y A_s jd \qquad M_c = 1/2\ f'_c\ bd^2 jk = 1/2\left(\frac{f'_c}{f_y}\right)\frac{jk}{p}\ (A_s f_y d)$$

$$k_2 = \frac{1}{1 + \frac{f_y}{f'_c}\ \frac{E_c}{E_s}} \qquad \rho_2 = 1/2\ k_2\ \frac{f'_c}{f_y} \qquad n = \frac{E_s}{E_c} = \frac{508}{\sqrt{f'_c}}$$

for $f'_c = 4{,}000$ psi $\quad f_y = 40{,}000$ psi $\quad n = 8$

$$k_2 = \frac{1}{1 + 10\ (1/8)} = 0.444 \qquad \rho_2 = 1/2\ (0.444)\ 4/40 = 0.022$$

$$j = 0.852$$

$$M_t = 0.852\ f_y A_s d$$

$$M_u = A_s f_y d\ [1-0.59(0.022)10] = 0.869\ A_s f_y d$$

$$\frac{M_u}{M_t} = \frac{0.869}{0.852} = 1.02$$

Also:

if $\rho < \rho_2$ (say $60\%\ \rho_2$)

$$\rho = 0.6\ (0.022) = 0.0132$$

$\alpha = 2\rho n = 2\ (0.0132)(8) = 0.211$

$$k = \left(0.211 + \frac{(0.211)^2}{4}\right)^{1/2} - \frac{0.211}{2} = 0.366$$

$j = 0.878$

$M_t = 0.878\ f_y A_s d$

$M_u = A_s f_y d\ [1\text{-}0.59\ (0.0132)10] = 0.922\ A_s f_y d$

$$\frac{M_u}{M_t} = 1.05$$

and; similarly,

if $\rho > \rho_2$ $\qquad \rho = 1.5\ \rho_2 = 1.5\ (0.022) = 0.033$

One finds M_c controls, and:

$$\frac{M_u}{M_c} = 1.26$$

For working stress design at balanced design

$$k_1 = \frac{1}{1 + 1.11\ (10/8)} = 0.419 \qquad \rho_1 = 0.45\ \frac{0.419}{(10)} = 0.0188$$

$j = 0.86$

$$M_t = 0.86\ f_s A_s d = 0.86\ \frac{f_y}{2}\ A_s d = 0.43\ A_s f_y d$$

$M_u = A_s f_y d\ [1\text{-}0.59(0.0188)10] = 0.889\ A_s f_y d$

$$\frac{M_u}{M_t} = 2.07$$

$$\therefore \frac{M_{allow}}{M_t} = \frac{\frac{0.9}{L.F.}\ M_u}{M_t} = \begin{cases} 1.33 \text{ if } L = 0 \\ 1.20 \text{ if } L = D \\ 1.09 \text{ if } D = 0 \end{cases}$$

Also:

if $\rho < \rho_1$ (say 60% ρ_2)

ρ = 0.6 (0.0188) = 0.0.0113 $\qquad \alpha = 2\ pn = 0.180$

$$k = \left(0.18 + \frac{(0.18)^2}{4}\right)^{1/2} - \frac{0.18}{2} = 0.344$$

j = 0.885

$$M_t = 0.885 A_s f_s d = 0.885\ A_s \frac{f_y}{2} d = 0.443\ A_s f_y d$$

$$M_u = A_s f_y d\ [1-0.59(0.0113)10] = 0.933\ A_s f_y d$$

$$\frac{M_u}{M_t} = 2.11$$

$$\therefore \quad \frac{M_{allow}}{M_t} = \begin{cases} 1.36 \text{ if } L = 0 \\ 1.22 \text{ if } L = D \\ 1.12 \text{ if } D = 0 \end{cases}$$

and:

if $\rho > \rho_1$ (say 1.5 ρ_1)

One finds concrete controls, and:

$$\frac{M_u}{M_c} = 2.43$$

$$\therefore \quad \frac{M_{allow}}{M_c} = \begin{cases} 1.56 \text{ if } L = 0 \\ 1.41 \text{ if } L = D \\ 1.29 \text{ if } D = 0 \end{cases}$$

In summary,

for yield limit design comparisons:

$$\frac{M_u}{M_t} = 1.02 \text{ to } 1.26$$

for working stress design comparisons:

$$\frac{M_{allow}}{M_t} = 1.09 \text{ to } 1.56$$

$\therefore$ Strength design allows beams to operate at a higher stress level. For these beams the older code is more conservative

Scale C

Example 2.

For Yield-Limit design at balanced design

for $f'_c = 3000$ psi $\qquad f_y = 36{,}000$ psi

$$k_2 = \frac{1}{1 + (12)(1/9)} = 0.429$$

$$\rho_2 = 1/2\ (0.429)\ 1/12 = 0.0179$$

$j = 0.857$

$M_t = 0.857\ A_s f_y d$

$M_u = A_s f_y d[1-0.59(0.0179)12] = 0.873\ A_s f_y d$

$$\frac{M_u}{M_t} = 1.02$$

Also:

if $\rho < \rho_2$ (say 60%)

$$\frac{M_u}{M_t} = 1.05$$

And:

if $\rho > \rho_2$ (say $\rho = 1.5\ \rho_2 = 0.0268$)

$$\frac{M_u}{M_c} = 1.26$$

<u>For Working Stress Design at balanced design</u>

$f'_c = 3$ ksi $\qquad f_y = 36$ ksi $\qquad n = 9 \qquad \frac{f_y}{f'_c} = 12$

$k_1 = 0.403 \qquad \rho_1 = 0.0151$

$$\frac{M_u}{M_t} = 2.06$$

$$\therefore \quad \frac{M_{allow}}{M_t} = \begin{cases} 1.32 \text{ if } L = 0 \\ 1.20 \text{ if } L = D \\ 1.09 \text{ if } D = 0 \end{cases}$$

Also:

if $\rho < \rho_1$ (say 60%)

$$\frac{M_u}{M_t} = 2.1$$

$$\therefore \quad \frac{M_{allow}}{M_t} = \begin{cases} 1.35 \text{ if } L = 0 \\ 1.22 \text{ if } L = D \\ 1.11 \text{ if } D = 0 \end{cases}$$

And:

if $\rho > \rho_1$ (say $1.5\ \rho_1$)

$$\frac{M_u}{M_c} = 2.58$$

$$\therefore \quad \frac{M_{allow}}{M_c} = \begin{cases} 1.66 & \text{if } L = 0 \\ 1.50 & \text{if } L = D \\ 1.36 & \text{if } D = 0 \end{cases}$$

In summary,

for yield limit design comparisons:

$$\frac{M_u}{M_t} = 1.02 \text{ to } 1.26$$

for working stress design comparisons:

$$\frac{M_{allow}}{M_t} = 1.09 \text{ to } 1.66$$

$\therefore$ Strength design allows beams to operate at a higher stress level. For these beams the older code is more conservative.

Scale C

In general, for designs controlled by flexure, beams designed by strength design methods will have higher stresses at service load levels than beams designed for the same service loads by working stress design methods.

CASE STUDY -4-

Ref AISC 1980 CODE

Subsection 2.4 Columns

"In the plane of bending of columns which would develop a plastic hinge at ultimate loading, the slenderness ratio $\frac{l}{r}$ shall not exceed Cc, ..."

where $C_c = \sqrt{\frac{2\pi^2 E}{F_y}}$

$E = 29 \times 10^3$ KSI

F_y = yield stress

Therefore $\frac{l}{r} \leq \frac{756.6}{\sqrt{F_y}}$

Ref AISC 1963 Code

Subsection 2.3 Columns

"In the plane of bending of columns which would develop a plastic hinge at ultimate loading, the slenderness ratio shall not exeed 120, ..."

$\frac{l}{r} \leq 120$

Which of the two codes is the more restrictive on l/r ratio depends on the yield strength of the steel used for the columns.

1) Both codes give $\frac{l}{r} = 120$ when

$$C_c = \frac{756.6}{\sqrt{F_y}} = 120$$

then,

$$F_y = 40 \text{ KSI}$$

2) The 1980 Code is 5% more conservative when

$$\frac{l}{r} = 114 = \frac{756.6}{\sqrt{F_y}}$$

then, $F_y = 44$ KSI

Conclusion:

	Scale
$F_y \leq 40$ KSI	C
$40 < F_y < 44$	B
$F_y > 44$	A

CASE STUDY -5-

Ref AISC 1980 Code

Subsection 1.10.5.3

" In girders designed on the basis of tension field action, the spacing between stiffeners at end panels, at panels containing large holes, and at panels adjacent to panels containing large holes shall be such that f_v does not exceed the value given" below

$$F_v = \frac{F_y}{2.89} C_v \leq 0.4 F_y$$

Where

$$C_v = \frac{45000k}{F_y (h/t)^2} \quad \text{when } C_v < 0.8$$

$$k = 4 + \frac{5.34}{(a/h)^2} \quad \text{when } a/h < 1.0$$

$$= 5.34 \quad \text{when } a/h > 1.0$$

Ref AISC 1963 Code

Subsection 1.10.5.3

" The spacing between stiffeners at end panels and panels containing large holes shall be such that the smaller panel dimensic a or h shall not exceed

$$\frac{1100t}{\sqrt{f_v}}$$ "

REF AISC SUB section 1.10.5.3

EXAMPLE

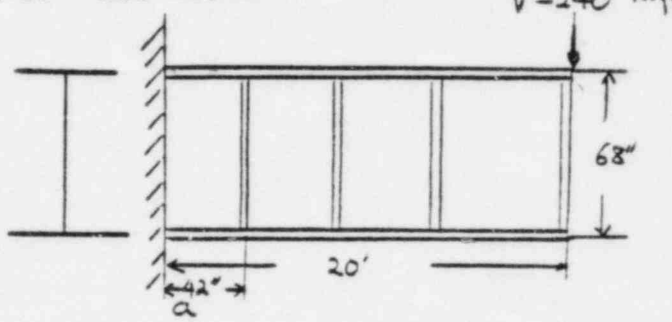
$h = 68''$

$t = .375''$

$A_w = 68 \times \frac{3}{8} = 25.5 \ \text{in}^2$

$V = 240$ Kips

$f_v = \frac{240}{25.5} = 9.06$ KSI



from 1.10.5.3 1963 Code

$$a \text{ or } h \not> \frac{11000\,t}{\sqrt{f_v}} = \frac{11000 \times 3/8}{\sqrt{9.06 \times 1000}} = 43 \text{ in}$$

Which is the distance from the end of the girder to the first transverse stiffener.

By considering the tension field action as specified in 1980 Code subsection 1.10.5.3

$f_v = 9.06$ KSI $\quad \frac{h}{t} = \frac{68}{.375} = 181 \quad \& \quad \frac{a}{h} = \frac{42}{68} = .618$

$$k = 4 + \frac{5.34}{(a/h)^2} = 4 + \frac{5.34}{(.618)^2} = 17.98$$

$$C_v = \frac{45000\,k}{F_y (h/t)^2} = \frac{45000 \times 17.98}{36\,(181)^2} = .686$$

$$F_v = \frac{F_y}{2.89} C_v \le .4 F_y$$

$$= \frac{36}{2.89} \times .686 = 8.54 \text{ KSI}$$

& from table 10.36 the Allowable shear stress $\approx$ 8.6 Ksi (checks Computed Value) however, lower than f_v of 9.06 Ksi

$\therefore$ Scale B for this example

Remarks

The following two figures show F_v vs. A/T for various values of A/H and F_y.
By knowing the shear stress F_v or F_v' the A/T value can be abtained and compared with the design A/T. Thus comparison should be examined on a case by case basis.

Franklin Research Center A Division of The Franklin Institute The Benjamin Franklin Parkway, Phila., Pa. 19103	Project: C5257		Page: C.5-5
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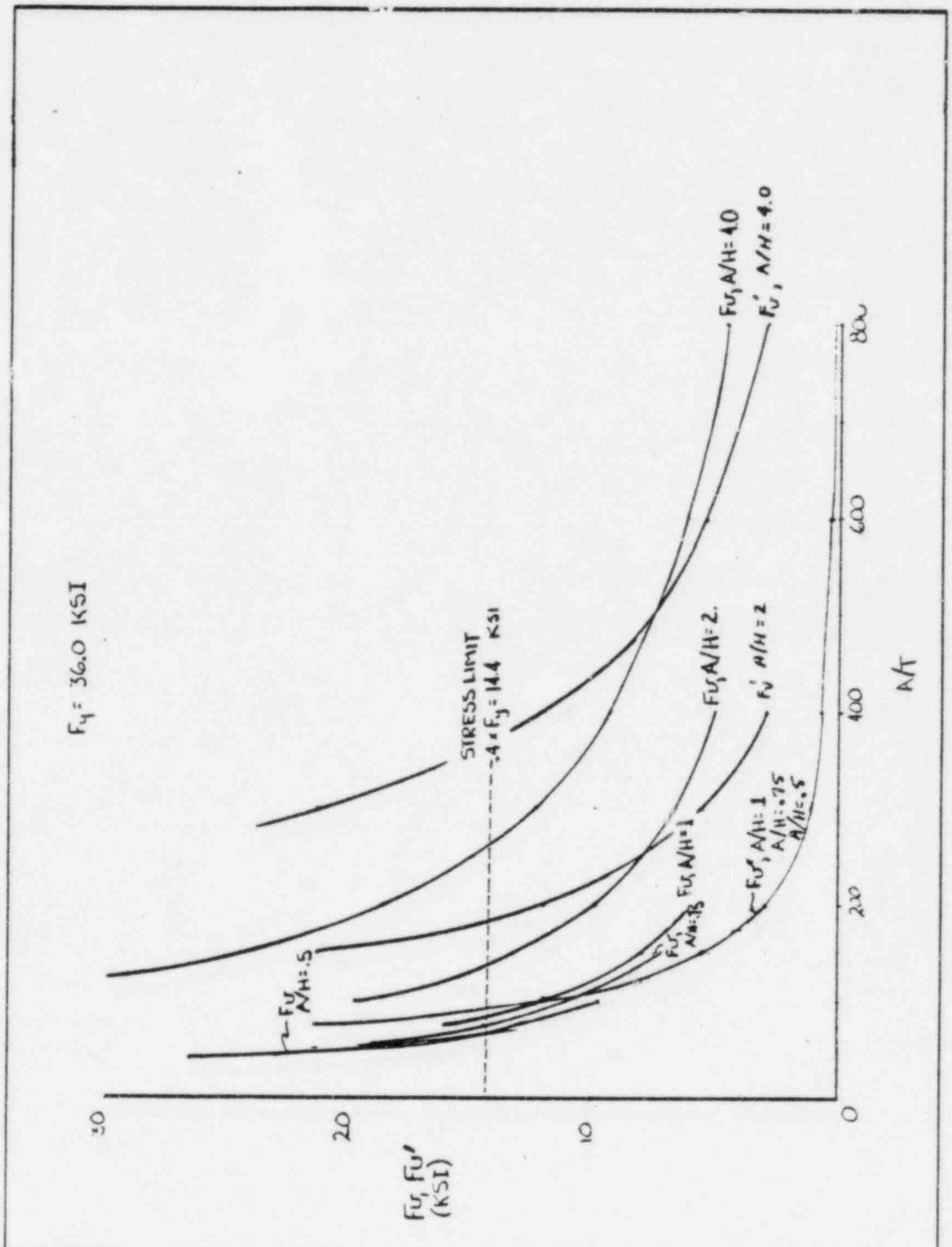

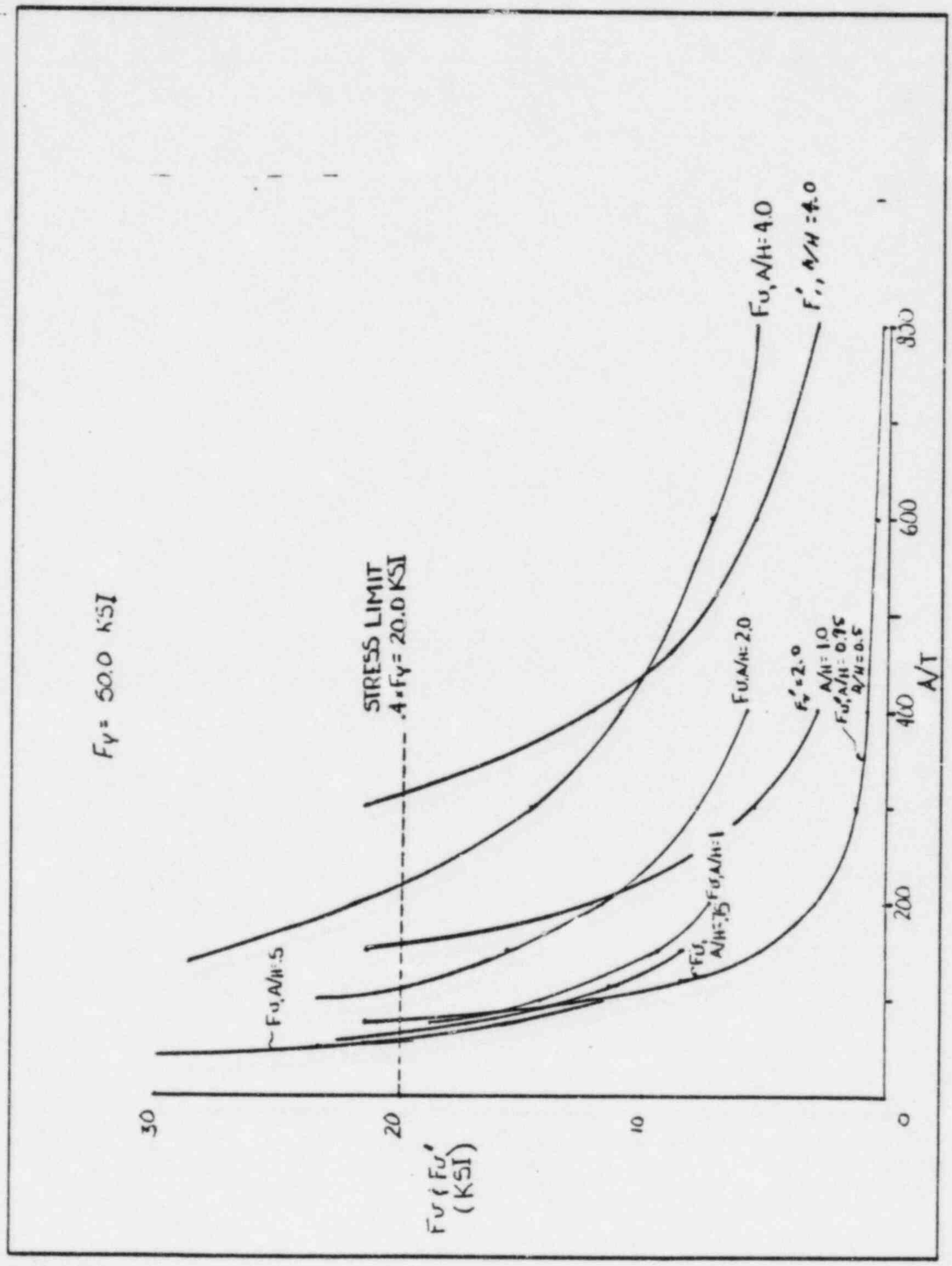

F_Y = 50.0 KSI
STRESS LIMIT
.4·F_Y = 20.0 KSI
Fu, A/H: 4.0
F'u, A/H: 4.0
Fu, A/H: 2.0
F'u = 2.0
Fu, A/H: 1
Fu, A/H: .75
Fu, A/H: .5
A/H: 1.0
Fu, A/H: 0.75
A/H: 0.5
30
20
10
0
200
400
600
800
A/T
Fu & Fu' (KSI)

CASE STUDY -6-

Ref AISC 1980 Code
Section 2.7

"The width-thickness ratio for flange of rolled W, M, or S shapes and similar built-up single-web shapes that would be subjected to compression involving hinge rotation under ultimate loading shall not exceed the following values:"

F_y, ksi	$b_f/2t_f$
36	8.5
42	8.0
45	7.4
50	7.0
55	6.6
60	6.3
65	6.0

"The width-thickness ratio of similarly compressed flange plates in box sections and cover plates shall not exceed $190/\sqrt{F_y}$"

Example

$$\frac{b}{t} = \frac{190}{\sqrt{F_y}}$$

F_y ksi	b/t
36	31.7
50	26.9
75	22
100	19

"The depth-thickness ratio of webs of members subjected to plastic bending shall not exceed ...."

$$d/t = \frac{412}{\sqrt{F_y}}\left(1 - 1.4\frac{P}{P_y}\right) \quad \text{when } \frac{P}{P_y} \leq 0.27$$

For $\frac{P}{P_y} = 0.0$

F_y	d/t
36	68.7
50	58.3
75	47.6
100	41.2

$$d/t = \frac{257}{\sqrt{F_y}} \quad \text{when } \frac{P}{P_y} > 0.27$$

F_y	d/t
36	42.8
50	36.3
75	30
100	25.7

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MD	SEPT. '81	E.M.D 10/81	

Ref AISC 1963 Code
Section 2.6

" Projecting element, that would be subjected to compression involving plastic hinge rotation under ultimate loading shall have width-thickness ratio no greater than the following: "

$b_f/2t_f \leq 8.5$ Rolled Shapes

$b_f/t_f \leq 32$ Box Sections

" The depth-thickness ratio of beam and girder webs subjected to plastic bending" is given by the following formula

$$43 \leq d/w \leq 70 - 100\frac{P}{P_y}$$

Remarks

The 1963 Code take into account material for A36 of $F_y = 36$ KSI or less (note that the two codes are the same for $F_y = 36$).
If the structure was designed using material having higher yield, the design might not be acceptable under present requirements.

$F_y \leq 36$ KSI (C)

$36 < F_y < 38$ KSI (B)

$F_y \geq 38$ KSI (A)

CASE STUDY -7-

Ref AISC 1980 Code
Section 2.9 Lateral Bracing

" Members shall be adequately braced to resist lateral and torsional displacements .... The laterally unsupported distance, l_{cr}, ... shall not exceed the value determined from "

$$\frac{l_{cr}}{r_y} = \frac{1375}{F_y} + 25 \quad \text{when} \quad 1.0 > \frac{M}{M_p} > -0.5$$

$$\text{or} \quad \frac{l_{cr}}{r_y} = \frac{1375}{F_y} \quad \text{when} \quad -0.5 \geq \frac{M}{M_p} > -1.0$$

example

l_{cr}/r_y	F_y = 36 KSI	50	75	100
$1 > \frac{M}{M_p} > -.5$	63.2	52.5	43.3	38.75
$-.5 \geq \frac{M}{M_p} > -1.0$	38.2	27.5	18.3	13.75

Ref AISC 1963 Code

Section 2.8 Lateral Bracing

When the moment definition is compatible with the 1980 code, the formula for l_{cr}/r_y becomes:

$$35 < \frac{l_{cr}}{r_y} = 60 + 40\frac{M}{M_p}$$

...ple

$\frac{M}{M_p}$	$\frac{l_{cr}}{r_y}$
1	100
0	60
-.5	40

CONCLUSIONS

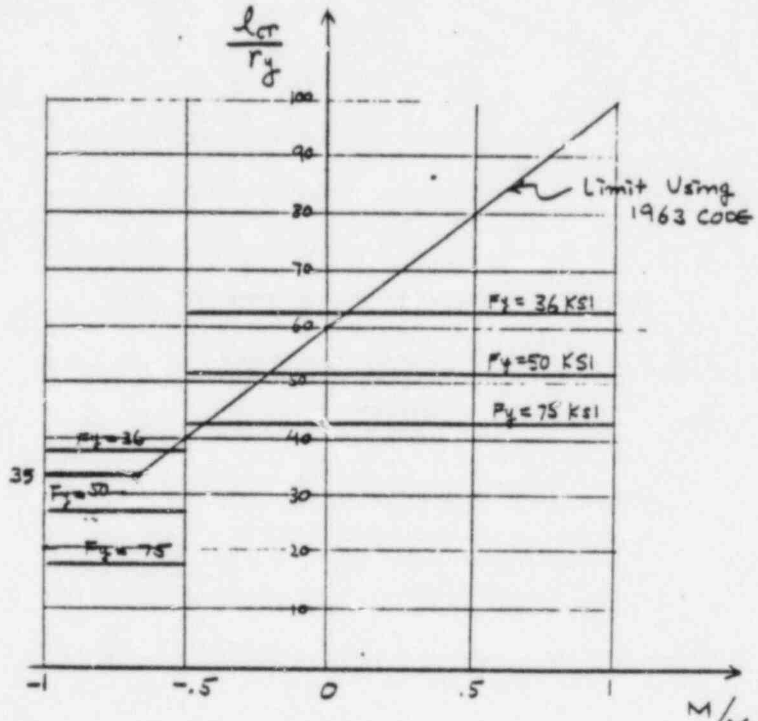
The figure which follows (l_{cr}/r_y vs. M/M_p) indicates that for A-36 steel ($F_y = 36$ ksi)

Scale

$0 < \frac{M}{M_p} < 1$ ——— (A)

$0 > \frac{M}{M_p} > -1$ ——— (C)

Note: The summary is based on material with $F_y = 36$, other material should be examined on a case by case basis.


$\frac{l_{cr}}{r_y}$
100
90
80
70
60
50
40
30
20
10
35
Limit Using 1963 CODE
Fy = 36 KSI
Fy = 50 KSI
Fy = 75 KSI
Fy = 36
Fy = 50
Fy = 75
-1
-.5
0
.5
1
M/Mp

CASE STUDY -8-

Comparison of Section 2.3, Columns (AISC, 1963) with Section 2.4, Columns (AISC, 1980)

AISC 1963	AISC 1980
1. Slenderness ratio for columns in continuous frames where sideway is not prevented, is limited by Formula (20) $$\frac{2P}{P_y} + \frac{l}{70r} \leq 1.0$$ This limits slenderness Ratio $\frac{l}{r} \leq 70$ and axial load not to exceed $0.5\,P_y$ for $\frac{l}{r} = 0$. Also limited by Formula (26) given below.	1. Slenderness ratio for Columns in continuous frames where Sidesway is not prevented, not limited to only 70. But limited by Formulas (2.9 - 1a) and (2.9 - 1b) given below and $\frac{l}{r}$ not to exceed C_c, as given below
2. For columns in braced frames the maximum axial load P shall not exceed $0.6\,P_y$.	2. The axial load in columns in braced frames not to exceed $0.85\,P_y$

(See Case Study 4 also, for Slenderness ratio)

3\. a) Slenderness ratio $\frac{l}{r}$ not to exceed 120

b) The allowable laterally unsupported distance

$l_{cr} = (60 - 40\frac{M}{M_p})r_y$,

Formula (26) But $l_{cr} \nless 35r_y$

c) $\frac{Kl}{r_{min}}$ not to exceed 200 in any case

3a. a Slenderness ratio $\frac{l}{r}$ not to exceed C_c

where $C_c = \sqrt{\frac{2\pi^2 E}{F_y}}$

and for $F_y = 36$ KSi,
$C_c = 126.1$

3 b. The laterally unsupported distance l_{cr} not to exceed the following

$$\frac{l_{cr}}{r_y} = \frac{1375}{F_y} + 25 \quad (2.9-1a)$$

When $+1.0 > \frac{M}{M_p} > -0.5$

And

$$\frac{l_{cr}}{r_y} = \frac{1375}{F_y} \quad (2.9-1b)$$

when $-0.5 \geq \frac{M}{M_p} > -1.0$

3c. $\frac{Kl}{r_{min}}$ not to exceed 200 in any case.

4(a) Interaction formulas for single curvature are

Formula (22)

$$\frac{M}{M_p} \leq B - G\left(\frac{P}{P_y}\right) \leq 1.0$$

$$M \leq M_p$$

and Formula (23)

$$\frac{M}{M_p} \leq 1.0 - H(P/P_y) - J(P/P_y)^2$$

Values of B, G, H and J listed in tables as a function of slenderness ratio and F_y.

(b) Interaction formulas for double curvature are

Formula (21)

$$M \leq M_p \quad \text{for } P/P_y \leq 0.15$$

$$\frac{M}{M_p} \leq 1.18 - 1.18(P/P_y) \leq 1.0$$

$$\text{for } P/P_y \geq 0.15$$

and Formula (22)

$$\frac{M}{M_p} \leq B - G\left(\frac{P}{P_y}\right) \leq 1.0 ;$$

$$M \leq M_p$$

4. Interaction formulas are

Formula (2.4-2)

$$\frac{P}{P_{cr}} + \frac{C_m M}{\left(1 - \frac{P}{P_e}\right) M_m} \leq 1.0$$

and Formula (2.4-3)

$$\frac{P}{P_y} + \frac{M}{1.18 M_p} \leq 1.0 ; \quad M \leq M_p$$

where $P_{cr} = 1.7 A F_a$

$$P_e = \frac{23}{12} A F_e'$$

F_a given by (1.5-1) and

F_e' given in Section 1.6.1

$M_m = M_p$ (braced in the weak direction)

$$= \left[1.07 - \frac{(l/r_y)\sqrt{F_y}}{3160}\right] M_p \leq M_p$$

(unbraced in weak direction)

a) For single curvature
$0.6 \leq C_m \leq 1.0$

b) For double curvature
$0.4 \leq C_m \leq 0.6$

For comparison of these specifications, graphs of P/P_y vs M/M_p are drawn for slenderness ratio of 30, 70 and 100. Typical Column 14 WF 150 with $F_y = 36$ ksi has been taken as an example for our purposes Separate graphs are drawn for single curvature ($0.6 \leq C_m \leq 1.0$) and double Curvature ($0.4 \leq C_m \leq 0.6$) cases.

For frames with sidesway ($C_m = 0.85$) allowed, graphs of P/P_y vs M/M_p are drawn for two types of columns 14 WF 150 and 12 WF 45, with $F_y = 36$ ksi. Columns assumed to be braced in the weak direction, for all graphs

It can be inferred from the graphs that in all cases, the major change is the limit of allowable axial load, which is increased from $0.5 P_y$ to $0.75 P_y$ for unbraced columns (Sidesway allowed) and $0.6 P_y$ to $0.85 P_y$ for braced columns. But the acceptable design region in both codes is almost same. For single curvature we notice for $\frac{Kl}{r} = 30$ the Formula (2.4-2) line for $C_m = 1.0$ is below the formula (23) line, but for $\frac{Kl}{r} = 70$, they overlap and for $\frac{Kl}{r} = 100$, the formula (2.4-2) for $C_m = 1.0$ is above the formula (23) line. Thus for $\frac{Kl}{r} = 30$ 1980 code being more conservative, while for $\frac{Kl}{r} = 100$, 1963 code seems to be <u>more conservative</u>. This change can thus be classified best as a. <u>B</u> change.

$F_y = 36$ ksi $\frac{kl}{r} = 30$ 14 W 150 SINGLE CURVATURE

Assume: braced in weak direction

∴ $M_m = M_p$

1963 Code

Formula (22) $\frac{M}{M_p} \leq B - G(P/Py) \leq 1.0$

$M \leq M_p$

Formula (23) $\frac{M}{M_p} \leq 1.0 - H(P/Py) - J(P/Py)^2$

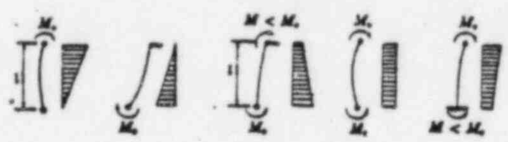
1980 Code

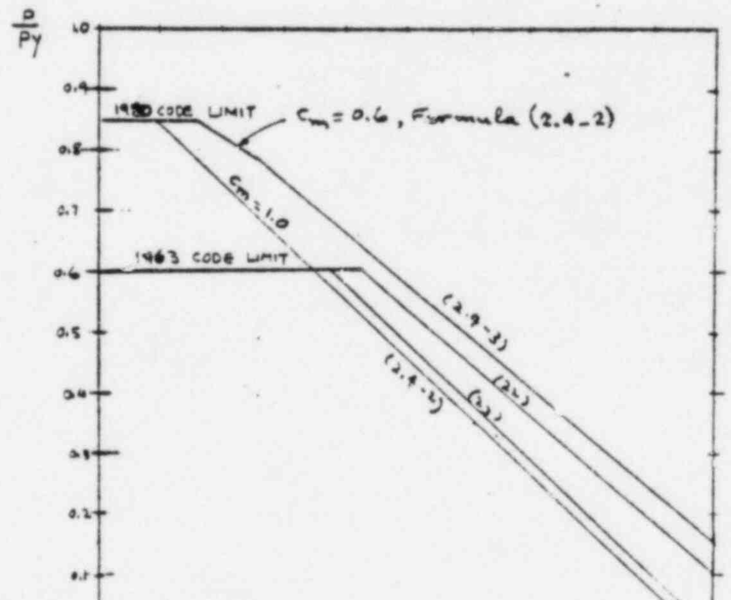
(2.4-2) $\frac{P}{P_{cr}} + \frac{C_m M}{(1 - \frac{P}{P_e})M_p} \leq 1.0$

$0.6 \leq C_m \leq 1.0$

(2.4-3) $\frac{P}{P_y} + \frac{M}{1.18M_p} \leq 1.0,\ M \leq M_p$

TYPICAL EXAMPLES





F_y = 36 ksi $\frac{kl}{r}$ = 30 14 w 150

DOUBLE CURVATURE
Assume braced in weak direction
$\therefore M_m = M_p$

<u>1963 Code</u>

Formula (21) $M = M_p$ when $P/Py \le 0.15$

$\frac{M}{M_p} \le 1.18 - 1.18(P/Py) \le 1.0$

Formula (22) $\frac{M}{M_p} \le B - G(P/Py) \le 1.0$

$M \le M_p$

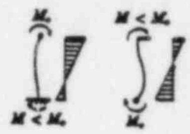
<u>1980 Code</u>

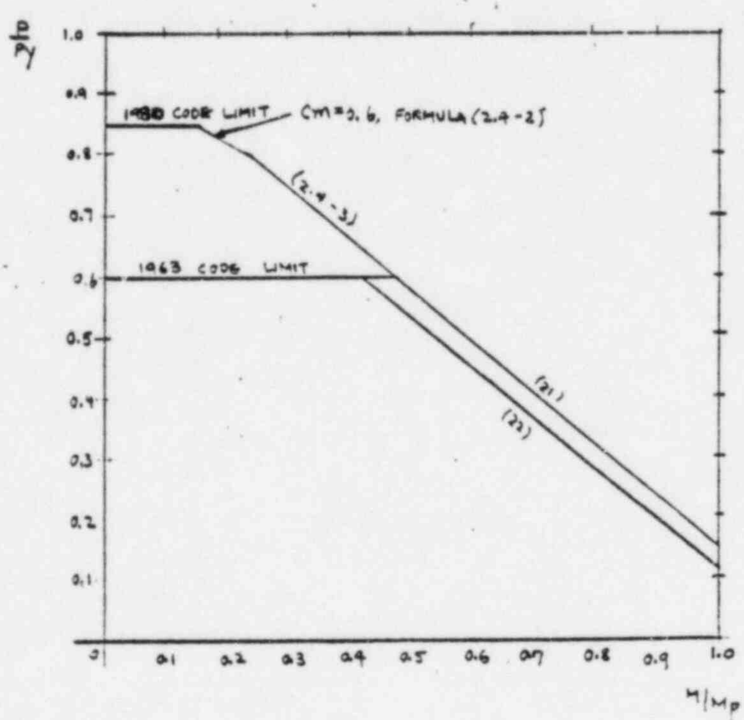
(2.4-2) $\frac{P}{P_{cr}} + \frac{C_m M}{(1 - \frac{P}{P_e})M_p} \le 1.0$

$0.4 \le C_m \le 0.6$

(2.4-3) $\frac{P}{P_y} + \frac{M}{1.18M_p} \le 1.0,\ M \le M_p$

<u>TYPICAL EXAMPLES</u>





$F_y = 36$ ksi $\frac{kl}{r} = 70$ 14 $\rightarrow$ 150 SINGLE CURVATURE
Assume braced in weak direction
i.e. $M_m = M_p$

1963 Code

Formula (22) $\frac{M}{M_p} \leq B - G(P/Py) \leq 1.0$

$M \leq M_p$

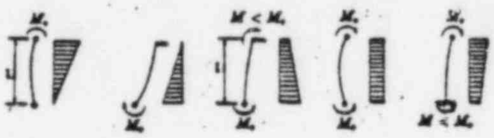
Formula (23) $\frac{M}{M_p} \leq 1.0 - H(P/Py) - J(P/Py)^2$

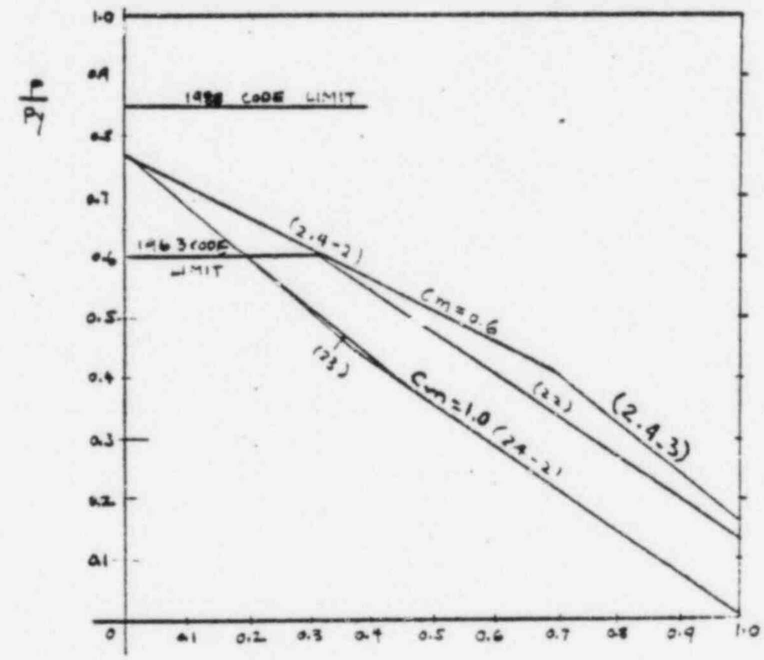
1980 Code

(2.4-2) $\frac{P}{P_{cr}} + \frac{C_m M}{(1 - \frac{P}{P_e})M_p} \leq 1.0$ $0.6 \leq C_m \leq 1.0$

(2.4-3) $\frac{P}{P_y} + \frac{M}{1.18M_p} \leq 1.0$, $M \leq M_p$

TYPICAL EXAMPLES





F_y = 36 ksi $\frac{kl}{r}$ = 70 14w 150 DOUBLE CURVATURE

Assume braced in weak direction
∴ $M_m = M_p$

<u>1963 Code</u>

Formula (21) $M = M_p$ when $P/Py \leq 0.15$

$$\frac{M}{M_p} \leq 1.18 - 1.18(P/Py) \leq 1.0$$

Formula (22) $\frac{M}{M_p} \leq B-G(P/Py) \leq 1.0$

$M \leq M_p$

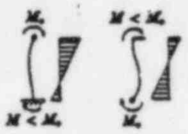
<u>1980 Code</u>

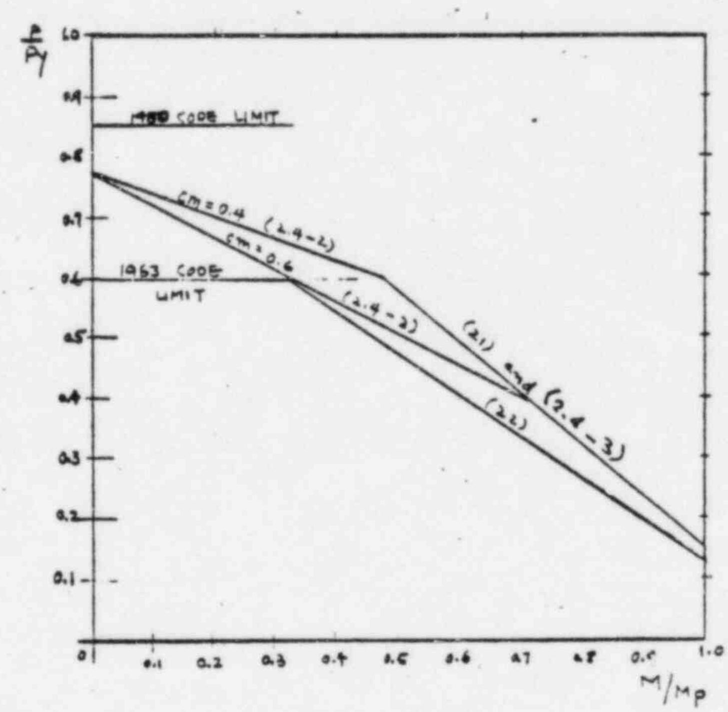
(2.4-2) $\frac{P}{P_{cr}} + \frac{C_m M}{(1 - \frac{P}{P_e})M_p} \leq 1.0$

$0.4 \leq C_m \leq 0.6$

(2.4-3) $\frac{P}{P_y} + \frac{M}{1.18M_p} \leq 1.0,\ M \leq M_p$

<u>TYPICAL EXAMPLES</u>





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F_y = 36 ksi $\quad \frac{kl}{r}$ = 100 14 W 150

SINGLE CURVATURE
Assume braced in weak direction
$\therefore M_m = M_p$

<u>1963 Code</u>

Formula (22) $\frac{M}{M_p} \leq B - G(P/Py) \leq 1.0$

$M \leq M_p$

Formula (23) $\frac{M}{M_p} \leq 1.0 - H(P/Py) - J(P/Py)^2$

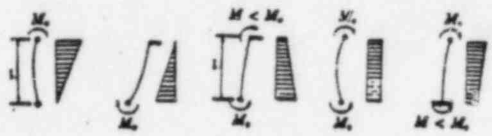
<u>1980 Code</u>

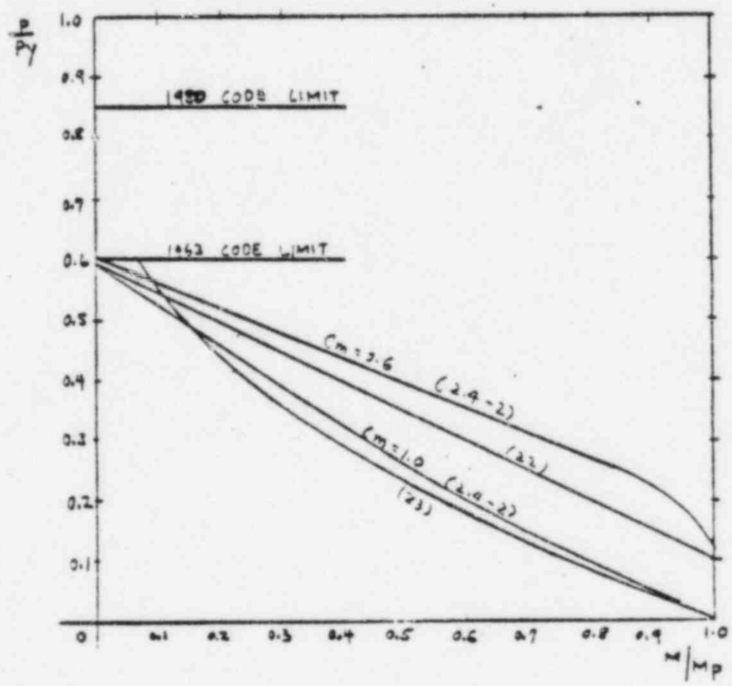
(2.4-2) $\frac{P}{P_{cr}} + \frac{C_m M}{(1 - \frac{P}{P_e})M} \leq 1.0$

$0.6 \leq C_m \leq 1.0$

(2.4-3) $\frac{P}{P_y} + \frac{M}{1.18M_p} \leq 1.0, \; M \leq M_p$

<u>TYPICAL EXAMPLES</u>





$F_y = 36$ ksi $\quad \frac{kl}{r} = 100$ 14W 150

DOUBLE CURVATURE
Assume braced in weak direction
$\therefore M_m = M_p$

1963 Code

Formula (21) $M = M_p$ when $P/Py \le 0.15$

$$\frac{M}{M_p} \le 1.18 - 1.18(P/Py) \le 1.0$$

Formula (22) $\frac{M}{M_p} \le B\text{-}G(P/Py) \le 1.0$

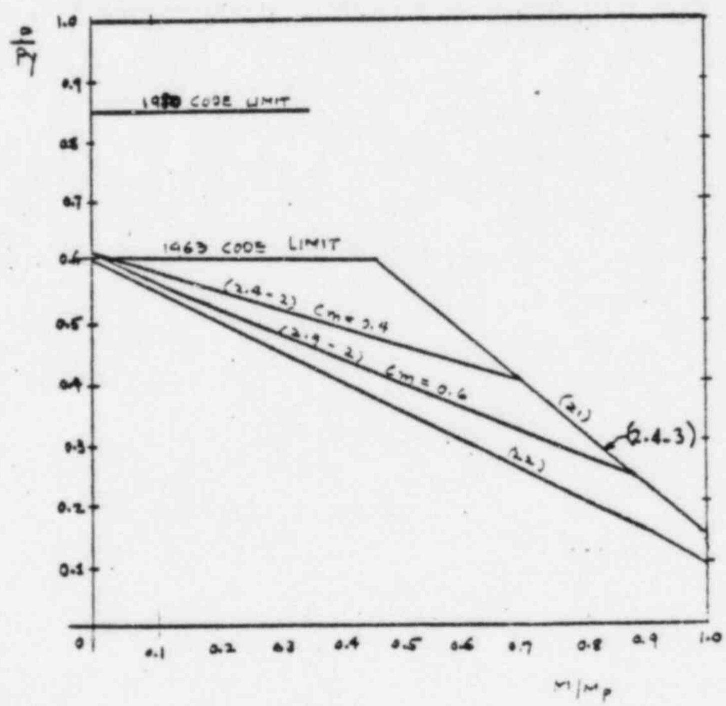
$$M \le M_p$$

1980 Code

(2.4-2) $\frac{P}{P_{cr}} + \frac{C_m M}{(1 - \frac{P}{P_e})M_p} \le 1.0$ $\quad 0.4 \le C_m \le 0.6$

(2.4-3) $\frac{P}{P_y} + \frac{M}{1.18M_p} \le 1.0,\ M \le M_p$

TYPICAL EXAMPLES



$F_y = 36$ ksi $\frac{kl}{r} = 30$ 12 WF 45

SIDESWAY ALLOWED
Assume braced in weak direction
$\therefore M_m = M_p$

1963 Code

Formula (21) $M = M_p$ when $P/Py \leq 0.15$

$$\frac{M}{M_p} \leq 1.18 - 1.18(P/Py) \leq 1.0$$

Formula (22) $\frac{M}{M_p} \leq B - G(P/Py) \leq 1.0$

$$M \leq M_p$$

Formula (23) $\frac{M}{M_p} \leq 1.0 - H(P/Py) - J(P/Py)^2$

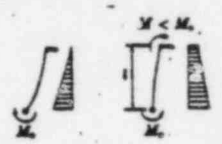
1980 Code

(2.4-2) $\frac{P}{P_{cr}} + \frac{C_m M}{(1 - \frac{P}{P_e})M_p} \leq 1.0$

$C_m = 0.85$

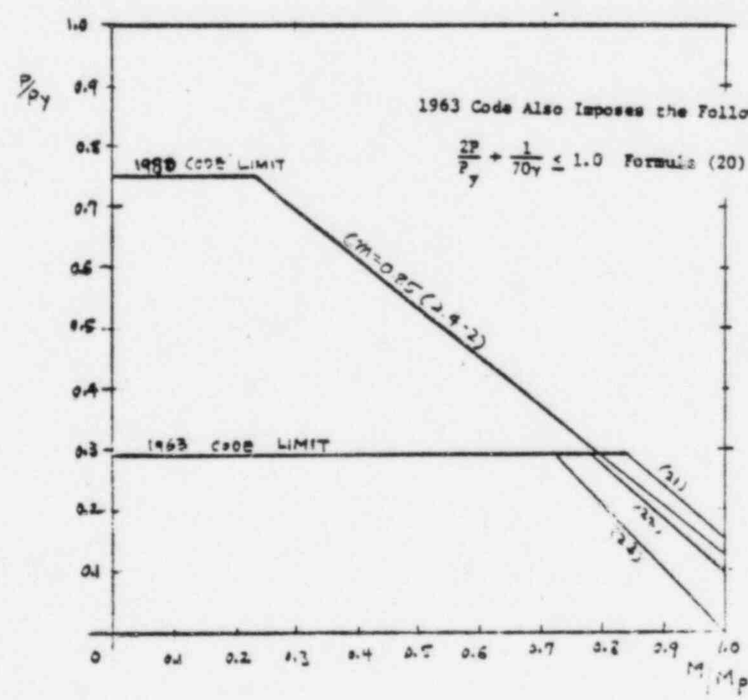
(2.4-3) $\frac{P}{P_y} + \frac{M}{1.18M_p} \leq 1.0,\ M \leq M_p$

TYPICAL EXAMPLES



1963 Code Also Imposes the Following Limit

$\frac{2P}{P_y} + \frac{1}{70r} \leq 1.0$ Formula (20)



$F_y = 36$ ksi $\quad \frac{kl}{r} = 30 \quad$ 14 WF 130 $\qquad$ SIDESWAY ALLOWED

Assume braced in weak direction

$\therefore M_m = M_p$

<u>1963 Code</u>

Formula (21) $M = M_p$ when $P/Py \leq 0.15$

$$\frac{M}{M_p} \leq 1.18 - 1.18(P/Py) \leq 1.0$$

Formula (22) $\frac{M}{M_p} \leq B - G(P/Py) \leq 1.0$

$$M \leq M_p$$

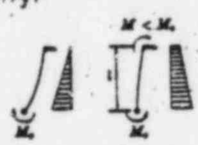
Formula (23) $\frac{M}{M_p} \leq 1.0 - H(P/Py) - J(P/Py)^2$

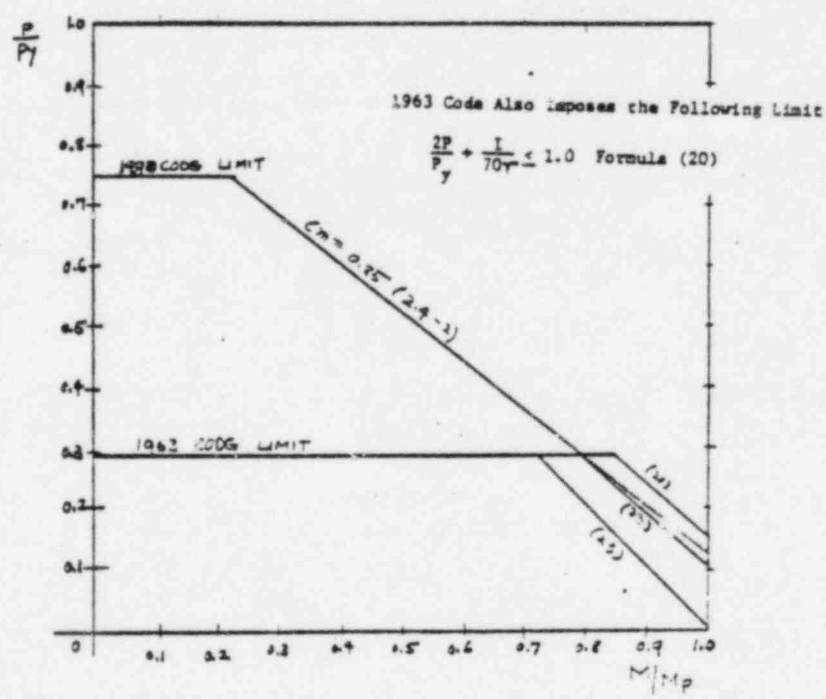
<u>1980 Code</u>

(2.4-2) $\frac{P}{P_{cr}} + \frac{C_m M}{(1 - \frac{P}{P_e})M_p} \leq 1.0$ $\qquad C_m = 0.85$

(2.4-3) $\frac{P}{P_y} + \frac{M}{1.18M_p} \leq 1.0, \; M \leq M_p$

<u>TYPICAL EXAMPLES</u>





CASE STUDY -9-

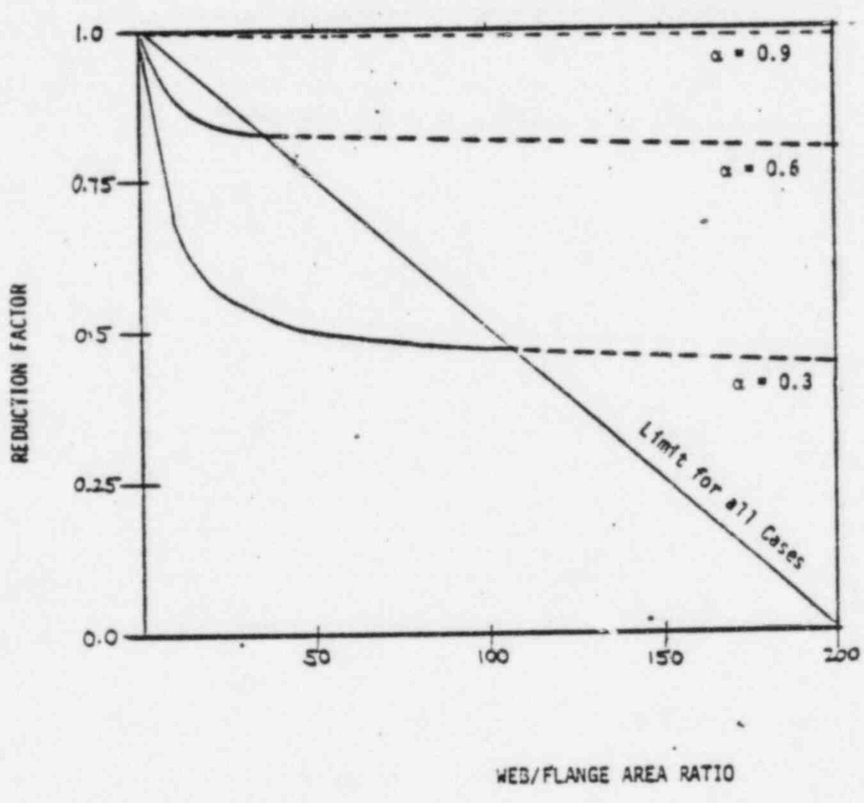
Comparison of AISC-1980 Section 1.10.6 with AISC-1963 Section 1.10.6, Reduction in Flange Stress, Hybrid Girders only.

The only change between the two codes is the introduction of Formula (1.10-6) for case of hybrid girder, in the 1980 code. Formula (1.10-5) of 1980 Code with Fb in ksi is identical to Formula (12) of 1963 with Fb in psi. Hybrid girder designed in 1963 would be designed in accordance with Formula (12) which is identical to (1.10-5) in 1980 Code. But a hybrid girder designed in accordance with 1980 has to conform to both Formulas (1.10-5) and (1.10-6). For F_b = 25 ksi and 50 ksi, we draw graphs of reduction Factor $\left(\frac{F_b'}{F_b}\right)$ Vs. Area of web to Area of Flange ratio (Aw/Af), using Formulas (1.10-5) and (1-10-6) for given $\alpha = 0.3$, 0.6, and 0.9 and for given h/t ratios (162, 172 & 182, for F_b = 25 ksi and 117, 127 & 137 for F_b = 50 ksi). We find in all six cases depending on Aw/Af ratio for $\alpha = 0.45$, Formula (1.10-6) in the 1980 code is quite conservative.

But for $0.45 < \alpha \leq 0.75$, Formula (1.10-6) or Formula (1.10-5) could be conservative as compared to each other depending on h/t ratio for given Fb. But for $\alpha > 0.75$, in any case, Formula (1.10-5) is more conservative. Thus we can make the following judgment on them.

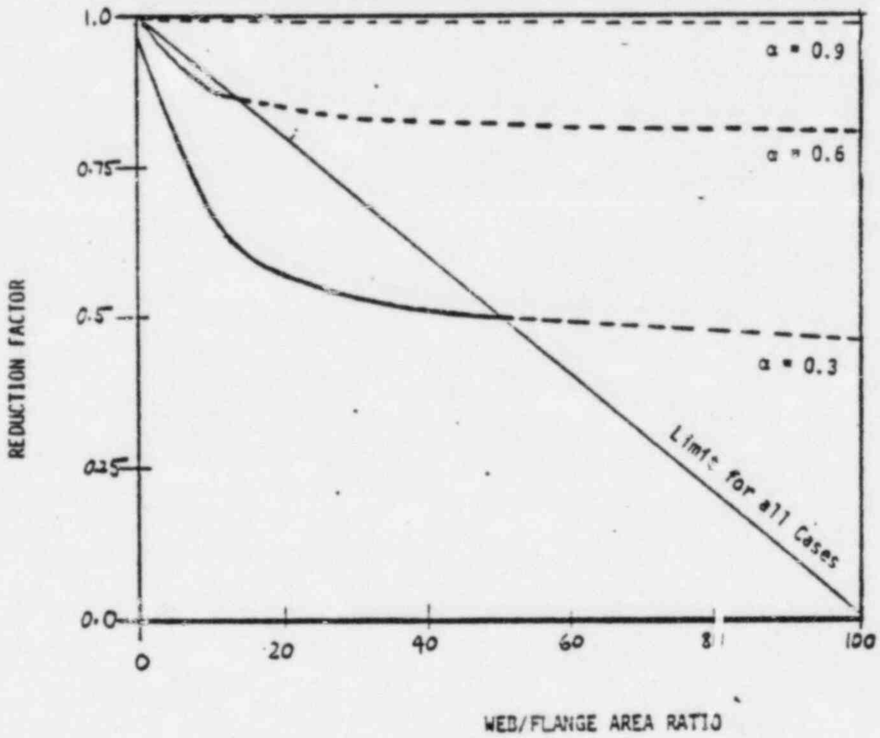
OLD Formulas	α	Scale
a) Formula (12), 1963 Code $F_b' \leq F_b\left[1.0 - 0.0005\frac{A_w}{A_f}\left(\frac{h}{t} - \frac{24000}{\sqrt{F_b}}\right)\right]$ with Fb in psi. b) Formula (1.10-5) 1980 code $F_b' \leq F_b\left[1.0 - 0.0005\frac{A_w}{A_f}\left(\frac{h}{t} - \frac{760}{\sqrt{F_b}}\right)\right]$, with Fb in ksi	≤ 0.45 and low A_w/A_f ratio	A
New Formula: Formula (1.10-6) 1980 code	0.45 to 0.75	B
$F_b' \leq F_b\left[\frac{12 + \left(\frac{A_w}{A_f}\right)(3\alpha - \alpha^3)}{12 + 2\left(\frac{A_w}{A_f}\right)}\right]$	≥ 0.75	C

AISC 1.10.6 1963/1980 CODE COMPARISON



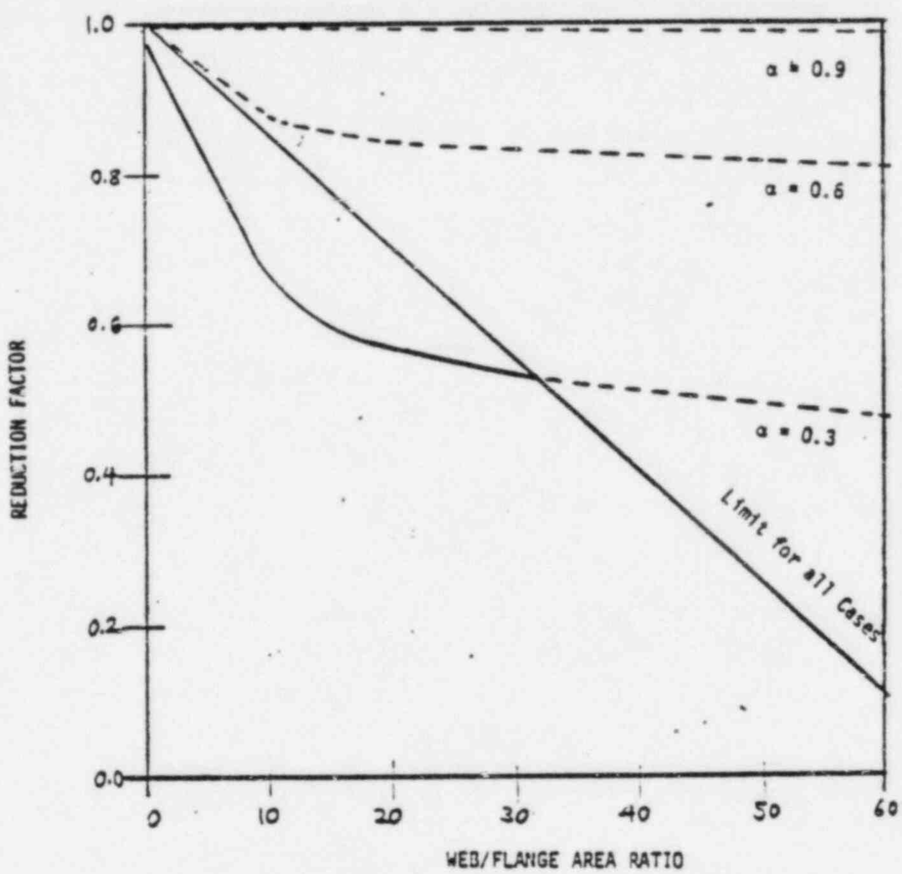
BENDING STRESS = 25KSI ALPHA=0.3, 0.6, 0.9, H/T RATIO = 162

AISC 1.10.6 1963/1980 CODE COMPARISON



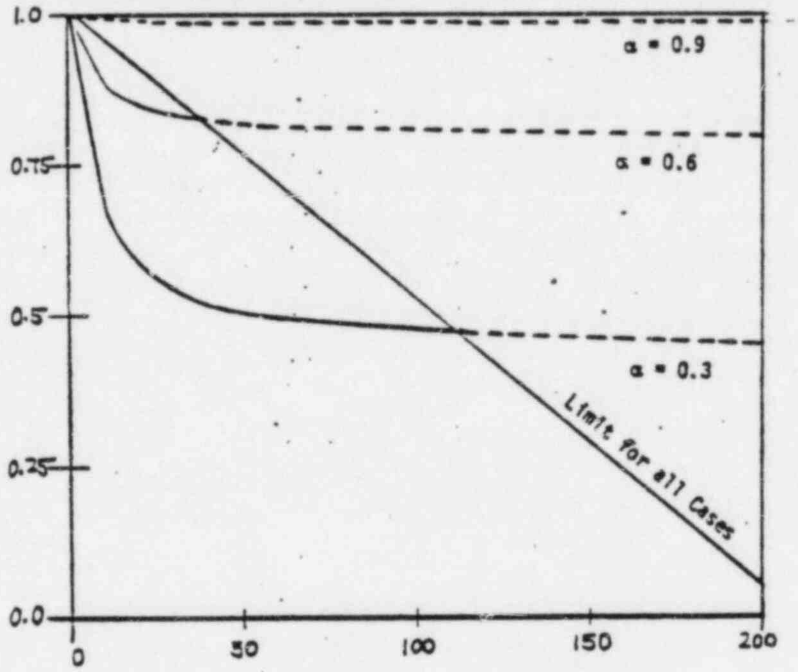
BENDING STRESS = 25KSI ALPHA=0.3, 0.6, 0.9, H/T RATIO = 172

AISC 1.10.6 1963/1980 CODE COMPARISON



BENDING STRESS = 25KSI   ALPHA=0.3, 0.6, 0.9, H/T RATIO = 182

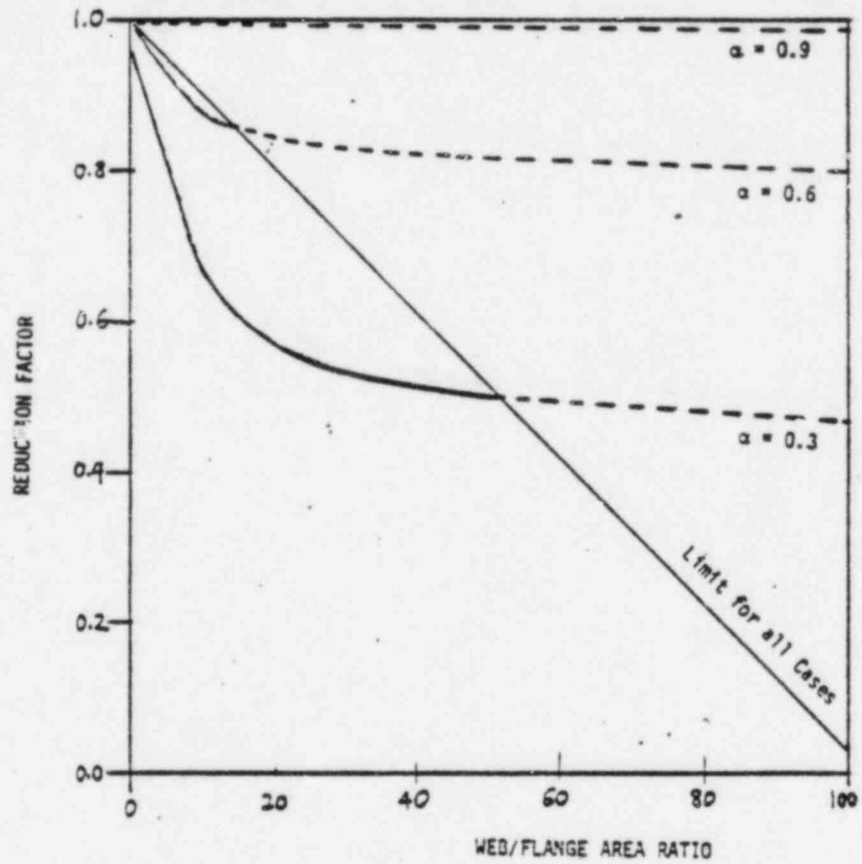
AISC 1.10.6 1963/1980 CODE COMPARISON



WEB/FLANGE AREA RATIO

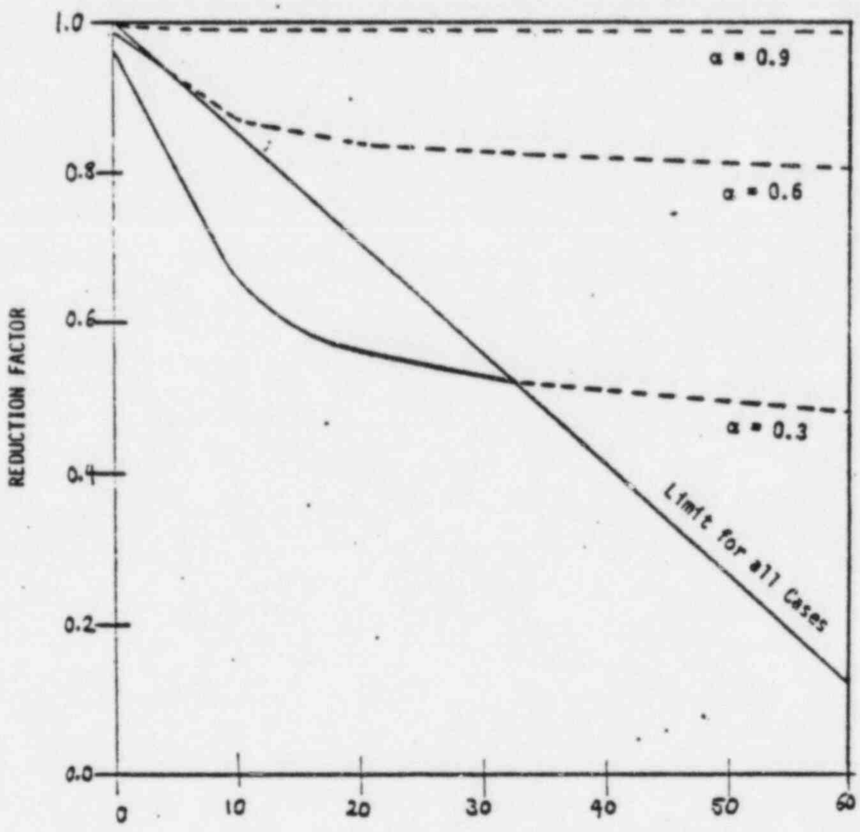
BENDING STRESS = 50KSI ALPHA=0.3, 0.6, 0.9, H/T RATIO = 117

AISC 1.10.6 1963/1980 CODE COMPARISON



BENDING STRESS = 50KSI ALPHA=0.3, 0.6, 0.9, H/T RATIO = 127

AISC 1.10.6 1963/1980 CODE COMPARISON



BENDING STRESS = 50KSI ALPHA=0.3, 0.6, 0.9, H/T RATIO = 137

CASE STUDY -10-

Comparison of Section (1.9.1.2) and Appendix C (AISC 1980) with Section 1.9.1 (AISC, 1963); width-thickness ratio of unstiffened elements subject to axial compression and compression due to bending.

In both sections the limit of width-thickness ratio is given for the following various cases.

CASE I: single-angle struts; double-angle struts with separators

CASE II: Struts comprising double angles in contact; angles or plates projecting from girders, columns, or other compression members; compression flanges of beams; Stiffeners on plate girders

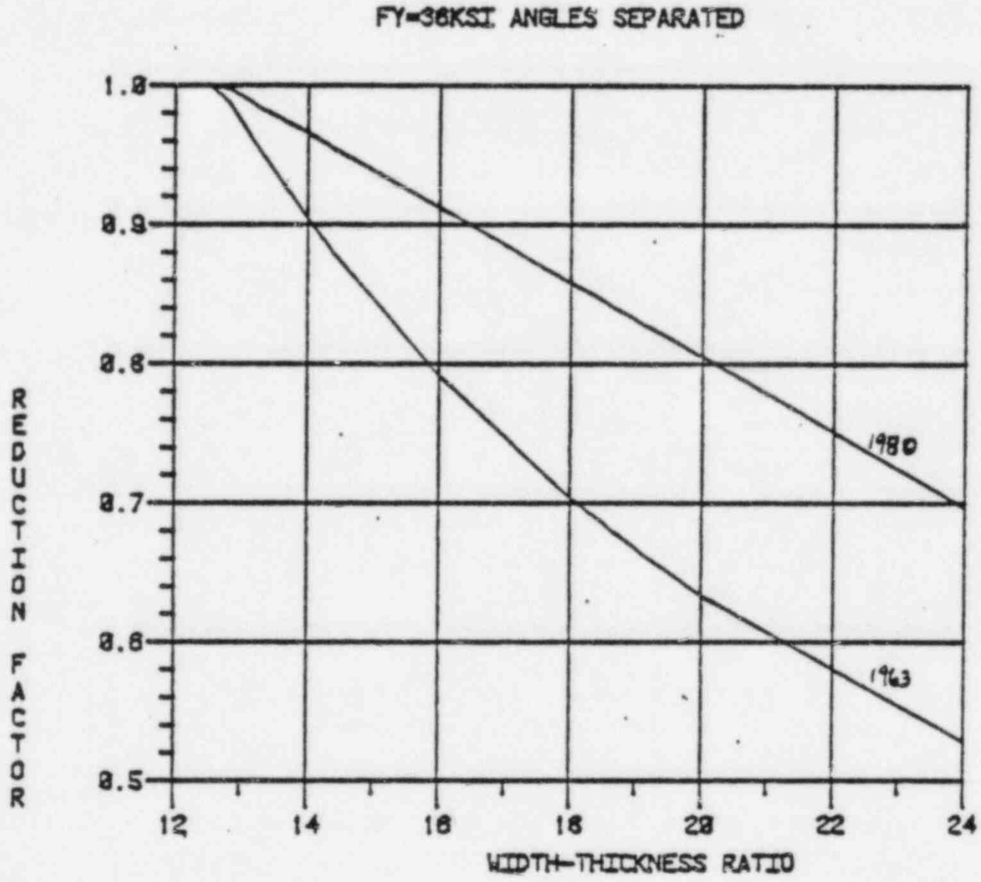
CASE III: Stems of tees

In AISC, 1980, according to the specifications for the above cases, when compression members exceed the allowable width-thickness ratio, the allowable stresses are reduced by a factor based on formulas given in appendix C which depends on yield stress (F_y) and the width-thickness ratio.

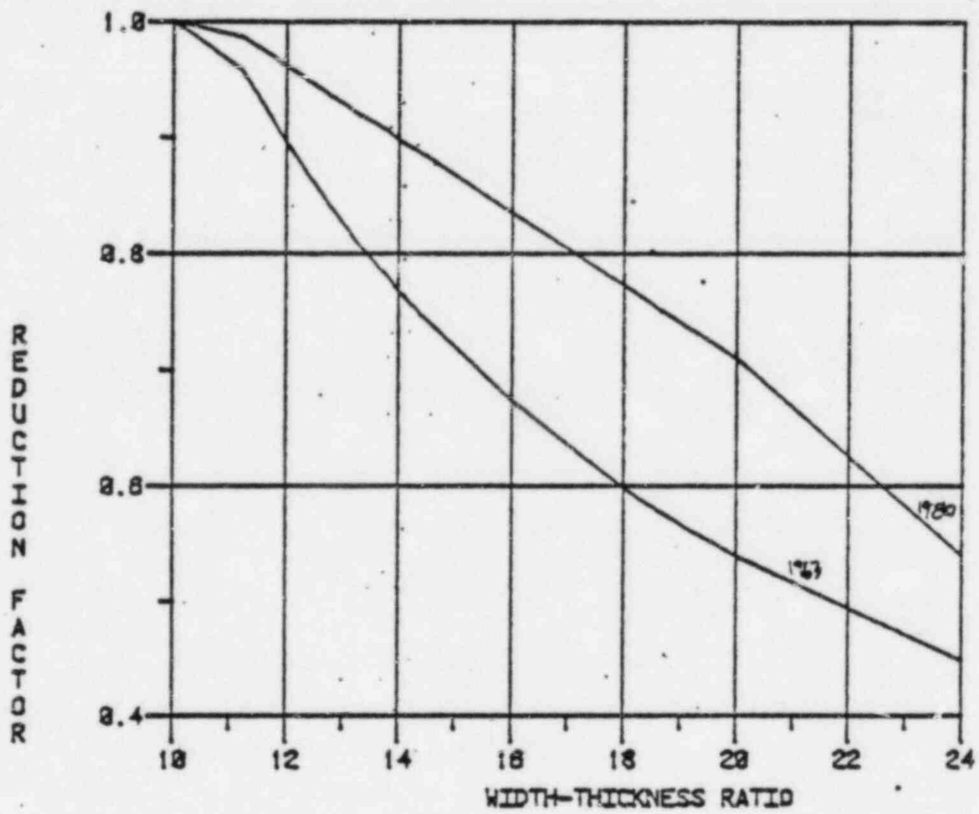
But according to AISC, 1963 Specifications, When compression members exceed the allowable width - thickness ratio, the member is acceptable if it satisfies the allowable stress requirements with a portion of width ie. effective width meets stress requirements.

For the case study, two values of Fy 36 ksi and 50 ksi are chosen. For the two values for typical angle section and T sections given in AISC Manual graphs have been plotted for Reduction Factor VS Width - thickness ratio.
Reduction Factor for AISC, 1980 Code is based on formulas given in appendix C and for AISC, 1963, reduction factor is the ratio of effective width to actual width of the section.

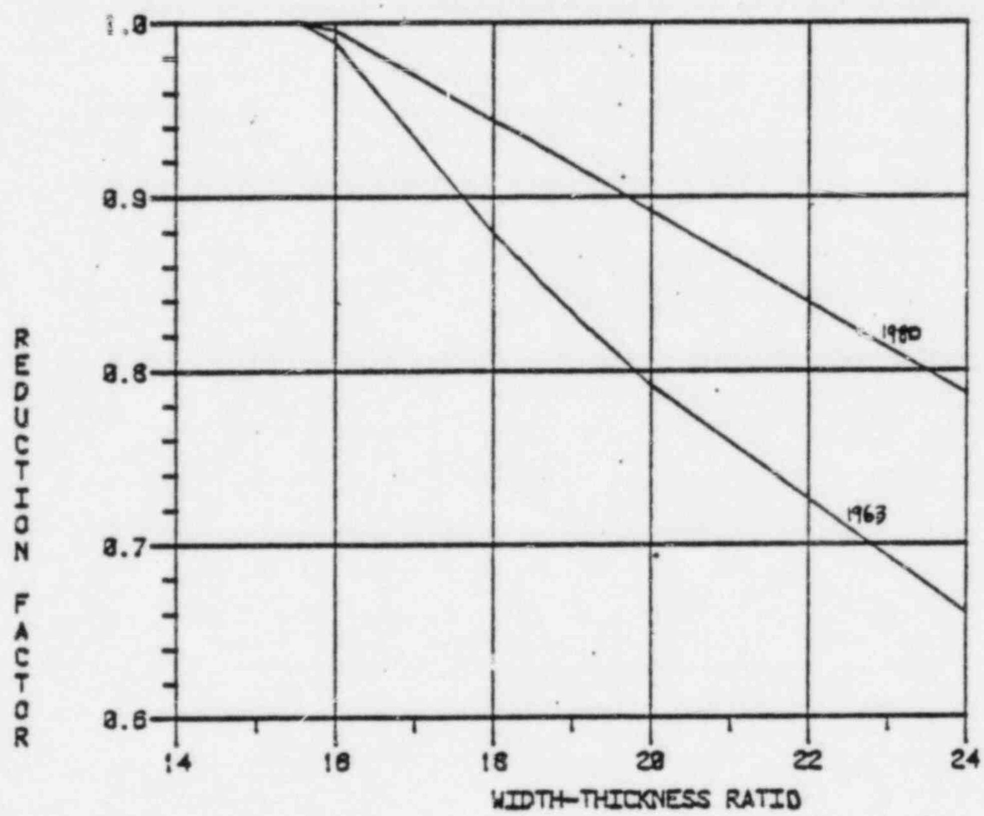
Based on the graphs, the change for case I and Case II at higher width/thickness ratio would be a C change, as Specifications were more conservative in 1963 code. But for Case III the change in Specification is A change as it is more conservative in 1980 Code, at higher width - thickness ratio.


FY=36KSI ANGLES SEPARATED
1.0
0.9
0.8
0.7
0.6
0.5
REDUCTION FACTOR
12
14
16
18
20
22
24
WIDTH-THICKNESS RATIO
1980
1963

FY=50KSI ANGLES SEPARATED



FY=36KSI ANGLES IN CONTACT



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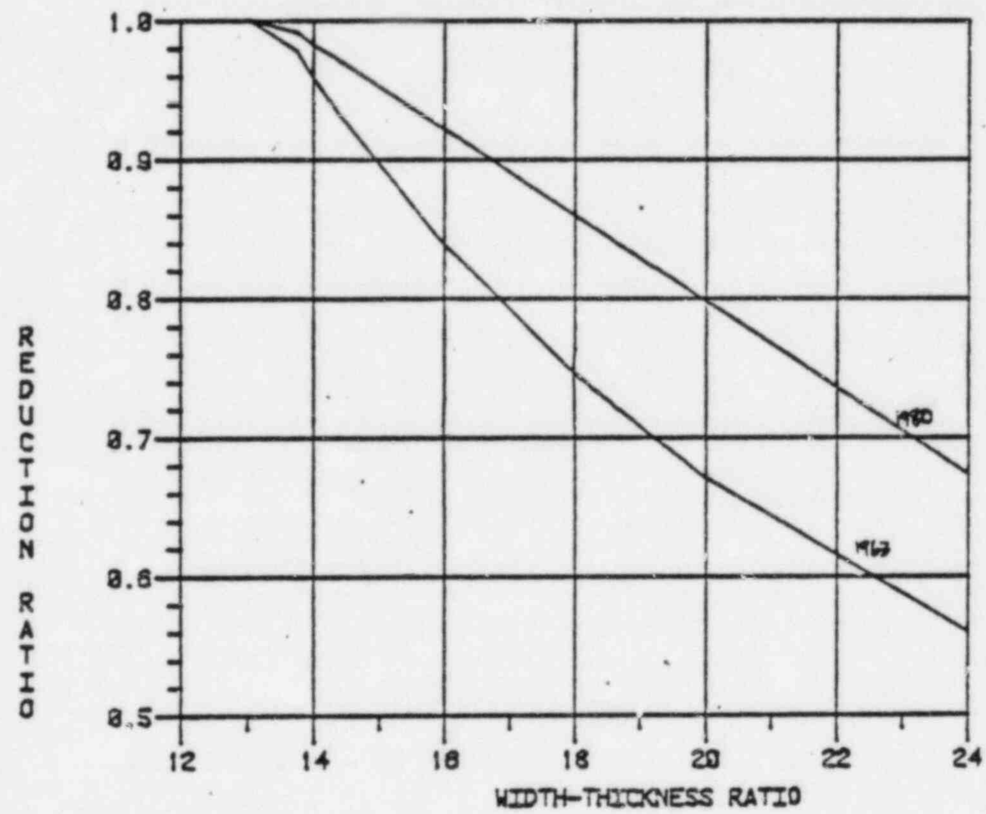
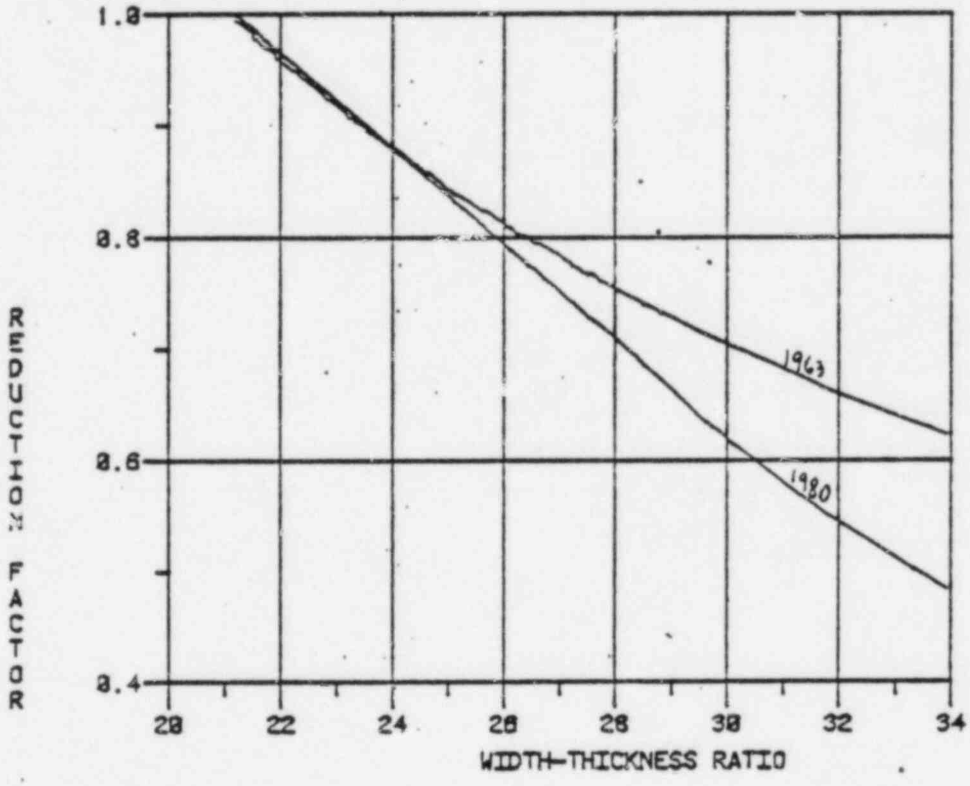
	By RA	Date SEPT'81	Ch'k'd EMW.	Date 10/81	Rev.	Date

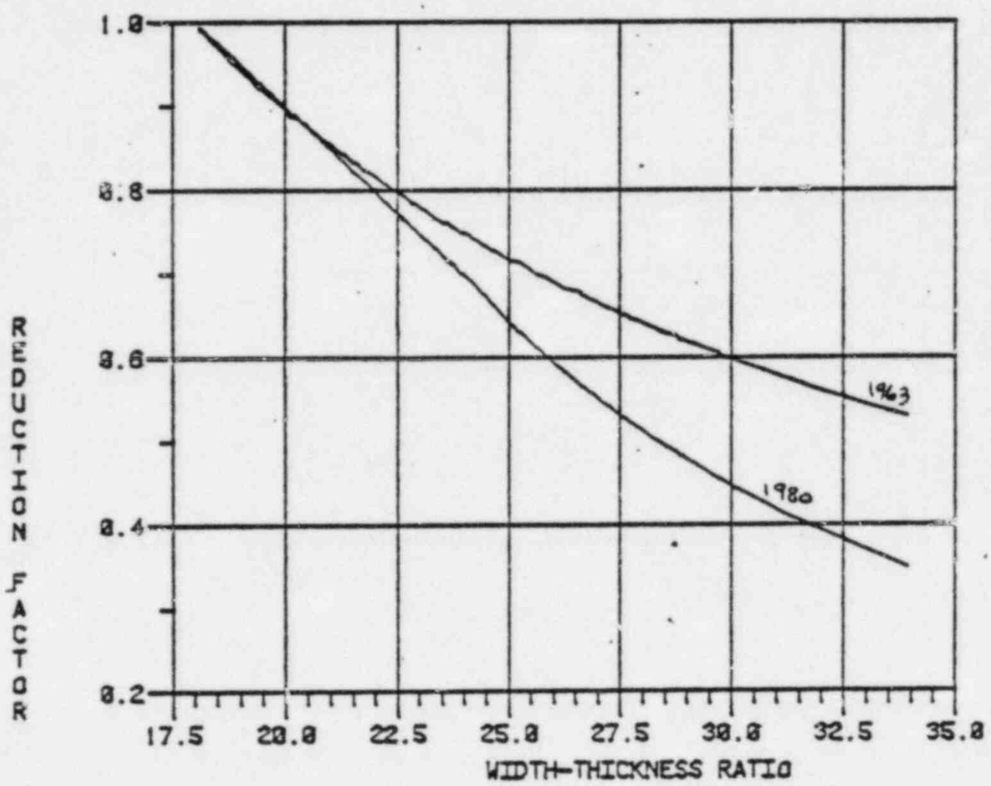
FY=50KSI ANGLES IN CONTACT



FY=36KSI T SHAPES



FY=50KSI T SHAPES



CASE STUDY -11-

Comparison of AISC 1980 Section 1.11.4 with AISC 1963 Section 1.11.4; Shear connectors for composite beams, where longitudinal reinforcing steel acts with beam.

According to AISC 1980, Formula (1.11-5)

$$V_h = A_{sr} F_{yr}/2 \quad (1.11\text{-}5)$$

is given for continuous composite beam where longitudinal reinforcing steel is considered to act compositely with the steel beam in the negative moment regions, to calculate the total horizontal shear to be resisted by shear connectors between an interior support and each adjacent point of contraflexure.

Whereas in AISC 1963 specifications, the total horizontal shear to be resisted between the point of maximum positive moment and each end or a point of contraflexure in continuous beams is given as the smaller value of Formula (18) and (19)

$$V_h = 0.85 \frac{f'_c A_c}{2} \quad (18)$$

and

$$V_h = \frac{A_s F_y}{2} \quad (19)$$

There is no separate formula for negative moment region in AISC, 1963. The above formulas are the same in AISC, 1980; Formula (1.11-3) and (1.11-4) for the positive moment region. Moreover in AISC, 1963, there is no consideration of reinforcing steel in concrete acting compositely with the steel beam in negative moment regions.

This implies that in computing the section modulus at the points of negative bending, reinforcement parallel to the steel beam, and lying within the effective width of slab may be included according to AISC, 1980. But it is not allowed to include reinforcing steel in computing the section modulus for the above case as per the specifications of AISC 1963. Thus design criteria is being liberalized in AISC 1980. Since the quantification of this liberal criteria is unknown, this change can best be classified as <u>C</u>. Any Composite beam designed as per AISC 1963 specifications will show more moment capacity when calculated according to AISC, 1980 Specifications.

CASE STUDY -12-

The allowable peripheral Shear Stress (Punching Shear Stress) as stated in the B & PV ASME Code Section III Div. 2, 1980 (ACI 359-80) Para. CC-3421.6 is limited to v_c where v_c shall be calculated as the weighted average of v_{ch} and v_{cm}

$$v_{ch} = 4\sqrt{f'_c}\sqrt{1+(f_m/4\sqrt{f'_c})}$$

$$v_{cm} = 4\sqrt{f'_c}\sqrt{1+(f_h/4\sqrt{f'_c})}$$

The ACI 318-63 Code Section 1707 states that the ultimate Shear Strength v_u shall not exceed $v_c = 4\sqrt{f'_c}$.

Comparing the above two cases the following is concluded;

When: **Scale**

1. Membrane stresses are compressive
   318-63 is more conservative (C)
2. Membrane stresses are tensile
   318-63 is less conservative (A)

Scale

3. Membrane stresses are zero
   318 - 63 is identical — No rating

4. Membrane stresses are opposite in sign
   318 - 63 could be less conservative — (A)

CASE STUDY -13-

The B & PV ASME Code Section III Division 2, 1980 (ACI 359-80) Para. CC-3421.7 states that the shear stress taken by the concrete resulting from pure torsion shall not exceed v_{ct} where

$$v_{ct} = 6\sqrt{f'_c}\sqrt{1 + \frac{f_h + f_m}{6\sqrt{f'_c}} + \frac{f_m f_h}{(6\sqrt{f'_c})^2}}$$

While the ACI 318-63 Code Section 1707 limits the ultimate Shear Strength v_u to

$$v_c = 4\sqrt{f'_c}$$

From the above two cases the following is concluded;

When:		Scale
1. Membrane stresses are compressive 318-63 is more conservative		(C)
2. Membrane stresses are tensile 318-63 is less conservative		(A)

Scale

3. Membrane stresses are zero
   318-63 is more conservative (C)

4. Membrane stresses are opposite in sign
   318-63 could be less conservative (A)

CASE STUDY -14-

Section 1301(c) - Allowable bond stresses - working stress design.

Allowable bond stresses for working stress design in the 318-63 code were newly described as functions of both the square root of concrete compressive strength and reinforcing bar diameter. The 318-56 code defined allowable bond stress as a linear function of concrete compressive strength only.

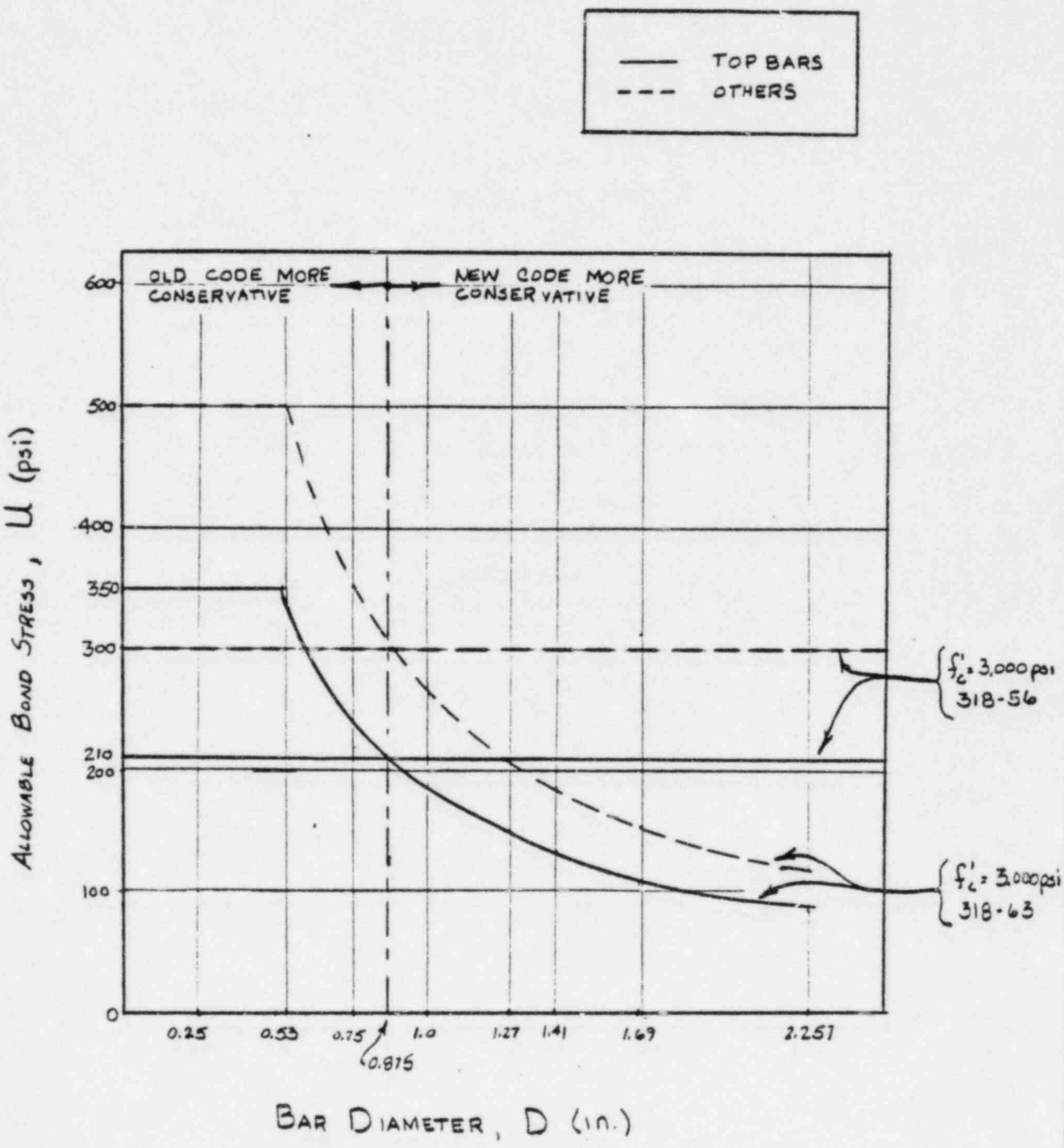
Plots for three commonly used concrete compressive strengths showing bond stress allowed by each code for deformed bars conforming to ASTM-A-305 plotted against bar diameter show that for small diameter bars the old code is more conservative and for large diameter bars the new code is more conservative. For bars No. 10, 11, 14 and 18 the new code is considerably more conservative.

Based on the plots shown, a reasonable interpretation of the code changes as regards scale rating is that for deformed bars conforming to ASTM-A-305:

1. For reinforcing bars with diameter less than or equal to 0.875 in. (No. 7 bar) - Scale C

2. For reinforcing bars with diameter greater than 0.875 in. (No. 7 bar) - Scale A

3. For deformed bars conforming to ASTM-A-408 for all diameters - Scale A


TOP BARS
OTHERS
OLD CODE MORE CONSERVATIVE
NEW CODE MORE CONSERVATIVE
600
500
400
350
300
210
200
100
0
ALLOWABLE BOND STRESS, U (psi)
0.25
0.53
0.75
1.0
1.27
1.41
1.69
2.257
0.875
BAR DIAMETER, D (in.)
f'_c = 3,000 psi 318-56
f'_c = 3,000 psi 318-63

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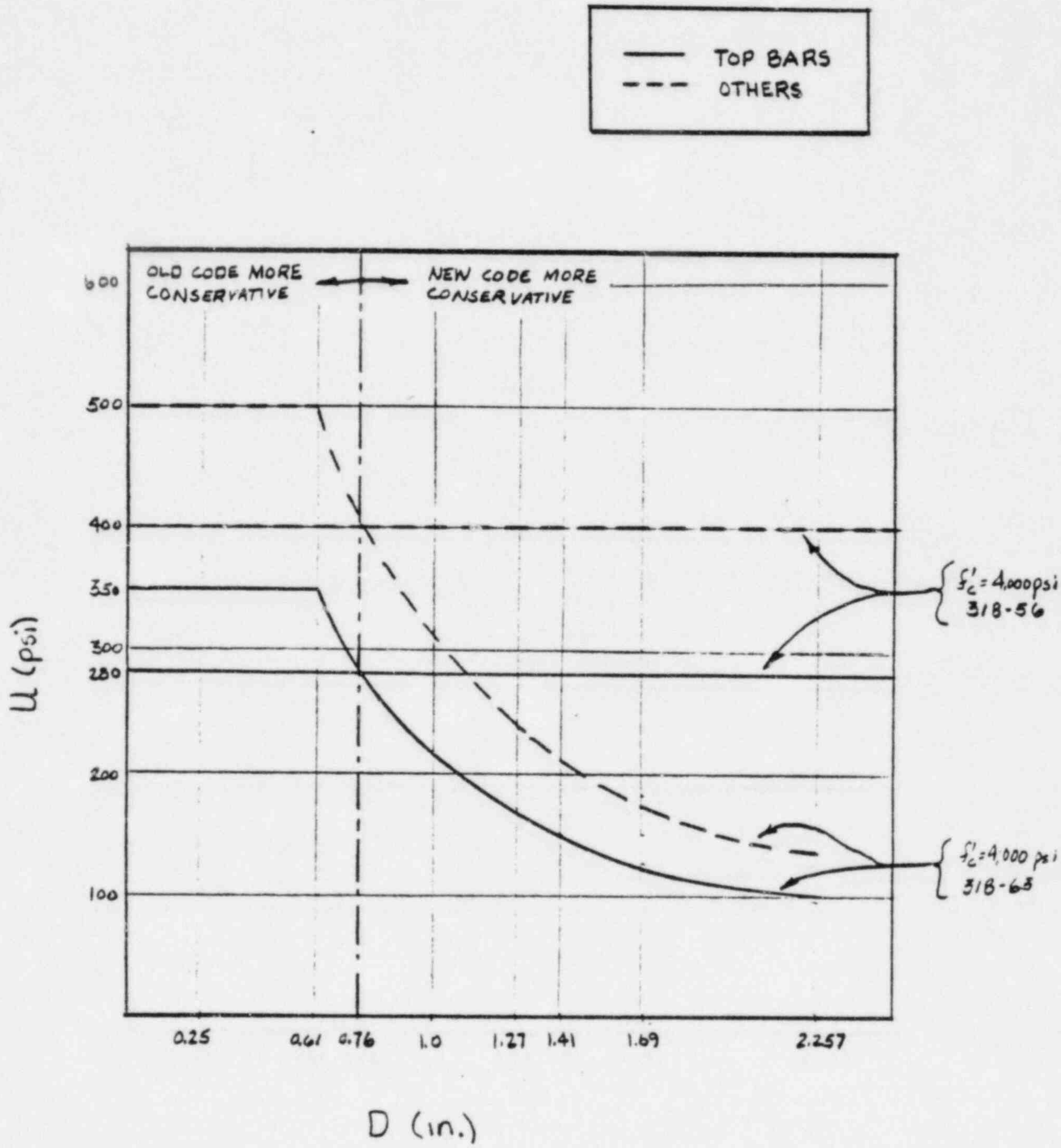
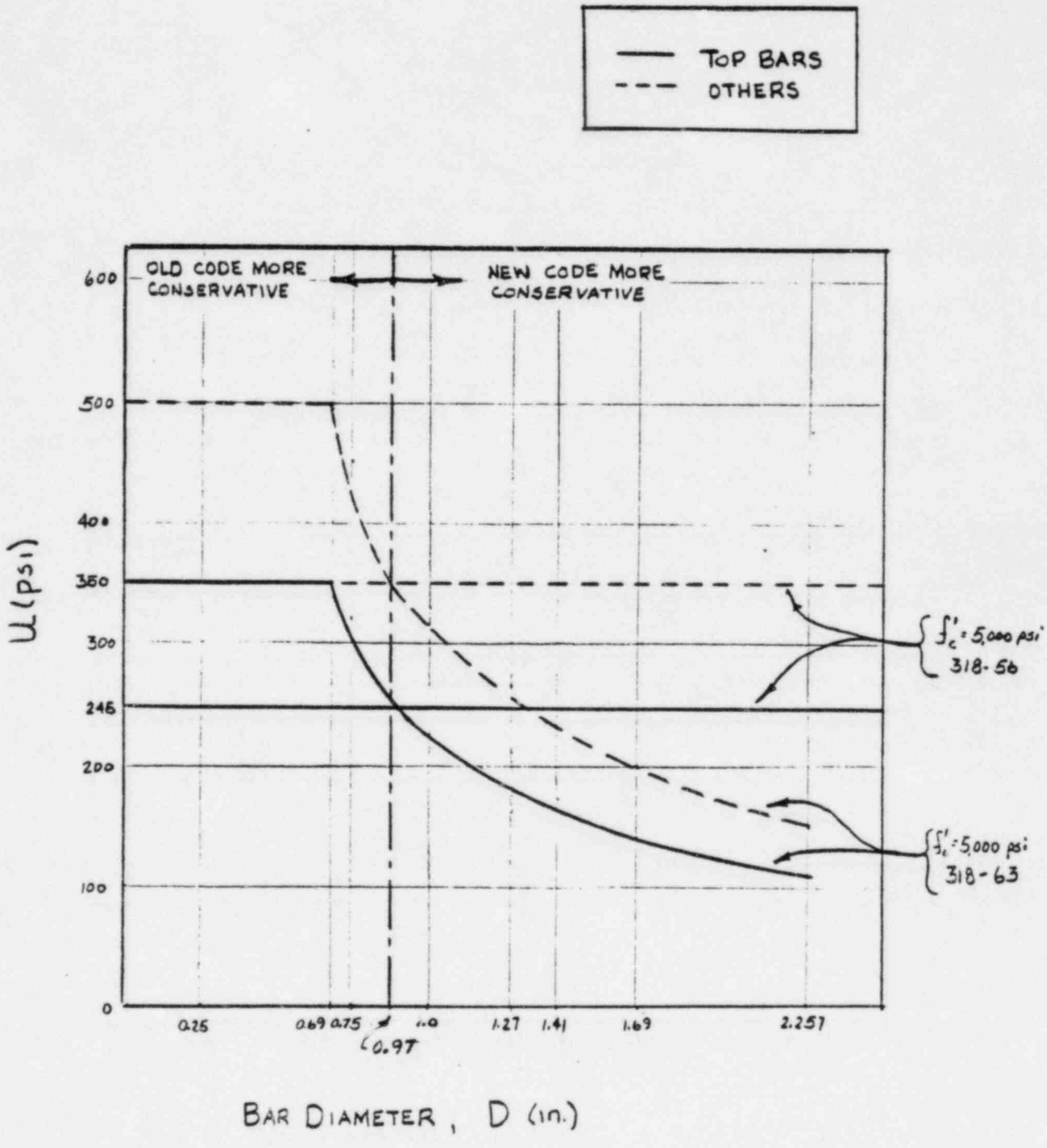
	By MLP Date April '82	Ch'k'd EMW Date 4/82	Rev. Date



FORM CS-FRC-81

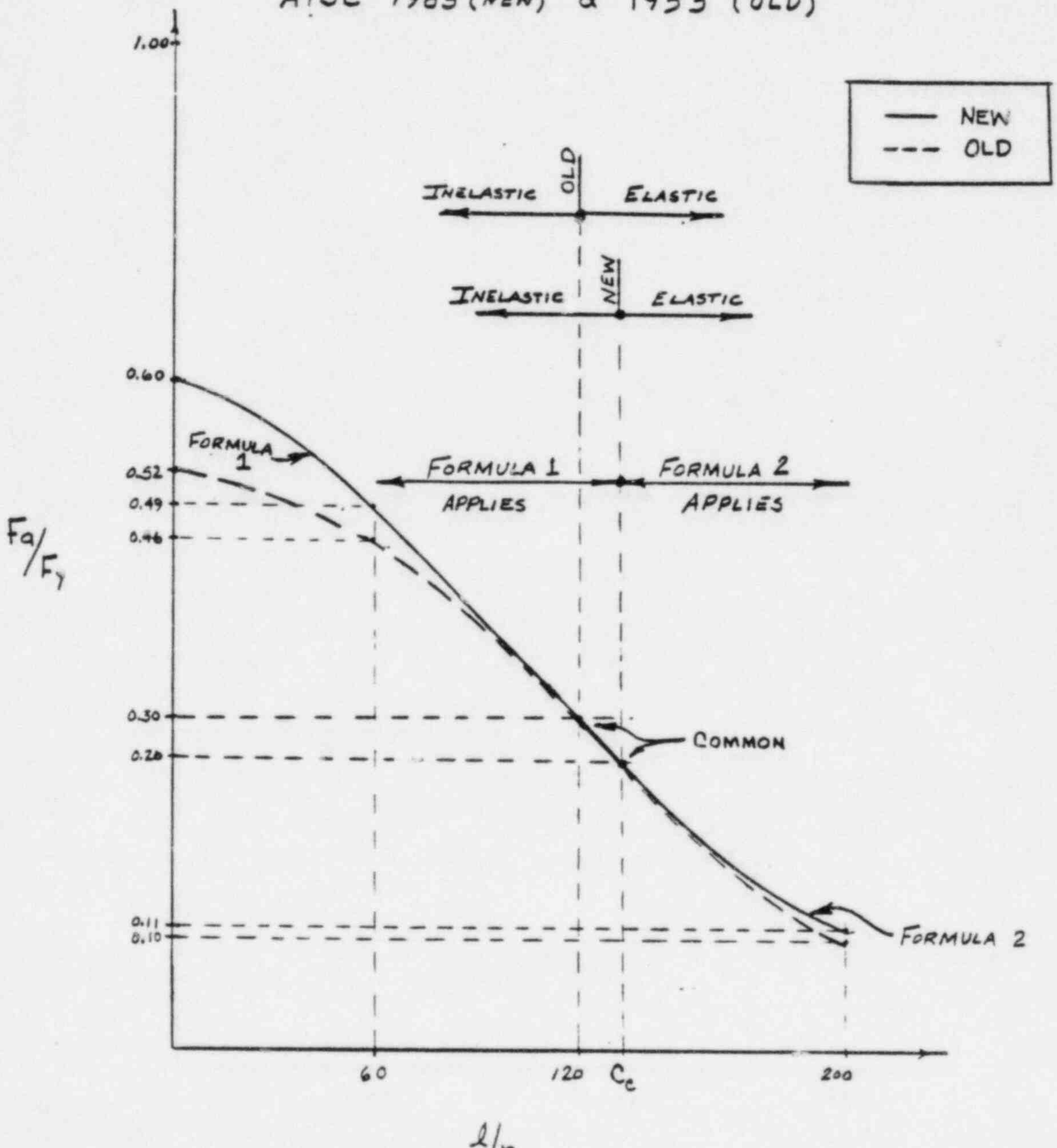

TOP BARS
OTHERS
OLD CODE MORE CONSERVATIVE
NEW CODE MORE CONSERVATIVE
U (psi)
600
500
400
350
300
245
200
100
0
0.25
0.69
0.75
1.0
1.27
1.41
1.69
2.257
0.97
f'_c = 5,000 psi
318-56
f'_c = 5,000 psi
318-63
BAR DIAMETER, D (in.)

CASE STUDY -15-

Ⓐ Subsection 1.5.1.3.1, 1963 code & 1st para. section 15(a)(2), 1953 code.

ALLOWABLE COMPRESSION ON MAIN MEMBERS

AISC 1963 (NEW) & 1953 (OLD)



SCALE RATING C

(B) Subsection 1.5.1.3.3, 1963 code & 2ND paragraph section 15(a)(2), 1953 code.

For axially loaded bracing and secondary members when l/r exceeds 120 both the new (Formula 3) and old allowable stress equations are shifted vertically up on the above type plot by the same function

∴ new allowables remain above the old

∴ Scale Rating C

APPENDIX D

ACI CODE PHILOSOPHIES

Franklin Research Center
A Division of The Franklin Institute
The Benjamin Franklin Parkway, Phila., Pa. 19103 (215) 448-1000

ACI CODE PHILOSOPHIES

The American Concrete Institute (ACI) Building Code Requirements for Reinforced Concrete delineate two philosophies of design which have long been in use: the so-called working stress method, which was in general acceptance and predominant use from early in this century to the early 1960's, and the ultimate strength method, which has been rapidly replacing working stress since about 1963.

Working Stress Method

The working stress method of design is referred to as the "alternate design method" by the most recent ACI code. By this method, the designer proportions structural elements so that internal stresses, which result from the action of service loads* and are computed by the principles of elastic mechanics, do not exceed allowable stress values prescribed by the code.

The allowable stresses as prescribed by the ACI code are set such that the stresses under service load conditions will be within the elastic range of behavior for the materials involved. As a result of this, the assumption of straight line stress-strain behavior applies reasonably for properly designed structural members. The member forces used in design by this method are those which result from an elastic analysis of the structure under the action of the service loads.

Ultimate Strength Design

The ultimate strength method is referred to as the "strength method" in the most recent ACI code. By this method, the proportioning of the members is based on the total theoretical strength of the member, satisfying equilibrium and compatibility of stress and strain, at failure. This theoretical strength is modified by capacity reduction factors which attempt to assess the variations to be encountered in material, construction tolerances, and calculation approximation.

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*Service loads are defined as those loads which are assumed to occur during the service life of the structure.

Strength Reduction Factor

In the present code, the capacity reduction factor (ϕ) varies for the type of member and is considered to account for the relative seriousness of the member failure as regards the overall integrity of the structure.

Load Factors

Also, by this method, the designer increases the service loads by applying appropriate load factors to obtain the ultimate design loads in an attempt to assess the possibility that the service loads may be exceeded in the life of the structure. The member forces used to proportion members by this method are based on an elastic analysis of the structure under the action of the ultimate design loads.

Importance of Ductility

A critical factor involved in the logic of ultimate strength design is the need to control the mode of failure. The present ACI code, where possible, has incorporated a philosophy of achieving ductility in reinforced concrete designs. Ductility in a structural member is the ability to maintain load carrying capacity while significant, large deformations occur. Ductility in members is a desired quality in structures. It permits significant redistribution of internal loads allowing the structure to readjust its load resistance pattern as critical sections or members approach their limiting capacity. This deformation results in cracking and deflections which provide a means of warning in advance of catastrophic collapse. Under conditions of loading where energy must be absorbed by the structure, member ductility becomes very important.

This concern for preserving ductility appears in the present code in many ways and has guided the changes in code requirements over the recent decades. Where research results have confirmed analysis and intuition, the code has provided for limiting steel percentages, reinforcing details, and controls--all directed at guaranteeing ductility. In those aspects of design where ductility cannot be achieved or insured, the code has required added strength to insure potential failure at the more ductile sections of structures.

Examples of this are evident in the more conservative capacity reduction factors for columns and in the special provisions required for seismic design.

Strength and Serviceability in Design

There are many reasons for the recent trend in reinforced concrete codes toward ultimate strength rather than working stress concepts. Research in reinforced concrete has indicated that the strain distributions predicted by working stress computations in general do not exist in the members under load. There are many reasons for this lack of agreement. Concrete is a brittle, non-linear material in its stress-strain behavior, exhibiting a down trend beyond its ultimate stress and characterized by a tensile stress-strain curve which in all its features is approximately on the order of one tenth smaller than its compressive stress-strain curve.

Time-dependent shrinkage and creep strains are often of significant magnitude at service load levels and are difficult to assess by working stress methods. While ultimate strength methods do not eliminate these factors, they become less significant at ultimate load levels. In addition, ultimate strength methods allow for more reasonable approximations to the non-linear concrete stress-strain behavior.

In the analyses of structures, the designer must, by necessity, make certain assumptions which serve to idealize the structures. The primary assumptions are that the structure behaves in a linearly elastic manner, and that the idealized member stiffness is constant throughout each member and constant in time.

Working stress logic does not lend itself well to accounting for variations in stiffness caused by cracking and variations in material properties with time. Although the ultimate strength method in the present code requires an elastic structural analysis to determine member forces for design, it recognizes these limitations and, in concept, anticipates the redistribution resulting from ductile deformation at the most critically stressed sections and in fact proportions members so that redistribution will occur.

In addition to strength, a design must satisfy serviceability requirements. In some designs, serviceability factors (such as excessive deflection, cracking, or vibration at service load) may prove to be more important than strength. Computations of the various serviceability factors are generally at service load levels; therefore, the present code uses elastic concepts in its controls of serviceability.

Factors of Safety

Factors of safety* are subjects of serious concern in this review. For working stress, the definition of the factor of safety is often considered to be the ratio of yield stress to service load stress. This definition becomes suspect or even incorrect where nonlinear response is involved. For ultimate strength, one definition of factors of safety is the ratio of the load that would cause collapse to the service or working load. As presented in the present code, a factor of safety is included for a variety of reasons, each of which is important but has no direct interrelation with the other.

The present ACI code has divided the provisions for safety into two factors; the overload factors and the capacity reduction factors (considered separately by the code) are both provisions to insure adequate safety but for distinctly different reasons. The code provisions imply that the total theoretical strength to be designed for is the ratio of the overload factor (U) over the capacity reduction factor (ϕ). The present ACI code has assigned values to the above factors such that the ratio U/ϕ ranges from about 1.5 to 2.4 for reinforced concrete structural elements.

*Factors of safety (FS) are related to margins of safety (MS) through the relation MS = FS - 1.

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

March 18, 1982

Docket No. 50-29
LS05-82-03-083

RECEIVED

Mr. James A. Kay
Senior Engineer - Licensing
Yankee Atomic Electric Company
1671 Worcester Road
Framingham, Massachusetts 01701

Dear Mr. Kay:

SUBJECT: SEP TOPIC III-7.D - CONTAINMENT STRUCTURAL INTEGRITY TESTING
YANKEE NUCLEAR POWER STATION

Enclosed is our final evaluation of SEP Topic III-7.D. This evaluation compares your facility, as described in your Safety Analysis Report dated November 18, 1981, and other information available on Docket 50-29, with criteria used in licensing new facilities.

The evaluation concludes that the structural integrity test performed at Yankee is satisfactory when compared to current criteria.

This evaluation will be a basic input to the integrated safety assessment for your facility and may be changed in the future if your facility design is changed or if NRC criteria relating to this topic are modified before the integrated assessment is completed.

Sincerely,

Ralph Caruso

Ralph Caruso, Project Manager
Operating Reactors Branch No. 5
Division of Licensing

Enclosure:
As stated

cc w/enclosure:
See next page

~~8203220210~~

Mr. James A. Kay

cc
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Yankee Atomic Electric Company
25 Research Drive
Westborough, Massachusetts 01581

Greenfield Community College
1 College Drive
Greenfield, Massachusetts 01301

Chairman
Board of Selectmen
Town of Rowe
Rowe, Massachusetts 01367

Energy Facilities Siting Council
14th Floor
One Ashburton Place
Boston, Massachusetts 02108

U. S. Environmental Protection
Agency
Region I Office
ATTN: EIS COORDINATOR
JFK Federal Building
Boston, Massachusetts 02203

Resident Inspector
Yankee Rowe Nuclear Power Station
c/o U.S. NRC
Post Office Box 28
Monroe Bridge, Massachusetts 01350

SYSTEMATIC EVALUATION PROGRAM

TOPIC III-7.D

YANKEE NUCLEAR POWER STATION

TOPIC: III-7.D, Assessment of Containment Structural Integrity Test

I. INTRODUCTION

The structural integrity test procedure and test results for the Yankee containment were evaluated against current criteria for such tests. The purpose of the evaluation was to verify that the containment structural integrity test was in compliance with the current requirements of 10 CFR 50 and thus provide assurance that the structure would perform its intended safety function.

II. REVIEW CRITERIA

References:

A. 10 CFR 50, Appendix A
B. NRC Standard Review Plan Section 3.8.2
C. ASME Boiler and Pressure Vessel Code, Section III, Division 1, Article NE-6000, 1980 Edition
D. Yankee SAR for SEP Topic III-7.D

References A, B, and C outline current criteria for conducting and evaluating containment structural integrity tests. Reference D describes the containment tests actually conducted at the Yankee Nuclear Power Station.

III. RELATED TOPICS AND INTERFACES

SEP Topic VI-3, "Containment Pressure and Heat Removal Capability" will provide an assessment of the adequacy of the original design pressure for the containment. The evaluation described herein is based on the original design and test pressure loading of the containment as presented in reference D.

IV. REVIEW GUIDELINES

The test procedure and results were compared with current NRC criteria for such tests in order to determine if any significant deviations existed. An available portion of the Yankee Final Hazards Summary Report (FHSR) was also studied for background information.

V. EVALUATION

The Yankee containment is a column-supported steel sphere (125 feet diameter) which was designed, fabricated, tested, and code stamped in accordance with the rules of the ASME Boiler and Pressure Vessel Code. The test pressure was 40 psig, which is 1.25 times design pressure, according to the information contained in reference D.

Current requirements indicate that the test pressure would be 1.1 times design (35.2 psig), thus the test pressure was quite conservative by today's standards.

According to the Yankee FHSR, the pressure was slowly raised to 20 psig and then raised by 4 psig increments to 40 psig. This pressure was held for a minimum of one hour and then reduced to 31.5 psig and bubble tested. Current criteria requires gradually applying the pressure to not more than one half of test pressure and then increased in steps of approximately one tenth of test pressure until test pressure is reached. Current criteria then requires holding the test pressure for a minimum of 10 minutes and then reducing it to the greater of the design pressure or .75 times test pressure and examining the vessel. The test procedure used at Yankee is conservative when compared to current criteria. The vessel is code stamped and thus it can be inferred that a qualified ASME inspector was present during the testing.

VI. CONCLUSION

Based on the review of the original structural integrity test in comparison to current test requirements, it is considered that the test was satisfactory and thus demonstrated that the structure is capable of performing its intended safety function.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555
December 29, 1981

Docket No. 50-029
LS05-81- 12-098

RECEIVED
DEC 31 1981
US NUCLEAR REGULATORY COMMISSION
DOCUMENT MANAGEMENT BR
TIDC

Mr. James A. Kay
Senior Engineer - Licensing
Yankee Atomic Electric Company
1671 Worcester Road
Framingham, Massachusetts 01701

Dear Mr. Kay:

SUBJECT: SYSTEMATIC EVALUATION PROGRAM TOPIC III-8.A, LOOSE PARTS MONITORING AND CORE BARREL VIBRATION PROGRAM - YANKEE ROWE

Enclosed is a copy of our evaluation of Systematic Evaluation Program Topic III-8.A.

You are requested to examine the facts upon which the staff has based its evaluation and respond either by confirming that the facts are correct, or by identifying errors and supplying the corrected information. We encourage you to supply any other material that might affect the staff's evaluation of this topic or be significant in the integrated assessment of your facility.

The need to actually implement a Loose Parts Monitoring Program will be determined during the integrated safety assessment.

Your response is requested as soon as possible. If no response is received by the time the next phase of the integrated assessment of your facility begins, we will assume that you have no comments or corrections.

Sincerely,

Walter A. Paulson

for Dennis M. Crutchfield, Chief
Operating Reactors Branch No. 5
Division of Licensing

Enclosure:
As stated

cc w/enclosure:
See next page

8201040056
PDR ADOCK

SYSTEMATIC EVALUATION PROGRAM
TOPIC III-8.A

YANKEE ROWE

TOPIC: III-8.A, LOOSE PARTS MONITORING AND CORE BARREL VIBRATION PROGRAM

I. INTRODUCTION

The purpose of this topic is to review the inservice surveillance program to detect loose parts and excessive motion of the main core support structure. The objective is to detect loose parts or excessive vibration before they can cause flow blockage or mechanical damage to the fuel or other safety related components.

II. REVIEW CRITERIA

Standard Review Plan (SRP) Section 4.4, Regulatory Guide (R.G.) 1.133.

III. RELATED SAFETY TOPICS AND INTERFACES

V-1 Compliance with codes and standards (10 CFR 50.55a).

IV. REVIEW GUIDELINES

See Evaluation.

V. Evaluation

1. LOOSE PARTS MONITORING:

R.G. 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," Revision 1, describes features for monitoring loose parts within the reactor coolant pressure boundary (RCPB). These features include sensors strategically located on the exterior surface of the RCPB capable of detecting acoustic disturbances, specifications for system sensitivity, alert levels, data acquistion modes and other system and procedural requirements. Yankee Rowe does not have a loose parts monitoring program that meet the criteria of this guide.

2. CORE BARREL VIBRATION:

This issue is only a problem for Combustion Engineering built plants.

VI. CONCLUSION

1. A Loose Parts Monitoring Program, (i.e., detection system and procedures as specified in Section C.2 and C.3 of R.G. 1.133, Rev. 1) as currently required for new facilities does not exist at the Yankee Rowe facility. The need to actually implement a Loose Parts Monitoring Program will be determined during the integrated assessment.

2. This issue has been deleted for Yankee Rowe.

SEP

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

November 15, 1979

Docket No. 50-29

Mr. Robert H. Groce
Licensing Engineer
Yankee Atomic Electric Company
20 Turnpike Road
Westboro, Massachusetts 01581

Dear Mr. Groce:

RE: SEP TOPIC III-8.C - IRRADIATION DAMAGE, USE OF SENSITIZED STAINLESS STEEL AND FATIGUE RESISTANCE

Enclosed is a copy of our draft evaluation of Systematic Evaluation Program Topic III-8.C. You are requested to examine the facts upon which the staff has based its evaluation and respond either by confirming that the facts are correct, or by identifying any errors. If in error, please supply corrected information for the docket. We encourage you to supply for the docket any other material related to these topics that might affect the staff's evaluation.

Your response within 30 days of the date you receive this letter is requested. If no response is received within that time, we will assume that you have no comments or corrections.

Sincerely,

for Thomas V. Wambach

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Enclosure:
Topic III-8.C

cc w/enclosure:
See next page

~~7911270514~~

cc w/enclosure:
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Westboro, Massachusetts 01581

Greenfield Community College
1 College Drive
Greenfield, Massachusetts 01301

K M C, Inc.
ATTN: Richard Schaffstall
1747 Pennsylvania Avenue, N. W.
Suite 1050
Washington, D. C. 20006

November 15, 1979

SYSTEMATIC EVALUATION PROGRAM
PLANT SYSTEMS/MATERIALS
YANKEE ROWE PLANT

Topic III-8.C - Irradiation Damage, Use of Sensitized Stainless Steel and Fatigue Resistance

The safety objective of this review is to determine whether the integrity of the internal structures of operating reactors has been degraded through the use of sensitized stainless steel.

The effect of neutron irradiation and fatigue resistance on material of the internal structures was eliminated from the safety objective of Topic III-8.C in memorandum to D. G. Eisenhut from D. K. Davis and V. S. Noonan dated December 8, 1978. The memorandum concluded that operating experience indicated that no significant degradation of the materials of the reactor internal structures had occurred as a result of either irradiation damage or fatigue resistance. Furthermore, the Standard Review Plan does not address neutron irradiation nor fatique resistance of the materials of the reactor internal structures.

Information for this assessment was obtained from the Final Safety Analysis Report, Hazards Analysis Reports, Safety Evaluation Reports to the ACRS, Licensee Event Reports and PWR Nuclear Power Experience for the Yankee Rowe plant. Our assessment is based on information in topical reports on the behavior of sensitized stainless steel in PWR nuclear steam supply systems, WAPD-SC-541 "PWR Hazards Summary Report for the Shippingport Reactor," WAPD-PWR-971, "Selection and Application of Materials for the PWR Reactor Plant," and conversations with materials engineers at Combustion Engineering, Westinghouse and General Electric Company.

The regulatory position is addressed in Section 4.5.2, "Reactor Internals Materials" of the Standard Review Plan. The areas currently reviewed in the applicants' SAR are materials specification and the controls imposed on the reactor coolant chemistry, fabrication practices and examination and protection procedures. The materials specification should comply with Section III of the ASME Boiler and Pressure Vessel Code and the fabrication procedures for the components should satisfy the recommendations of Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal" and Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel."

The Yankee Rowe reactor is generally similar to the Shippingport reactor. The control rods and core support structure are described in Sections 1 and 2 of the Final Safety Analysis Report. The functions of the core support structure are to support and orientate the fuel assemblies, maintain orientation and position of the control rods, and to provide passageway for the reactor coolant. The structure consists of an upper and lower support plate, an upper and lower core support barrel, core barrel, radial support and baffle structure. The internals are supported from the reactor flange.

Components of the reactor coolant pressure boundary were designed, fabricated and inspected to the requirements of Section VIII of the ASME Boiler and Pressure Vessel Code, 1956 Edition. Stress and deflection analyses were made by the licensee (YAEC-77 and YAEC-105).

The materials used for constructing the reactor internals were identified in the FSAR as Type 304 stainless steel with minor quantities of special purpose alloys, such as Inconel X, type 410 stainless steel, Armco 17-4 PH, and cobalt-base alloys. The type of materials used was specified in the Westinghouse Equipment Specification, which, in some cases, upgraded or modified the ASME Code requirements. Justification for the use and selection of these materials is presented in WAPD-PWR-971, "Selection and Application of Materials for the PWR Reactor Plant."

Insufficient information was included in the FSAR to ascertain compliance with the recommendations of Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel," and to assure proper control of welding materials and procedures. Therefore, we assume for this assessment that the reactor internal structures contained sensitized stainless steel.

Justification for the use of sensitized stainless steel in PWR quality coolant water was presented in a topical report, WCAP-7477-L, "Sensitized Stainless Steel in Westinghouse PWR Nuclear Steam Supply Systems," written by M. A. Golik, March, 1970. The report reviewed the nature of sensitized Types 304 and 316 stainless steel and the significant factors in the application of sensitized stainless steel in present and future nuclear steam supply systems. In reviewing the PWR operating experience with the Shippingport, BR-3, Saxton, Yankee Rowe, Selni, Connecticut Yankee, San Onofre and Zorita reactors the conclusion was reached that no general problems of intergranular or stress corrosion related to sensitized stainless steel have been encountered in PWR operating reactors. This conclusion was discussed with personnel at Westinghouse and Combustion Engineering who confirmed the conclusion in the report and updated to current PWR operating experience.

The operational experience of the Yankee Rowe Plant was reviewed in the Licensee Event Reports, the PWR Nuclear Power Experience, and the Hazards and Safety Evaluation Reports. The review revealed a number of minor problems with the reactor internals. These problems are described in the Appendix to this report. None of the events described were directly traceable to the use of sensitized stainless steel in the fabrication of the reactor internal structures.

The inservice inspection program for the reactor internal structure for the current inspection interval for the Yankee Rowe plant will be conducted to the requirements of Section XI, ASME Boiler and Pressure Code, 1974 Edition, including Summer 1975 Addendum. The program is in accordance with paragraph (g), Section 50.55a, 10 CFR Part 50.

We conclude from our review of the information submitted by the licensee and the operating information in the Licensee Event Reports together with the PWR Nuclear Power Experience that the integrity of the reactor internal structures for the Yankee Rowe Plant has not been degraded through the use of sensitized stainless steel. Furthermore, we conclude that the integrity of the internal structures will be assured by an inservice inspection program in accordance with the requirements of paragraph (g), Section 50.55a, 10 CFR Part 50.

APPENDIX

Topic III-8.C Review of Service Experience Yankee Rowe Reactor

The abnormal occurrences described for the Yankee Rowe plant related to the operating experience for the reactor internals are summarized in this Appendix.

1) The Hazards Analysis dated August 31, 1962, described two problems observed during inspection at the first refueling outage.

   (a) Wear occurred on certain parts of the control rod assemblies, and

   (b) Deterioriation of the nickel coating on the Ag-In-Cd control rod absorber section.

   The analysis concluded that these problems have not resulted in an unsafe operation of the reactor.

2) A defective thermal shield was evaluated in a Safety Evaluation dated October 28, 1965.

   "The thermal shield for the Yankee Rowe reactor consists of a hollow cylinder that is larger in diameter than the upper neck of the pressure vessel. It was installed in four sections that were fastened together at four half-lap joints by thirteen bolts. The structure is located in the primary coolant flow annulus between the core barrel and the vessel wall, and is supported by eight support lugs. Recent inspection of the thermal shield has revealed that some of the bolts have failed, and that two joints had separated radially at the upper end and three at the lower end. The maximum measured separation was 3/8-inch. Based on inspection of some of the failed bolts, it is believed that the bolts failed as a result of shear forces."

   "Yankee Rowe proposed to reinforce the bolted joints by the installation of four Joint Clamp Assemblies. Each assembly consists of two vertical clamps that grasp the upper and lower edges of adjacent thermal shield sections. The two sections of each clamp are joined at the top and bottom by bars which span the half-lap joints. The clamp assemblies have been designed to preclude vibration, and to withstand any anticipated static or dynamic stresses."

"Since the thermal shield is located in the primary coolant flow annulus, failure of this structure could cause a potential flow reduction. However, even if one or more joints should become completely separated, there is insufficient clearance between the support lugs and the core support barrel to permit any significant displacement. The maximum possible motion for a thermal shield section would be for it to tilt either against the core barrel or the vessel wall, and would present a flow restriction in only one-quarter of the available flow annulus. This condition would be detected by existing flow and temperature instrumentation, and would not result in any reactor damage. Therefore, we believe that installation of the Joint Clamp Assemblies will improve the integrity of the thermal shield joints and that the safety of reactor operations will not be adversely affected."

It was concluded that the proposed change does not present significant hazards considerations not described or implicit in the hazards summary report, and that there is reasonable assurance that the health and safety of the public will not be endangered.

3) The Hazards Analysis dated March 26, 1962, evaluated the request to install in the reactor up to twelve special tube assemblies containing encapsulated specimens of reactor vessel material. The assemblies were to be placed in eight holes provided in the upper flange of the core baffle and in four guide channels attached to the outer surface of the thermal shield.

One of the surveillance capsules became loose in the reactor and lodged between the core support and vessel cladding. The cladding was perforated as a result of the event. The event was attributed to fatigue of the fastener. This capsule as well as the remaining attached capsules were removed from the reactor during inspection at the first refueling operation.

4) The Safety Evaluation dated October 25, 1965, evaluated a proposed design change authorizing the installation of four Secondary Core Supports to limit movement of the reactor core in the unlikely event that the primary core support structure should fail.

"The Yankee Rowe core consists of 76 fuel elements, which weigh about 26 tons, and is contained within the core barrel. The core barrel is suspended by a flange from the top of the pressure vessel, and primary support for this structure is provided by a full penetration weld between the core barrel and the top support flange. This weld has adequate strength to support this structure, but is reinforced by twelve one-inch thick gusset plates spaced radially around the core barrel. The support welds were recently inspected by dye penetrant and ultrasonic testing techniques and found to be in sound condition."

"Yankee Rowe believes that failure of the primary core support is extremely unlikely, but has proposed to install four Secondary Core Supports to limit the downward motion of the core if a failure should occur. The secondary support consists of four stainless steel straps (3/4" x 7 3/4" in cross section) that are firmly attached to the top and bottom of the thermal shield. These straps provide four points of support under the lower core support plate and would limit the downward motion to 5/8 inch. The secondary supports have been designed to preclude vibration, and to absorb the impact of the core structure. The weight of the core structure would ultimately be supported by the thermal shield support lugs which are an integral part of the pressure vessel wall. We believe that the Secondary Core Supports have been adequately designed, and that their installation would provide a desirable additional safety factor for the Yankee Rowe reactor."

"Since the control rods are inserted from the top of the core, downward movement of the core would result in a reactivity addition. However, the Yankee Rowe reactor is operated as a chemical shim plant with only one group of control rods normally inserted. With this mode of operation, the reactivity addition would be negligible if the core should fall. The maximum potential reactivity addition (control rods at maximum worth) that could occur for this postulated accident, with the Secondary Core Supports in place, would be during shutdown or low power operation. This reactivity addition would be less than 0.005. Yankee Rowe has analyzed this reactivity addition, and we agree that the resulting transient would not result in any damage to the reactor."

It was concluded that the Proposed Change does not present significant hazards considerations not described or implicit in the hazards summary report, and that there is reasonable assurance that the health and safety of the public will not be endangered.

5) The Safety Evaluation dated September 29, 1966, evaluated the proposed change in design to authorize the installation of twenty-four inconel-clad silver-indium-cadmium control rods with zircaloy follower sections that are welded to the control rods. This change was requested to provide a control rod design that would preclude inadvertent disassembly of the follower section under certain reactor shutdown conditions.

"The Yankee Rowe control system contains 24 control rods that are cruciform in shape and contain zircaloy follower sections. The current complement of control rods consists of 20 hafnium rods, two Inconel-clad Ag-In-Cd rods, and two stainless steel clad Ag-In-Cd rods. Yankee Rowe has proposed to replace these rods with 24 new Inconel-clad Ag-In-Cd control rods. The physical dimensions and reactivity worth of these control rods are essentially identical to those currently in service. Two Inconel-clad Ag-In-Cd control rods have been in service during Cores III, IV, and V and inspections during refueling shutdowns have indicated no problems with respect to mechanical wear and corrosion."

"The replacement control rods will have follower sections that are welded to the absorber sections. The current design contains a snap joint that connects the absorber to the follower section by rotation about the vertical axis. Yankee Rowe has reported that inspection of the two control rods removed during the 1965 refueling indicated that some wear of the snap joint has occurred. This wear could allow the follower section to rotate and fall from the absorber during refueling operations when the upper core support plate is removed. However, such separation could not occur during reactor operation since relative rotation sufficient to permit separation of the two sections is precluded by the internal core structure. The use of follower sections that are welded to the absorber sections will prevent inadvertent disassembly of the control rods even during the refueling operation. In this respect we believe that the safety of reactor operations will be improved by the use of the proposed control rods."

The staff concluded that the proposed change does not present significant hazards considerations not described or implicit in the haz rds summary report, and that there is reasonable assurance that the health and safety of the public will not be endangered.

6) A Regulatory Operations inquiry report was filed on November 14, 1972, reporting a loose bolt lying on top of the lower core support plate. The event was described as follows:

"While performing the control rod interchange during the present scheduled outage, it was observed that a bolt was lying on top of the lower core support plate. Observations made with a T.V. camera through 40 feet of water indicated that there were two additional bolts lying loose. An examination of the bolts indicate that they are control rod shroud tie down bolts."

"Preliminary determination identifies two shroud tubes displaced. There are a total of 32 shroud tubes, 8 for shim rods and 24 for moveable control rods. All T.V. monitoring has been recorded on tape."

The licensee summarized the scope and results of the inspection program conducted on the reactor internals following the above described event which occurred after twelve years of reactor operation. The results of the inspection program were as follows:

"Complete visual inspections by the use of binoculars, underwater TV cameras, and boroscopes included the lower internals, significant areas of the upper internals, and areas inside the reactor vessel. Measurements performed by the use of specially designed tools and gaging devices included torque checks on every other lower core support plate to core barrel connecting bolt. These measurements confirmed that these important load carrying bolted connections remained tight. We conclude that this inspection program has identified the extent of the bolting failures in the lower shroud tube assembly and it has confirmed that the other internals components have not experienced significant structural degradation. In addition, during this inspection program an impression was made of the existing cladding defect inside the reactor vessel. A check of measurements of this impression does not indicate measureable changes from measurements made in 1965 and 1968 inspections. All dropped out bolts and locking devices were retrieved from the core support plate and the reactor vessel bottom, except for one bolt. Efforts will be continued to recover this missing bolt. A large foreign object, identified as part of an original low flux specimen holder was also retrieved from the underside of the lower core support plate."

The Hazards Analysis dated March 30, 1973, evaluated the proposed shroud tube design change of the lower internals and the supplemental information presented in support of the change. The event was attributed in part to flow induced vibration acting on the shroud tubes that had loose connecting flange bolts. The Hazards Analysis contained the following:

"We have reviewed your Proposed Change No. 106 and the supplemental items of information, including the results of your inspection program performed on the original shroud tube assembly and other internals; your evaluation of the bolting failures in the original shroud tube assembly; the summary of your mechanical, thermal, and hydraulic evaluation; the guides, codes, standards, and the quality assurance and audit program used in the design and fabrication of the replacement shroud tube assembly; and the preoperational and post-operational inspection and surveillance programs. We have concluded that: (a) the bolting failures were limited to a local area in the original shroud tube assembly and the other internals components have not experienced significant degradation during the period of 12 years of reactor operation, (b) significant design improvements have been incorporated into the new shroud tube assembly, (c) necessary modifications to facilitate installation of the new assembly will not significantly change important performance characteristics, and (d) the specified surveillance program for monitoring the integrity of the new shroud tube assembly is acceptable."

"We have concluded that there are no hazards considerations not described or implicit in the Safety Analysis Report. There is reasonable assurance that the health and safety of the public will not be endangered. Accordingly, pursuant to 50.59 of 10 CFR Part 50, Change No. 106 is hereby authorized as proposed."

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555
September 22, 1981

Docket No. 50-29
LS05-81-09-058

Mr. James A. Kay
Senior Engineer-Licensing
Yankee Atomic Electric Company
1671 Worcester Road
Framingham, Massachusetts 01701

Dear Mr. Kay:

SUBJECT: SEP TOPIC III-10.A, THERMAL-OVERLOAD PROTECTION FOR MOTORS OF MOTOR-OPERATED VALVES, SAFETY EVALUATION REPORT FOR YANKEE ROWE

The enclosed staff safety evaluation supplements our contractor's evaluation that has been made available to you previously. This evaluation supports the findings of the enclosed staff safety evaluation on Topic III-10.A that proposes modifications to the valve protective trips.

The need to actually implement these changes will be determined during the integrated plant safety assessment. This topic assessment may be revised in the future if your facility design is changed or if NRC criteria relating to this topic are modified before the integrated assessment is completed.

Sincerely, -

Dennis M. Crutchfield

Dennis M. Crutchfield, Chief
Operating Reactors Branch No. 5
Division of Licensing

Enclosure:
As stated

cc w/enclosure:
See next page

8109240453

Mr. James A. Kay

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SYSTEMATIC EVALUATION PROGRAM
TOPIC III-10.A

YANKEE ROWE

TOPIC: III-10.A, THERMAL-OVERLOAD PROTECTION FOR MOTORS OF MOTOR-OPERATED VALVES

I. INTRODUCTION

The primary objective of thermal overload relays is to protect motor windings of motor-operated valves (MOV) against excessive heating. This feature of thermal overload relays could, however, interfere with the successful functioning of a safety related system. In nuclear plant safety system application, the ultimate criterion should be to drive the valve to its proper position to mitigate the consequences of an accidents, rather than to be concerned with degradation or failure of the motor due to excessive heating.

II. REVIEW CRITERIA

The primary review criteria are:

1. IEEE Std. 279-1971, and
2. Regulatory Guide 1.106.

As a result of numerous operating plant events resulting from torque switch problems the following supplemental criterion was used:

"(3) In MOV designs that use a torque switch to limit the opening or closing of the valve, the automatic opening or closing signal should be used in conjunction with a corresponding limit switch."

III. RELATED SAFETY TOPICS AND INTERFACES

There are no safety areas related to the scope of this review that are addressed by other SEP Topics nor are any other topics dependent on the results of this review.

IV. REVIEW GUIDELINES

The review should assure that: (1) thermal overload protection, if provided, for MOV's should have the trip setpoint at a value high enough to prevent spurious trips due to design inaccuracies, trip setpoint drift, or variation in the ambient temperature at the installed location; (2) the circuits that bypass the thermal overload protection under accident conditions are designed to IEEE Std. 279-1971 criteria, as appropriate for the rest of the safety related system; and (3) in MOV designs that use a torque switch instead of a limit switch to limit the opening or closing of the valve, the automatic opening or closing signal should be used in conjunction with a corresponding limit switch and thermal overload should remain as backup protection over the first 10% of valve travel.

V. EVALUATION

The design provisions for motor-operated valve protection are described in EG&G Report 1662F, "Thermal-Overload Protection for Motors of Motor-Operated Valves." Thermal-overload protection for motor-operated valves at Yankee Rowe does not comply with current licensing criteria. Thermal-overload devices are not bypassed, no information is available to support adequacy of trip setpoints, and torque switches rather than limit switches are used to terminate valve travel.

VI. CONCLUSIONS

The Yankee Rowe design does not satisfy the current licensing criteria for safety related valve functions. Because poor valve reliability may lead to the failure of more than one valve during emergency conditions, and multiple valve failures have not been analyzed for their affect upon system performance and plant safety, the staff recommends that action should be taken to improve valve reliability (i.e., bypassing torque switches with a limit switch during automatic actuation of the valve and bypassing thermal overloads with an ECCS signal).

SEP

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

November 5, 1979

Docket No. 50-29

Mr. Robert H. Groce
Licensing Engineer
Yankee Atomic Electric Company
20 Turnpike Road
Westboro, Massachusetts 01581

Dear Mr. Groce:

RE: COMPLETION OF SEP TOPIC IV-1.A - YANKEE ROWE

A draft of the subject topic assessment was sent to you by letter dated April 18, 1979. We have been informed by representatives of your staff by telephone on October 23, 1979, that the facts upon which the NRC staff based its evaluation are correct. Accordingly, our review of SEP Topic IV-1.A is complete and this evaluation will be a basic input to the integrated safety assessment for your facility.

The subject assessment compares your facility design with the criteria currently used by the staff in licensing new facilities. This assessment may need to be re-examined if you modify your facility of if the criteria are changed before we complete our integrated assessment.

Sincerely,

Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

cc: See next page

~~7911280165~~

cc w/enclosure:
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K M C, Inc.
ATTN: Richard Schaffstall
1747 Pennsylvania Avenue, N. W.
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Washington, D. C. 20006

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

April 18, 1979

Docket No. 50-29

Mr. Robert H. Groce
Licensing Engineer
Yankee Atomic Electric Company
20 Turnpike Road
Westboro, Massachusetts 01581

Dear Mr. Groce:

RE: TOPIC IV-1.A OPERATION WITH LESS THAN ALL LOOPS IN SERVICE

Enclosed is a copy of our revised safety assessment of Topic IV-1.A, Operation With Less Than All Loops In Service. This revision includes consideration of the comments received on the assessment issued by our letter dated February 1, 1979. Your letter dated March 7, 1979, provided comments on this assessment.

This revision completes our assessment of Topic IV-1.A which will be used as input to the integrated review of the Yankee Rowe Station.

If there are any errors in the facts of this revised assessment, please supply corrected information within 30 days of the date you receive this letter. If no response is received within that time, we will assume that you have no further comments or corrections.

Sincerely,

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Enclosure:
Revised Assessment for
Topic IV-1.A

cc w/enclosure:
See next page

7905010363

cc w/enclosure:
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Yankee Atomic Electric Company
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Greenfield Community College
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1747 Pennsylvania Avenue, N. W.
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Washington, D. C. 20006

SYSTEMATIC EVALUATION PROGRAM

Topic IV-1.A Operation with Less Than All Loops In Service

Plant: Yankee Rowe

Discussion

The majority of the presently operating BWRs and PWRs are designed to permit operation with less than full reactor coolant flow. That is, if a PWR reactor coolant pump or a BWR recirculation pump becomes inoperative, the flow provided by the remaining loop or loops is sufficient for steady state operation at some definable power level, usually less than full power.

Plants authorized for long term operation with one reactor coolant loop out of service have submitted, and the staff has approved, the necessary ECCS, steady state and transient analysis. The remaining PWR and BWR licensees have Technical Specifications which require a reactor shutdown within 24 hours if one of the operating loops becomes inoperable and cannot be returned to operation within the time period.

Evaluation

The docketed material for Yankee Rowe has been reviewed with respect to operation with less than all loops in service. The search of our docketed material revealed that on January 29, 1976, the licensee submitted an application for operation of the Yankee Rowe facility in the N-1 loop configuration. The application transmitted Appendix C, an analysis, entitled "Idle Loop Operation" dated January 28, 1976. The licensee requested that Appendix C be reviewed in conjunction with the information and ongoing review of proposed change number 125 entitled "Yankee Nuclear Power Station Core XII Performance Analysis", dated

July 14, 1975. However, to date, documented findings of the staff's review can not be found.

The licensees' application for authorization to operate in the N-1 loop mode was the result of a loss of flow incident that occurred on January 13, 1976. The cause of the incident was determined to be an electrical fault in the motor of one of the four main coolant pumps. Repair or replacement of the pump was expected to involve significant effort and time requiring a reactor shutdown until the problem could be resolved. Therefore, the licensee submitted the N-1 loop analysis in an effort to obtain approval for 3 loop operation for an interim period with a further commitment to perform a more detailed analysis if resumption of 4 loop operation were not possible because of their inability to restore the inoperable pump. Yankee Rowe obtained a replacement pump from the Indian Point Unit 1 plant and resumed full flow operation within a two week period. At this time, the NRC was verbally requested by the licensee to discontinue the review of their N-1 loop submittal. In a telephone conversation with the licensee (telecon P. A. DiBenedetto, NRC, R. Groce, YAEC, January 17, 1979), the fact that the application was verbally withdrawn was confirmed. Additionally, it was recognized, because of the hardware and logic modifications performed on the ECCS since tendering the N-1 loop analysis, that the submittal was no longer valid and that authorization for N-1 loop operation would have to be supported by a new analysis.

The Yankee Rowe facility is presently restricted (Technical Specifications 3.2.1 and 3.0.3) to operate with no less than all loops in service. Under Section 3/4.2, Power Distribution Limits of the Technical Specifications, if a Linear Heat Generation Rate (LHGR) limit is exceeded, several action statements permit the licensee to operate the facility at reduced power thereby reducing the LHGR to within the appropriate limits; however, the limiting condition for operation (LCO, Technical Specification 3.2.1) is footnoted. The footnote states that operation in the 3 loop mode is not permitted until appropriate LOCA analyses for this mode have been approved by the NRC. This prohibits the licensee from satisfying the action statements and reducing power to maintain the LHGR conducive to that of 3 loop operation. When an action statement cannot be satisfied, Technical Specification 3.0.3 requires that the licensee place the facility in at least hot standby within one hour and within 30 hours be in cold shutdown unless corrective measures are completed.

The restriction of Technical Specification 3.2.1 renders Technical Specifications 3.4.1.1, 3.4.1.2 and 3.4.1.3 (issued with Amendment No. 49 May 30, 1978) ineffective. These Technical Specifications provide the LCO's for reduced loop operation. Although not authorized to operate in this mode, the Technical Specifications were issued to facilitate the ease of future amendments by minimizing the number of changes to the Technical Specifications. Therefore, if an analysis for N-1 loop operation is performed by the licensee and approved by the NRC, only the footnote of Technical Specification 3.2.1 has to be removed.

Based on our review, we find that the restrictions presented in Technical Specification 3.2.1 and the requirements of 3.0.3, which prohibited extended operation with N-1 loops in lieu of approved analytical support for N-1 loop operation, are consistent with our criteria. Therefore, we conclude that Topic IV-1.A is completed and acceptably resolved for Yankee Rowe.

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

April 18, 1979

Docket No. 50-29

Mr. Robert H. Groce
Licensing Engineer
Yankee Atomic Electric Company
20 Turnpike Road
Westboro, Massachusetts 01581

Dear Mr. Groce:

RE: TOPIC IV-1.A OPERATION WITH LESS THAN ALL LOOPS IN SERVICE

Enclosed is a copy of our revised safety assessment of Topic IV-1.A, Operation With Less Than All Loops In Service. This revision includes consideration of the comments received on the assessment issued by our letter dated February 1, 1979. Your letter dated March 7, 1979, provided comments on this assessment.

This revision completes our assessment of Topic IV-1.A which will be used as input to the integrated review of the Yankee Rowe Station.

If there are any errors in the facts of this revised assessment, please supply corrected information within 30 days of the date you receive this letter. If no response is received within that time, we will assume that you have no further comments or corrections.

Sincerely,

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Enclosure:
Revised Assessment for
Topic IV-1.A

cc w/enclosure:
See next page

7905020363

cc w/enclosure:
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K M C, Inc.
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1747 Pennsylvania Avenue, N. W.
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SYSTEMATIC EVALUATION PROGRAM

Topic IV-1.A Operation with Less Than All Loops In Service

Plant: Yankee Rowe

Discussion

The majority of the presently operating BWRs and PWRs are designed to permit operation with less than full reactor coolant flow. That is, if a PWR reactor coolant pump or a BWR recirculation pump becomes inoperative, the flow provided by the remaining loop or loops is sufficient for steady state operation at some definable power level, usually less than full power.

Plants authorized for long term operation with one reactor coolant loop out of service have submitted, and the staff has approved, the necessary ECCS, steady state and transient analysis. The remaining PWR and BWR licensees have Technical Specifications which require a reactor shutdown within 24 hours if one of the operating loops becomes inoperable and cannot be returned to operation within the time period.

Evaluation

The docketed material for Yankee Rowe has been reviewed with respect to operation with less than all loops in service. The search of our docketed material revealed that on January 29, 1976, the licensee submitted an application for operation of the Yankee Rowe facility in the N-1 loop configuration. The application transmitted Appendix C, an analysis, entitled "Idle Loop Operation" dated January 28, 1976. The licensee requested that Appendix C be reviewed in conjunction with the information and ongoing review of proposed change number 125 entitled "Yankee Nuclear Power Station Core XII Performance Analysis", dated

April 18, 1979

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555
December 23, 1981

Docket No. 50-029
LS05-81-12-078

Mr. James A. Kay
Senior Engineer - Licensing
Yankee Atomic Electric Company
1671 Worcester Road
Framingham, Massachusetts 01701

Dear Mr. Kay:

SUBJECT: SEP TOPIC IV-2, REACTIVITY CONTROL SYSTEMS
DRAFT SAFETY EVALUATION - YANKEE ROWE NUCLEAR
POWER PLANT

Enclosed is a copy of our draft safety evaluation of SEP Topic IV-2, Reactivity Control Systems for Yankee Rowe. This assessment compares your facility, as described in Docket No. 50-029, with the criteria currently used by the regulatory staff for licensing new facilities. Please inform us if your as-built facility differs from the licensing basis assumed in our assessment.

Your response within 30 days of the date you receive this letter is requested. If no response is received within that time, we will assume that you have no comments or corrections. This evaluation should be a basic input to the evaluation of Topic XV-8 and the integrated safety assessment for your facility unless you identify changes needed to reflect the as-built conditions at your facility. This assessment may be revised in the future if your facility design is changed or, if NRC criteria relating to this subject are modified before the integrated assessment is completed.

Sincerely,

Thomas V. Wambach
for Dennis M. Crutchfield, Chief
Operating Reactors Branch No. 5
Division of Licensing

Enclosure:
As stated

cc w/enclosure:
See next page

~~8112280330~~

Mr. James A. Kay

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SAFETY EVALUATION REPORT

SEP TOPIC IV-2, REACTIVITY CONTROL SYSTEMS

INCLUDING FUNCTIONAL DESIGN AND

PROTECTION AGAINST SINGLE FAILURES

YANKEE ROWE NUCLEAR POWER PLANT

DOCKET NO. 50-029

I. INTRODUCTION

The purpose of this evaluation is to insure that the design basis for the Yankee Rowe reactivity control systems is consistent with analyses performed to verify that the protection system meets General Design Criterion 25. General Design Criterion 25 requires that the reactor protection system 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 of control rods. Reactivity control systems need not be single failure proof. However, the protection system must be capable of assuring that acceptable fuel design limits are not exceeded in the event of a single failure in the reactivity control systems. The review criterion, covered in this evaluation, is addressed in Section II. Review areas that are not covered, but are related and essential to the completion of this topic are covered by other SEP topics addressed in Section III. The scope of the SEP topics is defined in the "Report on the Systematic Evaluation of Operating Facilities" dated November 25, 1977.

This report is limited to the identification of inadvertent control rod withdrawals and malpositioning of controls rods which may occur as a result of single failures in the control rod drive system.

II. REVIEW CRITERION

The review criterion for this topic is based upon Section 7.7, Part II of the NRC Standard Review Plan. In the specific case of the reactivity control systems a single failure shall not cause plant conditions more severe than those for which the reactor protection system is designed.

III. RELATED SAFETY TOPICS

The following listed review areas are not covered in this report, but are related and essential to the completion of this topic. These review areas are covered by other SEP topics as indicated below.

1. Analyses of the consequences of control rod withdrawals and

the malpositioning of control rods which may occur as a result of single failures in the electrical circuits of the reactivity control systems are covered by SEP Topic XV-8, "Control Rod Misoperation (System Malfunction or Operator Error").

2. Analyses of reactivity insertions occurring as a result of inadvertent boron dilutions are covered in SEP Topic XV-10, "Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant."

IV. REVIEW GUIDELINES

The purpose of this evaluation is to identify inadvertent control rod withdrawals and malpositioning of control rods which may occur as a result of single failure in the control rod drive system for the Yankee Rowe Nuclear Power Plant.

V. EVALUATION

Information was provided in Yankee Atomic Electric Company letter dated September 17, 1981, describing single failures within the control rod drive system which can cause control rod withdrawals and malpositioning of control rods at the Yankee Rowe Nuclear Power Plant. Also included was a description of design features which limit reactivity insertion rates and rod malpositionings resulting from single failures. Based upon an audit review of the information provided by the licensee we conclude that the following may occur as a result of single failures:

1. A single rod may drop into the core.
2. A single rod may not move when movement is commanded.
3. A single rod may be inadvertently moved or malpositioned.
4. An entire group may drop into the core.
5. An entire group may not move when movement is commanded. This includes both automatic in (based on T_{ave}) and manual commands.
6. An entire group may be inadvertently moved or malpositioned. This includes the simultaneous movement of two groups when only one group is commanded and the inward movement of a group when the outward movement is selected.
7. All rods may drop into the core.
8. All rods may not move in when movement commanded. This includes any number of groups failing to move in when all rods are commanded to move.
9. All rods may be inadvertently moved in or malpositioned.
10. An entire group of rods may be withdrawn beyond the 485 MWT limit.

The conclusion is based upon the design of the control rod drive system. The analysis performed by the licensee found many of the results and the remaining items were noted during the audit.

VI. CONCLUSION

The licensee should revise the evaluation of SEP Topic XV-8 to include the ten items above or should show why these types of failures cannot occur at Yankee Rowe.

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

July 13, 1982

Docket No. 50-29
LS05-82-07-019

Mr. James A. Kay
Senior Engineer - Licensing
Yankee Atomic Electric Company
1671 Worcester Road
Framingham, Massachusetts 01701

Dear Mr. Kay:

SUBJECT: SEP TOPIC V-5, REACTOR COOLANT PRESSURE BOUNDARY LEAKAGE DETECTION - YANKEE NUCLEAR POWER STATION

Enclosed is a copy of our final evaluation of SEP Topic V-5 for the Yankee Nuclear Power Station. This assessment was based on a comparison of the facility, as described in Docket No. 50-29, with the criteria currently used by the regulatory staff for licensing new facilities. This evaluation factors in the information contained in the Yankee Technical Specifications and information in SEP Topic V-10.A, and your June 15, 1982 comments on our draft evaluation. We now consider this topic to be complete.

This evaluation will be a basic input to the integrated safety assessment for your facility unless you identify changes needed to reflect the as-built conditions at your facility. This assessment may be revised in the future if your facility design is changed or if NRC criteria relating to this subject are modified before the integrated assessment is completed.

Sincerely,

Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing

Enclosure:
As stated

cc w/enclosure:
See next page

8207190332

Mr. James A. Kay

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ENCLOSURE

SYSTEMATIC EVALUATION PROGRAM
TOPIC V-5

YANKEE NUCLEAR POWER STATION

TOPIC: V-5, Reactor Coolant Pressure Boundary (RCPB) Leakage Detection

I. INTRODUCTION

The safety objective of Topic V-5 is to determine the reliability and sensitivity of the leak detection systems which monitor the reactor coolant pressure boundary to identify primary system leaks at an early stage before failures occur.

II. REVIEW CRITERIA

The acceptance criteria for the detection of leakage from the reactor coolant pressure boundary is stated in the General Design Criteria of Appendix A, 10 CFR Part 50. Criterion 30, "Quality of Reactor Coolant Pressure Boundary," requires that means shall be provided for detecting and, to the extent practical, identifying the location of the sources of leakage in the reactor coolant pressure boundary.

III. REVIEW GUIDELINES

The acceptance criteria are described in the Nuclear Regulatory Commission Standard Review Plan Section 5.2.5, "Reactor Coolant Pressure Boundary Leakage Detection." The areas of the Safety Analysis Report and Technical Specifications are reviewed to establish that information submitted by the licensee is in compliance with Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems."

IV. EVALUATION

Safety Topic V-5 was evaluated in this review for compliance of the information submitted by the licensee with Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems." The information in the Final Hazards Summary Report, the Final Safety Analysis Report, the Technical Specifications and SEP Topic V-10.A was reviewed.

Regulatory Guide 1.45 recommends that at least three separate detection systems be installed in a nuclear power plant to detect unidentified leakage from the reactor coolant pressure boundary to the primary containment of one gallon per minute within one hour. Leakage from identified sources must be isolated so that the flow rates may be monitored separately from unidentified leakage. The detection system should be capable of performing its function following certain seismic events and capable of being checked in the control room. Of the three separate leak detection methods recommended, two of the methods should be (1) sump level and flow monitoring, and (2) airborne particulate radioactivity monitoring. The third method may be either monitoring of condensate flow rate from air coolers or monitoring of airborne gaseous radioactivity. Other detection methods, such as humidity, temperature and pressure, should be considered to be alarms of indirect indication of leakage to the containment. In addition, provisions should be made to monitor systems interfacing with the reactor coolant pressure boundary for signs of intersystem leakage through methods such as radioactivity and water level or flow monitors. Plant incorporated systems and their corresponding features are tabulated and attached to this evaluation. Detailed guidance for the leakage detection system is contained in Regulatory Guide 1.45.

Based upon our review of the referenced documents and the summaries presented in the tables, we have determined that:

(1) the leakage detection systems employed to detect leakage from the reactor coolant pressure boundary to the vapor container do not meet all of the recommendations of Regulatory Guide 1.45 (refer to table 1). Specifically, only two of the three recommended detection systems are present. There is no system to directly monitor airborne gaseous radioactivity or the flow rate of condensate from the air coolers. However, the low pressure surge tank level is monitored hourly for the amount of makeup to the reactor coolant pressure boundary. None of the systems employed is seismically qualified. It is our judgement that a one gallon per minute primary system leak can be detected, but in a time of greater than one hour from the time of initiation of the leak.

(2) adequate provisions have been made to monitor reactor coolant leakage to the secondary system and to the component cooling system.

(3) a primary system water balance is performed once every 24 hours, with the data collected once per shift. The water balance is performed in accordance with Operating Procedure No. OP-4220. This technique is capable of detecting small unidentified leaks.

(4) the vapor container is also equipped with atmosphere pressure, temperature and humidity monitoring systems. Although the leak detection sensitivity of these systems is poor, they do serve as a means of indirect leakage detection and can serve as an alert to the operators that a potential problem exists.

V. CONCLUSIONS

Our review indicated that the systems employed at Yankee to measure reactor coolant pressure boundary leakage do not meet all of the recommendations of Regulatory Guide 1.45. Specifically, the systems are not seismically qualified and, while the systems can detect a one gallon per minute leak, the time required to detect most leaks is greater than one hour. All of the other recommendations of the Guide have been met or equivalent alternatives have been provided.

The necessity for any leakage detection system modification will be considered during the integrated safety assessment.

BOUNDARY LEAKAGE DETECTI STEMS

uide 1.45 Requirements

: YANKEE NUCLEAR POWER STATION

Req'd Achieve itivity	Earthquake For Which Function Is Assured	Control Room Indication For Alarms & Indicators	Documentation Reference	Testable During Normal Operation
3 hrs	-	YES	FSAR 5.2.4	YES -
-	-	-	-	-
2 hrs	-	YES	FSAR 5.2.4 and 11.4	YES -
-	-	-	FHSR 216:3	-
-	-	-	FSAR 9.4	-
-	-	-	FHSR 216:3	-
-	-	YES	FSAR 5.2.4	-
1 hr	-	YES	YAEC ltr 6/15/82 Topic V-5	YES

SURE BOUNDARY LEAKAG. ... TION SYSTEMS

ry Guide 1.45 Requirements

it: YANKEE NUCLEAR POWER STATION

Time Req'd to Achieve Sensitivity	Earthquake For Which Function Is Assured	Control Room Indication For Alarms & Indicators	Documentation Reference	Testable During Normal Operation
-	-	YES	FHSR 215:2a	-
-	-	YES	FHSR 215.34	-
-	-	YES	NRC Ltr. to YAEC 3/21/79 Topic V-10.A	-
-	-	YES	YAEC ltr 6/15/82 Topic V-5	

Table 3:

RCS Inventory Balance

1) Leak Rate Sensitivity: 1 gpm

2) Time Required to Achieve Sensitivity: 24 hours

3) Documentation Reference: Yankee Technical Specification 3/4.4.5

4) Description of Inventory Balance Procedure: See Yankee Operating Procedure No. OP-4220

DCrutchfield
5-15

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

March 5, 1980

Docket No. 50-29

Mr. James A. Kay
Senior Engineer-Licensing
Yankee Atomic Electric Company
25 Research Drive
Westborough, Massachusetts 01581

Dear Mr. Kay:

RE: COMPLETION OF SEP TOPIC V-6, REACTOR VESSEL INTEGRITY
YANKEE ROWE ATOMIC POWER STATION

Enclosed herewith is a report entitled, "Evaluation of the Integrity of SEP Reactor Vessels," (NUREG-0569) for your information. This report documents our review of the design specifications and quality assurance programs used in reactor vessel construction. It also provides the status of material surveillance programs, pressure-temperature operating limits, inservice inspection programs and other vessel related issues for plants in the Systematic Evaluation Program (SEP).

This evaluation completed our review of SEP Topic V-6 and will be used as input to the integrated review of Yankee Rowe Atomic Power Station.

Sincerely,

Thomas V. Wambach
for Dennis L. Ziemann, Chief
Operating Reactors Branch #2
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Enclosure:
NUREG-0569

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NUREG-0569

Evaluation of the Integrity of SEP Reactor Vessels

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ABSTRACT

This report includes a documented review of the integrity of the 11 reactor pressure vessels covered in the Systematic Evaluation Program. This review deals primarily with the design specifications and quality assurance programs used in the vessel construction and the status of material surveillance programs, pressure-temperature operating limits, and inservice inspection programs of the applicable plants. Several generic items such as PWR overpressurization protection and BWR nozzle and safe-end cracking also are evaluated. The 11 vessels evaluated include Dresden Units 1 and 2, Big Rock Point, Haddam Neck, Yankee Rowe, Oyster Creek, San Onofre 1, LaCrosse, Ginna, Millstone 1, and Palisades.

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EVALUATION OF THE INTEGRITY OF SEP REACTOR VESSELS

1.0 INTRODUCTION

In March 1977, the Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, established a Systematic Evaluation Program (SEP) Review Group. During Phase I of the SEP, this group developed a list of topics to be used as the basis for performing systematic evaluations of operating reactors (Ref. 1).

The group recommended that during Phase II of the SEP those plants for which operating licenses had been issued prior to 1969 should be reviewed. Indian Point Unit 1 and Humboldt Bay Unit 3 were omitted from this list because of the uncertainty associated with their future operation. There are six plants in this category. Subsequently, five facilities operating under Provisional Operating Licenses (POLs) were added to the program since the SEP review would efficiently accomplish conversion from POL to Full-Term Operating License. The plants under SEP review are:

Plant Name	Docket No./License No.
*Dresden Unit 1	50-10/DPR-2
*Yankee Rowe	50-29/DPR-3
*Big Rock Point	50-155/DPR-6
*San Onofre 1	50-206/DPR-13
*Haddam Neck	50-213/DPR-61
*Oyster Creek	50-219/DPR-16
Dresden Unit 2	50-237/DPR-19
Ginna	50-244/DPR-18
Millstone Unit 1	50-245/DPR-21
Palisades	50-255/DPR-20
*La Crosse	50-409/DPR-45

As a part of the SEP review, the NRC staff investigated the topic of reactor vessel integrity. The staff assessment included compliance with appropriate sections of 10 CFR Part 50 including fracture toughness, neutron irradiation, surveillance programs, pressure-temperature operating limits, inservice inspection programs, transient analysis, and vessel nozzle and safe-end flaws.

NUREG-0081, issued by the NRC in June 1976, documented a review of the integrity of reactor vessels that were not designed to ASME Code, Section III (Ref. 2). Seven reactor vessels now in SEP were evaluated in this earlier review and are indicated by an asterisk in the preceding list. This study

*Denotes vessels designed to ASME Boiler and Pressure Vessel Code, Sections I and/or VIII, and reviewed in NUREG-0081. These Codes are hereinafter referred to as the ASME Code, Section I, Section III, Section VIII, and Section XI.

was used as part of the basis for the present SEP review. The information contained in that report was expanded to include all the SEP vessels and several items not included in NUREG-0081 and was updated to January 1, 1979.

The NRC document NUREG-0081 was published in response to concerns expressed by the Advisory Committee on Reactor Safeguards on the integrity of reactor vessels not constructed of SA-533 and SA-508 steels, designed to ASME Boiler and Pressure Vessel Code, Sections I and/or VIII, and where only limited conformance to ASME Code Section XI is practical (Ref. 3). From this review it was concluded that all the pre-Section III reactor vessels, except Indian Point 1, have approximately the same degree of reliability as the vessels designed to meet Section III discussed in WASH-1318 (Ref. 4).

2.0 DESIGN

The quality level initially built into a nuclear reactor pressure vessel is a very important factor affecting the integrity of the vessel throughout its service life. This quality level is directly related to the materials and construction practices used in the manufacture of the vessel. The current requirements governing these practices are given in the ASME Code, Section III. The minimum initial quality of nuclear pressure vessels built to this section is higher than that of vessels designed solely to Sections I or VIII. The main reasons for this are summarized below.

1. For nuclear service (Section III design), the properties of materials are more strictly defined and much more attention is paid to make certain that the proper environmental conditions are considered.

2. Design practices are much more highly refined and more emphasis is placed on the careful analysis of design details in preference to reliance only on the nominal membrane stresses and gross factor of safety. Comparison between the requirements of Section III and those of Section I and Section VIII shows that the nonnuclear sections gain all of their conservatism by increased wall thickness requirements, evaluated solely for membrane loads and with a safety factor of four between design and ultimate strength values. In contrast, although Section III requires a nominal factor of only three, this section of the Code provides rules for treating design features affected by fatigue loads and other secondary loading conditions known to be major contributors to failure of vessels.

3. Section III specifies that the vessel function, design conditions, and environmental loads be carefully defined in the Owner's Design Specification so that designers can account for all anticipated conditions that the vessel may experience over its service lifetime. Nonnuclear sections do not have such a requirement, and the majority of nonnuclear vessel failures reported were associated with conditions not considered during design.

4. Fabrication and installation methods are much more carefully controlled under Section III than for most nonnuclear service applications, because of the extensive quality assurance requirements of Section III.

Only four SEP reactor vessels were designed to Section III: Dresden 2, Palisades, Ginna and Millstone 1. The other seven vessels were designed to Sections I and/or VIII. However, the quality of these pre-Section III reactor vessels is much better than that of nonnuclear vessels designed to these sections since their design procedures were supplemented by requirements of various nuclear code cases, the Navy Code, and supplementary requirements of the vessel fabricator or vendor. The various nuclear code cases and the Navy Code used in the design of these seven nuclear vessels are discussed in Appendix N.

Comparing the seven SEP nuclear vessels designed to Sections I and/or VIII with the vessels designed to Section III with respect to the four design aspects expressed above, we conclude that the initial quality of these vessels is very close to that of Section III vessels. A summary of this comparison follows:

1. All of the pre-Section III reactor vessels were made of SA-302, Grade B, plate. SA-302 is very similar to the SA-533, Grade B, Class 1, material used for modern reactor vessels and has approximately the same tensile and toughness properties. Also, these vessels have thicker walls and therefore have lower primary stresses than a Section III designed vessel. These lower stresses reduce the probability of a brittle failure. Finally, SA-302 is an acceptable material for Code Class 1 vessels according to Section III.

2. All of these pre-Section III vessels except Dresden 1 and Yankee Rowe used the Navy Code in their design. The Navy Code requires the stress analysis to include fatigue and thermal transient effects similar to that required by Section III. One vessel, Haddam Neck was reevaluated to the ASME Code Section III rules. It was determined that the vessel met the Code rules and in some areas even exceeded them (Ref. 5).

3. All the SEP pre-Section III vessels were designed to meet conditions they might be expected to encounter during their lifetimes, such as plant heatup and cooldown, step load transients, loss of load, and turbine trip with SCRAM.

4. The supplementary requirements of the vessel manufacturers or vendors and the various code cases utilized required quality assurance programs very similar to those specified in Section III. The inspections performed are compared to Section III requirements for each of these vessels in tabular form in the discussion of the subject plants. (See appendices.)

Appendices A through K discuss the following topics: the design criteria and materials used in vessel construction; the status of material surveillance programs, pressure-temperature operating limits, inservice inspection programs; and the resolution of various generic safety items affecting vessel integrity.

Appendix L summarizes the design and material data for these vessels. Appendix M discusses the design criteria in ASME Code Sections I and VIII. Appendix N discusses the Navy Code and the code cases used in design of these plants.

The initial quality of Dresden 2, Palisades, Ginna, and Millstone 1 is considered acceptable since these vessels were designed to ASME Code Section III (1965 Edition). Class 1 vessels, including reactor vessels, designed to the 1965 Edition of Section III, are essentially the same as vessels designed to later editions of this code. The basic philosophy in Section III has not changed from 1965 to the present day. The most important changes in Section III have been to expand it to include design criteria for components other than the reactor vessel such as supports, valves, pumps, etc. The design criteria in the 1965 Edition were primarily concerned with the reactor vessel design. However, there have been some changes in the Code that affect the reactor vessel design that should be mentioned. The most important change is the additional requirements for fracture toughness testing and the prevention of brittle fracture introduced into the Code in 1972. Vessels designed before this criteria became effective did not perform all the tests required by it. However, because fracture toughness is so important, the staff has developed procedures for analyzing the fracture toughness of these older vessels to provide the same margin of safety as for newer vessels (Ref. 8). Another important change is the requirement to leave access to permit Section XI inservice inspections (discussed in more detail in Section 6.0 of this report). Other changes that will affect the reactor vessel to a lesser extent are the inclusion of more detailed procedures for performing quality assurance examinations, more detailed administrative requirements for quality assurance programs, and the addition of requirements for performing an analysis for faulted and emergency operating conditions.

The design criteria utilized for all the reactor vessels in SEP are considered acceptable. The design criteria used provide assurance that the initial integrity of these vessels is acceptable.

3.0 MATERIALS

Reactor vessels being built today are constructed from SA-533, Grade B, Class 1, plate material and SA-508, Class 2, forging material. ASME Code Section II contains the specifications for these materials. The specifications include requirements for chemical composition, mechanical properties, and methods for steelmaking, plate preparation, and the forging process. Additional requirements for the materials in the beltline region, including weld metal, are contained in Appendix G of ASME Code, Section III. The selection of these materials is based on their adequate strength, high resistance to unstable crack extension under load ("toughness"), and their availability in the required sizes and thicknesses. In addition, the materials must allow the production of high-quality weldments and be compatible with the stainless steel cladding.

All of the 11 SEP reactor vessels except Ginna were made of SA-302 B plate material. Ginna was made from SA-508, Class 2, forging material. Both of these materials are acceptable by today's ASME Code Section III standards. Five of these early vessels (Yankee Rowe,* Palisades, Millstone 1, Oyster Creek, and Dresden 2) were made from SA-302 B material modified by the addition of 0.4% to 0.7% nickel. The modified SA-302 B metal is identical to SA-533, Grade B, Class 1, material. The nickel is added to increase hardenability; i.e., to produce higher fracture toughness properties through the thickness of plates in the quenched and tempered condition.

The most significant difference between present day vessel materials and materials used for older vessels is the control of residual elements. It has been shown conclusively that residual copper and phosphorus impurities are highly detrimental to nuclear radiation embrittlement resistance (Ref. 6). In vessels constructed in the 1960s and early 1970s, copper content in base and weld metal often exceeded 0.30% and the phosphorus content was higher than 0.02%. For vessels being constructed today, copper and phosphorus contents are required to be maintained below specified

*The nickel content in several of the Yankee Rowe plates was only about 0.2%.

levels (generally about 0.12% for copper and 0.017% for phosphorus). The effect of this control on the material chemistry can be seen from the predicted radiation damage curves in USNRC Regulatory Guide 1.99, Revision 1. These prediction curves show that the degree of radiation damage (in terms of the drop in upper shelf Charpy energy or the increase in RT_{NDT}*) is reduced by about 50% when copper and phosphorus are reduced from 0.30% and 0.02% to 0.15% and 0.017%.

The materials used in the construction of these 11 reactor vessels are considered acceptable. They are acceptable by the current ASME CODE, Section III standards.

4.0 MATERIAL SURVEILLANCE PROGRAM

Irradiation causes a decrease in upper shelf toughness values and an increase in the ductile to brittle transition temperature of reactor vessel materials. The amount of degradation is difficult to predict since it is influenced to a large degree by the chemical composition of the material and its environment. Therefore, a material surveillance program is required to monitor the effects of irradiation on the mechanical properties of the reactor vessel materials. NRC regulations, Appendix H to 10 CFR Part 50, require that surveillance programs contain tensile and Charpy specimens from vessel beltline plate, heat-affected zone and weld material. These specimens are to be removed periodically from the vessel and tested in accordance with ASTM Specification E 185-73, "Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels." The base and weld metal to be included in the program should represent the material that limits the operation of the reactor during its lifetime (limiting material). The bases for selecting the limiting material are initial transition temperature, and the predicted amount of radiation damage considering chemical composition (particularly copper and phosphorus) and neutron fluence. The material surveillance programs of nuclear plants applying for a construction permit today are carefully reviewed to make certain these programs comply with Appendix H, 10 CFR Part 50. However, for older plants, such as all of the SEP plants, the surveillance programs were initiated prior to the issuance of Appendix H. Therefore, we can not expect the surveillance programs of these older vessels to completely meet all the requirements of Appendix H. The method of compliance with specific provisions of Appendix H are reviewed and determined on a case-by-case basis (Refs. 7, 8).

The most common deficiencies, or areas of nonconformance, found in the programs of older vessels are (a) Charpy specimens oriented in the strong direction, (b) insufficient number of Charpy specimens, (c) lead factors higher than three, (d) no weld material specimens, and (e) surveillance materials not including the limiting vessel material. With proper care in planning tests and analyzing the results, most of these nonconformances will not affect the validity and usefulness of the surveillance program to a great extent. For example, test results from longitudinally oriented (strong direction) specimens reduced to 65% of their value provide a conservative estimate of values expected from transversely oriented (weak direction) specimens (Ref. 8). Futhermore, weld metal, which is isotropic, is generally the limiting vessel material. Paragraph II.B of Appendix H requires that surveillance programs comply with ASTM E 185-73. This Standard Recommended Practice requires 12 Charpy V-notch specimens each from the limiting base, weld and heat affected zone (HAZ) materials. Some of the early vessels have programs with only 8 or 10 Charpy specimens. In such cases, we

*The reference nil-ductility temperature, RT_{NDT}, is the highest of the nil-ductility temperatures established from drop weight tests or 60°F less than the temperature at which the material exhibits at least 35 mils lateral expansion and 50 ft-lbs absorbed energy in Charpy V-notch tests.

recommend that tests be conducted first at relatively high temperatures to establish the upper-shelf energy* with a minimum of three data points before proceeding to lower temperature tests in the transition region. When tests are started at low temperatures with only eight specimens, upper shelf values may never be reached (Ref. 9). Appendix H requires that lead factors (the ratio of neutron flux received by specimens to the maximum flux received by the vessel inner surface) be no greater than three. The reason for this provision was the concern over the rate effect on damage. However, recent research work has indicated that varying the rate over several orders of magnitude will have little effect on the amount of radiation damage incurred (Ref. 10). Therefore, lead factors up to at least 10 appear satisfactory.

Regarding the lead factor, it should be pointed out that paragraph II.C.2 of Appendix H states that accelerated capsules are acceptable provided that the surveillance program contains the minimum number of wall capsules as specified in paragraph II.C.3. The required number of wall capsules, three to five, depends on the value of the adjusted reference temperature at end of life. In the preceding discussion, a wall capsule is defined as one with a lead factor of three or less. Since many of these older vessels had more than the required number of capsules (some as many as 20), their programs usually contained the required number of capsules with a lead factor of three or less even though some of their capsules exceeded the required lead factor of three.

The most significant deficiency found in some of the older surveillance programs is that they do not contain any weld metal or that they do not include the limiting vessel material. Programs with these nonconformances must be carefully evaluated to determine their effectiveness to predict radiation damage. In some cases it is possible to supplement the data from the deficient program with data from other surveillance programs or from research programs (Ref. 10). When data on a particular material are not available, the staff uses NRC Regulatory Guide 1.99, Revision 1, to obtain conservative predictions of radiation damage.

Although all of the SEP reactor vessels were designed and built prior to the issuance of Appendix H, 10 CFR Part 50, they all have ongoing material surveillance programs that generally conform to Appendix H with the exception of Yankee Rowe. Yankee Rowe had a surveillance program that was terminated due to a structural failure of the capsule holder fixtures. However, prior to termination, five capsules had been removed from the vessel and tested. These results are considered to be sufficient to monitor the effects of radiation on the vessel materials throughout service life. Since all of the SEP surveillance programs were established prior to the issuance of Appendix H, they all have some areas where they do not completely conform to Appendix H requirements. The areas of nonconformance for each plant are discussed in detail in Appendices A through K. From our review of these programs, we conclude that they all provide a satisfactory means of monitoring radiation damage on vessel materials throughout service lifetime.

The preceding discussion has pointed out some of the major deficiencies found in the surveillance programs of older plants. It should also be pointed out that these older programs often have areas that exceed Appendix H requirements. Many of the surveillance programs of older plants contain specimen types and materials not presently required by Appendix H. Examples of additional specimens

*Upper-shelf energy is defined as the energy to fracture Charpy specimens at temperatures in which the fracture mode is 100% shear.

are WOL (wedge opening loading) and compact tension specimens. These types of specimens are used to obtain fracture toughness properties such as K_{IC}.* Older surveillance programs also often contain specimens from a correlation monitor material. The object of performing tests on a correlation material is to correlate the test results of one program with the surveillance results of other programs, including those from test reactors. Also, many of these older programs contain more capsules (hence more total specimens) than required by Appendix H.

5.0 PRESSURE-TEMPERATURE OPERATING LIMITS

Pressure-temperature limits for reactor vessel operation provide a means of assuring vessel integrity throughout its operating life. Operation in accordance with NRC regulations will ensure that, in the normal operating range, the vessel will operate in the upper-shelf region of its material toughness. The required operating limits also provide assurance that the fracture toughness of vessel materials during heatup and cooldown transients will be adequate to prevent rapid crack propagation (brittle fracture).

Pressure-temperature limits for inservice testing, heatup and cooldown, and core operation are required to be in compliance with the rules of Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements." When first published in 1971, Appendix G used a transition temperature approach to establish safe operating limits. Appendix G was revised in 1973 to require a fracture mechanics approach. The fracture mechanics approach usually gives more conservative operating limits. The fracture mechanics approach relies on a fracture mechanics characterization of the material and its stress environment. Using this characterization, the stress in any portion of the vessel, in conjunction with any assumed flaw, can be compared with the stressed-flaw tolerance of the material, a material parameter such as K_{IC}. Using this parameter, the stress in the vessel can be limited such that, in the presence of an assumed flaw size so large as to ensure detection, no rapid crack propagation can occur. Above NDT, the fracture toughness of the materials used in nuclear reactor vessels increases greatly. Thus, the crack tolerance of the material at the normal operating temperatures is high. Under this system of fracture control, prevention of rapid fracture is assured by the control of stresses and flaw sizes. For nuclear vessel materials of normal shelf fracture toughness (according to Appendix G, 10 CFR 50, a Charpy upper-shelf energy of 50 ft-lb is required), very large cracks would be required to cause the onset of rapid crack propagation at operating temperature and pressure. In regions of high local stresses, such as nozzle corners, ductile tearing could commence at smaller cracks or lower pressure but, as the tear extended into a region of lower nominal stress such as the vessel wall, rapid fracture would again require very large cracks. Appendix G also states that a smaller postulated defect may be used provided that it can be justified. For example, Appendix G states that a quantitative evaluation of the fracture toughness requirements for nozzles is not feasible at this time, but preliminary data indicate that the design defect size for nozzles (considering the combined effects of internal pressure, external loading and thermal stresses) may be a fraction of that postulated for the vessel shell. Nondestructive examination methods shall be sufficiently reliable and sensitive to detect these smaller defects.

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*K_{IC} is the plane strain fracture toughness of a material.

The specific methods to calculate the pressure-temperature operating limits are contained in Appendix G to ASME Code Section III. For regions remote from discontinuities (the beltline region), the stress intensity factors calculated in the development of these operating limits are based on a postulated sharp, surface flaw penetrating to a depth of 1/4 of the vessel wall thickness and having a length 1½ times the section thickness. Since the maximum size flaw that might escape detection in a preservice or inservice inspection is much smaller than this assumed flaw size, the combination of inspections and conservative pressure-temperature limits provides a high degree of assurance for vessel integrity throughout service life. For nozzles, flanges, and shell regions near discontinuities, a smaller defect size may be used. The smaller defect size must be justified and nondestructive examination methods must be sufficiently reliable and sensitive to detect these smaller defects. The procedures to calculate the stress intensity factors for these regions provide margins of safety comparable to those required for the beltline region. Appendix G provides methods to calculate stress intensities for membrane tension stress, bending stress, and stresses resulting from thermal gradients, and lists the safety factors to be applied to these stress intensities.

Irradiation degrades material toughness causing its RT_{NDT} to increase. Since the pressure-temperature limits are based on a temperature above RT_{NDT}, these limits must be revised periodically to reflect the changes in toughness. Since the postulated flaw penetrates to 1/4 the wall thickness, the increase in RT_{NDT} is based on the fluence at the 1/4 thickness location. Increases in RT_{NDT} are usually obtained from the results of the vessel's material surveillance program. If these results are for some reason not considered applicable or valid, the staff uses Regulatory Guide 1.99, Revision 1, to obtain conservative radiation damage values.

As of January 1, 1979, all of the SEP plants have pressure-temperature operating limits that are in conformance with Appendix G, 10 CFR Part 50. The NRC staff will continue to review the results of material surveillance programs of the SEP reactor vessels as well as those of later vintage vessels. These results will be used to update the operating limits of these plants. As a minimum, the staff will require that these limits meet the requirements of Appendix G, 10 CFR Part 50. In general, the same criteria will be used to evaluate the operating limits of these older reactor vessels as will be used for later vintage vessels.

In the review of the pressure-temperature operating limits, Regulatory Guide 1.99, Revision 1, was used to evaluate the amount of radiation damage, change in RT_{NDT}, on vessel materials. This guide has been approved by NRC's Regulatory Requirements Review Committee as a backfit item. The guide was used to conservatively predict radiation damage on materials that are not included in the vessel's material surveillance program. The guide was also used to check the validity of test results from the various SEP material surveillance programs by comparing them to the predictions in the guide. No major or unexpected deviations from the Regulatory Guide predictions were noted in these reviews.

6.0 INSERVICE INSPECTION PROGRAM

Following the completion of fabrication, testing, examination, and certification in accordance with the construction rules of ASME Code Section III, and as part of the requirements of ASME Code Section XI, modern reactor vessels are required to be further examined nondestructively prior to, and as a condition for, placement into nuclear power plant service. These examinations are totally unlike those generally applied to the inspection of fossil-fueled power plant boiler drums. Inservice inspection programs for steam boiler drums (fossil vessels), which have been formalized by the National Board Inspection Code, have evolved as a result of specific requirements imposed by state jurisdictional authorities. Visual examinations and hydrostatic tests are, in general, the extent of inspection performed in su[illegible] cases. The preservice examinations required by Section XI require volumetric examination of [illegible] of the vessel that are subject to periodic inservice inspections. These examinations cover es[illegible]y 100% of the pressure-retaining welds. These preservice examinations provide a record of any indications, such as extremely small flaws, within the limits permitted by the allowable indication standards of Section XI. Also, the mapping of small flaws provides a basis for future comparison with the results of subsequent periodic inservice examinations to determine the flaw growth during service. Evaluation and analysis of the influence on the structural integrity of any detected change in size of flaws provide, in turn, a means to assess the safety of the vessel for continued service.

Section XI identifies the principal vessel areas and the extent of those areas that are periodically examined during the entire service lifetime of the reactor vessel. The principal areas of the vessel selected for examination are (a) those that are more highly stressed (e.g., nozzle-shell junction), (b) those components for which a representative examination sampling provides an assessment of the overall structural co[illegible]dition of the component materials (e.g., shell weld joints), and (c) those portions of the vessel where environmental effects and irradiation could influence the properties of the vessel materials (e.g., reactor beltline region). The inspections required by Section XI have resulted in many changes in vessel design to facilitate inspection and to provide access for the conduct of examinations. The examinations required by Section XI are required to be completed during each 10-year interval of service (inspection interval). Each inspection interval is divided into three 40-month inspection periods. A specified percentage of the total required inspections on each component is required to be performed during each inspection interval.

ASME Code Section XI requires nondestructive inservice examinations utilizing volumetric, surface, and visual examin[illegible]tion methods. As the name implies, a visual examination is a visual observation of a component for signs of leakage, corrosion, or deterioration. S[illegible]face examinations, such as dye penetrant and magnetic particle, are used to reveal flaws at or near the surface inspected. Generally, the most effective method of detecting flaws is by volumetric methods such as ultrasonic, radiography, and eddy current [illegible]amination techniques. A volumetric examination examines the entire volume of a material for flaws. Rules governing the examination procedures for the three preceding volumetric examination techniques are given in Section XI. However, Section XI does not preclude the use of other examination techniques provided they can be justified (Article IWA-2240). The most effective alternative volumetric examination method (a method not specifically described in Section XI) appears to be acoustic emission. One of the important advantages of acoustic emission is that it can be used to detect flaws in regions where lack of accessibility prevents the use of other techniques. The main disadvantages of acoustic emission are (a) that it has not been tested sufficiently to prove its effectiveness, and (b) there are no codes approved by NRC to govern its use.

Many operating plants have reactor vessels that were designed prior to the initial issuance of Section XI. The design of these vessels does not provide access to permit the conduct of all examinations now required by Section XI. Furthermore, prior to Se[illegible]tember 1, 1976, there was no requirement in NRC regulations (10 CFR Part 50) governing the inse[illegible]vice inspection program for plants that received a construction permit prior to January 1, 1971. Thus, the inspection programs formulated by older plants were generally nonuniform. However, on September 1, 1976, a change to paragraph 50.55a(g) became effective which requires that, throughout the service life of a nuclear facility, the reactor vessel be inspected in accordance with editions of ASME Code Section XI and addenda that become effective to the extent practical within the limitations of design, geometry and material of construction of the vessel. Thus, the inservice inspection programs for older plants will be, as far as practical, the same as for newer plants.

Under the revised requirements of paragraph 50.55a(g), inservice examinations conducted during successive 40-month periods throughout service life will comply to the extent practical with the requirements in editions of Section XI and addenda in effect no more than 6 months prior to the start of each 40-month period. For older plants the initial period will be that period starting after September 1, 1976, based on successive periods commencing at the start of facility commercial operation. The date of the first inspection period for the SEP plants under this provision follows:

Ginna	-	9/1/76
Palisades	-	10/31/76
Dresden 1	-	3/4/77
Millstone 1	-	8/28/77
Haddam Neck	-	1/1/78
San Onofre 1	-	1/1/78
Yankee Rowe	-	3/1/78
Big Rock Point	-	9/1/78
Dresden 2	-	2/9/79
LaCrosse	-	11/1/79
Oyster Creek	-	12/79

An updated inservice inspection program in accordance with 10 CFR 50.55a(g) has been submitted for all plants in SEP. The program for Big Rock Point was originally due by mid-1978. However, NRC granted, by exemption, the licensee a 7-month delay. The program for Big Rock was submitted in December 1978 and is currently under review. The NRC staff has completed its review of updated inservice inspection programs for Ginna, Palisades, Dresden 1, Millstone 1, and San Onofre 1. Safety Evaluations, including grants for relief from performing inspections on certain components in accordance with Section XI requirements where we have determined that Code requirements are impractical, for these plants have been issued. The review of the inservice inspection programs for the other SEP plants is expected to be completed by early 1980.

Since the reactor vessels in SEP were designed prior to the initial issuance of Section XI, some inspections now required by Section XI cannot be performed because of the design geometry of the plant. The most important category where relief is requested is the Category B-A* welds in the vessel core region. The only plants in SEP that can inspect these welds are San Onofre, Haddam Neck, Ginna, and Palisades. Dresden 1 management has committed to attempt an inspection of these beltline welds. In view of the importance of these welds and the fact that many older plants cannot inspect them, it is recommended that alternative inspection techniques be investigated by NRC. The technique that appears to be most promising is acoustic emission. NRC is currently funding research programs to develop acoustic emission examination methods (Ref. 11).

*Examination Category is that used in Table IWB-2600 of Section XI.

Commercial companies are also conducting research on acoustic emission. Exxon Nuclear Company has performed an acoustic emission examination of the German KRB plant nuclear steam supply system (Ref. 12). This inspection detected three significant acoustic sources and nine minor sources. The areas of significant sources were reexamined by ultrasonic techniques and flaws were found that closely resembled those predicted by acoustic emission. Electric Power Research Institute recently initiated an acoustic emission examination of the LaSalle reactor vessel during a shop hydro test. The results of this examination were compared to those obtained during the preservice ultrasonic examination. The comparison of results indicates that acoustic emission examination methods are feasible. Dunnigan/Endevco reviewed the progress that has been made in acoustic emission monitoring of nuclear plants (Ref. 13). This report also concludes that acoustic emission examination techniques are feasible for monitoring reactor vessels. Reference 13 also lists acoustic emission examinations that have been performed on commercial nuclear power plants. The examinations performed to date on nuclear plants gives credence to the acoustic emission method. It is recommended that the NRC staff formulate a set of rules to govern acoustic emission examinations so that utilities would receive credit for examinations conducted by this method.

By the end of 1979, all reactor vessels in SEP will have updated inservice inspection programs in accordance with the provisions of 10 CFR 50.55a(g). These programs will require that the examination of all vessel components be performed in accordance with updated editions of Section XI to the maximum extent possible. These inspection programs will continue to be updated throughout service life. Examinations in accordance with these inspection programs will provide assurance that the integrity of the reactor vessels in SEP will be maintained at acceptable levels throughout service life.

7.0 GENERIC SAFETY ITEMS

The status of several generic safety items that affect the integrity of the reactor vessel are discussed in this report. These items include low upper-shelf Charpy energy, PWR overpressurization protection system, BWR feedwater nozzle cracking, BWR control rod drive return line nozzle cracking, and sensitized stainless steel safe end cracking.

7.1 Low Upper-Shelf Charpy Energy

The NRC staff has determined that some operating plants may have reactor vessels with material that will have marginal toughness after several years of operation. PWR vessels are more likely to have a low upper-shelf problem than BWR vessels because their materials are exposed to much higher fluence levels. However, since some of the pre-Section III BWR vessels are small and will see relatively high fluences, the upper-shelf energy status of all SEP vessels is being reviewed as part of the SEP review.

The technical aspect of the problem is that Appendix G, 10 CFR Part 50, specifies minimum toughness requirements for operation that are based on the assumption that the upper-shelf toughness is at least 50 ft-lbs, as measured by Charpy impact tests. Recent surveillance data indicate that the toughness of materials in some plants may fall below 50 ft-lbs after comparatively short periods of operation.

Although the 50 ft-lb minimum was derived by empirical methods, it is not considered overly conservative by most experts in the field. This level was very carefully reviewed at the time Appendix G was formulated, and is supported by the best empirical correlations available. At this time, we have only limited data that justify operation with lower toughness than is indicated by the 50 ft-lb Charpy level and, except for the smallest vessels, 45 ft-lbs is about as low as we expect to be justified with current technology. However, some justification to permit operation with Charpy upper shelf as low as 33 ft-lbs is contained in Reference 14. This justification is based on an Appendix G type of analysis and on the fact that the stress levels in some vessels are lower than those permitted in ASME Code Section III.

Appendix G, 10 CFR Part 50, describes the steps that must be taken to justify continued operation in cases of low toughness (i.e., under 50 ft-lbs). It specifies that all the following must be done:

1. Augmented inservice inspection of the reactor vessel beltline.

2. Additional fracture toughness information from different types of tests, such as K_{ID} determinations using special specimens. K_{ID} is the fracture toughness value similar to K_{IC} except that it is determined from a dynamic test.

3. A fracture mechanics analysis proving that sufficient safety margins are provided.

Category A Technical Activity No. A-11 will develop criteria to evaluate low upper shelf vessels (Ref. 15). At present it appears that only two SEP vessels (Ginna and Yankee Rowe) may have an upper-shelf Charpy energy that falls below 50 ft-lbs. For Ginna, this will not occur until at least 11 EFPY (Effective Full Power Years) of operation. By this time, Activity A-11 will be completed and Ginna will be reevaluated in accordance with the resulting criteria. It is estimated that the upper-shelf energy of plate materials in the Yankee Rowe vessel will fall to about 42 ft-lbs at end of life. However, because of the low stresses in the vessel, this value of upper shelf energy may be shown to be acceptable. This item will be reviewed again following the completion of Activity A-11, when the licensee provides submittals in accordance with Appendix G.

7.2 Overpressurization Protection System

As of January 1, 1979, PWR licensees have reported 33 incidents of reactor coolant system pressure transients in excess of the Technical Specification or Appendix G pressure-temperature limits (Ref. 16). The majority of cases have occurred during reactor startup or shutdown when the reactor coolant system was in a water solid condition. Of the 30 events, 10 reached a pressure of 1000 psig or more, 4 reached a pressure of 1500 psig or more, 3 reached a pressure of 2000 psig or more, and 1 exceeded 3000 psig. However, half of these incidents occurred prior to initial criticality of the reactor. Since there was no core decay heat or fission products, these events did not pose a potential hazard to the public health and safety. All of the pressure transients were such that fracture mechanics and fatigue calculations indicate that the reactor vessels were not damaged and that continued operation of these vessels was acceptable.

The pressure transient events reported to date have affected only pressurized water reactors (PWRs). Boiling water reactors (BWRs) never operate in a water solid condition except during some hydrostatic tests. During cold shutdown conditions for BWRs, a letdown path is maintained through the reactor water cleanup system to remove the water added to the reactor through control rod drive seals. This flow is controlled to maintain reactor water levels within a narrow range. Thus, the upper region of the reactor vessel always contains vapor (steam) or gas (air). This provides a significant capability to accept volume surges with only small pressure changes. Also, the BWR reactor is pressurized for normal operation by heatup of the coolant and follows a water saturation pressure line. Thus, high pressures are not produced unless the vessel temperature is sufficient to satisfy Appendix G pressure-temperature limits.

Because of the relatively high frequency of these pressure transients, the NRC staff has concluded that administrative procedures and overpressure protection devices should be upgraded in an appropriate time frame to reduce the likelihood of future pressure transient events. As of January 1, 1979, all SEP PWRs have upgraded their overpressurization protection systems and submitted proposed Technical Specification changes to minimize the possibility of inadvertent pressure transients. Licensing action, including the Safety Evaluation and required Technical Specification changes, has been completed for Haddam Neck, Ginna, Palisades, and Yankee Rowe. Licensing action is expected to be completed on San Onofre by the end of 1979. However, the licensee is administratively complying with their proposed Technical Specifications that contain testing and administrative procedures to protect against an overpressurization incident.

Criteria used for the review of overpressurization protection systems in operating PWRs is outlined in NRC Report NUREG-0224 (Ref. 16). These criteria differ slightly from those used to evaluate plants undergoing a construction permit or operating license review. Differences are mainly in the areas of administrative controls, and electrical and seismic design requirements of the systems. These differences are not considered to be significant in terms of the degree of protection that will be provided.

7.3 BWR Feedwater Nozzle Cracking

The feedwater nozzles of essentially all operating BWRs inspected to date have been found to have blend radius cracks, some of which propagated through the cladding into the base metal (Refs. 17, 18). In several reactors, similar cracks were found in the nozzle bore.

The deepest cracks found to date were in the nozzle bore and were of a total depth of about 1½ inches. Analyses performed by the NRC staff, which are in agreement with those done by GE and field data from operating BWRs, indicate that the initial crack growth rate is high up to crack depths of about 1/4 to 1/2 inch. Further growth is slow but would accelerate with increasing depth. Eventually, the cracks will present a repair problem if, in removing them by grinding, the ASME Code limits on nozzle reinforcement should be exceeded. The crack depth equated with the reinforcement limit will depend on the details of nozzle dimensions (see NB-3330 in Section III, ASME Code).

Feedwater nozzle cracks are of concern to the NRC staff because (a) reactor pressure vessel integrity is extremely important to safety, (b) there are uncertainties about the rate at which the cracks might grow, (c) current nozzle repair procedures require that cracks be removed, thus removing metal from a relatively high stressed region of the reactor vessel, and (d) considerable radiation exposure is received by personnel performing inspections of the nozzle region and repairing cracks in the nozzles.

The NRC staff is in general agreement with GE as to the mechanisms responsible for crack initiation and growth. Crack initiation is believed to be the result of high cycle thermal fatigue caused by rapid fluctuations in water temperature within the vessel in the sparger-nozzle region during periods of low feedwater temperature when the flow may also be unsteady and perhaps intermittent. Once initiated, the cracks are believed to be driven deeper by the larger, relatively low frequency startup/shutdown pressure and thermal cycles. The latter result from significant changes in feedwater temperature during flood-up of the reactor vessel and when feedwater heaters are put into, or taken out of, service. During normal power operation, the plant feedwater heaters maintain the feedwater temperature at about 180°F below the reactor water temperature. At low power, when the feedwater heaters are not in service, the temperature differential can be 400°F or more. We believe that the basic cause of the thermal fatigue cracking problem is this relatively large temperature differential between cold incoming feedwater and the hot reactor vessel water during low power/flow and flood-up operations.

Because of the current incomplete status of studies and design efforts to resolve the nozzle cracking issue, and because hardware changes and other long-term remedial measures will require considerable time to implement at operating facilities, certain interim revisions in operational practice are desirable.

In general, the NRC staff has concluded that BWR facility operators should monitor feedwater temperature and flow during low power operation (Ref. 18). In addition, operating procedures should be revised to minimize rapid changes in feedwater flow and/or temperature, to minimize the duration of cold feedwater injection, to avoid conditions that may lead to inadvertent or unnecessary high pressure coolant injection (HPCI) system actuation, and to avoid the introduction of cold water from the reactor cleanup system. Reactor operators should attempt to limit the temperature differential between water entering the feedwater nozzles and the reactor vessel water to no greater than the normal differential at full power. They should also avoid feedwater temperature transients to the extent practicable. It has been demonstrated that by carefully bringing feedwater heaters into service, the magnitude of feedwater temperature transients can be significantly reduced. While these steps are not expected to eliminate the nozzle cracking problem, we believe that they should help to minimize the extent of cracking until permanent changes are made.

Inspection of SEP vessel nozzles has revealed crack depths ranging from 1/4 to 1/2 inch at Millstone 1, Dresden 2, and Oyster Creek. Because of their design, the other SEP BWRs Dresden 1, LaCrosse and Big Rock Point vessel nozzles are not subject to this type of nozzle flaw. To date, all of the SEP plants having this nozzle cracking problem have taken steps to alleviate it. All affected SEP plants have completed at least one dye-penetrant examination and have removed all cracks detected in the feedwater nozzles. Oyster Creek has replaced the sparger/sleeve with an advanced design component and has removed the nozzle cladding. Millstone 1 and Dresden 2 have installed interference-fit sleeves as an interim solution and are currently planning their future long-term resolution of this problem. This interim action by the licensees is considered satisfactory. Final resolution of this item will be made upon completion of NRC Technical Activity A-10 (Ref. 14).

7.4 Control Rod Drive Return Line Nozzle Cracking

There is usually one control rod drive (CRD) return line nozzle in BWR reactor vessels, generally located from 68 inches to 100 inches above the top of the active fuel. The return line is typically

4 inches in diameter. As early as 1974, a General Electric task force on austenitic stainless steel piping noted the large measured thermal gradient in CRD return line (CRD RL) nozzles. Based on the unexpectedly high top to bottom thermal gradients in the nozzle, particularly at low flows, crack initiation susceptibility was cited and rerouting the return line was considered. In addition, recent experience with BWR feedwater nozzles has demonstrated the occurrence of crack initiation in nozzles from thermal cycling and further suggested the need to examine CRD return line nozzles. Flaws approximately 1 inch deep (including the cladding) have been reported in the return line nozzles. GE issued Service Information Letter (SIL) No. 200 in October 1976 recommending inspection of the nozzle and rerouting of the return line (Ref. 19). This SIL was amended in March 1977 to provide for valving out the return line as an interim fix.

Dye penetrant (PT) inspections of the CRD return line nozzles to date at BWR plants have revealed cracks in the majority of plants inspected. Cracking has been observed in both the blend radius and bore regions of the CRD RL nozzle. Even though most plants have a thermal sleeve in the CRD RL nozzle, which would be expected to reduce the amount or extent of cracking, cracks have been found at plants both with and without sleeves.

The cause of crack initiation appears to be thermal fatigue, similar to that experienced with BWR feedwater nozzles. The thermal cycling results from the low temperature (50°F to 100°F) condensate water that enters the reactor vessel through the CRD RL nozzle during normal operation. Although crack initiation mechanisms for the feedwater and CRD RL nozzles apear to be the same, there is a substantial difference in the steady-state stresses that ultimately affect crack growth rates. CRD RL nozzle crack growth appears to be enhanced by the existence of a continuous large thermal gradient from the top to the bottom of the nozzle (550°F at the top, 50°F at the bottom), that yields high thermal stresses.

Effective long-term solutions to this problem require that the thermal cycling in the CRD RL nozzle be eliminated. Accordingly, the General Electric Company has made recommendations, both interim and final, involving system modifications to accomplish this goal.

The initial recommended General Electric fix involved (a) valving off the CRD return line to the reactor vessel, (b) reducing CRD RL system flow, (c) raising the exhaust water pressure to a level sufficient to permit the return water to enter the reactor vessel via leakage past the sealing rings in the control rod drives rather than via the return line, (d) adding exhaust water filters, and (e) testing the modified system to verify that the control rod drives would operate properly.

The interim General Electric system modification proposed rerouting the CRD return line in conjunction with the repair and capping of the nozzle. For BWR/2 plants, GE recommended that the return line be rerouted to the feedwater system outside the primary containment and downstream of all motor operated isolation valves. For BWR/3, 4 and 5 plants, the return line could be directed to the reactor water cleanup system downstream of the last motor-operated isolation valve. A final solution involving only cutting and capping the nozzle without reroute is still under NRC review. Although in agreement with the GE proposed interim solution for return line rerouting, the NRC staff also recommended an augmented inspection of these nozzles and the reactor vessel wall below these nozzles (Ref. 18).

Dresden 1, Big Rock Point, and LaCrosse do not have a CRD R' nozzle. Therefore, this item is not applicable to these plants. Each of the applicable SEP facilities (Millstone 1, Dresden 2, and Oyster Creek) has taken action to inspect the CRD RL nozzle. Dresden 2 is currently operating with the return line valved out. Millstone 1 is operating with the return line rerouted to the feedwater system. Oyster Creek, after inspecting and finding no crack indications, is operating with the original design configuration (thermal sleeve welded to the nozzle safe end). The above interim action is presently considered to be satisfactory. However, some further modifications may be required after completion of NRC Technical Activity A-10 (Ref. 14).

7.5 Sensitized Stainless Steel Safe Ends

Stainless steel safe ends can become sensitized during welding operations or stress relieving heat treatments. Sensitization occurs when a nonstabilized stainless steel containing over about 0.02% carbon is held for a period of time at a temperature ranging from 800°F to 1600°F. The maximum allowable carbon content in 304 stainless steel is 0.08%. For 304L stainless steel the carbon is intentionally kept low, to a maximum of 0.035%, and its resistance to stress corrosion cracking is relatively good. It should be pointed out that since no minimum carbon values are required, a 304 stainless steel can have less carbon than a 304L material. Sensitization is generally attributed to the precipitation of a complex carbide in the grain boundaries. In supplying chromium for the carbide precipitate, the chromium content in the immediate vicinity of these carbides drops below a critical limit and, therefore, the material becomes subject to severe attack by corrosive media. Chlorides and fluorides are the most important contaminants, although oxygen, low pH, and elevated temperature generally must also be present for cracking to occur. When a sensitized stainless steel is subjected to corrosive environments, cracking usually occurs in the grain boundaries (intergranular). The rate of this corrosion cracking in sensitized stainless steels increases as the applied stress is increased. In safe ends, the stresses are generally low except for regions near welds where the residual stresses may be high. Thus, the major cracking is usually found in the areas around welds.

Cracking in sensitized stainless steel material resulting from stress corrosion generally occurs only in BWRs. In PWRs, controls on the water chemistry of the primary coolant provide a noncorrosive atmosphere (Refs. 20, 21). Not only are halides monitored and controlled to low levels, but oxygen concentration is also automatically held to practically zero during plant operation by the use of hydrogen overpressure. Even with the use of boric acid, the effective pH is high. No stress corrosion cracking of either sensitized or nonsensitized stainless steel is expected under these conditions. Extensive testing experiences under operating conditions have borne this fact out. Therefore, only the BWR vessels in SEP have been reviewed in this area.

All the BWRs in SEP except Big Rock Point have taken steps to resolve the sensitized stainless steel cracking problem. The Big Rock Point primary safe ends are sensitized but the carbon content of the 304 stainless steel used was held to low values. As pointed out above, this increases its resistance to stress corrosion cracking. To date, no cracks have been detected in these safe ends. LaCrosse and Millstone 1 have replaced all their sensitized stainless steel safe ends with nonsensitized material. Dresden 2 originally had 27 sensitized stainless steel safe ends. Five of these have been replaced with nonsensitized material. The remaining 22 safe ends are inspected at each refueling outage. All the safe ends in Oyster Creek are cladded with weld overlay on their inside diameter. The sensitized safe ends on Dresden 1 are made of 304L stainless steel. Since

304L has low carbon, it has a good resistance against stress corrosion cracking. So far no cracks have been reported in the Dresden 1 safe ends. During the forthcoming decontamination outage, a complete examination of these safe ends will be performed. If any large flaws are detected in this examination, the staff will require the licensee to take other steps to resolve this issue. The above inspection and repair procedures are in accordance with the recommendations of the NRC Pipe Crack Study Group (Ref. 22).

8.0 CONCLUSIONS AND RECOMMENDATIONS

From this review, including SEP Topics V-3 (Overpressurization Protection) and V-6 (Reactor Vessel Integrity), it is concluded that the integrity of all 11 reactor vessels in SEP is currently acceptable. Safety Evaluations on the overpressurization protection systems for Haddam Neck, Ginna, Palisades, and Yankee Rowe have been issued. The review of the San Onofre system is expected to be completed by the end of 1979. The integrity of these 11 vessels was evaluated by the same criteria that are used to evaluate the latest operating reactor vessels. Acceptable inservice inspection and material surveillance programs, conservative pressure-temperature operating limits, and the use of materials having acceptable fracture toughness properties provide assurance that the integrity of such vessels will be maintained at acceptable levels throughout the remainder of their service life. Based on the test results of the Ginna and Yankee Rowe surveillance programs, we conclude that some of the beltline materials of these two reactor vessels will fall below current fracture toughness requirements prior to end of life. The Ginna vessel materials are expected to fall below current standards at about 11 to 13 EFPY. Prior to this time, the NRC staff will have completed NRC Technical Activity A-11. Based on the criteria developed from this activity, the Ginna reactor vessel will be reevaluated before its materials drop below current standards. The NRC performed a fracture analysis on the Yankee Rowe reactor vessel in accordance with paragraph V.C.3 of Appendix G, 10 CFR Part 50. Because of the low stresses in this vessel, the analysis showed that the levels of upper-shelf toughness of the Yankee Rowe vessel materials throughout service life are acceptable. This topic will be reviewed again in accordance with the criteria to be developed from Technical Activity A-11. The extent of inservice examinations on Dresden 1 have been limited by high radiation levels in the primary system. However, Dresden 1 is currently shut down for an extended period during which time the primary system is being decontaminated. Upon completion of this decontamination, an extensive examination will be performed on vessel components. The integrity of the Dresden 1 reactor vessel will be established by the results of this post-decontamination inspection. The integrity of the remaining eight reactor vessels is projected to be acceptable throughout service life. This conclusion is based on the assumption that NRC reviews of safety-related topics, such as updated inservice inspection programs, are completed, including implementation, expeditiously and properly (Ref. 23).

Finally, it is noted that many of these reactor vessels in SEP, as well as many other older operating vessels, do not have the required access to permit the conduct of all examinations required by ASME Code Section XI. For such vessels, it is recommended that acoustic emission methods be considered as an alternative examination method. Rules governing the performance of acoustic emission examinations and acceptance standards should be formulated.

Detailed conclusions and recommendations for each reactor vessel are presented in the appendices.

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APPENDIX B

YANKEE ROWE ATOMIC POWER PLANT
REACTOR VESSEL

Design

The nuclear steam supply system, consisting of four loops, was designed by Westinghouse. The reactor vessel was designed and fabricated by Babcock and Wilcox in accordance with ASME Code Section VIII and Westinghouse specifications (Refs. 1, 2, 3). In addition to the minimum Code requirements, the following quality enhancing factors were employed:

1. The vessel stress analysis included the following:

   a. A discontinuity analysis was performed on the vessel closure head.

   b. Calculations were performed to account for nozzle stresses due to piping forces and moments on the reactor vessel primary coolant inlet and outlet nozzles.

   c. A stress analysis of the reactor vessel support lugs was performed.

   d. A stress analysis was performed to establish the allowable heatup and cooldown rates for the vessel.

2. The vessel was constructed of the following materials:

   a. SA-302, Grade B, modified slightly in chemistry for improved toughness. This material was Charpy V-notch impact tested and is essentially equivalent to the SA-533, Grade B, Class 1, material being used today. It was used for the vessel heads and the cylindrical region opposite the core.

   b. The primary coolant nozzles are SA-182 forgings modified to meet the mechanical and chemical properties of SA-302, Grade B. The material was Charpy V-notch impact tested and is very similar to the SA-503, Class 3, material being used in some vessels today.

   c. The vessel flanges are SA-105, Grade II, material and were Charpy V-notch impact tested. This material is similar to SA-508, Class 1, material. It is slightly lower in alloy content and lower in strength than the SA-508. Class 2 material being used for most reactor vessel flanges today and accordingly must be used in greater thicknesses than SA-508, Class 2, for equivalent service.

3. All plate forging material forming the pressure boundary of the vessel was ultrasonically inspected.

4. All weld preps of ferritic materials were magnetic particle inspected prior to deposition of weld metal.

5. The surfaces of all completed ferritic welds were magnetic particle inspected.

A tabulation of fabrication quality control inspections performed for the Yankee Rowe reactor vessel, as compared to those required by ASME Section VIII and Section III, is included in Table B-1.

In the beltline region the vessel ID is 109 inches and the thickness is 7-7/8 inches. The operating pressure is 2000 psig and the design pressure is 2500 psig. The maximum estimated end-of-life fluence on the vessel wall ID is 2.5×10^{19} n/cm^2. The date of commercial operation is July 1961. As of January 1, 1979, Yankee Rowe has operated for about 13 EFPY.

Materials

The hemispherical bottom head of the reactor vessel is formed of carbon steel plate clad on the inside with a sheet of 304 stainless steel. The cylindrical shell section of the reactor vessel, except for the upper shell, is made up of two rolled courses of carbon steel plate and is clad on the inside with sheets of 304 stainless steel. Each rolled course has a longitudinal weld seam and the two courses are joined together by a circumferential weld seam. Twenty equally spaced brackets for supporting the thermal insulation are welded to the outside of the lower shell section. The cylindrical upper shell section of the reactor vessel is made up of three formed segments of carbon steel plate clad on the inside with weld deposited 308L stainless steel. The three formed segments are joined together by three longitudinal weld seams. The stainless steel sheet cladding is 0.109 inch thick and was attached to the base material by spot welding. The weld deposited cladding is 1/4-inch thick.

The plates used to construct the vessel are made of SA-302B steel modified by the addition of about 0.2% nickel (Ref. 4). This is slightly less than the 0.3% to 0.7% amount required for SA-533 material. However, it is expected that this nickel additive will increase the hardenability of the material. The copper and phosphorus content of these plates varies from 0.18% to 0.20% and 0.012% to 0.02%, respectively. These values are not excessively high, and therefore these materials are expected to provide adequate resistance to radiation damage. Welds were made by the submerged arc process. The type of weld wire and flux used is unknown. No chemical analysis was made on weld material.

For both plate and weld material, RT_{NDT} is estimated to be 10°F by the procedures outlined in Standard Review Plan, Section 5.3.2. Charpy tests were performed on all vessel plate and weld materials at a test temperature of 10°F. The Charpy energy varied from 36 to 41 ft-lbs for the beltline weld metals and were about 32 ft-lbs for plate materials (strong direction).

TABLE B-1

INSPECTION AND ACCEPTANCE STANDARDS FOR YANKEE ROWE

	ASME Section III		Yankee Rowe		ASME Section VIII	
Materials	Exam.*	Extent	Exam.*	Extent	Exam.*	Extent
Plates	U.T.	100% Volume	U.T.	100% Volume	N.S.	-
Forgings	U.T.	100% Volume	U.T.	100% Volume	N.S.	-
Fabrication						
Weld Grooves	M.T. or P.T.	100% Surface	M.T.	100% Surface	Visual	N.S.
Weld Joints	M.T. or P.T.	100% Surface	M.T.	100% Surface	-	-
Shell and Head	R.T.	100% Volume	R.T.	100% Volume	R.T.	100% Volume
Nozzle Welds	R.T.	100% Volume	R.T.	100% Volume	R.T.	100% Volume
	M.T. or P.T.	100% Surface	M.T.	100% Surface	N.S.	-
Partial Penetration Welds						
Progressive	M.T. or P.T.	100% Surface	M.T. or P.T.	100% Surface*	Visual	N.S.
Final Surface	M.T. or P.T.	100% Surface	M.T. or P.T.	100% Surface	Visual	N.S.

*Examination Notations
R.T. - Radiography
U.T. - Ultrasonic Examination
M.T. - Magnetic Particle Examination
P.T. - Liquid Penetrant Examination
N.A. - Not Available; performance of this examination could not be substantiated from review of documentation available at NRC.
N.S. - Not Specified

Material Surveillance Program

The Yankee Rowe reactor pressure-vessel material surveillance program consists of two elongated specimen capsules, designated as "vessel-wall capsules," located between the thermal shield and pressure-vessel wall, and eight additional and physically similar capsules, designated "accelerated capsules," located inside the thermal shield and adjacent to the shroud surrounding the fuel core (Ref. 5). The surveillance capsules were installed in the vessel during the loading of the second core. Each surveillance capsule contains Charpy V-notch and tensile specimens from an A302-B steel fabrication test plate made from the runout of one plate for the upper vessel shell course. Charpy V-notch specimens from a 6-inch A302-B plate of a reference heat were also included in each capsule. Specimens were machined from 1/4 T material and were oriented in the longitudinal direction. No weld or HAZ specimens were included in the Yankee Rowe program. The program is based on the requirements of ASTM E 185 and also generally conforms to the requirements of Appendix H to 10 CFR Part 50. The main areas where this program does not conform to Appendix H requirements are:

1. Capsules contained only 7 Charpy specimens instead of the presently required 12.

2. The program did not contain any weld or HAZ material specimens.

3. Charpy specimens were oriented in the strong direction.

4. The program has been terminated.

Since data from capsules removed from the vessel and tested provide sufficient data to obtain transition temperature shifts and upper-shelf energy at fluences the vessel materials will be exposed to throughout service life, the low number of Charpy specimens does not detract from the program. Thus, the program has generated sufficient data to monitor the effect of radiation on vessel plate material throughout service life. Since the program did not contain samples from weld metal, the irradiated properties of the vessel weld metal will be conservatively estimated from Regulatory Guide 1.99, Revision 1.

The Yankee Rowe material surveillance program was terminated in 1965 due to structural failure of the capsule holding fixtures (Ref. 6). All capsules have been removed from the vessel. Five capsules have been tested and the remaining capsules are in storage. Because of the quality and quantity of data generated, the termination of the program is not unacceptable. From the data already obtained from the tested capsules along with the use of Regulatory Guide 1.99, the NRC staff feels that there is an adequate basis to predict the irradiated properties of the Yankee Rowe reactor vessel throughout its service life.

At the end of the second core cycle four accelerated capsules were removed from the vessel and tested (Ref. 5). At the end of the fourth core cycle one wall capsule was removed and tested (Ref. 7). All tests were conducted at the Naval Research Laboratory. The materials in these capsules were exposed to neutron fluences of 2.2×10^{18}, 5.0×10^{19}, 7.0×10^{19} and 9.0×10^{19} n/cm^2. Based on the results of these tests, we calculate that the plate material at end-of-life will have an RT_{NDT} of, 280°F and an upper shelf energy (strong direction) of 52 ft-lbs at the 1/4 T location. NRL also conducted experiments on these samples to determine the effects of post-irradiation annealing on the mechanical properties. Annealing temperatures studied were 640°F, 750°F and 850°F.

Pressure-Temperature Operating Limits

In a letter dated March 15, 1975, Yankee Atomic Electric Company submitted a proposed amendment to the Yankee Rowe Technical Specifications regarding pressure-temperature operating limits (Ref. 8). Additional information on the methods used to calculate these proposed limits was supplied in letter dated June 11, 1976 (Ref. 9). Data from the material surveillance program was used to obtain irradiated values of RT_{NDT} for plate materials. Regulatory Guide 1.99 was used to estimate RT_{NDT} for weld materials. The vessel weld metal was determined to be the limiting material. Operating limit curves for inservice hydrostatic tests, heatup and cooldown, and critical core operation were calculated for the fluence estimated at the end-of-core cycle 12. The proposed Technical Specifications require that, at each refueling outage following core cycle 12, these operating limits be updated to account for further radiation damage (increase in RT_{NDT}). To predict this radiation damage, a curve is included in the specifications that plots changes in RT_{NDT} as a function of operating time. The staff approved these operating limits and incorporated them into the Technical Specifications as License Amendment No. 27 (Ref. 10). These limits were reviewed again as part of this present review and were determined to be in accordance with Appendix G, 10 CFR Part 50.

Inservice Inspection Program

An inservice inspection program in accordance with ASME Code Section XI was initiated in 1970. A preservice examination on the Yankee Rowe primary coolant system was conducted by Southwest Research Institute in accordance with the rules of the 1965 Edition of Section III. The data obtained from this inspection will be used as a baseline to which inservice examination results will be compared. Inservice inspections have been performed on the reactor vessel in 1970, 1972, 1974 and 1977 (Refs. 11, 12). These examinations generally conformed to the requirements of the 1971 Edition of Section XI. There were many welds that could not be examined because of lack of accessibility, such as Category B-A and B-B vessel welds, Category B-D nozzle to vessel welds, and Category B-F safe end welds. Items inspected include the closure head, closure head cladding, flange to vessel weld, head to flange weld, ligaments, and closure studs and nuts. The only reportable indications were several small flaws in the closure head cladding.* No inspection has been performed, or is contemplated on Category B-A vessel welds.

Several inspections were performed in the 1960s that were not in accordance with Section XI. For example, during the 1965 refueling outage, all fuel was unloaded and the core barrel and core support were removed to permit a general inspection of the vessel interior. This examination revealed that one material surveillance capsule holding fixture had broken loose from the vessel. The capsule itself was also broken and surveillance specimens were released. This debris settled in the lower head region of the vessel. The debris caused several small worn areas in the northwest quadrant of the lower head (Ref. 13). Although there were numerous scratches and gouges in the cladding, the carbon steel was exposed only in two small adjacent areas. The total area of carbon steel exposed was about 2 square inches. The defect penetrated into the base metal about 1/8 inch. The defect in the area of penetration was smooth, i.e., no sharp cracks or scratches were detected in this area.

*Components of the reactor vessel were examined during the 1978 refueling outage (October to December). Particular attention was paid to the known clad defect. The defect was examined and photographed, and no change in the size of the defect was found. (Reference YAEC letter, R. Groce, to NRC dated March 15, 1979.)

It thus appears that the defect grew by a wear type of mechanism. In the flaw region, the minimum original thickness of the base material was more than 0.3 inch greater than required by ASME Code, Section III. Thus, there is still a safety margin on thickness requirements. Casts of the worn areas were made from a silicone rubber compound. These casts were used in subsequent inspections to determine if any flaw growth was occurring. So far, inspections have revealed no flaw growth (Ref. 14). Westinghouse evaluated these flaws and concluded that they do not present a safety or operational problem (Ref. 15). The staff currently agrees that these flaws present no safety hazard. However, we do recommend that the cladding in the area of these flaws be inspected for any signs of further deterioration during each inspection interval.

By letter dated December 19, 1977, Yankee Atomic Electric Company submitted a revised inservice inspection program for Yankee Rowe in accordance with the requirements of 10 CFR 50.55a(g) (Ref. 16). This revised program is for the second 40-month inspection period of the second 10-year inspection interval (March 1, 1978 to June 30, 1981). The proposed program is based on the 1974 Edition of ASME Code, Section XI and Addenda through Summer 1975. Relief from the Code examination requirements was requested for the following welds: Category B-A welds in the beltline region, some Category B-B vessel welds, some Category B-D nozzle to vessel welds, some Category B-F nozzle to safe end welds, Category B-H vessel supports, and Category B-I closure head cladding. The Category B-A and B-B welds are inaccessible due to the neutron shield tank outside the vessel and a nonremovable thermal shield inside the vessel. For the other welds that relief was requested, the licensee stated that examinations would be performed on a best effort basis. This proposed inspection program is currently being reviewed by the staff. From this SEP review, it is recommended that acoustic emission be used to examine the vessel welds that cannot be examined by ultrasonic methods.

Generic Safety Items

The generic safety items applicable to Yankee Rowe are low upper-shelf toughness and overpressurization protection. From our review of the vessel materials it is concluded that the upper-shelf energy of plate materials in the vessel beltline region is currently below 50 ft-lbs. Appendix G, 10 CFR Part 50, requires that vessel materials have an upper-shelf energy of at least 50 ft-lbs for material in the transverse (weak) orientation throughout service life. Since the vessel materials in the beltline region are subjected to high fluences, the upper-shelf energy values are expected to further degrade with operating time. Therefore, the vessel integrity, as far as low upper-shelf energy applies, will be analyzed for conditions at end-of-life. The maximum fluence on the vessel wall at the 1/4 T location is calculated to be 1.5×10^{19} n/cm^2 at the end-of-life. Based on the results of the Yankee Rowe material surveillance program, the upper-shelf energy of plate material at this fluence will be about 52 ft-lbs for material oriented in the longitudinal direction. Using the procedures outlined in NRC Standard Review Plan, Section 5.3.2, this corresponds to a value of about 35 ft-lbs for material oriented in the transverse direction.

Several abnormalities were found in this review that should be considered in the evaluation. First, the irradiated value of upper-shelf energy is based primarily on data from accelerated capsules removed from the vessel early in life. The results from accelerated capsules may give lower energy values than would be obtained on materials irradiated at slower rates. This is due to possible annealing effects that may increase toughness properties with increased exposure time. Second, during the early operation of Yankee Rowe, the operating temperature was generally below the normal temperature - often falling below 500°F (Ref. 7). Research programs have shown that irradiation at temperatures from 450°F to 500°F produces more damage (a greater decrease in upper-shelf energy) than radiation at normal operating temperatures of about 550°F (Ref. 17). Thus, it is concluded that the amount of radiation damage on the Yankee Rowe surveillance specimens may have been augmented by these abnormalities.

Test results from research programs have also shown that the decrease in upper-shelf energy from irradiation is less for material oriented in the transverse direction than for material oriented in the longitudinal direction. This is especially true at relatively high fluence levels and when the upper-shelf energy falls below 50 ft-lbs (Ref. 18). This behavior may be due to the fact that there is a low-shelf value that the upper shelf energy will not go below regardless of the amount of radiation.

Because of the above abnormalities and test data, an attempt was made to obtain the irradiated upper-shelf energy of Yankee Rowe vessel plate material from the test results on the correlation material contained in the Yankee Rowe material surveillance program. This correlation material is SA-302, Grade B, steel. Its chemical composition is almost identical to that of the Yankee Rowe vessel plate material. Also, its mechanical properties, both irradiated and unirradiated, are very similar to the vessel materials. These properties are compared in Table B-2. The advantage to using the correlation material is that this material was tested not only in the Yankee Rowe vessel but also in a test reactor at the Naval Research Laboratory (NRL). In the NRL test program, material was irradiated and tested in both the longitudinal and transverse orientation directions. The NRL test results show that the upper-shelf energy (transverse orientation) of this correlation material drops from about 50 ft-lbs (unirradiated) to about 42 ft-lbs at a fluence of 1.5×10^{19} n/cm^2 (Ref. 19). The above data were obtained for irradiation at 550°F. Since this value of upper shelf energy is still below the 50 ft-lbs required by Appendix G, an evaluation of the minimum acceptable energy level for the Yankee Rowe vessel materials was made. The minimum acceptable energy for Yankee Rowe may be less than 50 ft-lbs because of the low stresses in the vessel beltline region.

This evaluation basically consisted of calculating the fracture toughness, K_{IC}, from the Charpy upper-shelf energy and performing an Appendix G, ASME Code Section III, analysis to obtain the minimum required fracture toughness (Ref. 20). The Rolfe-Novak equation was used to calculate the fracture toughness from the Charpy energy (Ref. 21). This equation is an empirical expression based on test results on many heats and types of steels. The Rolfe-Novak equation can be written:

$$K_{IC} = \left[5\, S_y \left(C_v - \frac{S_y}{20}\right)\right]^{\frac{1}{2}}$$

where: S_y is the yield strength in ksi

C_v is the Charpy impact energy in ft-lbs

For a yield strength of 90 ksi, the following results are calculated:

TABLE B-2

SUMMARY OF TEST RESULTS ON THE YANKEE ROWE MATERIAL SURVEILLANCE SPECIMENS (LONGITUDINAL ORIENTATION)

Irradiation Capsule	Material	Neutron Fluence	Upper Shelf Energy (ft-lb)	Change in RT_{NDT} (°F)
Preirradiation Condition	Yankee	0.0	76	--
	ASTM	0.0	71-87	--
Vessel Wall	Yankee	2.2×10^{18}	62	110
	ASTM	2.2×10^{18}	62	85
Accelerated Capsule 8	Yankee	5×10^{19}	46	320
	ASTM	5×10^{19}	46	225
Accelerated Capsule 6	Yankee	7×10^{19}	45	360
	ASTM	7×10^{19}	44	260
Accelerated Capsule 2	Yankee	9×10^{19}	45	380
	ASTM	9×10^{19}	42	310

C_v	K_{IC}
50	143
45	135
40	126
35	117

The Appendix G analysis was made using the vessel design pressure and assuming a heatup/cooldown rate of 100°F per hour. This calculation showed that a value of 113 ksi$\sqrt{in}$ is required for K_{IC}. This value is low compared to that required for more modern vessels because of the low stresses in the Yankee Rowe vessel. From the end-of-life Charpy upper-shelf energy (42 ft-lbs) and the Rolfe-Novak correlation, it is estimated that the fracture toughness of vessel plate material at end-of-life will be approximately 130 ksi$\sqrt{in}$. Since the fracture toughness of the Yankee Rowe vessel plate material at end-of-life is well above that required by an Appendix G analysis, it is concluded that the plate material has acceptable fracture toughness properties.

Appendix G, 10 CFR Part 50, requires that all vessel materials, including welds, have an upper-shelf energy of at least 50 ft-lbs. Since the Yankee Rowe surveillance program contained no weld metal specimens, the properties of weld metal (irradiated) must be estimated from test results on unirradiated weld material. Charpy energy was obtained on vessel weld metal at one temperature, 10°F. Values varied from 36 to 41 ft-lbs. Based on test data from other surveillance programs having weld metal samples with similar unirradiated Charpy energy, it is concluded that the Yankee Rowe vessel beltline weld metal will not fall below 45 ft-lbs at a fluence of 1.5×10^{19} n/cm^2. By the same analysis used for the vessel plate material, it is concluded that this value of upper shelf energy is acceptable. Therefore, it is concluded that the Yankee Rowe reactor vessel materials will have acceptable toughness properties throughout service life.

Although no low temperature overpressurization incidents have been experienced on Yankee Rowe (Ref. 22), NRC requested that Yankee Atomic Electric Company take steps, including design modifications, necessary to preclude exceeding the limits of Appendix G, 10 CFR Part 50, during inadvertent pressure transients (Ref. 23). By letters dated December 1, 1976, May 27, 1977, and April 28, 1978, the licensee submitted to NRC plant-specific analyses in support of their proposed reactor vessel low-temperature overpressurization protection system (Ref. 24). Low-temperature overpressure protection is provided by the pressurizer solenoid operated relief valve (SORV), the pressurizer air, steam or nitrogen bubble, and the two SCS safety valves (SVS). The licensee has modified the actuation circuitry of the SORV to provide a low pressure setpoint of 500 psig. Whenever the RCS temperature is below 324°F, the SORV setpoint is manually switched to the low value (500 psig), and whenever the RSC temperature is below about 300°F, the SCS is placed in service with the accompanying two SCS SVS. A pressure transient caused by mass addition (HPSIP, LPSIP or charging pump) or heat addition (decay heat, pressurizer heaters or starting a reactor coolant pump with a hot steam generator) is terminated below the Appendix G limits by automatic operation of these valves or filling the pressurizer steam or air space. However, the mass addition from the inadvertent operation of a train of ECCS equipment (one HPSIP and one LPSIP) when the RCS temperature is between 300°F and 324°F, and the mass addition from a single LPSIP when the RCS temperature is below 200°F, results in the RCS pressure exceeding allowable limits. The licensee has proposed an equipment modification to make these events less likely (Ref. 25). The staff has completed its review of the Yankee Rowe OPS and has found it acceptable. The Technical Specifications have been revised to include testing and administrative procedures recommended by the staff. These specifications, including the related

Safety Evaluations, were incorporated into the Technical Specifications by Amendment 59 dated September 14, 1979.

Conclusions and Recommendations

The Yankee Rowe reactor vessel was designed to ASME Code Section VIII. However, the requirements of this Code were supplemented by additional quality control measures that are essentially in accordance with those required by ASME Code Section III. The design and quality control measures utilized in the fabrication of this vessel provide assurance that the initial integrity of the vessel is acceptable. This conclusion is borne out by the fact that the vessel has operated for approximately 13 EFPY without a major malfunction or flaw indication. The primary stresses in the vessel beltline region are low; about 70% of those permitted by Section III. However, in this review it was determined that the upper-shelf energy of plate materials in the vessel beltline region is below that required by Appendix G, 10 CFR Part 50. From the results of the Yankee Rowe material surveillance program and the results obtained on similar materials in research programs, it is estimated that the vessel plate material will have a Charpy upper-shelf energy of about 42 ft-lbs at end-of-life. A value of 50 ft-lbs is required by Appendix G, 10 CFR Part 50. Because of the low stresses in the vessel beltline region, it is concluded that a value of 42 ft-lbs for upper-shelf energy is acceptable. This conclusion is based on present-day technology and should be reviewed in the early 1980s as recommended below. The Yankee Rowe surveillance program was terminated in 1965 due to a structural failure. Prior to termination, five capsules were removed from the vessel and tested. Some of these capsules were subjected to fluences higher than those expected on the vessel wall ID at end-of-life. Therefore, the program is considered to have accomplished its purpose. Since the surveillance program did not contain any weld metal samples, the staff will use Regulatory Guide 1.99 to predict the irradiated properties of the vessel weld materials. Inservice inspections have been performed on components of the reactor vessel, except where limited by lack of accessibility, in accordance with ASME Code Section XI rules since 1970. We will require that future inservice examinations be performed on vessel components in accordance with Section XI rules to the maximum extent permitted by the vessel design. The reactor vessel is currently operating with pressure-temperature operating limits that are in accordance with Appendix G, 10 CFR Part 50. The staff will continue to review and update these operating limits to account for further radiation damage on the vessel beltline materials. The amount of radiation damage will be predicted from the results of tests on the Yankee Rowe surveillance specimens and the damage predictions obtained from Regulatory Guide 1.99. The combination of inservice inspections, conservative pressure-temperature operating limits, low vessel stresses, and the use of materials having adequate fracture toughness provide assurance that the vessel integrity will be maintained at acceptable levels throughout service life. The generic safety items applicable to Yankee Rowe (low upper-shelf energy and overpressurization protection) have been successfully resolved. However, since the conclusions regarding low upper-shelf energy are based on present-day technology and involve the use of empirical equations, this item should be rereviewed in the early 1980s.

The following recommendations are made:

1. Samples from several vessel welds made by the same technique as the vessel beltline welds should be taken and a chemical analysis performed on them. If possible, these samples should be taken from welds made from the same batch of weld wire and flux as the vessel beltline welds. Particular attention should be devoted to getting the copper and phosphorus content of

these samples. This data will allow the staff to more accurately predict radiation damage from Regulatory Guide 1.99.

2. Since many of the vessel welds cannot be examined in accordance with Section XI rules, it is recommended that the use of acoustic emission be considered as a means of verifying the integrity of such welds. This is considered especially important for this vessel since the upper-shelf energy of some of the vessel materials are below the requirements of Appendix G, 10 CFR Part 50.

3. The adequacy of an upper-shelf energy value of 42 ft-lbs is based on present-day technology. This item should be rereviewed upon the completion of NRC Category A Technical Activity A-11 and fracture toughness tests in the HSST program. The results of these programs should enable the staff to more accurately predict values of fracture toughness, K_{IC}, from Charpy test data. Following completion of Activity A-11, the licensee should submit a report to NRC covering the items required by Appendix G, 10 CFR Part 50, for vessels having materials under 50 ft-lbs.

References (Appendix B)

1. Yankee Nuclear Power Station FSAR, Vol. I, Section 5.**
2. Letter, Yankee Atomic Electric Company (D. Vandenberg) to NRC dated March 10, 1975.**
3. Westinghouse Specification YTR Project, Reactor Vessel, May 15, 1957.
4. Letter, Yankee Atomic Electric Company to NRC dated October 19, 1977.**
5. C. Z. Serpan, et al., NRL Report 6179, November 24, 1964.*
6. Letter, Yankee Atomic Electric Company (R. Coe) to AEC dated November 23, 1965.**
7. C. Serpan and J. Hawthorne, NRL Report 6616, September 29, 1967.*
8. Letter, YAEC (J. French) to NRC dated March 15, 1976.**
9. Letter, YAEC (J. French) to NRC dated June 11, 1976.**
10. Letter, NRC (A. Schwencer) to YAEC dated July 14, 1976.**
11. Southwest Research Institute Reports 17-2935 dated February 23, 1971, and 17-3331 dated June 1972.**

*Available at the National Technical Information Service, Springfield, VA 22161.
**Available in NRC PDR for inspection and copying for a fee.

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

November 5, 1979

Docket No. 50-29

Mr. Robert H. Groce
Licensing Engineer
Yankee Atomic Electric Company
20 Turnpike Road
Westboro, Massachusetts 01581

Dear Mr. Groce:

RE: COMPLETION OF SEP TOPIC V-10.A - YANKEE ROWE

A draft of the subject topic assessment was sent to you by letter dated March 21, 1979. We have been informed by representatives of your staff by telephone on October 23, 1979, that the facts upon which the NRC staff based its evaluation are correct. Accordingly, our review of SEP Topic V-10.A is complete and this evaluation will be a basic input to the integrated safety assessment for your facility.

The subject assessment compares your facility design with the criteria currently used by the staff in licensing new facilities. This assessment may need to be re-examined if you modify your facility of if the criteria are changed before we complete our integrated assessment.

Sincerely,

Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

cc: See next page

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

March [illegible], 1979

Docket No. 50-29

Mr. Robert H. Groce
Licensing Engineer
Yankee Atomic Electric Company
20 Turnpike Road
Westboro, Massachusetts 01581

Dear Mr. Groce:

Enclosed is a copy of our revised safety assessment of Topic V-10.A, Residual Heat Removal System Heat Exchanger Tube Failures. This revision includes consideration of the comments received on the initial assessment issued by our letter dated January 17, 1979. This revision completes our assessment of Topic V-10.A which will be used as input to the integrated review of the Yankee Rowe Plant.

It should be noted, however, that we have a difference of opinion on Comment 5 of your February 13, 1979 letter which provided comments on the initial assessment of this topic. This matter will be resolved in the context of the integrated plant review.

If there are any errors in the facts of this revised assessment, please supply corrected information within 30 days of the date you receive this letter. If no response is received within that time, we will assume that you have no further comments or corrections.

Sincerely,

Thomas V. Wambach
for Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Enclosure:
Revised Assessment for
Topic V-10.A

cc w/enclosure:
See next page

7904100600

SYSTEMATIC EVALUATION PROGRAM

PLANT SYSTEMS MATERIALS

YANKEE ROWE

Topic V-10.A Residual Heat Removal System Heat Exchanger Tube Failures

The safety objective of this review is to assure that impurities from the cooling water system are not introduced into the primary coolant in the event of shutdown cooling system heat exchanger tube failure. This was expanded to assure that adequate monitoring exists to assure no leakage of radioactive material in the other direction - into the service water and thus to the environment.

Information for this assessment was gathered from plant personnel during the safe shutdown review site visit and from related telephone conversations. Information was also taken from Yankee Rowe system drawings, the Yankee Rowe Technical Specifications, and the Yankee Rowe Final Safety Analysis Report.

The bases for the review of these cooling systems on today's plants include: (1) the NRC's Standard Review Plan (SRP) 9.2.1, which requires that the service water system include the capability for detection and control of radioactive leakage into and out of the system and prevention of accidental releases to the environment; (2) SRP 9.2.2, which requires that auxiliary cooling water systems (such as the shutdown cooling system) include provisions for detection, collection and control of system leakage and means to detect leakage of activity from one system to another and preclude its release to the environment; and (3) SRP 5.2.3, which discusses compatibility of materials with reactor coolant and required monitoring and [illegible] of the primary coolant system. These Standard Review Plans were [illegible] the comparison of Yankee Rowe and [illegible] licensed plants.

[illegible] operating requirements which must be met [illegible] [illegible] viable means of [illegible] the stated goals.

The Shutdown Cooling System (SCS) at Yankee Rowe consists of a single heat exchanger, a pump, and associated valves and piping. However, there is another system, containing the low pressure surge tank cooling pump and heat exchanger, which is connected in parallel to the SCS and can serve as its backup. Maximum system design conditions for both systems are set at 405 psig and 370°F, according to page 6.5-2 of the Yankee Rowe Final Safety Analysis Report. However, according to a Yankee Rowe plant procedure, the SCS shall not be started up until the main coolant system has been cooled down to less than 330°F and depressurized to less than 300 psig. There is, therefore, margin between the design limits and the operating limits.

Pressure of the primary coolant in the SCS heat exchanger, after the primary system is depressurized and the reactor head removed, varies between 88 psig and 102 psig. The component cooling water (CCW) system at this heat exchanger operates at a pressure between 110 psig and 115 psig and has a relief valve setpoint of 125 psig (CCW also cools the low pressure surge tank cooler). Thus, if leakage were to occur at SCS system startup, differential pressures would be such that radioactive material from the primary coolant system would flow to the CCW system. Conversely, when the primary coolant system is depressurized, any leakage would be from the CCW system into the primary coolant system.

In addition to the margins noted above between the design and operating pressures and temperatures, the SCS and low pressure surge tank heat [illegible] tested as part of the Yankee Rowe [illegible] [illegible], adding additional assurance that the [illegible] of [illegible] high and low level alarms to alert the operators to any leakage either into or out of the system, and the CCW system has a radiation monitor on the suction of the CCW pumps. This monitor will detect radioactive leakage from the SCS heat exchanger (or any other component cooled by the CCW system) and alert the operator to such leakage. Plant procedures also require a weekly sample for CCW radioactivity.

As further assurance of protection against undetected leakage into the primary coolant system, Yankee Rowe Technical Specification 3.4.6 requires sampling the primary coolant, when shutdown and operating, at least once every 72 hours for chloride and fluoride. Oxygen sampling is added to the chloride and fluoride samples when the plant is operating at average coolant temperatures greater than 250°F.

The two CCW system heat exchangers are cooled by the service water system, which forms the ultimate heat sink. Although it would take simultaneous [illegible] one (or both) of the CCW heat exchangers and the [illegible] of [illegible] primary system components to result in a discharge of [illegible] material, SRP 9.2.1 requires that the service water system incorporate a monitor and alarm to alert plant operators to leakage of radioactive material to the environment. There is no such monitor at Yankee Rowe. In addition, there are no technical specification requirements for the operability and surveillance of the CCW system monitor. Thus, although we are satisfied that Yankee Rowe meets the intent of [illegible] criteria regarding protection of the primary system, we believe that action should be taken to provide protection of the environment. [illegible]

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

February 1, 1982

Docket No. 50-29
LS05-02-008

Mr. James A. Kay
Senior Engineer - Licensing
Yankee Atomic Electric Company
1671 Worcester Road
Framingham, Massachusetts 01701

Dear Mr. Kay:

SUBJECT: YANKEE ROWE - SEP TOPICS V-10.B RHR SYSTEM RELIABILITY, V-11.B RHR INTERLOCK REQUIREMENTS AND VII-3 SYSTEMS REQUIRED FOR SAFE SHUTDOWN (SAFE SHUTDOWN SYSTEMS REPORT)

Enclosed is the revised evaluation of Safe Shutdown Systems incorporating comments from your letter dated November 24, 1981. Changes from the previous issue are marked by a line in the margin.

We now consider the safe shutdown systems topic evaluation to be complete. This evaluation will be a basic input to the integrated safety assessment for your facility. This assessment may be revised in the future if your facility design is changed or if NRC criteria relating to this subject are modified before the integrated assessment is completed.

Also enclosed are the staff conclusions concerning items for consideration in the integrated assessment.

Sincerely,

Dennis M. Crutchfield

Dennis M. Crutchfield, Chief
Operating Reactors Branch No. 5
Division of Licensing

Enclosures:
As stated

cc w/enclosures:
See next page

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Enclosure 1

SEP Review
of
Safe Shutdown Systems
for the
Yankee Rowe Nuclear Power Plant

(Revision 1)

January 1982

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Franklin Research Center
A Division of The Franklin Institute

CONTENTS

1. INTRODUCTION

The Systematic Evaluation Program (SEP) review of the "safe shutdown" subject encompassed all or parts of the following SEP topics, which are among those identified in the November 25, 1977 NRC Office of Nuclear Reactor Regulation document entitled "Report on the Systematic Evaluation of Operating Facilities":

1. Residual Heat Removal System Reliability (Topic V-10.B)
2. Requirements for Isolation of High and Low Pressure Systems (Topic V-11.A)
3. Residual Heat Removal Interlock Requirements (Topic V-11.B)
4. Systems Required for Safe Shutdown (Topic VII-3)
5. Station Service and Cooling Water Systems (Topic IX-3)
6. Auxiliary Feedwater System (Topic X).

The review was primarily performed during an onsite visit by a team of SEP personnel. This onsite effort, which was performed from June 13 to June 16, 1978, afforded the team the opportunity to obtain current information and to examine the applicable equipment and procedures, and it also gave the Licensee (Yankee Atomic Electric Company) the opportunity to provide input into the review.

The review included specific system and equipment requirements for remaining in a hot standby condition (defined as the reactor subcritical with T_{AVE} greater than or equal to 330°F) and for proceeding to a cold shutdown condition (defined as reactor coolant temperature less than or equal to 200°F). The review for transition from reactor operation to hot standby considered the requirement for the capability to perform this operation from outside the control room. The review was augmented as necessary to assure resolution of the applicable topics, except as noted below:

Topic V-11.A (Requirements for Isolation of High and Low Pressure Systems) was examined only for application to the residual heat removal (RHR) system. Other high pressure/low pressure interfaces were not investigated in this review.

Topic VII-3 (Systems Required for Safe Shutdown) was completed except for determination of design adequacy of the system.

Topic IX-3 (Station Service and Cooling Water Systems) was only reviewed to consider redundancy and seismic and quality classification of cooling water systems that are vital to the performance of safe shutdown system components.

Topic X was reviewed only to address design adequacy for heat removal. Other aspects are considered as part of the design basis event review or under implementation of the TMI Action Plan.

The criteria applied to the safe shutdown systems and components in this review are taken from the Standard Review Plan (SRP) 5.4.7, "Residual Heat Removal (RHR) System"; Branch Technical Position RSB 5-1, Revision 1, "Design Requirements of the Residual Heat Removal System"; and Regulatory Guide 1.139, "Guidance for Residual Heat Removal." These documents represent current staff criteria and are used in the review of facilities being processed for operating licenses. This comparison of the existing systems with current licensing criteria led naturally to at least a partial consideration of design criteria that will be pertinent to SEP Topic III-1, "Classification of Structures, Components and Systems (Seismic and Quality)." This report will also be reviewed for its application to the resolution of other topics.

As noted above, the six topics were examined while possible interactions with other topics, systems, and components not directly related to safe shutdown were neglected. For example, Topics II-3.B (Flooding Potential and Protection Requirements), II-3.C (Safety-Related Water Supply), III-4.C (Internally Generated Missiles), III-5.A (Effects of Pipe Break on Structures, Systems and Components Inside Containment), III-6 (Seismic Design Considerations), III-10.A (Thermal Overload Protection for Motors of Motor-Operated Valves), III-11 (Component Integrity), III-12 (Environmental Qualification of Safety-Related Equipment), and V-1 (Compliance with Codes and Standards) could be affected by the results of the safe shutdown review or could affect the safety of the systems that were reviewed. These effects will be reviewed later. Further, this review did not cover in any significant detail either the reactor protection system or the electrical power distribution, both of which will also be reviewed later.

The staff considers that the ultimate decision concerning the safety of any of the SEP facilities is based upon the ability of the facility to withstand the SEP design basis events (DBEs). The SEP topics provide a major input to the DBE review, from the standpoint of assessing both the probability and the consequences of the event. As examples, the safe shutdown topics pertaining to the listed DBEs are provided in Table B.1 (the extent of applicability will be determined during the plant-specific review).

Completion of the safe shutdown topic review (limited in scope as noted above), as documented in the attached report, significantly contributes to an assessment of the existing safety margins.

Table B.1

TOPIC	DBE GROUP		IMPACT UPON PROBABILITY OR CONSEQUENCES OF DBE
V-10.B	VII	(Spectrum of Loss-of-Coolant Accidents)	Consequences
V-11.A	VII	(Defined above)	Probability
V-11.B	VII	(Defined above)	Probability
VII-3	All	(Defined as a generic topic)	Consequences
IX-3	III	(Steam Line Break Inside Containment) (Steam Line Break Outside Containment)	Consequences
	IV	(Loss of AC Power to Station Auxiliary) (Loss of all AC Power)	Consequences
	V	(Loss of Forced Coolant Flow) (Primary Pump Rotor Seizure) (Primary Pump Shaft Break)	Probability
	VII	(Defined above)	Consequences
X	II	(Loss of External Load) (Turbine Trip) (Loss of Condenser Vacuum) (Steam Pressure Regulator [closed]) (Loss of Feedwater Flow) (Feedwater System Pipe Break)	Consequences
	III	(Defined above)	Consequences
	IV	(Defined above)	Consequences
	V	(Defined above)	Consequences
	VII	(Defined above)	Consequences

Piping System Passive Failures

The NRC staff normally postulates piping system passive failures as (1) accident-initiating events in accordance with staff positions on piping failures inside and outside containment, (2) system leaks during long-term coolant recirculation following a LOCA, and (3) failures resulting from hazards such as earthquakes and tornado missiles. In this evaluation, certain piping system passive failures have been assumed beyond those normally postulated by the staff, e.g., the catastrophic failure of moderate energy systems. These assumptions were made to demonstrate safe shutdown system redundancy in the event of complete failures of these systems and to facilitate future SEP reviews of DBEs and other topics that will use the safe shutdown evaluation as a source of data for the SEP facilities. SRP 5.4.7 and BTP RSB 5-1 do not require the assumptions of piping system passive failures.

Credit for Operating Procedures

For the safe shutdown evaluation, the staff may give credit for facility operating procedures as alternate means of meeting regulatory guidelines. Those procedural requirements identified as essential for acceptance of an SEP topic on DBEs will be carried through the review process and considered in the integrated assessment of the facility. At that time, the staff will decide which procedures are so important that an administrative method must be established to ensure that, in the future, these operating procedures are not changed without appropriate consideration of their importance to the SEP topic evaluation.

2. DISCUSSION

2.1 Normal Plant Shutdown and Cooldown

A normal shutdown from full power to hot standby is accomplished with the use of operating procedure OP-2104, "Scheduled Plant Shutdown to Hot Standby." The shutdown from power is accomplished by reducing the generator load using the turbine control system and following with control rod insertion to control T_{AVE}. The load reduction is performed at a rate of 8 MWe per 5 minutes, and changes in main coolant average temperature are controlled at a rate of 2°F per 5 minutes. The reactor is borated using the charging pumps to the amount necessary to maintain the control rod bank above the low insertion limit and ensure that the axial flux difference will remain within its target band.

The first main feedwater pump and condensate pump are removed from service when the generator load has been reduced to less than 140 MWe. When the generator load has been reduced to less than 60 MWe, the second main feedwater pump and condensate pump are removed from service. The power reduction is continued using the remaining operating boiler feed pump to provide feed to the steam generators. When the load on the generator has decreased to less than 30 MWe, station service loads are transferred to the auxiliary transformers fed from the offsite power supply and condensate recirculation is established back to the condenser hot well. Manual control of the turbine bypass is taken when the generator load is reduced to less than 15 MWe. The turbine is tripped just before the generator load reaches 0 MWe. Normally, the plant can be maintained in a hot standby condition (main coolant average temperature at 514°F, 2000 psig) by using main coolant pump heat, decay heat, and discharging steam to the main steam header.

During the plant shutdown to hot standby, a control rod group remains withdrawn to a height sufficient to provide a reactivity worth of 1% for emergency shutdown capability. If for any reason a control rod group cannot be withdrawn to provide a reactivity worth of 1%, then the main coolant system is borated to 5% Δk/k shutdown margin. Prior to proceeding to hot shutdown and cold shutdown, the main coolant system is borated to the 5% Δk/k shutdown margin. The main coolant system is borated using the charging and volume control system. The charging pumps take suction from the boric acid mixing and storage tank. If any main coolant loops are isolated, they too are borated to the shutdown margin.

The plant cooldown is limited to 50°F per hour, and cooling is accomplished by continuing the bypass of steam to the main condenser. At least two main coolant loops are tied to the reactor vessel until the shutdown cooling system is in operation. Pressurizer level is now manually controlled using the charging pumps to provide makeup for contraction caused by the cooldown of main coolant system water. Pressurizer temperature and pressure are controlled to maintain the reactor vessel within nil ductility transition temperature range.

When the cooldown from hot standby is initiated, all main coolant pumps may be in operation. At 370°F, main coolant pump operation is limited to two pumps. When main coolant temperature is between 300°F and 330°F and pressure is less than 300 psig shutdown cooling is initiated. If the main coolant system is to be depressurized, then the remaining main coolant pumps are secured. Pressurizer temperature and and pressure reduction is performed by charging pump flow through the auxiliary spray line to the pressurizer spray, while simultaneously draining to the low pressure surge tank. When the main coolant system temperature reaches 200°F, the charging rate is increased to the main coolant system in order to fill the pressurizer. When pressurizer temperature reaches 200°F, the pressurizer vent is opened to depressurize the main coolant system. Shutdown cooling continues until the main coolant system reaches about 140°F, where it is maintained by the shutdown cooling system, which is cooled by the service water system. The service water system takes cold water from the river, circulates it through the component cooling system heat exchangers and returns the warmer water to the river. Thus, heat is transferred from the main coolant system to the river to accomplish cooldown and decay heat removal.

2.2 Shutdown and Cooldown with Loss of Offsite Power

Operating Procedure "Loss of A.C. Supply" defines the action to be taken following a total loss of ac power to provide emergency electrical power to vital equipment.

Following a loss of offsite power and turbine trip, the main condenser circulating water pumps cannot be powered from onsite sources. With the loss of circulating water pumps, main condenser vacuum cannot be maintained and the main condenser becomes unavailable for heat removal. With the loss of normal heat sink, the main steam safety valves will lift to vent steam to atmosphere. The operator is directed to verify closing of the steam dump valve on loss of condenser vacuum and automatic starting of the three emergency power diesel generators. In addition, the operator is directed to perform the necessary electrical switching to remove connections to the offsite power lines and to start the emergency boiler feed pump and commence feeding the steam generators. The steam supply valve to the large hogger is opened and set to maintain an inlet steam pressure of 300 psig. Electrical power is restored to the 480-V buses, and pressurizer heaters Nos. 5 through 8 are energized to restore main coolant system pressure control. The operator establishes a minimum of 200 psig overpressure on the main coolant system. A service water pump and component cooling water pump are then started to supply plant equipment cooling requirements.

Operating Procedure "Loss of Condenser Vacuum" delineates the action to be taken if a loss of condenser vacuum occurs while the plant is operating at power. One of the immediate actions is to initiate maximum feed and bleed and to increase low pressure surge tank cooling, if required. Subsequent operator action is to line up the following equipment to provide main coolant heat removal to control temperature as necessary:

a. atmospheric steam dump
b. hogger air ejections
c. steam drains to atmosphere
d. throttle line steam drains to the auxiliary boiler blowdown tank.

At this point, the plant's essential equipment is being supplied through the operation of the emergency diesel generator. Reactor decay heat is transferred to the steam generators and dissipated by lifting of the main steam safety valves and operation of various other vent paths.

Operating Procedure "Plant Cooldown from Hot Standby" delineates the steps required for a plant cooldown with or without the main condenser in service. The operator is directed to do the following when the main condenser is not available:

- o borate the main coolant system to the shutdown margin
- o initiate maximum feed with supplemental low pressure surge tank cooling
- o adjust the atmospheric steam dump valve to achieve the desired cooldown rate but not greater than 50°F/h
- o remove the emergency core cooling system from service when the main coolant system pressure is less than 1000 psig
- o initiate the shutdown cooling water system when the main coolant system pressure is less than 300 psig and temperature is between 300 and 330°F.

Cooling with the shutdown cooling system is then accomplished in the same manner as was discussed in Section 2.1.

3. CONFORMANCE WITH BRANCH TECHNICAL POSITION 5-1 FUNCTIONAL REQUIREMENTS

The functional requirements stated in Branch Technical Position (BTP) 5-1 for the safe shutdown systems are:

1. The design shall be such that the reactor can be taken from normal operating conditions to cold shutdown* using only safety-grade systems. These systems shall satisfy General Design Criteria 1 through 5.

2. The system(s) shall have suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation 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) the system function can be accomplished assuming a single failure.

3. The system(s) shall be capable of being operated from the control room with either only onsite or only offsite power available with an assumed single failure. In demonstrating that the system can perform its function assuming a single failure, limited operator action outside of the control room would be considered acceptable if suitably justified.

4. The system(s) shall be capable of bringing the reactor to a cold shutdown condition,* with only offsite or onsite power available, within a reasonable period of time following shutdown, assuming the most limiting single failure.

Compliance of the Yankee Rowe safe shutdown systems with these criteria is discussed below.

3.1 Background

The BTP 5-1 requirements are stated with respect to plant shutdown and cooldown with only offsite or only onsite power available. The staff evaluated the plant's ability to conduct a shutdown with only offsite power available and determined that the "only onsite power available" case is more

*Processes involved in cooldown are heat removal, depressurization, flow circulation, and reactivity control. The cold shutdown condition, as described in the Standard Technical Specifications, refers to a subcritical reactor with a reactor coolant temperature no greater than 200°F.

limiting. The plant electrical system is sufficiently versatile to allow energizing of all necessary equipment from only offsite power. Therefore, the staff concentrated its evaluation of the Yankee Rowe safe shutdown systems to shutdown following a loss of offsite power.

A "safety-grade" system is defined, in the NUREG-0138 [1] discussion of issue No. 1, as one which is designed to seismic Category I (Regulatory Guide 1.29) Quality Group C or better (Regulatory Guide 1.26) and is operated by electrical instruments and controls that meet Institute of Electrical and Electronics Engineers Criteria for Nuclear Power Plant Protection Systems (IEEE Std 279-1971). Yankee Rowe received its Full Term Operating License on June 23, 1961 prior to the issuance of Regulatory Guides 1.26 and 1.29 (as Safety Guides 26 and 29 on March 23 and June 7, 1972, respectively). Also, the proposed IEEE Std 279, dated August 30, 1968, was not used in the design of the facility. Therefore, for this evaluation, systems which should be "safety-grade" are the shutdown systems classified in Table 3.1 and those tabulated in the minimum list of safe shutdown systems that follows.

General Design Criteria (GDC) 1 through 4 [2] require that systems, structures, and components important to safety (1) be constructed to quality standards, and (2) be protected from the effects of natural phenomena (earthquakes, etc.) and other conditions (fires, pipe breaks, etc.). GDC 5 requires that systems important to safety not be shared among other nuclear power units unless such sharing does not significantly impair the performance of system safety functions. For Yankee Rowe systems and equipment, the various aspects of GDC 1 through 5, including the systems required for safe shutdown, will be evaluated elsewhere under several SEP topics.

In order to accomplish a plant shutdown and cooldown following a loss of offsite power, certain "tasks" must be performed, such as core decay heat removal, steam generator makeup, and component cooling. The staff and Licensee developed a "minimum list" of systems necessary to perform these tasks, considering a loss of offsite ac power and the most limiting single failure. Although other systems may be used to perform shutdown and cooldown functions, the following list is the minimum number of systems required to fulfill the BTP RSB 5-1 criteria:

1. steam relieving paths involving main steam, auxiliary steam, and heating steam systems
2. auxiliary feedwater system
3. water sources (demineralized water storage tank, primary makeup tank, and safety injection tank)
4. shutdown cooling system
5. component cooling system
6. service water system
7. pressure control and relief system
8. chemical and volume control system
9. control air system
10. emergency power system
11. instrumentation for shutdown and cooldown.*

The staff's evaluation of each of these systems, with respect to the BTP 5-1 functional requirements, is given in Section 3.2. The power supplies and location of major safe shutdown components are also provided.

* For a list of safe shutdown instrumentation, see Section 3.3.

TABLE 3.1 CLASSIFICATION OF SHUTDOWN SYSTEMS - YANKEE ROWE

Components/Subsystems	Quality Group R.G. 1.26	Quality Group Design	Seismic R.G.1.29	Seismic Plant Design	Remarks
Emergency Feedwater System					
Pumps	ASME III Class 3	Note 2	Category I	Note 2	
Piping from PWST/DWST to pump suction	ASME III Class 3	Note 2	Category I	Note 2	
Discharge piping	ASME III Class 3	Note 2	Category I	Note 2	
Main Steam System					
Main steam headers from steam generators up to and including the non-return valves, steam supply to the EFBP and connecting piping up to and including the first valve that is normally closed or capable of automatic closure	ASME III Class 2	ASA B31.1	Category I	Note 1	
Emergency Boiler Feed Pump	ASME III Class 3	Mfr. Std.	Category I	Note 1	
EBFP piping from discharge of pump to main feed lines including EBFP relief	ASME III Class 3	ASA B31.1	Category I	Note 1	
Main feed piping from and including valves MOV-1003 through 1006, CV-1000A, CV-1100A, CV-1200A, and CV1300A, up to valves CV-1000, 1100, 1200, and 1300	ASME III Class 3	ASA B31.1	Category I	Note 1	

TABLE 3.1 (Continued)

Components/Subsystems	Quality Group R.G. 1.26	Quality Group Plant Design	Seismic R.G. 1.29	Seismic Plant Design	Remarks
Main feed piping from and including CV-1000, 1100, 1200, and 1300 up to the steam generators and connecting piping up to and including the first valve that is normally closed or capable of automatic closure	ASME III Class 2	ASA B31.1	Category I	Note 1	
EBFP piping from suction of pump to and including the DWST and/or the PWST and connected piping up to and including the first valve that is either normally closed or capable of automatic closure	ASME III Class 3	ASA B31.1	Category I	Note 1	Refer to Technical Specification 3.7.1.3
Shutdown Cooling System					
Pump	ASME III Class 3	Mfr. Std.	Category I	Note 1	
Heat exchanger (shell side)	ASME III Class 3	ASME VIII (1956)	Category I	Note 1	Heat exchanger also constructed in accordance with the 1956 edition of Standards of the Tubular Exchanger Mfr's. Association
(tube side)	ASME III Class 2				

TABLE 3.1 (Continued)

Components/Subsystems	Quality Group: R.G. 1.26	Quality Group: Plant Design	Seismic: R.G. 1.29	Seismic: Plant Design	Remarks
SCS piping from MOV-552 through the SCS pump and heat exchanger to MOV-551 and connected piping up to the first normally closed valve or valves capable of automatic closure	ASME III Class 2	ASA B31.1 (1955) Sect. 1 and 6	Category I	Note 1	
Component Cooling Water					
Pumps (2)	ASME III Class 3	ASME VIII (1956)	Category I	Note 1	
Heat exchangers (tube side) (shell side)	ASME III Class 3	ASME VIII (1956)	Category I	Note 1	
CCW piping and connected piping up to and including containment the first valve that is either normally closed or capable of automatic closure	ASME III Class 3	ASA B31.1 (1955) Sect. 1 and 6	Category I	Note 1	Note: Piping which penetrates up to the outermost containment isolation valve should be ASME III, Class 2
CCW surge tank	ASME III Class 3	ASME VIII (1956)	Category I	Note 1	
CCW valves and fittings	ASME III Class 3	ASA B16.5 (1957)	Category I	Note 1	

TABLE 3.1 (Continued)

Components/Subsystems	Quality Group R.G. 1.26	Quality Group Plant Design	Seismic R.G. 1.29	Seismic Plant Design	Remarks
Service Water System					
Pumps (3)	ASME III Class 3	Mfr. Std.	Category I	Note 1	
SWS piping and connected piping up to and including the first valve that is either normally closed or capable of automatic closure	ASME III Class 3	ASA B31.1	Category I	Note 1	Note: Piping which penetrates containment up to the outermost containment isolation valve should be ASME III Class 2
Pressure Control					
Solenoid-operated pressurizer relief valve	ASME III Class 1	B31.1 ASME Sec I and B16.5	Category I	Note 1	
Pressurizer relief valve	ASME III Class 1	B16.5	Category I	Note 1	
Main pressurizer spray flow isolation valve	ASME III Class 1	B16.5	Category I	Note 1	
Pressurizer heaters	N/A	N/A	Category I	Note 1	
Chemical and Volume Control System					FSAR Sections 203 and 204
Pumps (3)	ASME III Class 2	ASME III (1956)	Category I	Note 1	Note: The system boundary includes connecting piping up to and including the first valve that is either normally closed or capable of automatic closure
Low pressure surge tank	ASME III Class 2	ASME VIII (1956)	Category I	Note 1	

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TABLE 3.1 (Continued)

Components/Subsystems	Quality Group R.G. 1.26	Quality Group Plant Design	Seismic R.G. 1.29	Seismic Plant Design	Remarks
Piping and valves from pump discharge to CH-V-617 and CH-V-611	ASME III Class 2	ASA 31.1 (1955) Sect. 1 and 6	Category I	Note 1	
Piping from and including CH-V-617 and CH-V-611 to the main coolant system	ASME III Class 1	ASA B31.1 (1955) Sect. 1 and 6	Category I	Note 1	
Letdown piping from the main coolant system to and including the orifice isolation valves	ASME III Class 1	ASA B31.1 (1955) Sect. 1 and 6	Category I	Note 1	
Feed and bleed heat exchangers	ASME III Class 1	ASME VIII (1956)	Category I	Note 1	
Letdown piping from orifice isolation valves to pump suctions via LPST	ASME III Class 2	ASA 31.1 (1955) Sect. 1 and 6	Category I	Note 1	
Piping from safety injection tank to charging pumps via MOV-540 up to and including valve CS-V-630	ASME III Class 2	ASA 31.1 (1955) Sect. 1 and 6	Category I	Note 1	Note: Boration is performed by CVCS pumps using borated water from SI tank or BAMT

TABLE 3.1 (Continued)

Components/Subsystems	Quality Group R.G. 1.26	Quality Group Plant Design	Seismic R.G. 1.29	Seismic Plant Design	Remarks
Piping from CS-V-630 to boric acid mix tank MOV-529	ASME III Class 3	ASA 31.1 (1955) Sect. 1 and 6	Category I	Note 1	BAMT was fabricated to ASME VIII
CVCS valves and fittings	As above for piping	ASA B16.5 (1953)	Category I	Note 1	
Emergency Power System					
Diesel generators (3)	NA	–	Category I	Note 1	
DC power system			Category I	Note 1	
Distribution lines, switchgear, control boards, motor control centers			Category I	Note 1	
Diesel generator fuel oil system	ASME III Class 3	ASA B31.1	Category I	Note 1	
Control Air					
Air compressors and associated equipment	Quality Group D	Note 1	Non-Seismic Category	Note 1	Air systems required to perform safety functions (e.g., accumulator and piping to a safety-related valve) are seismic Category I.
Service Air					
Air compressors	Quality Group D	Note 1	Non-Seismic Category	Note 1	

Note 1: Plant design information is not known.

Note 2: Newly installed system

3.2 Functional Requirement

STEAM RELIEVING PATHS

Task: Removal of core decay heat and main coolant system sensible heat by venting steam from the main steam system directly to atmosphere.

Discussion

Immediately after the loss of offsite ac power, turbine trip, and reactor scram, the main steam safety valves automatically actuate to control steam system pressure and main coolant system temperature. However, the main steam safety valves are not normally used at pressures below their lift pressure, although a lifting lever is furnished on each valve for manual operation. The cooldown of the Yankee Rowe main coolant system following a loss of offsite ac power would be accomplished using the atmospheric dump valve (ADV) and several other steam flow paths. The following paragraphs will briefly describe each vent path.

The air-controlled ADV* vents steam from any of the four 14-inch (outside diameter [OD]) main steam lines between the vapor containment and the turbine building. The ADV vents steam from a steam header pressurized by manually operated 1-inch isolation valves from any or all of the four main steam lines. The piping system is arranged such that the ADV can remove energy from any or all steam generators.

The Licensee calculated the capacity of the ADV based on 775 psig saturated steam and critical flow. The mass flow rate out the ADV is about 29,500 lbm/h or about 9100 Btu/sec (based on 1199 Btu/lbm hg and 88 Btu/lbm h_f**).

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*Air is supplied to the ADV diaphragm from either the instrument air system or from a newly installed (dedicated) N_2 bottle.

**The staff notes that most of the feedwater inside the steam generator is at T_{sat}, and therefore an h_f of about 500 Btu/lbm would have been more appropriate initially. However, even use of this enthalpy does not account for the difference between the calculated and measured energy removal rate of the ADV.

Actual measurements, however, indicated that the capacity of the ADV is considerably less. The tests were conducted by calculating the cooldown rate with the ADV fully open, knowing the heatup rate with the ADV shut. The tests showed the ADV able to remove only about 3100 Btu/sec.

Based on actual measurements of these tests, the Licensee considered it necessary to provide another flow path for energy removal. The flow path created allows steam to pass from the pressurized steam header, as with the ADV, through two manually operated valves (AS-V-720 and AS-V-721) and then to atmosphere through a 1-inch pipe. Both valves are normally closed. The Licensee's calculated energy removal rate using this path is about 9661 Btu/sec. This value was calculated by assuming a steam pressure at 775 psig, critical flow from a 1-inch pipe, and a feedwater inlet of 120°F.

The plant operating procedure for a "Loss of A.C. Supply," OP 3251 (discussed in Section 2), directs the operator to start the emergency boiler feed pump (EBFP) and line up steam to the large "hogger." The large and small hoggers at Yankee Rowe are single-stage venturi-type air ejectors which draw from the condenser and exhaust directly to atmosphere (unlike the main air ejectors which exhaust to the shell side of a condenser cooled by condensate). The hoggers are normally used for removing large amounts of air and gasses from the condenser during startups. (During startups, steam for the hoggers comes from the main steam lines.)

The hoggers can be used to remove energy from the steam generators by bleeding steam from the main stem system to the hoggers via the auxiliary steam system. In this mode of operation, the suction valves to the main condenser are shut. Main steam is throttled at the nozzle inlets to maintain about 100 psig on the large hogger and 60 psig on the small hogger. Since there is no automatic pressure regulator, as main steam pressure drops during the reactor coolant system (RCS) cooldown, the throttle valve setting must be manually adjusted.

The EBFP is utilized to provide feedwater to the steam generators during the loss of offsite ac power and is described in the following section. The

EBFP can be used to remove energy from the steam generators, since the EBFP uses steam from the main steam system via the auxiliary steam and heating steam systems. Steam pressure is automatically maintained at 100 psi by pressure control valve PCV-305.

The earliest time following a loss of offsite ac power and reactor shutdown when each component's energy release rate equals the core decay heat generation rate is provided below.

Component	Energy Removal Rate (Btu/h)	Time (hours)
ADV	13.5×10^6	~10
1-in vent	42.3×10^5	~0.2
Large hogger	5.50×10^6	>36
Small hogger	1.12×10^6	>36
EBFP	2.27×10^6	>36

Staff scoping calculations assumed that steam generator pressure remained at 935 psig (lowest main steam safety valve setpoint) until the component energy removal rate equals the decay heat generation rate. The time calculated represents the approximate time when (1) plant cooldown commences if the component is used and (2) intermittent main steam safety valves lifting would stop.

Redundancy

To establish the degree of redundancy provided by the various components discussed above, the staff and licensee calculated the main coolant system cooldown times using various combinations of the components. The staff's calculations are summarized below:

Component(s)	Results
ADV	494°F in 50 hours
1-in vent	370°F in 50 hours
ADV + 1-in vent	351°F in 50 hours (~ 330°F in 95 hours)
ADV + 1-in vent + EBFP	330°F in 48 hours

Using these results to establish the redundancy, it is apparent that, even if all steam vent paths are considered, the main coolant system cannot be cooled down within a reasonable period of time (defined as 36 hours in Standard Review Plan 5.4.7). A single failure within the vent paths of the ADV or 1-inch vent would extend the time required.

The staff also performed scoping calculations to determine the dependence of RCS cooldown time on the initiation time. It was found that, if the cooldown were delayed 4 hours, the time to reach 330°F would be the same as if the cooldown began immediately (as soon as possible after the scram). Since the core decay heat is less at 4 hours and the energy removal rate is the same, the cooldown rate is higher initially, but then decreases as the energy removal rate (determined mainly by steam pressure) decreases.

Although the assumptions and calculating methods varied between the Licensee and staff analysis, the Licensee's results support the staff conclusion that the steam vent paths do not have sufficient capacity or redundancy to satisfy the functional requirements of BTP RSB 5-1. In a March 26, 1981 letter [3], the Licensee proposed changes to provide automatic quick closure of the four main steam line non-return valves. This modification necessitated the installation of a new steam supply line to the steam-driven emergency feedwater pump and installation of additional steam dump capacity. During a March 27, 1981 discussion [4], the Licensee indicated that an additonal manually operated dump valve would be installed on each steam line upstream of the non-return valve. Each of these valves is to have the ability to remove approximately 60,000 lbm/h. The Licensee stated that the new dump valves are intended to be the main method of decay heat removal following the loss of offsite ac power.

The staff performed scoping calculations to assess the plant cooldown capability based on the proposed modification. A single failure was postulated to one manually operated dump valve and a decay time of 4 hours prior to commencing cooldown was assumed. Based on these assumptions, a main coolant system temperature of 330°F. was attained in approximately 3.5 hours. The results demonstrate that the proposed modifications as described would afford sufficient capacity and redundancy to satisfy the functional requirements of BTP RSB 5-1.

Based on the above discussion, the staff concluded that the steam relieving paths did not conform to Criterion 4 and that the proposed modifications (as described above) would provide sufficient capacity and redundancy to satisfy the functional requirements of BTP RSB 5-1. These modifications were installed during the summer 1981 outage.

Location and Operation

The staff evaluated the equipment discussed above with respect to its location and operability during a loss of offsite ac power. Table 3.2-1 shows the equipment's location, the points from which it may be operated, and its power supply. The design of the electrical instrumentation and controls for this and other safe shutdown equipment will be evaluated in the electrical portion of the staff's review of Topic VII-3.

AUXILIARY FEEDWATER SYSTEM

Task: Provide steam generator makeup inventory whenever the main coolant system temperature is greater than 330°F and the feedwater system is inoperable.

Discussion

While the main coolant system temperature is above 330°F, the core decay heat is removed by bleeding steam from the steam generators using the various flowpaths discussed in previous sections. The condensate and feed pumps are powered from the 2400-V bus which is normally supplied from offsite power. Following a loss of offsite ac power these pumps will not be available.

Table 3.2-1

EQUIPMENT	LOCATION	CONTROL POINTS	ELECTRIC
Large and small hoggers	Turbine building, adjacent to condenser hotwell about 50 ft from feedwater regulating valves, and one flight of stairs below control room.	Local manual operation only (Open/shut steam inlet valves.	No electrical power is needed.
Atmosphere dump valve	Outside, in the vicinity of the MSSVs, accessible by catwalk about 10 ft above ground level.	Control room operation and Local operation using nitrogen bottle pressure in the lower level of the turbine building.	No electrical power is needed.
1-in vent pipe	Valves to lineup to control this path are located in the heating boiler room; the 1-in vent pipe goes to atmosphere just outside the boilder room.	Local operation only using the manual control valves.	No electrical power is needed.
EBFP	See following section.	See following section.	See following section.
New atomospheric dump valves		Manual	No electrical power needed.

The auxiliary feedwater system is designed to supply water to the steam generators for main coolant system decay heat removal when the normal feedwater system is not available. The auxiliary feedwater system is not normally used for other plant operations such as startup or shutdown. The auxiliary feedwater system is initiated by starting the emergency boiler feed pump (EBFP) locally and by opening four normally closed manual valves in parallel discharge lines to each steam generator. The valve operation is also accomplished locally.

The EBFP is a turbine-driven reciprocating pump that provides a minimum of 80 gpm directly into the four feedwater lines immediately upstream of the air-operated feedwater regulating valves (FRVs). Steam for the turbine is supplied from either main steam header via an automatic reducing valve or from auxiliary boilers. Turbine exhaust is directed to atmosphere. The EBFP is lined up to receive water from the demineralized water storage tank and can also receive water directly from the primary water storage tank. The EBFP discharges to the main feed system via a 2-inch (OD) header. The header divides into four 1.5-inch (OD) lines, each of which pressurizes one of the four normal feed lines downstream of the motor-operated isolation valves (MOVs). Each 1.5-inch (OD) line has a manual isolation valve that is opened to pressurize the four feed lines. Steam generator level control is performed using the individual FRVs* from the local station after the four MOVs are shut.

Redundancy

The auxiliary feedwater system consisting of a single EBFP and piping train is susceptible to single failures. A backup method of supplying feedwater to the steam generators in the event of failure in the auxiliary feedwater system is the charging pumps with a total capacity of approximately

*The FRVs are normally air operated but can be manually operated (during a loss of instrument air) using a handwheel.

100 gpm. Two of these pumps have variable-speed motors. The charging and volume control system (CVCS) is connected permanently by a spool piece to the feedwater system. The operation of ten valves (two drains and eight isolation valves including manual valves CH-V-692, CH-V-751, CH-V-642, CH-V-641, and CH-V-689) is required to initiate flow from this source. The CVCS is also connected to the steam generator blowdown piping. Manual operation to open valves CH-V-741, VD-V-1093, VD-V-1094, VD-V-1095, and VD-V-1096 is required to establish a feed path to the steam gnerators. Both of these paths use non-nuclear system (NNS) piping. The water supply to the charging pump is the 135,000-gallon primary water storage tank or the 30,000-gallon demineralized water storage tank.

The high pressure safety injection (HPSI) and low pressure safety injection (LPSI) pumps provide two additional methods of supplying feedwater to the steam generators. The first path is from the safety injection discharge header through normally closed motor-operated valves SI-MOV-514 and -515 and manual throttle valve SI-V-645 to the charging header. Therefore, safety injection water can be directed to the charging header and distributed to steam generators by the CVCS connections to the feedwater of blowdown piping. Discharge from the HPSI and LPSI pumps can also be directed through manual valves SI-V-700, VD-V-1093, VD-V-1094, VD-V-1095, and VD-V-1096 to establish a feed path to the steam generators through the blowdown system. The flow available from the HPSI and LPSI sources is 200 gpm per train (three trains available).

Power for the charging pumps and motor-operated valves is supplied from separate nonsafety 480-V ac buses, which are capable of being fed by the emergency 480-V ac buses by remote manual operation of circuit breakers. The HPSI and LPSI pumps are connected to the 480-V emergency buses. Following a loss of offsite ac power and a single failure in the auxiliary feedwater system, the charging pumps would not be available unless operator action occurs to initiate manual operation of circuit breakers to supply emergency power. In a December 21, 1979 letter [5], the Licensee concluded that there is not sufficient emergency diesel capacity to provide normal power to the charging pumps and simultaneously supply the existing emergency core cooling requirements. Since the charging pumps are not supplied from emergency buses, the safety injection system may be required to fulfill functions normally assigned to the charging pumps (i.e., boration and primary plant makeup). In addition, the safety injection tank functions as a source of borated water similar to a refueling water storage tank in other Westinghouse reactor facilities. If the safety injection tank is used as an alternate source of water, the dissolved boron will be concentrated in the steam generators by the release of steam. The volume of water in the safety injection tank cannot be considered as an alternate source of water for the steam generators except under very extreme conditions. Consideration of severe conditions warranting such use of the safety injection tank is outside the intent of the safe shutdown review.

Since the charging pumps cannot be considered available as a backup method to feed the steam generators following a loss of offsite ac power and the safety injection system does not have a suitable source of water for steam generator feed, the auxiliary feedwater system as currently designed does not satisfy the functional criteria of BTP RSB 5-1.

However, in Reference 5, the Licensee described proposed auxiliary feedwater system design changes to provide redundancy to the system. As proposed, the revised system consists of two 100-percent, safety-class electric pumps driven from redundant power sources. The preferred flow path is to the existing auxiliary feedwater header. An alternate flow path is proposed utilizing the containment penetrations provided by the steam generator blowdown pipes. Check valves in the blowdown lines direct auxiliary feed to the feed nozzle and prevent flow from entering the steam generator at the blowdown connection. The motor-driven pumps can take a suction on either the DWST or the PWST. The new pumps, located in the primary auxiliary building, are capable of being started either locally or from the control room.

The Licensee has indicated that the existing steam-operated EBFP will be restrained but its intended emergency function will be modified to mitigation of station blackout only. In addition, the new auxiliary feedwater pumps are capable of being powered by the existing emergency diesel generators by remote manual operation of circuit breakers.

Based on the above discussion, the staff concludes that the current* auxiliary feedwater system does not meet the functional requirements of BTP RSB 5-1 but that the recently installed modifications satisfy the functional requirements of BTP RSB 5-1, except that the components are not connected to diesel-backed buses, although the capability exists for manual connection. The reliability of the auxiliary feedwater system is being further evaluated by the staff under TMI Action Plan Item II.E.1.1.

Location and Operation

The staff evaluated the equipment discussed above with respect to its location and operability during a loss of offsite ac power. Table 3.2.2 shows the equipment location, the points from which it may be operated, and its power supply.

WATER SOURCES (DEMINERALIZED WATER STORAGE TANK, PRIMARY MAKEUP TANK, AND SAFETY INJECTION TANK)

Task: Provide a source of auxiliary feedwater, primary makeup, and borated water.

Discussion

The EBFP takes a suction from the demineralized water storage tank (DWST) via a 10-inch (OD) line which also serves as the hotwell makeup and rejection line. This line leaves the bottom of the DWST and from there branches into the following:

1. a 10-inch hotwell rejection line (i.e., flow from hotwell using condensate pumps and a level control valve)
2. a 10-inch hotwell makeup line
3. a 3-inch EBFP suction line
4. a 4-inch LPST makeup and charging pump suction line
5. a 4-inch auxiliary boiler makeup line.

*Pre core-XV configuration

Table 3.2-2

EQUIPMENT	LOCATION	CONTROL POINTS	ELECTRIC
Emergency boiler feed pump	NW corner of heating boiler room floor, which is a partitioned part of the turbine building.	Local operation only. Once at proper rpm, governor maintains speed.	No electrical power is needed.
Charging to feed system spool piece	Spool pieces and piping flanges (and bolts) are in charging pump cubicle. Valves connecting to feed system must be opened in lower level of turbine building, in vicinity of BFPs (6 ft north of BFP motors).	Local manual only.	No electrical power is needed.
Charging pumps and valves	Pumps are located in separate cubicles in PAB. Valves are under PAB floor, with reach rods.	Pumps operated from control room or locally at their controllers (open door and use jumpers). Valves are local manual only.	CP#1 - MCC 4, Bus 1 (480) CP#2 - MCC 2, Bus 1 (480) CP#3 - MCC 4, Bus 2 (480)
Motor-driven Auxiliary Feed-water pumps	PAB	Control room or local.	2400 V Bus 2 2400 V Bus 3
EDG	See emergency power system discussion below.	See emergency power system discussion below.	See emergency power system discussion below.

Condensate and demineralized water are stored in the DWST and the primary water storage tank (PWST). The DWST is an aluminum 30,000-gallon tank that is normally filled from the water treatment plant. The DWST is sized to handle all expected transients in the condensate/feedwater system. This is accomplished by providing makeup to and accepting rejected water from the condenser hotwell.

The EBFP can also take a suction from the PWST via a 4-inch (CD) line which also serves as an alternate supply of water to the charging pumps. The PWST provides demineralized water for the primary plant as well as for various demands in the primary auxiliary building, the radwaste building, and the spent fuel storage area. It is the supply for the low pressure surge tank makeup pumps and, as such, serves the above areas. The PWST is constructed of aluminum and has a capacity of 125,000 gallons. An inner floating roof prevents aeration of the tank contents. The tank receives makeup water directly from the water treatment plant.

Technical Specification 3.7.1.3 requires there to be a minimum combined volume of 85,000 gallons available from the PWST and the DWST. The service water system (discussed later), which receives fresh water from Sherman Pond, supplies the water treating (WT) plant for PWST and DWST makeup. The WT plant* is sized to provide 40 gpm of demineralized water on a continuous basis and 80 gpm maximum, based on the average chemical analysis of Sherman Pond water obtained over a 1-year period.

The safety injection tank (SIT) is sized to provide a source of borated water to the safety injection pumps following a loss of coolant accident. Another function of the SIT is to provide a source of water for flooding the shield tank cavity during refueling operations.

The SIT also provides a source of borated water for the reactivity control system. Technical Specification 3.1.2.11 (Limiting Condition for Operation of Borated Water Sources) requires that the SIT be operable with:

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*The WT plant is not included in the list of "minimum systems" but would probably be available since it is essentially a passive system which is pressurized using the service water system.

- a minimum contained borated water volume of 117,000 gallons of water, equivalent to a tank level of ≥ 25.5 feet
- a minimum boron concentration of 2200 ppm
- a minimum solution temperature of 40°F.

Redundancy

The staff calculated the maximum length of time the plant can stay at hot shutdown following the loss of offsite ac power, using the initial steam generator water inventory and a maximum DWST level of 30,000 gallons. The calculations show that approximately 20.2 hours water supply are available for main coolant system temperature control; after that, the EBFP suction must be shifted to the PWST. The staff also calculated that the total water inventory required by Technical Specifications (85,000 gallons) is enough to keep the plant at hot shutdown for about 73 hours. In Appendix B Part 2, Safe Shutdown Water Requirements, the staff determined the time required to complete a shutdown to the point of shutdown cooling system operation. Assuming (1) no credit for the initial steam generator inventory, (2) no condensate in the hotwell, (3) no single failure, and (4) use of the atmospheric dump valve, the 1-inch vent, the large and small hoggers, and the EBFP (all steam vent paths), the staff determined that a water inventory of 72,000 gallons is sufficient to conduct the cooldown in 48 hours. Further calculations show that, if the plant stayed at hot shutdown for 4 hours and then plant cooldown was initiated, the cooldown rates would be higher, but the time to cool the main coolant system to 330°F would remain the same.

Since the condensate pump motor is much smaller than the boiler feed pump motor (250 hp versus 700 hp), a single condensate pump can be started to pump the contents of the condenser hotwell back to the DWST for EBFP usage. The EBFP would not have to be stopped during this operation since its suction would just be augmented by the condensate pump (the condensate pump rejection line would pressurize the EBFP suction and fill the DWST). The hotwell has a capacity of 15,000 gallons and a normal operating level of about 10,000 gallons. However, the hotwell contents following a loss of offsite ac power and subsequent feed and condensate pump trips cannot be predetermined since event and component coast down times cannot be accurately predicted. Therefore, no credit can be given for this inventory; however, it is likely that there would be a significant quantity of condensate available and usable.

The basis of the technical specifications for reactivity control systems states that the maximum boration capability requirement occurs at the end of core life from full power equilibrium xenon conditions and requires 9,192 gallons of 2200 ppm borated water from the safety injection tank. Since 117,000 gallons of 2200 ppm borated water is available in the SIT, the staff concludes that sufficient borated water capacity is provided to satisfy BRP RSB 5-1 functional requirements.

The amount of main coolant system makeup during the cooldown (and filling of the pressurizer) from 539°F to 330°F was calculated by the staff to be about 6,000 gallons. Since the cooldown to shutdown cooling system initiation used 72,000 gallons and since 85,000 gallons is available per technical specification, the staff concludes that sufficient primary makeup water is available to satisfy BTP RSB 5-1 functional requirements.

Location and Operation

The staff evaluated the equipment discussed above with respect to its location and operability during a loss of offsite ac power. Table 3.2-3 shows the equipment's location, the points from which it may be operated, and its power supply.

SHUTDOWN COOLING SYSTEM

Task: Removal of core decay heat and main coolant system sensible heat to cool the system from 330°F to 140°F.

Discussion

The shutdown cooling system is placed in service after the main coolant temperature has been reduced to approximately 330°F and the pressure to less

Table 3.2-3

EQUIPMENT	LOCATION	CONTROL POINTS	ELECTRIC
DWST (Tank 1)	Outside, north (true) of the VC, between VC and auxiliary boiler room.	Filling and draining using valves below.	No electrical power is needed.
PWST (Tank 39)	Outside, east (true) of the VC in tank farm area.	Filling and draining using valves below.	No electrical power is needed.

Valve No.:		DWST to Charging Pumps	DWST to Condensate Pump Suction	DWST to EBFP	DWST to Hotwell	PWST to EBFP[1]	PWST to Charging Pumps	PWST to Hotwell	PWST to Condensate Pump Suction	
DW-V-608	Auxiliary boiler room	X	O	X	X	X	X	X	X	No electrical power is needed.
DW-V-609	Auxiliary boiler room	O	X	O	O	X	X	X	X	No electrical power is needed.
DW-V-630	PAB	X	X	O	X	O	X	O	O	No electrical power is needed.
DW-V-631	PAB	X	X	X	X	X	X	X	X	No electrical power is needed.
DW-V-662	PAB	X	X	O	X	X	O	X	X	No electrical power is needed.
DW-V-699	At PWST	X	X	X	X	X	X	X	X	No electrical power is needed.
DW-V-777	Auxiliary boiler room	X	X	O	X	O	X	O	O	No electrical power is needed.
DW-V-632	PAB	X	X	X	X	O	O	O	O	No electrical power is needed.
DW-V-697	At PWST	X	X	X	X	O	O	O	O	No electrical power is needed.
DW-V-793	Auxiliary boiler room	X	X	X	X	O	X	X	X	No electrical power is needed.

[1] Filling DWST at the same time

X = shut

O = open

than 300 psig. The shutdown cooling system then reduces the main coolant temperature to 140°F or less and operates continuously to maintain this temperature as long as is required by maintenance or refueling operations.

The shutdown cooling system consists of a heat exchanger, circulating pump, piping, valves, and instruments arranged in a low pressure auxiliary loop parallel with the main coolant loops. The shutdown cooling pump takes suction from the hot leg of the main coolant piping on the reactor side of the loop stop valves and recirculates main coolant water through the tube side of the shutdown cooler and back into the cold leg of the main coolant piping, which is also on the reactor side of the loop stop valves. The main coolant is contained in a closed system, and reactor decay heat load is transferred through the shutdown cooler to the component cooling system which in turn is cooled by river water. This arrangement of providing the intermediate cooling medium of the component cooling system was selected in order to assure that any possible leakage of radioactive main coolant would not enter the river water.

The shutdown cooling system is designed to remove the reactor decay heat about 5 hours after shutdown following 10,000 full power hours of operation. According to the Licensee's estimates, about 16.2×10^5 Btu/h are generated by the reactor and transferred to the main coolant system.*

Redundancy

Although the shutdown cooling system consists of a single cooler and cooling pump, a complete backup of this system is provided by the low pressure surge tank pump and cooler. The coolers and pumps are identical. The low pressure surge tank cooler and pump are connected in parallel with the shutdown cooler and pump. By employing double valving in the inlet and outlet lines of the main coolant piping, any combination of pump or cooler can be used to maintain decay heat removal. Normally the shutdown cooler and

*Draft ANS 5.1 decay heat curve predicts a $P/P_o \simeq 0.008$ at $t = 18.0 \times 10^3$ sec, or about 16.34×10^6 Btu/h.

pump are aligned to the main coolant system and the low pressure surge tank cooler and pump are aligned to cool the low pressure surge tank. By manual valve operation, a failed component in the shutdown cooling system (SCS) can be replaced by a similar component in the low pressure surge tank cooling subsystem; therefore, the SCS has redundancy of components. Because of the sharing of common suction and discharge piping, the SCS is susceptible to passive piping failures, as well as failure of one of the motor-operated valves.

Failure to open of either a suction or discharge valve would result in loss of shutdown cooling system operation. As discussed in Section 5.1, alternative methods of cooling exist, therefore, this deviation from the review criteria is considered acceptable.

Based on the above discussion, the staff concludes that the SCS satisfies the functional requirements of BTP RSB 5-1 except that the SCS and LPST cooling pumps are not normally powered from diesel-supplied electrical buses. The staff will evaluate the significance of this in the SEP integrated assessment of Yankee Rowe.

Location and Operation

The staff evaluated the equipment discussed above with respect to its location and operability during a loss of offsite ac power. Table 3.2.4 shows the equipment's location, the points from which it may be operated, and its power supply.

COMPONENT COOLING WATER SYSTEM

Task: Provide cooling water to the SCS and/or LPST coolers and to other essential equipment.

Discussion

The component cooling system is necessary to remove reactor decay heat from the shutdown cooling system heat exchanger (or the low pressure surge tank cooler) and to provide cooling to equipment necessary for plant cooldown.

The component cooling system consists of two coolers, two circulating pumps, a surge tank, a chemical addition tank and associated piping, system and instrumentation piping, valves, fittings, and instruments. This equipment is connected to two main piping headers. One supplies vapor container

Table 3.2-4

EQUIPMENT	LOCATION	CONTROL POINTS	ELECTRIC
SCS and LPST cooling pumps and coolers	Cubicles in the PAB, lower level.	The SCS and LPST cooling pumps are operated from the control room and can be operated locally during an emergency by jumping the cell switch and using the test control switch at the switchgear cubicle. The coolers require local, manual valve operation such that operation from the control room is not possible.	SCS pump: Bus 5-2 (480 V) LPST cooling pump: Bus 6-3 (480 V)
SCS valves	See Sections 4.1 and 4.2.	See Sections 4.1 and 4.2.	See Sections 4.1 and 4.2.

components, the other supplies equipment outside the containment. Independent lines, provided with isolation valves located outside the vapor container, are connected from the vapor container supply header to the various components inside the vapor container.

A surge tank (4,000 gallons) is used in the component cooling system to provide makeup water for the system, to accommodate the expansion and contraction of the water in the system as temperature changes, and to act as a receiver for the safety valves in the component cooling lines. The water level in the tank is maintained at approximately 2,500 gallons. The surge tank is equipped with a vent to the primary vent stack and a safety valve which discharges into the vapor container drain tank.

Level controls and alarms are provided on the surge tank. A low pressure alarm and control switch is located on the common pump discharge header and pressure indicators are provided in the outlet of the component cooling pumps. A pressure switch starts the standby pump on low pressure.

The common cooler inlet and outlet pipes are provided with local and remote temperature indicators. The inlet pipe has a high temperature alarm, and the outlet has a flow meter, and a low flow alarm. Controls for the component cooling pumps are located on the nuclear auxiliary panel in the main control room.

Two motor-driven centrifugal circulating pumps are provided. The capacity of each pump is approximately 2,000 gpm, with a total dynamic head of 190 feet of water and a design discharge shutoff pressure of 110 psig. The switches may be set in "Close", "Auto", or "Trip" position with provision for "Pull-out" in the trip position.

The two component coolers are of the shell and tube design and are provided to transfer heat from the component cooling water (CCW) to the service cooling water. The tubes are made of admiralty metal.

During a cooldown of the main coolant system following a loss of offsite ac power, the SCS is used to circulate the hot main coolant through the SCS cooler (tube side). The shell side of the cooler is furnished with CCW, and component cooling should both CCW pumps be inoperable or if a rupture in the system has occurred. The procedure (OP3115 - Loss of Component Cooling) directs the operator to attempt first to hook up to the portable fire hose from the fire system to the CCW system, then, if unable to use the fire system, to use the service water system. Thus, there are redundant and diverse means to provide component cooling.

Based on the above discussion, the staff concludes that the CCW system satisfies the functional requirements of BTP RSB 5-1, except that the electrical components are not powered from diesel supplied electrical buses. The staff will evaluate the significance of this in the SEP integrated assessment of Yankee Rowe.

Location and Operation

The staff evaluated the equipment discussed above with respect to its location and operability during a loss of offsite ac power. Table 3.2-5 shows the equipment's location, the points from which it may be operated, and its power supply.

SERVICE WATER SYSTEM

Task: Provide cooling water to the component cooling water coolers and the SCS pump and/or LPST cooling pump coolers.

Discussion

The service water system consists of three 2,500 gpm vertical deep well type pumps which obtain their suction from a common intake well in the circulating water pump house. The pumps discharge to a common 12-inch header which branches into two 12-inch supply headers. The supply headers run parallel to the southern wall of the turbine room basement. The two 12-inch supply headers furnish the various components with service water via separate taps from one or both of these two main supply headers. The headers can be (manually) cross-connected so that any combination of pumps supplies the necessary loads.

Table 3.2-5

EQUIPMENT	LOCATION	CONTROL POINTS	ELECTRIC
CCW pumps (2)	Bottom floor level of the PAB, NW end of building, under CCW coolers.	Operable from the control room. Can be operated locally in an emergency by jumping the cell switch and using the test control switch at the 2400 V breaker.	CCW pump #1 - Bus #3 (2400 V) CCW pump #2 - Bus #2 (2400 V)
CCW coolers (2)	Side-by-side, upper level of PAB, SW end of building, adjacent to CCW surge tank.	Local-manual operation of valves.	No electrical power is needed.
CCW hose fittings and portable hoses	Located at various places in the SW end of the PAB, all within about 50 ft.	Local-manual operation only.	No electrical power is needed.

The greatest heat load on the system occurs when the SCS is first placed in operation. A total of 2500 gpm of 60°F cooling water is required at that time. This same flow is required at any time when the main coolant system water chemistry requires operating the purification system at its maximum capacity of 100 gpm. There is adequate capacity in the service water pumps to meet these special operating conditions.

Redundancy

Normally, two pumps will be in operation, with one pump on standby. If the pressure in the discharge header falls below a preset value, the standby pump will start and simultaneously an alarm will be given at the main control board. The pressure switch for initiating this standby operation is located in the turbine room and is set at approximately 50 psig.

The 2400-V power supplies to service water pumps #1, 2, and 3 are Bus #3, Bus #1, and Bus #2, respectively. These buses can be separated so a fault in one would not disable any more than one service water pump.

Should all pumps fail due to electrical problems, localized damage in the screen house, or loss of suction from Sherman Pond or if a break affecting certain portions of the service water header should occur, selected service water loads can be provided with cooling water from the fire system. The fire system could be supplied by either the installed fire pumps or from portable fire pumps connected in series taking a suction from the river or from Sherman Pond. Also, the potable water system can supply selected service water loads with cooling water. The plant procedure (OP-3009, Loss of Service Water) describes which components may receive fire water or potable water and the locations of the necessary connections.

Based on the above discussion, the staff concludes that the service water system satisfies the functional requirements of BTP RSB 5-1, except that the electrical components are not powered from diesel-supplied electrical buses. The staff will evaluate the significance of this in the SEP integrated assessment of Yankee Rowe.

Location and Operation

The staff evaluated the equipment discussed above with respect to its location and operability during a loss of offsite power. Table 3.2-6 shows the equipment's location, the points from which it may be operated, and its power supply.

PRESSURE CONTROL AND RELIEF SYSTEM

Task: To maintain a system overpressure during hot standby and/or natural circulation cooling and to depressurize the main coolant system to permit the initiation of the shutdown cooling system and to cool down the pressurizer.

Discussion

The pressure control and relief system primarily functions to maintain the required main coolant pressure at the reactor outlet during steady-state operation, to limit to an allowable range the pressure changes caused by main coolant thermal expansion and contraction during normal load transients, and to prevent the pressure in the main coolant system from exceeding the design pressure. The pressure control and relief system consists of a pressurizer vessel containing a two-phase mixture of steam and water, immersion heaters, safety and relief valves, spray system, interconnection piping, valves, and instrumentation.

Depressurization of the main coolant system in pressurized water reactors is generally achieved by the pressurizer in conjunction with one or more of the following: (1) the main pressurizer spray, (2) the auxiliary pressurizer spray, or (3) the pressurizer relief valve. The pressurizer spray nozzle is located in the manway at the top of the pressurizer. The spray pipe is connected to the main coolant system inlet pipe on the reactor side of loop number 2 isolation valve. This connection is in the form of a scoop inside the coolant piping, so that the velocity head plus the static pressure difference between this connection and the surge pipe connection provide the maximum possible driving force for spray flow. A motor-operated valve on the spray can be operated by a switch on the main control board.

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TABLE 3.2-6

EQUIPMENT	LOCATION	OPERATION	POWER SUPPLY
SWPs (3)	Circulating water pump house	Operable from the control room; can be operated locally in an emergency by jumping the cell switch and using the test control switch at the 2400v breaker	SWP #1 - Bus #3 (2400v) SWP #2 - Bus #1 (2400v) SWP #3 - Bus #2 (2400v)
Hose connections to SW system	Inlet and outlet to the CCW coolers	Local-manual	No electrical power is needed

Following a loss of offsite ac power, the main coolant pumps are not available to sustain primary coolant flow; therefore, normal pressurizer spray flow is not available for main coolant system depressurization. As a consequence, depressurization of the main coolant system must be achieved with either the auxiliary pressurizer spray or the pressurizer solenoid-operated relief valve, PR-SOV-90. The solenoid-operated relief valve can be operated manually by a switch in the control room. A motor-operated valve placed on the solenoid valve inlet piping is provided for isolation of the solenoid valve. The steam and/or water discharged from the solenoid-operated relief valve will discharge directly to the containment atmosphere via a rupture disc in the line. For this reason, this method would be used as a last resort.

The auxiliary spray line is located in the chemical and volume control system. It is connected to the feed line downstream of the feed and bleed heat exchangers. This arrangement permits charging of water by the high pressure charging pumps into the top of the pressurizer.

Following a loss of offsite ac power, main coolant system pressure control is necessary to maintain an adequate subcooling margin and assure no disruption of the natural circulation flow. Once natural circulation is achieved, system pressure control would be accomplished by maintaining a system overpressure with the pressurizer through use of the chemical and volume control system or pressurizer heaters. There are 48 pressurizer heaters combined into 24 groups. The 24 groups are combined into eight 3-phase groups of 37.5 kW capacity.

Redundancy

In an April 9, 1980 letter [6], the Licensee indicated that operating experience at Yankee Rowe has demonstrated that the operation of one group of pressurizer heaters (37.5 kW) is required to meet the heat loss from the pressurizer with normal spray flow through the pressurizer at hot standby conditions. The ability to maintain natural circulation under emergency conditions would require less capacity than for normal operations. The Licensee also indicated that Westinghouse had performed a study to determine minimum heater requirements without offsite power and that extrapolation of the results of this study to the Yankee Rowe pressurizer confirmed the required heater capacity.

The Westinghouse study also determined that the capability to supply emergency power to the heaters within 4 hours would prevent loss of subcooling in the primary system following a loss of offsite power. The Yankee Rowe facility has four groups of pressurizer heaters connected to 480-V Bus 6-3 and four groups connected to 480-V Bus 5-2. Each of these buses is connected to an emergency bus via two circuit breakers in series, and these tie breakers are operated from the main control room. To supply the required heater capacity from the emergency bus, Bus 6-3 and Bus 5-2 are cleared and the buses are re-energized by closing the tie breakers to the emergency buses. The Licensee indicates that the time required to accomplish this, giving due consideration to all requirements of plant operating procedures, is 15 minutes from the occurrence of the loss of offsite power.

In a March 19, 1981 letter [7], the Licensee described design features of a proposed alternative safety shutdown system (ASSS). The Licensee indicated that the ASSS is designed in accordance with requirements of Appendix R to 10CFR50 as further clarified in the NRC Generic Letter 81-12, dated February 20, 1981. The proposed design has one group of pressurizer heaters being powered by #3 diesel generator through a new 480-V ASSS motor control center such that main coolant system pressure can be maintained. One group of heaters is capable of maintaining a hot shutdown condition.

As described, the pressure control and relief system has two methods of depressurization. A single failure of the solenoid-operated relief valve or its blocking valve would not preclude the capability to depressurize, provided auxiliary pressurizer spray flow is available from the chemical and volume control system. The availability of this flow path is assessed in the discussion of the chemical and volume control system.

Based on the above discussion, the staff defers evaluation of the adequacy of the pressure control and relief system to satisfy BTP RSB 5-1 pending resolution of current staff reviews of applicable TMI-2 action items and fire protection requirements. The staff will determine the effect that the completion of these reviews will have on the safe shutdown topic during the integrated assessment.

Location and Operation

The staff evaluated the equipment discussed above with respect to its location and operability during a loss of offsite ac power. Table 3.2-7 shows the equipment's location, the points from which it may be operated, and its power supply.

CHEMICAL AND VOLUME CONTROL SYSTEM

Task: Provide main coolant system makeup (due to the contraction of the coolant during the cooldown), to provide a flow path for borating the main coolant system to the necessary shutdown margin, and to provide a means for depressurization of the main coolant system.

Discussion

The chemical and volume control system consists of three positive displacement charging pumps, feed and bleed heat exchangers, pressure reducing orifices, LPST, LPST cooling pump, LPST cooler, LPST makeup pumps, and associated piping, valves, fittings and instruments.

During normal operation, bleed flow passes from No. 1 loop T_c line, through the tube side of the feed and bleed heat exchangers, through the vari-orifice, and finally into the LPST through an eductor. Charging flow passes from the purification pump discharge through the charging pumps, through the shell side of the feed and bleed heat exchangers, and into No. 4 loop T_h line. In addition, charging flow can be lined up to the individual loops via the safety injection system.

Each charging pump is a positive displacement reciprocating pump rated at 33 gpm, 2500 psig and driven by a 50-hp motor. No. 1 and 3 pumps have variable speed drives. No. 2 pump is directly coupled to its motor and its

TABLE 3.2-7

EQUIPMENT	LOCATION	OPERATION	POWER SUPPLY
PR-SOV-90 Solenoid Operated Pressurizer Relief Valve	Vapor Container	Control Room	Battery #1
PR-MOV-512 Pressurizer Relief Blocking Valve	Vapor Container	Control Room	480V Emergency MCC 1
Pressurizer Heaters	Vapor Container	Control Room	Bus 5-2 (4 groups) Bus 6-3 (4 groups)
PR-MOV-191 Main Pressurizer Spray Flow Isolation Valve	Vapor Container	Control Room	MCC1 Bus 2

constant speed. No. 2 pump could put out a variable flow by throttling CH-V-690 between the discharge and suction of the pump.

Charging pump suction can be from the following sources:

1. LPST (gravity flow)
2. Purification system (IX-gravity)
3. Purification system (pumps)
4. Boric acid mix tank (gravity)
5. Safety injection tank (gravity)
6. WT system (via LPST makeup pumps)
7. PWST (gravity or LPST makeup pump)
8. DWST (gravity).

Boration of the RCS is accomplished by injecting borated water from either the boric acid mix tank (1500 gal at 12.0 to 12.5% by weight-min) or the safety injection tank (117,000 gal at 2200 ppm-min). The SI tank is normally used since it provides finer reactivity control because of its lower boron concentration; however, the boric acid mix tank is also available.

A means for depressurization of the main coolant system is provided by the auxiliary spray line. It is connected to the feed line downstream of the feed and bleed heat exchangers. To initiate auxiliary spray flow, motor-operated valves CH-MOV-524 and PR-MOV-191 are closed and manual valve CH-V-613 is opened and throttled. CH-MOV-524 and PR-MOV-191 are operated remote manually from the control room; however, access to the vapor container is required to locally operate CH-V-613.

Redundancy

To ensure that the pressurizer level can be controlled during the most rapid cooldown (i.e., ensure sufficient charging pump discharge) the staff used the calculations of the main coolant system cooldown with the 1-inch vent after a wait time of 4 hours. This cooldown rate was initially (i.e., at T_{RCS} = 540°F) slightly greater than 50°F/h. The staff calculated that the liquid contraction rate due to the cooldown at about 50°F/h is less than the input rate available from each charging pump. Therefore, the pressurizer level can be raised by only one charging pump during this cooldown, and the remaining pumps provide further redundancy.

Boration of the main coolant system cannot be accomplished at Yankee Rowe without the charging pumps unless primary system pressure is reduced to allow use of the low pressure and high pressure safety injection pumps. The shutoff head of the high pressure safety injection pumps is 1350 feet, while that of the low pressure pumps is 1530 feet. Assuming that the low pressure safety injection pump is used as a booster pump for the high pressure safety injection pump, the shutoff head of the safety injection system corresponds to approximately 1470 psi. In order for the safety injection system to be a viable path for borated water addition, depressurization of the main coolant system would be required. As discussed in the previous section, two means are available to depressurize. If a single failure is postulated in the pressure control and relief system (i.e., solenoid-operated relief valve does not open on demand), the auxiliary spray flow from the chemical and volume control system is required. In order to ensure the availability of this depressurization method, the charging pumps must be available.

A backup path for charging flow is from the charging pump discharge via CTT-MOV-522 to the loop fill headers. Motor-operated valves in one of the safety injection lines would be opened to direct the charging flow into a cold leg. These valves are remote-manually operated from the control room.

Primary coolant can be letdown via sample and drain lines. All valves are operable from the control room or at the sample sink.

The charging pumps are not powered from the 480-V emergency buses. The Yankee Rowe facility has three charging pumps powered from three different 480-V buses. Each of these buses is connected to an emergency bus via two circuit breakers in series, and these tie breakers are operated from the main control room. To supply the charging pumps from an emergency bus, the non-emergency buses are cleared and then re-energized byclosing the tie breakers to the emergency buses. In Reference 7, the Licensee proposed that a charging pump and associated motor-operated valves be powered by #3 diesel generator through a new 480-V ASSS motor control center.

Based on the above discussion, the staff concludes that the chemical volume and control system does not meet the functional requirements of BTP RSB 5-1 in that the charging pumps and other electrical components are not powered from diesel-supplied electrical buses. The staff will evaluate the significance of these deviations during the SEP integrated assessment of Yankee Rowe.

Location and Operation

The staff evaluated the equipment discussed above with respect to its location and operability during a loss of offsite power. Table 3.2-8 shows the equipment's location, the points from which it may be operated, and its power supply.

CONTROL AIR SYSTEM

Tank: Provide compressed air for instrumentation and the control of air-operated valves in other safe shutdown systems.

Discussion

The No. 3 control air compressor is a two stage rotary screw compressor rated at 435 scfm. at 100 psig. This compressor is directly driven by a 1800 rpm, 100 hp, 480-V motor. This compressor normally supplies both instrument air and service air requirements.

Two 125 scfm, 600 rpm control air compressors with V-belt drive and 25 hp motors provide backup air at 100 psig to the instrument and control air system. Each compressor can operate in one of two modes. A local control switch with "Off-Hand-Auto" positions actuates the starter for each compressor. In "Hand" position, the compressor runs continuously with the compressor loading and unloading automatically to maintain receiver pressure. In "Auto" position, the compressor motor is started and stopped automatically to maintain receiver pressure. Each compressor is sized to provide 100% of the station's compressed air requirements. The vertical, single stage, double acting, reciprocating, water-cooled compressors are of the non-lubricated carbon or teflon ring type and are installed with aftercoolers, air receivers, and intake filters.

The discharge from each control air receiver supplies one header of a double header piping system that runs throughout the station. The two control air headers are cross-connected at the receivers in the turbine area and in the primary auxiliary building. Air from each header is supplied through reducing valves, as required, to each instrument or control air supply manifold in the turbine area and primary auxiliary building. The reduced air station within the main control board has a low pressure alarm at 25 psig. The control air header low pressure alarm is set at 75 psig. A solenoid-operated bypass valve that opens at 65 psig connects receiver and header directly.

TABLE 3.2-8

EQUIPMENT	LOCATION	OPERATION	POWER SUPPLY
Charging Pumps (3)	(See EBFP discussion)	(See EBFP discussion)	(See EBFP discussion)
Boric Acid Mix Tank	Upper level of PAB in general vicinity of the component cooling water surge tank	Local manual operation only (filling, etc.)	Mechanical agitator is powered from MCC4 Bus 1, trace heaters from MCC4 Bus 2, and redundant trace heaters from emergency Bus 1.
Safety Injection Tank	Outside of the safety injection building, west of the waste disposal building	Tank is filled by lining up various valves in the PAB. Suction path to SIS is automatically aligned	There is a small heat exchanger and circulating pump which keeps the water between 120-130°F, but these not necessary following the loss of AC. Therefore, no electrical power is needed.
CH-MOV-524	Inside vapor containment	Control room	480V Emergency MCC1
CH-MOV-523	In PAB next to charging pumps	Control room	480V Emergency MCC1
CH-MOV-525	Inside vapor containment	Control room	480V MCC1 Bus 1
LCV-222	In PAB	Control room	

Redundancy

Following a loss of offsite ac power, the control air compressors would not be available, since they are powered from non-emergency 480-V buses. Loss of the air compressors eventually results in a complete loss of control air to safe shutdown equipment. Operating Procedure 3002, "Loss of Control Air Supply", defines the immediate operator actions required to place the plant in a hot shutdown condition.

Upon loss of offsite ac power and control air supply, safe shutdown components are affected as follows:

- o PCV-451 (steam supply to emergency boiler feed pump) will close if open. This valve supplies steam to the turbine driven emergency feed pump.
- o TV-405 (auxiliary steam trip valve) will close. This valve supplies steam to the large and small hoggers.
- o TCV-200 (cooling water return CCW system) will open. This valve controls component cooling water flow through the LPST cooler and the SDC cooler.
- o TCV-205 (cooling water return CCW system) will close. (VC)
- o LCV-222 (letdown line level control valve to LPST) will close. This valve controls letdown flow rate to LPST.
- o TV-411 (atmospheric dump valve) will close. This valve provides a path for sensible and decay heat removal.
- o CV-1100A, B, C, D (bypass valves around feedwater blocking valves) will lock in position. These valves provide a bypass path around motor-operated feedwater blocking valves.
- o CV-1100, CV-1200, CV-1300, and CV-1400 (feedwater regulating valves) will lock in position. These valves control feedwater to the steam generators.
- o Wide and narrow range steam generator water level transmitters will indicate low steam generator level.
- o Variable speed charging pumps will not control from the main control board. (One charging pump controller is locked in the full speed position.)
- o Indications from pneumatic level instruments will be erratic, including pressurizer pressure.

Following a loss of control air, the operator can valve in emergency nitrogen supply to the atmospheric dump valve and the auxiliary steam trip valve. In addition, the operator would control steam generator levels by using manual bypass valves. With erratic pneumatic steam generator levels, the operator must rely on the electrically transmitted steam generator level signals in the switchgear room, or the electric level indication on or inside the main control board can be used.

The service air system is another source of pressurized air. Service air is provided by one 514 scfm service air compressor with V-belt drive and 100-hp motor. The vertical, single-stage, double-acting, reciprocating, water-cooled compressor is of the lubricated type and is installed with intake filter and air receiver. The 100-hp, 440-V motor is powered from non-emergency bus 4-1 and controlled by an air circuit breaker with 125-V dc control in the 440-V switchgear. Operation of the circuit breaker is controlled from a locally mounted 3-position switch. The type of automatic control for this compressor duplicated that provided for the control air compressors.

The discharge from the service air receivers supplies a single header piping system which runs throughout the station. This system is interconnected with the control air system. The service air system can function as a backup to the control air system during normal operation. Following a loss of offsite power and/or a single failure, the system cannot be relied upon to function and therefore is not considered to be a safe shutdown system.

The control air system provides pressurized air for necessary valve control functions within safe shutdown systems and pneumatic signals in essential instrumentation. Loss of the compressed air system will not prevent reaching a safe shutdown condition, but it is detrimental from the standpoint of causing numerous manual valve operations and erroneous indication to the operators. These additional actions are beyond the limited operator actions that may result from a single failure.

Based on the above discussion, the staff concludes that the control air system does not satisfy the functional requirements of BTP RSB 5-1 in that a reliable source of control air is not available and significant operator action outside the control room is required to effect a safe shutdown. The staff will evaluate the significance of this in the SEP integrated assessment of Yankee Rowe.

Location and Operation

The staff evaluated the equipment discussed above with respect to its location and operability during a loss of offsite ac power. Table 3.2-9 shows the equipment's location, the points from which it may be operated, and its power supply.

EMERGENCY POWER SYSTEM

Task: Supply a reliable source of ac power to run the necessary equipment.

Discussion

The three emergency diesel generators (EDGs) are each rated for continuous operation at 500KVA, 480V, 0.8 pf, and 1800 rpm. The engines are fast-starting, V-16 (cylinders), two-cycle, water-cooled engines that are directly coupled to an air-cooled synchronous generator.

Each engine has a closed, self-contained water cooling cycle and is started with a 125-V dc cranking motor that is supplied with power from an independent battery.

Air for operation of the engine and for cooling the generator and engine radiator is obtained from roof intake vents. The cooling air exhausts to the outside atmosphere, and the engine exhaust is via a muffler.

Each EDG has a 275-gallon fuel oil supply tank which contains enough fuel for 11.5 hours at full load. A 30,000-gallon fuel oil storage tank can supply any supply tank via gravity flow. The storage tank Technical Specification minimum (8,000 gallons) can supply enough fuel for all EDGs at required load for more than 7 days. HI-LOW level in the three supply tanks is annunciated in the control room.

TABLE 3.2-9

EQUIPMENT	LOCATION	OPERATION	POWER SUPPLY
* Two control air compressors	First floor of turbine building	Local control switch	(1) 480V MCC 2 Bus 1 (2) 480V MCC 1 Bus 1
* One service air compressor	First floor of turbine building	Locally mounted switch	480V station service switchgear Bus Sect. 4-1
** One rotary non-lube air compressor	First floor TB	Locally mounted switch	480-V station service switchgear Bus Sect. 4-1

* Standby equipment

** Normally supplies both service and instrument control air

Redundancy

Each of the three EDG's are rated at about 536 hp (400 kw), which is sufficient to supply the necessary electrical loads during shutdown and cooldown of the plant. However, since the emergency buses, powered by the EDG's, are normally connected to safety injection loads, manual operator action is required to disconnect these loads and feed the shutdown loads.

The EDG's are further evaluated in the resolution of SEP Topics VII-3 (electrical portion) and VIII-2.

Location and Operation

The staff evaluated the equipment discussed above with respect to its location and operability during a loss of offsite ac power. Table 3.2-10 shows the equipment's location, the points from which it may be operated, and its power supply.

3.3 Safe Shutdown Instrumentation

Table 3.3-1 lists the instruments required to conduct a safe shutdown. The list includes those instruments which provide information to the control room operator from which the proper operation of all safe shutdown systems can be inferred. These instruments are the RCS pressure and temperatures, pressurizer level, and steam generator level. Improper trending of these parameters would lead the operator to investigate the potential causes. Other instruments listed in the table provide the operator with (1) a direct check on safe shutdown system performance and (2) an indication of actual or impending degradation of system performance. The list of instruments satisfies the requirement of BTP RSB 5-1 for safe shutdown. The DBE evaluations, which in many cases are not based on the same assumptions as this review, may determine that additional instrumentation is required to achieve and maintain a safe shutdown following a DBE. The design of the instrumentation and controls used for safe shutdown will be evaluated later in the electrical portion of the resolution of SEP Topic VII-3.

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TABLE 3.2-10

EQUIPMENT	LOCATION	OPERATION	POWER SUPPLY
Emergency diesel generators	Diesel generator building	Control room	Start #1 125V DC Bus 1 Start #2 125V DC Bus 2 Start #3 Battery Dist. Switchboard 3

TABLE 3.3-1 SAFE SHUTDOWN INSTRUMENTS

COMPONENT/ SYSTEM	INSTRUMENT	INSTRUMENT LOCATION
Steam Generator	Steam Generator Level (LT&LI FW-1001, 1101, 1201, 1301 (WR); FW-1003, 1103, 1203, 1303 (NR))	Control Room (Indication and alarm)
	Steam Generator Pressure (PIT-MS-403, 4, 5, 6)	Control Room (Indication)
Auxiliary Feed System	Demineralized Water Storage Tank Level (LIT-405)	Control Room (Indication and alarm)
	Auxiliary Feedwater Flow	Control Room (Indication)
Chemical and Volume Control System	Charging Flow (FI&FT-2) Letdown Flow (FI&FT-1)	Control Room (Indication and alarm) Control Room (Indication)
Shutdown Cooling System	SCS Flow (FI-204)	Local (Indication)
Component Cooling Water System	CCW Flow (FI&FT-201)	Control Room (Indication and alarm)
	Cooling Water Supply Temperature (TI-222)	Control Room (Indication) Control Room (Indication and alarm)
Service Water System	SWS Flow	Local (Indication)
Main Coolant System	Pressurizer Level (WR) (PR-LD-8)	Control Room (Indication and alarm)
	Pressurizer Pressure (PR-PD-6, PR-PT-700)	Control Room (Indication and alarm)
	Main Coolant System (MC-PD-9)	Control Room (Indication and alarm)
Safety Injection Tank	SIT Level (SI-LT-1)	Control Room (Indication and alarm)

TABLE 3.3-1 (Continued)

COMPONENT/ SYSTEM	INSTRUMENT	INSTRUMENT LOCATION
Primary Water Storage Tank	PWST Level	Control Room (Indication and alarm)
Diesel Generator	Generator Output (voltage, current, frequency)	Control Room (Indication)
Emergency Power System	480V ac buses (status)	Control Room (Indicating Lights and Voltmeter)
	2400V ac buses (status) 125V dc buses (status)	Control Room (Ammeter and Voltmeter)
Air	pressure	Control Room (Indication and alarm)

4. SPECIFIC RESIDUAL HEAT REMOVAL AND OTHER REQUIREMENTS OF BRANCH TECHNICAL POSITION 5-1

Branch Technical Position 5-1 contains the functional requirements discussed in Section 3 and the detailed requirements applied to specific systems or areas of operation. Each requirement is presented below along with a description of the Yankee Rowe system or component applicable to the requirement.

4.1 RHR Isolation Requirements

Requirement

The following shall be provided in the suction side of the RHR system to isolate it from the RCS.

1. Isolation shall be provided by at least two power-operated valves in series. The valve positions shall be indicated in the control room.

2. The valves shall have independent diverse interlocks to prevent the valves from being opened unless the RCS pressure is below the RHR system design pressure. Failure of a power supply shall not cause any valve to change position.

3. The valves shall have independent diverse interlocks to protect against one or both valves being open during an RCS increase above the design pressure of the RHR system.

Evaluation

1. The Yankee Rowe shutdown cooling system (SCS)* suction line has two power-operated isolation valves which do not have position indication in the control room.

2. Neither of the two SCS suction valves are provided with "open permissive" interlocks. The opening of these valves is administratively controlled. The controls for the valves are located in the primary auxiliary building (PAB). Key lock switches control each MOV, and the key is in the custody of the shift supervisor.

   The two suction valves, MOV 552 and MOV 554, are powered from MCC 1, Bus #1. A failure of power supply will not effect the position of these valves (either open-to-close or close-to-open).

3. Neither of the two SCS suction valves are provided with "autoclosure" interlocks. The SCS pressure is controlled by the RCS pressure and

*The SCS functions at the residual heat removal (RHR) system at Yankee Rowe.

SCS (LPST) pump performance when the two systems are connected. To ensure that the SCS is not overpressured, the RCS overpressure protection system, which includes the SCS relief valves, is provided. This is discussed further in Section 4.2.

The staff has concluded that the deviations regarding the independent diverse interlocks for the SCS isolation valves that prevent opening until pressure has decreased below SCS design should be corrected. The staff's position on these deviations is given in Section 5.2.

The deviation from the BTP regarding lack of automatic closure for SCS isolation valves is acceptable because of the combination of adminstrative controls and alarms provided on the SCS system. These alarms provide additional assurance that the operator action required by procedure will be taken to shut the isolation valves when RCS pressure is increasing toward SCS design pressure.

The staff has concluded that the deviations regarding SCS isolation valve position indication in the control room should be corrected. The staff's position on these deviations is given in Section 5.2.

Requirement

One of the following shall be provided on the discharge side of the RHR system to isolate it from the RCS:

1. The valves, position indicators, and interlocks described in Section 4.1.

2. One or more check valves in series with a normally closed power-operated valve. The power-operated valve position shall be indicated in the control room. If the RHR system discharge line is used for an ECCS function, the power-operated valve is to be opened upon receipt of a safety injection signal once the reactor coolant pressure has decreased below the ECCS design pressure.

3. Three check valves in series, or

4. Two check valves in series, provided that there are design provisions to permit periodic testing of the check valves for leaktightness and the testing is performed at least annually.

Evaluation

The Yankee Rowe SCS has two motor-operated isolation valves in series on the system discharge. The position of these valves is not indicated in the control room.

Like the two SCS suction MOVs discussed in Section 4.1, neither SCS discharge MOV control circuitry is provided with an "open permissive" or "auto-closure" interlock. The opening/closing of these valves is administratively controlled. The controls for these valves are adjacent to the controls for the suction valves. Like the SCS suction MOV control switches, these are key lock switches with the key under the control of the shift supervisor.

The two SCS discharge valves, MOV-551 and 553, are powered from MCC 1 Bus #1. A failure of this power supply will not affect the position of these valves (either open-to-close or close-to-open).

4.2 Pressure Relief Requirements - Overpressure Protection

Requirement

To protect the RHR system against accidental overpressurization when it is in operation (not isolated from the RCS), pressure relief in the RHR system shall be provided with relieving capacity in accordance with the ASME Boiler and Pressure Vessel Code. The most limiting pressure transient during the plant operating condition when the RHR system is not isolated from the RCS shall be considered when selecting the pressure relieving capacity of the RHR system. For example, during shutdown cooling in a PWR with no steam bubble in the pressurizer, inadvertent operation of an additional charging pump or inadvertent opening of an ECCS accumulator valve should be considered in selection of the design bases.

Evaluation

All operating PWRs have been required to modify plant operating procedures and install the necessary hardware to ensure that the RCS when in a cold and shutdown condition is not overpressurized. The RCS low temperature overpressure protection system (LTOPS) must be capable of mitigating the most limiting mass and energy input events. The LTOPS will also afford protection for the shutdown cooling system (SCS), which at Yankee Rowe is the equivalent of the RHR system.

The RCS and SCS can be connected whenever RCS temperature is below 330°F and the RCS pressure is below about 300 psig. There are no interlocks associated with the two suction or two discharge MOVs, and their position is under administrative control. The SCS design pressure is 425 psig, and the system has two spring-loaded safety valves set to open at 425 psig.

The RCS low temperature overpressure protection system is designed to prevent exceeding the 10CFR50 Appendix G (isothermal curve) limit during the design basis mass and energy input events. The LTOPS utilizes the two SCS safety valves and the pressurizer solenoid-operated relief valve (SORV) with a manually enabled low pressure setpoint of 500 psig. By procedure, the SORV is switched to the low setpoint when the RCS pressure has decreased to below 450 psig.

The LTOPS, while being specifically designed to maintain the RCS pressure within the Appendix G limits, is available for overpressure protection of the SCS. Each credible mass and energy input event is listed in Table 4.2-1 along with the peak RCS (hence SCS) pressure. In a September 14, 1979 [8] safety evaluation the staff reviewed the Yankee Rowe LTOPS. Information concerning mass and energy input events as well as additional discussion of the LTOPS equipment, its employment, testing, and associated technical specifications are further discussed in the staff's evaluation.

The SCS design limit of 440 psia is based on the pressure limit for the bellows seals employed in certain system valves. If the bellows failed, the valve stem packing would be subject to system pressure, and even if the packing itself failed, the SCS would not experience total loss of function. The governing standard for the allowable pressure on the pipes and other major components of the SCS is American Standard ASA B31.1 1955. This standard allows the imposed stress of 115 percent of design during 10 percent of the operating period and 120 percent of design during 1 percent of the operating

TABLE 4.2-1 LTOPS ENERGY AND MASS ADDITION EVENTS

Heat Input Source	RCS Temperature	LTOPS Lines of Defense	Single Failure	Peak Pressure (psig)
Energy Addition Events				
Core decay heat and RCP (thermal)	T<300°F	1 SV[1] + SORV	1 SV	515
All heaters	T<300°F	2 SV + SORV	1 SV	470
RCP startup	T_{RCS} = 50° ΔT = 100°F (Note 2)	2 SVs + SORV	2 SVs[3]	513
	T_{RCS} = 100° ΔT = 100°F	2 SVs + SORV	2 SVs[3]	520
	T_{RCS} = 100° ΔT = 100°F	2 SVs + SORV	SORV	452
	T_{RCS} = 150° ΔT = 100°F	2 SVs + SORV	2 SVs[3]	531
	T_{RCS} = 200° ΔT = 100°F	2 SVs + SORV	2 SVs[3]	538
	T_{RCS} = 100° ΔT = 150°F	2 SVs + SORV	2 SVs[3]	536

1. Note 1: One SV is assumed initially unavailable since an SCS MOV closure is assumed to initiate the event. The closure of an SCS suction MOV makes the SCS suction side SV unavailable.

2. Note 2: The ΔT indicated is the differential temperature between the steam generator secondary water and the coldest water anywhere in the RCS.

3. Note 3: The Licensee's analyses assumed only the availability of the SORV, and took no credit for the SCS SVs. The staff has found no failure which would disable both SCS SVs.

TABLE 4.2-1 (Cont.)

Mass Input Source	Mass Input Rate (gpm)	RCS Temperature	Lines of Defense	Single Failure	Peak Pressure (psig)
Mass Addition Events					
1 CCP	30	200°F<T<300°F	2 SVs + SORV	1 SV	425
1 CCP	30	T<200°F	2 SVs + SORV	1 SV	425
1 HPSIP	220	200°F<T<300°F	2 SVs + SORV	1 SV	450
1 HPSIP	220	T<200°F	2 SVs + SORV	1 SV	450
1 LPSIP	1100	200°F<T<300°F	2 SVs + SORV	1 SV	700
1 LPSIP[4,5]	1100	T<200°F	2 SVs + SORV	1 SV	700
1 Train[6]	1100	200°F<T<300°F	2 SVs + SORV	1 SV	700

4. Note 4: Only the mass addition from one LPSIP is postulated in this temperature band since all HPSIPs and LPSIPs are de-energized, but an operator error during testing of the LPSIP could result in mass addition.

5. Note 5: The peak RCS pressure in this event is above the Appendix G (isothermal) limits.

6. Note 6: An ECCS train is composed of one LPSIP and one HPSIP operating in series. The mass addition from a single train of ECCS equipment is postulated since only one train is energized in this temperature band.

period.* The Licensee states that Stone and Webster (AE for Yankee Rowe) specifications for the SCS are based on these ASA B31.1 requirements. These specifications are given below.

System Temperature	Allowable Pressure**
300°F	680 psig
200°F	700 psig
100°F	720 psig

Since these limits are not exceeded during any of the postulated transients given above, the staff concludes that the SCS piping and major components are adequately protected for the LTOPS design base transients. The relief protection used, however, is not in accordance with the ASME code since an active component (SORV) is utilized. The staff does not consider this a significant deviation and concludes that the overall SCS pressure relief requirements of BTP 5-1 are met.

This evaluation of SCS overpressure protection also applies to the low pressure surge tank (LPST) cooling loop since the LPST loop design is identical to the SCS.

Requirement

Fluid discharged through the RHR system pressure relief valves must be collected and contained such that a stuck-open relief valve will not:

1. result in flooding of any safety-related equipment
2. reduce the capability of the ECCS below that needed to mitigate the consequences of a postulated LOCA
3. result in a nonisolable situation in which the water provided in the RCS to maintain the core in a safe condition is discharged outside of the containment.

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* ASA B31.1, 1955, paragraph 123(b).

** It should be noted that these pressures are above the allowable pressures (at comparable temperatures) required by Appendix G (isothermal curve) for the RCS.

Evaluation

1. The SCS relief valves (2) can discharge to either the low pressure surge tank (LPST) or to the primary drain collecting tank (PDCT). During SCS operation, the SCS relief valve discharge is valved directly to the LPST. A common 6-inch (OD) header directs relief discharge from several sources to two eductors under water. The LPST has a capacity of 750 ft^3, and a level control system keeps the tank about half full. The tank and water level control is designed to take three pressurizer steam volumes before tank pressure reaches 75 psig. The LPST has six safety valves which relieve to a common header. The header has a rupture disc which opens at 25 psig and relieves directly to containment.

   If one of the SCS relief valves stuck open, then approximately 101 gpm* would be lost out the RCS (and SCS) system. In about 29 minutes, the LPST would overflow out the open rupture disc.** In this situation, the following alarms would alert the operator: LPST level and pressure downstream of LPST safety valves. Since there is no safety-related equipment in the containment sump or on the containment floor where the LPST safeties and rupture disc would relieve, no flooding of ECCS-related equipment would occur.

2. The SCS is not used during either the injection or the recirculation phases following a LOCA. Therefore, a stuck-open SCS relief valve does not reduce the capability of the ECCS equipment.

3. The SCS relief valves are outside containment but relieve to the LPST, which relieves back inside the vapor container (VC); therefore, on a stuck-open SCS relief, there is no net loss of RCS or ECCS fluid.

Requirement

If interlocks are provided to automatically close the isolation valves when the RCS pressure exceeds the RHR system design pressure, adequate relief capacity shall be provided during the time period while the valves are closing.

Evaluation

As discussed in Sections 4.1 and 4.2, the SCS isolation valves (two suction valves and two discharge valves) are not furnished with auto closure features. Therefore, this requirement is not applicable.

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*101 gpm at 465 psig (110% of setpoint pressure).

**The safety valves and rupture disc would open in about 15 min.

4.3 Pump Protection Requirements

Requirement

The design and operating procedures of an RHR system shall have provisions to prevent damage to the RHR system pumps due to overheating, cavitation, or loss of adequate pump suction fluid.

Evaluation

There are no automatic trip or other features associated with the SCS or LPST cooling pumps that are designed to protect these pumps from overheating, cavitation, or loss of adequate pump suction fluid. The 480-V breakers supplying power to the SCS and LPST cooling pumps are equipped with the following protective devices:

1. inverse time magnetic overcurrent trip (adjustable from 60-160% of 100 amps coil rating)
2. instantaneous trip (5-12 times the overcurrent coil rating of 100 amps).

These features are designed to protect the power supplies from an equipment fault, but under certain circumstances (e.g, overheating), the trips may protect the pump motors. The Licensee has not evaluated these features with respect to cavitation, overheating, or loss of suction fluid.

The following indications could alert the operator(s) to an abnormal situation in the SCS:

1. MOV-554, -552, Position Indication
2. MOV-551, -553, Position Indication
3. LPST level
4. LPST pressure
5. SCS inlet temperature
6. SCS or LPST pump discharge pressure
7. SCS or LPST cooler temperature (discharge)
8. SCS discharge (to RCS) flow
9. SCS or LPST cooler control valve (TICV 200) position.

4.4 Test Requirements

Requirement

The isolation valve operability and interlock circuits must be designed so as to permit on-line testing when operating in the RHR mode. Testability shall meet the requirements of IEEE Standard 338 and Regulatory Guide 1.22.

The preoperational and initial startup test program shall be in conformance with Regulatory Guide 1.68. The programs for PWRs shall include tests with supporting analysis to (a) confirm that adequate mixing of borated water added prior to or during cooldown can be achieved under natural circulation conditions and permit estimation of the times required to achieve such mixing, and (b) confirm that the cooldown under natural circulation conditions can be achieved within the limits specified in the emergency operating procedures. Comparison with performance of previously tested plants of similar design may be substituted for these tests.

Evaluation

The procedure used to test the operability of SCS isolation valves MOV-551 through -554 requires the stopping of the SCS pump prior to cycling the valves. Alternately, the operability of these valves could be checked by transferring SCS cooling requirements to the low pressure surge tank cooling system (feed and bleed). Since there are no "open permissive" interlocks associated with any of the four MOVs (two suction valves and two discharge valves), it is not necessary to bypass interlocks.

Yankee Rowe has conducted plant cooldowns using RCS natural circulation, but has not performed any tests regarding flow measurement, cooldown rates, or boron mixing. However, the staff believes that, with the boric acid concentrations used for shutdown, adequate boron mixing will occur under natural circulation flow.

4.5 Operational Procedures

Requirement

The operational procedures for bringing the plant from normal operating power to cold shutdown shall be in conformance with Regulatory Guide 1.33. For pressurized water reactors, the operational procedures shall include specific procedures and information required for cooldown under natural circulation conditions.

Evaluation

The Licensee has procedures to perform safe shutdown operations including shutdown to hot standby, operation at hot standby, hot shutdown, operation at hot shutdown, and cold shutdown including long-term decay heat removal. The Licensee has also provided its operating staff with procedures for shutting down the reactor and for decay heat removal under abnormal and emergency conditions. These procedures describe operator action in the event of loss of system or parts of system functions normally needed for shutdown and cooling the core. Procedures for the operation of individual systems used in safely shutting down the reactor are also included in the plant operating procedures. These procedures were reviewed and are in conformance with Regulatory Guide 1.33. In addition, Section 8, "Procedures," of the Licensee's Technical Specifications assures establishment of written procedures in accordance with NRC standards and Regulatory Guides (including Regulatory Guide 1.33).

4.6 Auxiliary Feedwater Supply

Requirement

The seismic Category I water supply for the auxiliary feedwater system for a PWR shall have sufficient inventory to permit operation at hot shutdown for at least 4 hours, followed by cooldown to the conditions permitting operation of the RHR system. The inventory needed for cooldown shall be based on the longest cooldown time with either only onsite or only offsite power available with an assumed single failure.

Evaluation

The main coolant system cooldown rates and auxiliary feedwater supply inventories under varying conditions are discussed in Section 3.2 and Appendix B, Part 2.

5. RESOLUTION OF SEP TOPICS

The SEP topics associated with safe shutdown have been identified in the introduction to this assessment. The following discussions evaluate the degree to which the safety objectives of these topics are fulfilled at the Yankee Rowe plant.

5.1 Topic V-10.B - RHR System Reliability

The safety objective of this topic is to ensure reliable plant shutdown capability using safety-grade equipment subject to the guidelines of SRP 5.4.7 and BTP RSB 5-1. The Yankee Rowe PWR systems have been compared with the criteria of BTP 5-1, and the results of these comparisons are discussed in Sections 3 and 4 of this assessment. Section 3 discusses the way the functional requirements are met and Section 4 discusses the shutdown cooling system (SCS), which performs the function identified in BTP RSB 5-1 as residual heat removal.

Redundancy to the SCS is provided by the low pressure surge tank (LPST) system. The LPST system is physically arranged in parallel with the SCS. The components (pump and heat exchanger) of both the LPST system and SCS are identical and share a common suction and discharge line in the shutdown cooling mode. Both the suction and discharge lines are isolated by two motor-operated valves in series. The staff finds this degree of redundancy acceptable; however, the following deviations exist which could impair the reliability of the system:

1. The SCS suction and discharge motor-operated isolation valves do not have position indication in the control room. The valves are operated from the primary auxiliary building (PAB) and cannot be operated from the control room.

2. There are no provisions to prevent damage to the SCS pump or LPST system cooling pump due to overheating, cavitation, or loss of adequate suction fluid.

3. In order to cool the reactor coolant system to the SCS initiation point and to initiate SCS operation, significant operator action must be performed from outside the control room.

The first deviation as it relates to the potential overpressurization of the SCS or RCS is addressed under Topic V-11.A.

The first two deviations also relate to interrupting the operation of the SCS while the plant is shut down and being maintained at a temperature equal to or less than 330°F and at a pressure less than 300 psig. The consequences of an inadvertent valve closure or pump failure are that the cooldown would terminate and the plant would start to heat up. Installation of valve position indicators and pump protective trips would alert the operator of the abnormal condition but would not preclude it from occurring. Other plant parameters that are monitored continously in the control room are available to indicate the status of the cooldown to the operator. In the event that the cooldown has been terminated due to a pump failure, the redundant pump and heat exchanger from the LPST system can be put into service.

Two modes of plant status must be considered when evaluating the overall effects of a loss of the SCS function and the acceptability of the deviations: (1) plant shutdown with the temperature being maintained at less than 330°F and at some pressure greater than atmospheric but less than 300 psig; (2) the plant shut down and cooled down to less than 200°F, the reactor vessel head removed, and the system pressure at atmospheric.

In the first case, if the SCS were disabled due to a pump failure, a second pump, from the LPST system, would be available for continued cooldown. If the disruption of SCS were due to valve problems, an alternate method of maintaining the cooldown would have to be employed. One such method would be to let the plant heat up and to remove the heat generated through the steam generators (feed to the steam generators can be obtained from a variety of sources). This provides an acceptable method in which to restore the heat removal from the primary system. In the second case, as defined above, if shutdown cooling were interrupted due to valving failures, adequate cooling of the reactor could be accomplished by keeping the core covered with water.

Based on the discussions above, the staff concludes that, although deviations from current licensing practice exist, the Yankee Rowe SCS can reliably perform its cooldown function; in the unlikely event of a pump or valve failure, acceptable alternatives exist to maintain the plant in a condition which will not endanger the public health and safety.

The third deviation relates to the amount of operator action required to establish shutdown cooling. Branch Technical Position 5-1 states that a limited amount of operator action from outside the control room is permissible. In the case of Yankee Rowe, substantial effort is required of the operators from outside the control room to decrease main coolant temperature and pressure to a point where the SCS can be placed in operation. Most of the equipment that requires manipulation for cooldown is located and controlled from outside the Yankee Rowe control room. The staff evaluation shows the time available before any operator action is necessary to be on the order of 1 hour or more, i.e., without any operator action, from inside or outside the control room, the facility can sustain itself with the water inventory at hand.

The amount of operator action required is not compatible with the intent of the topic criteria. The staff will consider the need for increased control room operability of cooldown systems during the integrated assessment.

5.2 <u>Topic V-11.A - Requirements for Isolation of High and Low Pressure Systems and Topic V-11.B - RHR Interlock Requirements</u>

The safety objective of these topics is to assure that adequate measures are taken to protect low pressure systems connected to the primary system from being subjected to excessive pressure, which could cause failures and in some plants could cause a LOCA outside containment. The current criteria for RHR isolation and pressure relief are discussed in Sections 4.1 and 4.2.

The Yankee Rowe SCS suction and discharge (isolation) valves do not have any open permissive interlocks or automatic closure features, and valve position indication is not provided in the control room. This deviation involves violating a pressure boundary between a high pressure system (RCS) and a low pressure system (SCS or LPST). The interlock and automatic closure features are required whenever the RCS is at a pressure greater than the design pressure of the SCS or LPST (300 psig). The most limiting case is when the RCS is at operating temperature and pressure. The SCS/LPST is isolated from the RCS at both the suction and discharge sides by two key-locked motor-operated valves in series. An inadvertent opening of the pair of suction or discharge valves could cause overpressurization of the low pressure system, which could cause a pipe or system failure, thereby creating a loss of coolant accident (LOCA) outside containment. Current criteria require open permissive interlocks, which prevent opening the valves when a specific pressure differential exists across the valves. In lieu of the open permissive interlock, Yankee Rowe has key-operated valves, operated locally in the primary auxiliary building, with the keys maintained under administrative control. Due to the potential severity of SCS overpressurization, the Licensee will be required to provide (1) interlocks to prevent opening of SCS isolation valves until the main coolant system pressure is below SCS design pressure and (2) valve position indication for the isolation valves in the control room.

The SCS isolation valves do not have automatic closure interlocks to close the valves during slow increases in RCS pressure. This is to prevent RCS pressurization with any SCS isolation valves in the open position. Rapid increases in RCS pressure are discussed in the Section 4.2 evaluation of the low temperature overpressure protection (LTOP) system. Some of these rapid pressure increases occur sufficiently fast that an automatic closure interlock would not respond in time to prevent overpressurization of the SCS. However, the staff concluded that the LTOP provides acceptable SCS and LPST cooling loop pressure relief for these rapid transients. The staff has determined that the installation of automatic closure interlocks would not be desirable since two of the three LTOP relief valves are on the SCS, and automatic isolation of the SCS from the RCS would render the LTOP system inoperable. However, in the SEP integrated assessment the staff will evaluate the potential need for additional measures, such as control room valve indications, to prevent RCS startup and pressurization with any SCS isolation valves in the open position.

5.3 Topic VII-3 - Systems Required for Safe Shutdown

The safety objectives of this topic are:

1. to assure the design adequacy of the safe shutdown sytem to (a) initiate automatically the operation of appropriate systems, including the reactivity control systems, such that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences or postulated accidents, and (b) initiate the operation of systems and components required to bring the plant to a safe shutdown

2. to assure that the required systems and equipment, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, are located at appropriate locations outside the control room and have a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures

3. to assure that only safety grade equipment is required for a PWR plant to bring the reactor coolant system from a high pressure condition to a low pressure cooling condition.

Safety objective 1(a) will be resolved in SEP Design Basis Event reviews. These reviews will determine the need for automatic initiation of safe shutdown systems to mitigate the consequences of accidents and transients.

Objective 1(b) relates to control room availability of the control and instrumentation systems needed to initiate the operation of safe shutdown systems and assures that the control and instrumentation systems in the control room are capable of following the plant shutdown from its initiation to its conclusion at cold shutdown conditions; this does not apply to Yankee Rowe, since the entire operation of shutdown cooling is performed outside the control room.

Safety objective 2 requires the capability to shut down to both hot shutdown and cold shutdown conditions using systems, instrumentation and controls located outside the control room. Yankee Rowe has procedures which identify several methods of tripping the plant and methods to cooldown, provide adequate instruction for determining the operability and condition of the essential plant equipment, and indicate the surveillance instrumentation and instructions needed for interpreting the information.

The review team visited each designated operator station and assessed the capability of the plant staff to perform the necessary operations. The staff concludes that the plant can perform these shutdown operations.

Conformance to safety objective 3 was not evaluated but will be completed in part under SEP Topic III-1, "Classification of Structures, Components, and Systems (Seismic and Quality)," and in part under Design Basis Event reviews.

5.4 Topic X - Auxiliary Feed System (AFS)

The safety objective for this topic is to assure that the AFS can provide adequate cooling water for decay heat removal in the event of loss of all main feedwater using the guidelines of SRP 10.4.9 and BTP ASB 10-1.

The EBFP system and backup method were compared with SRP 10.4.9 and BTP ASB 10-1 with the following conclusions:

1. The Yankee Rowe Nuclear Plant including the AFS will be reevaluated during the SEP with regard to internally and externally generated missiles, pipe whip and jet impingement, quality and seismic design requirements, and earthquakes, tornadoes, and floods.

2. The AFS conforms to General Design Criteria (GDC) 45 ("Inspection of Cooling Water Systems") and GDC 46 ("Testing of Cooling Water Systems"). GDC 5 ("Sharing of Structures, Systems, and Components") is not applicable.

3. The Yankee Rowe AFS is not automatically initiated. The need to provide for automatic AFS initiation in accordance with Lessons Learned Task Force recommendations are under staff review.

4. The Yankee Rowe AFS does not have capability to automatically terminate feedwater flow to a depressurized steam generator and provide flow to the intact steam generator. This is accomplished by local, manual valve operation. The effect of this deviation will be assessed in the main steam line break evaluation for the plant.

5. In 1967, the Licensee made modifications to the Yankee Rowe plant to prevent the occurrence of feed system waterhammer. The staff is continuing its evaluation of feed system waterhammer on a generic basis. SEP Topic V-13, "Waterhammer," applies.

6. The technical specifications for the AFS will be reevaluated against current requirements under SEP Topic XVI, "Technical Specifications."

6. REFERENCES

1. Staff Discussion of Fifteen Technical Issues in Attachment to November 3, 1976 Memorandum from Director, NRR, to NRR Staff, NUREG-0138, November 1976.

2. Appendix A to Part 50 of the Code of Federal Regulations, Title 10.

3. L. H. Heider (YAEC)
Letter to Office of Nuclear Reactor Regulation (NRC)
Subject: Core XV Refueling.
26 March 1981

4. E. McKenna (NRC)
Telephone Conversation with B. Jones (YAEC)
Subject: Safe Shutdown for Yankee Rowe.
27 March 1981

5. D. E. Moody (YAEC)
Letter to D. G. Eisenhut (NRC)
Subject: NRC Requirements for Auxiliary Feedwater Systems at Yankee Rowe Nuclear Power Station.
21 December 1980

6. J. A. Kay (YAEC)
Letter to D. L. Ziemann (NRC)
Subject: Resolution of TMI "Category A" Implementation Audit Outstanding Items.
9 April 1980

7. L. H. Heider (YAEC)
Letter to H. R. Denton (NRC)
Subject: Compliance with Appendix R to 10CFR50.
19 March 1981

8. D. L. Ziemann (NRC)
Letter to R. H. Groce (YAEC)
Subject: Amendment No. 59 to Facility Operating License No. DPR-3 for the Yankee Nuclear Power Station (Yankee Rowe).
14 September 1979.

APPENDIX B, PART 2

SAFE SHUTDOWN WATER REQUIREMENTS

Introduction

Standard Review Plan (SRP) 5.4.7, "Residual Heat Removal (RHR) System," and Branch Technical Position (BTP) RSB 5-1, Rev. 1, "Design Requirements of the Residual Heat Removal System," are the current criteria used in the Systematic Evaluation Program (SEP) evaluation of systems required for safe shutdown. BTP RSB 5-1 Section A.4 states that the safe shutdown system shall be capable of bringing the reactor to a cold shutdown condition, with only offsite or onsite power available, within a reasonable period of time following shutdown, assuming the most limiting single failure. BTP RSB 5-1 Section G, which applies specifically to the amount of auxiliary feed system (AFS) water of a pressurized water reactor available for steam generator feeding, requires the seismic Category I water supply for the AFS to have sufficient inventory to permit operation at hot shutdown for at least four hours, followed by cooldown to the conditions permitting operation of the RHR system. The inventory needed for cooldown shall be based on the longest cooldown time needed with either only onsite or only offsite power available with an assumed single failure. A reasonable period of time to achieve cold shutdown conditions, as stated in SRP 5.4.7 Section III.5, is 36 hours.

For a reactor plant cooldown, water is the medium for transfer of heat from the plant to the environs. Two modes of heat removal are available. The first mode involves the use of reactor plant heat to boil water and the venting of the resulting steam to the atmosphere. The water for this process is typically demineralized "pure" water stored onsite and, therefore, is limited in quantity. The systems designed to use this mode of heat removal (boiloff) are the steam generators for a pressurized water reactor (PWR) and the emergency (isolation) condenser for a boiling water reactor (BWR). The second heat removal mode (blowdown) involves the use of power-operated relief valves to remove heat in the form of steam energy directly from the reactor coolant system. Since it is not acceptabl, to vent the reactor coolant system directly to the atmosphere, the steam is typically vented to the containment building, from which containment cooling water systems transfer the heat to an ultimate heat sink - usually a river, lake, or ocean. When the blowdown mode is used, reactor coolant system makeup water must be continuously supplied to keep the reactor core covered with coolant to compensate for the loss of coolant inventory. Systems employing the blowdown heat removal mode have been designed into or backfitted onto most BWRs. The efficacy of the blowdown mode for PWRs has received increased staff attention since the Three Mile Island Unit 2 accident in March 1979. Additional studies are planned or in progress.

This evaluation of cooling water requirements for safe shutdown and cooldown is based on the use of the system identified in the SEP Review of Safe Shutdown Systems which has been completed for each SEP facility in accordance with SRP 5.4.7 and BTP RSB 5-1 criteria. It should be noted that the SEP Design Basis Events (DBE) reviews, now in progress, may require the use of systems other than those evaluated in this report for reactor plant shutdown and cooldown. In those cases, the water requirements for safe shutdown will have to be evaluated using the assumptions of the DBE review.

Discussion

The requirement in BTP RSB 5-1 and SRP 5.4.7 that a plant achieve cold shutdown conditions within approximately 36 hours is based mainly on the desire to be able to activate the RHR system and transfer the plant heat to an ultimate heat sink prior to the exhaustion of the limited amount of onsite-stored pure water available for the AFS of a PWR. A sustained hot shutdown condition, with reactor coolant systems temperature and pressure in excess of RHR initiation limits, requires continued boiling off of pure water to remove reactor core decay heat. A BWR relying on the emergency condenser system for cooldown under similar conditions would also potentially exhaust onsite-stored pure water.

If water stored onsite is depleted, raw water, for example from a river, lake, or ocean, can usually be tapped to supply the boiloff systems. However, raw water can accelerate the corrosion of boiloff system materials in the steam generator and emergency condenser tubes even if the water is fresh. Raw fresh water can cause caustic stress corrosion cracking of both stainless steel and inconel tubes in less than 72 hours through NaOH concentration. Seawater can cause chloride stress corrosion cracking of the tubes well with one week. Plant cooldown and depressurization would help reduce the rate of tube cracking by reducing the stresses in the tube materials and would also reduce the leakage rate of reactor coolant through cracks that do occur.

The original design criteria for the SEP facilities did not require the ability to achieve cold shutdown conditions. For these plants, and for the majority of operating plants, safe shutdown was defined as hot shutdown. Therefore, the design of the systems used to achieve a cold shutdown condition was determined by the reactor plant vendor and was not necessarily based on safety concerns. Safe shutdown reviews have pointed out a difference in vendor approach to system design for cold shutdown reflected in the Standard Technical Specification definition of cold shutdown: for a BWR, cold shutdown requires reactor coolant temperature to be $\leq$212°F; for a PWR, the temperature is $\leq$200°F. This difference in cold shutdown temperatures requires additional systems for PWR cooling not needed for a BWR. For example, a BWR could use isolation condenser alone to reach 212°F (although the approach to the final temperature would be asymptotic); but a PWR, in addition to the steam generators, must use RHR and supporting systems to cool to 200°F.

Evaluation

Table 1 provides plant-specific data and assumptions used in the staff calculation of safe shutdown water requirements for the Yankee Rowe nuclear plant. Table 2 presents the results of the calculation.

Four hours after the reactor trip, the decay heat rate is 1.774×10^7 Btu/h, and the integrated heat over the 4-hour period is 1.149×10^8 Btu. To maintain a constant reactor coolant temperature of 538°F for the 4 hours, the staff calculated that 12,270 gallons of auxiliary feedwater are required to remove the integrated heat. Following a 4-hour delay period, steam is released through five steam vent paths:

1. atmospheric dump valve (ADV)
2. 1-in vent
3. large hogger
4. small hogger
5. emergency boiler feed pump (EBFP).

Assuming that the cooldown rate does not exceed the administrative limit of 50°F/h and that no single failure event occurs, cooldown to 330°F requires an additional 44 hours and consumes an additional 59,700 gallons of makeup water. Final cooldown to 200°F is accomplished by manually actuating the shutdown cooling system.

Based on the staff calculation, Yankee Rowe's existing steam vent paths do not have sufficient heat removal capacity to achieve cold shutdown conditions within the Standard Review Plan 5.4.7 requirement of 36 hours assuming loss of offsite power and a single failure. However, sufficient water inventory is available to conduct the plant cooldown.

In a March 26, 1981 letter [3], the Licensee proposed changes to provide automatic quick closure of the four main steam line non-return valves. This modification necessitated the installation of a new steam supply line to the steam-driven emergency feedwater pump and installation of additional steam dump capacity. During a March 27, 1981 discussion [4], the Licensee indicated that an additional manually operated dump valve would be installed on each steam line upstream of the non-return valve. Each of these valves is to have the ability to remove approximately 60,000 lbm/h. The staff repeated the safe shutdown water requirement calculation for these new steam vent paths assuming a single failure of one atmospheric dump valve. Table 3 presents the results of the calculation.

As in the previous case, the decay heat rate 4 hours following the reactor trip is 1.774×10^7 Btu/h and the integrated heat over the 4-hour period is 1.149×10^8 Btu. A total of 12,270 gallons of auxiliary feedwater is expended to remove the integrated heat. Steam is then vented through the three atmospheric dump valves. Assuming that the cooldown rate does not exceed the administrative limit of 50°F/h, cooldown to 330°F requires an additional 4.35 hours and consumes an additional 7,320 gallons of makeup water. Based on the above calculations, Yankee Rowe's proposed steam vent paths have sufficient capacity to conduct a plant cooldown in accordance with BTP RSB 5-1.

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

July 27, 1982

Docket No. 50-029
LS05-82-07-070

Mr. James A. Kay
Senior Engineer - Licensing
Yankee Atomic Electric Company
1671 Worcester Road
Framingham, Massachusetts 01701

Dear Mr. Kay:

SUBJECT: SEP TOPIC V-11.A, REQUIREMENTS FOR ISOLATION OF HIGH AND LOW PRESSURE SYSTEMS AND V-11.B, RHR INTERLOCK REQUIREMENTS FINAL SAFETY EVALUATION REPORT FOR YANKEE

The enclosed staff final safety evaluation report has been revised to reflect the comments provided by your letter of June 18, 1982. This evaluation is consistent with the findings in our contractor's evaluation of Topics V-11.A and V-11.B. As a result of our safety evaluation of Topics V-11.A and V-11.B, we propose modifications to the RHR isolation valve control circuitry.

The need to actually implement these changes will be determined during the integrated plant safety assessment. This topic assessment may be revised in the future if your facility design is changed or if NRC criteria relating to this topic are modified before the integrated assessment is completed.

Sincerely,

Dennis M. Crutchfield for
Ralph Caruso, Project Manager
Operating Reactors Branch No. 5
Division of Licensing

Enclosure:
As stated

cc w/enclosure:
See next page

8207300216

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SYSTEMATIC EVALUATION PROGRAM

YANKEE ROWE

TOPICS: V-11.A, REQUIREMENTS FOR ISOLATION OF HIGH AND LOW PRESSURE SYSTEMS
V-11.B, RHR INTERLOCK REQUIREMENTS

I. INTRODUCTION

Several systems that have a relatively low design pressure are connected to the reactor coolant pressure boundary. The valves that form the interface between the high and low pressure systems must have sufficient redundancy and interlocks to assure that the low pressure systems are not subjected to coolant pressures that exceed design limits. The problem is complicated since under certain operating modes (e.g., shutdown cooling and ECCS injection) these valves must open to assure adequate reactor safety.

II. REVIEW CRITERIA

The review criteria are presented in Section 2 of EG&G Report 1350F, "Electrical Instrumentation and Control Features for Isolation of High and Low Pressure Systems."

III. RELATED SAFETY TOPICS AND INTERFACES

The scope of review for this topic was limited to avoid duplication of effort since some aspects of the review were performed under related topics. The related topics and the subject matter are identified below. Each of the related topic reports contain the criteria and review guidance for its subject matter.

V-3	Overpressurization Protection
V-10.B	RHR Reliability
VI-4	Containment Isolation
XV-19	Loss of Coolant Accidents

IV. REVIEW GUIDELINES

The review guidelines are presented in Section 7.3 of the Standard Review Plan.

V. EVALUATION

As noted in EG&G Report 1350F, Yankee Rowe has two systems with a lower design pressure rating than the RCS that are directly connected to the RCS. These are the Reactor Heat Removal (RHR) and the Chemical Volume Control (CVCS) Systems.

The RHR system and CVCS are not in compliance with current licensing requirements for isolation of high and low pressure systems as noted below.

(1) The RHR system isolation valves do not have any interlocks to prevent opening when RCS pressure exceeds RHR system design pressure as required by BTP RSB 5-1;

(2) No interlocks are provided to automatically close any RHR system isolation valves if RCS pressure increased above RHR system design pressure during RHR system operation as required by BTP RSB 5-1; and

(3) The isolation valves for the CVCS do not have interlocks to prevent CVCS overpressurization as required by BTP EICSB-3.

The CVCS letdown is isolated by two remote, motor operated, manually controlled valves in series inside of containment and an air operated automatic valve outside of containment. The air operated valve automatically closes on low pressurizer level.

The positive displacement charging pumps and the charging pump discharge line are designed for reactor system pressure. The discharge line is isolated by two remote, motor operated, manually controlled valves, one valve is inside of containment. The other is outside of containment.

The design of the CVCS letdown and charging system is similar to Palisades and Ginna. The radiological consequences of a break in the CVCS system was evaluated under Topic XV-19 for these plants and found to be acceptable. The valving arrangements in these plants was also found to be acceptable.

VI. CONCLUSIONS

The RHR system isolation valve control circuitry should be modified to prevent opening when RCS pressure exceeds RHR system design pressure as required by BTP RSB 5-1.

Interlocks to close these valves if RCS pressure increases above RHR system design pressure during RHR system operation are not necessary because of the overpressure protection system.

Pending a detailed review under SEP Topics VI-4 and XV-19, the isolation of the CVCS system is acceptable based on previous reviews of similar systems.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555
August 3, 1981

Docket No. 50-29
LS05-81-08-002

Mr. James A. Kay
Senior Engineer - Licensing
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25 Research Drive
Westborough, Massachusetts 01581

Dear Mr. Kay:

SUBJECT: YANKEE ROWE - SEP TOPIC II-2.A, SEVERE WEATHER PHENOMENA

By letter dated December 17, 1980, we forwarded to you our draft evaluation of SEP Topic II-2.A, "Severe Weather Phenomena". Your response dated February 10, 1981, stated that you concurred with our assessment except for the evaluation of snow and tornado loadings.

Regarding the snow loadings you stated that the local terrain effects would result in values much lower than those provided in our evaluation and requested that we re-evaluate the snow loadings accordingly.

The snow load value for the site region has been recently studied as part of a NOAA Study, "Estimating Water Equivalent Snow Depth from Related Meteorological Variables," (NUREG/CR-1389, May 1980) which shows the 100 year snow load in that area to be from 50 to 60 lb/ft^2. Cautions are advised in the use of interpolation of the isopleths, in this mountainous area, since local topographic influences can affect snowfall totals. Similarly, structural effects on drifting snow can result in higher or lower loads than shown for ground level snow loads in this NOAA report.

Additional research on snow loading is being completed through use of actual field studies on power plant structures in the northeast and the Great Lakes areas. Thus, subject to additional information resulting from these studies, the snow loads defined in our December 17, 1980 evaluation, i.e., normal of 40 lbs/ft^2 and snow load combined with probable maximum winter precipitation of 125 lbs/ft^2, should be used for structural response analysis.

Your comments regarding the design basis tornado are similar, indicating that terrain effects will significantly effect the tornado wind loadings and frequency.

The tornado analysis by McDonald (Enclosure 2 of our December 17, 1980 letter) considered 224 tornadoes, during the period from 1950-1978, that occurred in the 3x3 degree square area roughly centered about the Yankee

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Rowe site. This analysis suggests an upper bound 95 percentile tornado wind speed, with a 10^{-7} probability level of approximately 300 mph is appropriate. This value is the suggested wind speed at the site, for the analysis of structural design capability. Inherent in the McDonald tornado analysis are weighting factors for population density, storm intensity and topographic features that could effect the observation of any tornado. Selection of a 300 mph upper 95 percentile wind speed provides a conservatism beyond the expected value of approximately 250 mph for the 10^{-7} probability level. In addition, the outbreak of tornadoes in Kentucky and Tennessee during April 3 and 4, 1974 indicates that tornadoes can occur in mountainous terrain.

Based on the studies mentioned above, we do not feel that you have provided sufficient justification for us to change our earlier evaluation provided on December 17, 1980. Therefore, we consider that evaluation to be final and Topic II-2.A to be completed.

The evaluation will be a basic input to the integrated safety assessment for your facility. The assessment may be revised in the future if your facility design is changed or if the NRC criteria relating to this subject is modified before the integrated assessment is complete.

Sincerely,

Dennis M. Crutchfield

Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing

cc: See next page

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555
September 4, 1981

Docket No. 50-029
LS05-81-09-013

RECEIVED
SEP 09 1981
U.S. NUCLEAR REGULATORY COMMISSION

Mr. James A. Kay
Senior Engineer-Licensing
Yankee Atomic Electric Company
1671 Worcester Street
Framingham, Massachusetts 01701

Dear Mr. Kay:

SUBJECT: SEP TOPIC II-2.C, ATMOSPHERIC TRANSPORT AND DIFFUSION CHARACTERISTICS FOR ACCIDENT ANALYSIS - YANKEE ROWE

Enclosed is the staff's final evaluation of SEP Topic II-2.C "Atmospheric Transport and Diffusion Characteristics for Accident Analysis" for Yankee Rowe. The staff has concluded that the values provided in your safety analysis dated June 30, 1981 are appropriate. Therefore, the values provided in the enclosed evaluation are to be used for all accident radiological consequence calculations.

This evaluation will be a basic input to the integrated safety assessment for your facility unless you identify changes needed to reflect the as-built conditions at your facility. This assessment may be revised in the future if your facility design is changed or if NRC criteria relating to this subject are modified before the integrated assessment is completed.

Sincerely,

Dennis M. Crutchfield

Dennis M. Crutchfield, Chief
Operating Reactors Branch No. 5
Division of Licensing

Enclosure:
As stated

cc w/enclosure:
See next page

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SEP TOPIC II-2.C
ATMOSPHERIC TRANSPORT AND DIFFUSION CHARACTERISTICS FOR ACCIDENT ANALYSIS

YANKEE ROWE

I. INTRODUCTION

The safety objective of this review is to determine the appropriate on-site and near-site atmospheric transport and diffusion characteristics necessary to establish conformance with the 10 CFR Part 100 guidelines. In particular, the short-term relative ground-level air concentrations (x/Q) are determined for use in estimating offsite exposures resulting from postulated accidents.

II. REVIEW CRITERIA

Section 100.10 of 10 CFR Part 100, "Reactor Site Criteria," states that meteorological conditions at the site and surrounding area should be considered in determining the acceptability of a site for a power reactor.

III. RELATED SAFETY TOPICS

Topic II-1.A, "Exclusion Area Authority and Control" provides the proper exclusion boundary distance over which the licensee has control. Various section XV topics utilize the atmosphere dispersion coefficients to determine the offsite radiological consequences of postulated accidents.

IV. REVIEW GUIDELINES

The atmospheric dispersion factors were calculated using a modified Gaussian dispersion model as outlined below. In order to account for the valley terrain at the site, dilution factors were calculated using a 10-sector downwind wind rose for both the EAB and LPZ. For all winds from the S clockwise through WSW cardinal wind direction sectors, it was assumed that effluents would remain in the valley. As such, winds from these four cardinal direction sectors were assumed to affect one "upstream" downwind sector. Likewise, winds from the N clockwise through ENE cardinal wind direction sectors were also assumed to remain in the valley and affect one 'downstream' downwind sector. Winds from the other eight cardinal wind direction sectors, W clockwise through NNW and E through SSE, were assumed to be cross-valley flows which affected the E through SSE and W through NNW downwind sectors, respectively.

The procedure for determining the dilution factors for the design basis accident evaluation reflects variations in atmospheric dispersion that occur as a function of wind direction frequencies and downwind receptor distances. Dilution factors were computed for each sequential hour of measured meteorological data and for receptors positioned in each of the ten downwind sectors. These hourly x/Q values were calculated using a modification of the Gaussian dispersion model outlined in Regulatory Guide 1.145. Plume centerline values were used to determine the short-term dilution factors (up through eight hours) and sector average values were used for the longer term dilution factors. The dispersion model for the plume centerline x/Q values considered the following effects:

1) Plume horizontal and vertical standard deviations were adjusted to account for building wake effects.

2) Lateral plume meander was allowed during periods of low wind speed and neutral and stable atmospheric conditions.

3) Lateral dispersion in the upstream and downstream downwind sectors was limited by the valley walls and included an increase in concentration due to multiple eddy reflections from the valley walls.

In addition, the sector width used to determine the hourly sector average x/Q values for the upstream and downstream downwind sectors was adjusted to account for the limited lateral dispersion potential due to the valley walls.

V. EVALUATION

The staff has reviewed and evaluated the Yankee Atomic Electric Company's (licensee) assessment dated June 30, 1981. Meteorological data collected at the Yankee ~~Rowe~~ site from January 1, 1980 through December 31, 1980 were used to determine the atmospheric transport and diffusion characteristics.

Short-term x/Q values for a ground-level release have been computed for various time intervals at the exclusion area boundary (EAB), a circle with a radius of 3,100 feet, and the outer boundary of the low population zone (LPZ), an approximately S-shaped boundary reflecting the fact that releases from the plant under certain meteorological conditions will remain within the Deerfield River valley.

Estimates of effluent plume dispersion and transport are complicated by the plant's location in the Deerfield River Valley whose sides rise over 800 feet above plant grade. There is evidence that the 32-foot wind sensors are often affected by localized nocturnal drainage winds flowing down the east slope of the river valley, thus biasing the lower level wind rose frequencies toward the east. As such, the 196-foot wind direction values were used to determine whether the wind flow for any given hour followed the valley or was cross-valley. The 32-foot wind speed values were used in the analysis. Vertical atmospheric stability was determined from the vertical temperature gradient between the 32-foot and 196 foot levels. Horizontal atmospheric stability was defined by fluctuations of the 196-foot horizontal wind direction (sigma theta) when winds were greater than 1.5 meters per second (mps) and by the vertical temperature gradient between the 32-foot and 196-foot levels when the wind speed was less than 1.5 mps.

Using the hourly x/Q values calculated as described above, average x/Q values for each downwind sector were then determined for successive overlapping time intervals of 1, 2, 8, 24, 96, and 720 hours corresponding to time periods of zero to one hour, one to two hours, zero to eight hours, eight to 24 hours, one to four days and four to 30 days, respectively. For each selected downwind sector and interval size, the averaging process began with the first hourly dilution value on record and was then repeated for the same interval size starting with each subsequent hour of dispersion data. In the averaging process, the only non-zero values within a given time interval which were considered in evaluating the average dilution factor for the interval were those hours during which the wind was blowing into the downwind sector of interest. The averaged x/Q values were then classified into groups as a function of interval size and downwind sector, and corresponding cumulative frequency distributions of non-zero values for each group were prepared. The x/Q value which was exceeded 0.5% of the total time was then determined from each group, and the maximum 0.5% downwind sector value from each time interval was chosen as the design-basis x/Q value for that time interval.

The follwing x/Q values were determined using the above model for an assumed ground level release for the various accident time intervals at the EAB and LPZ:

Time Period	Distance & Direction	CHI/Q (sec/m^3)
0 - 1 hours	EAB (3100 feet upstream)	2.8×10^{-4}
1 - 2 hours	EAB (3100 feet downstream)	2.3×10^{-4}
0 - 8 hours	LPZ (2 miles upstream)	2.8×10^{-5}
8 - 24 hours	LPZ (6 miles downstream)	1.9×10^{-5}
24 - 96 hours	LPZ (6 miles downstream)	1.6×10^{-5}
96 - 720 hours	LPZ (6 miles downstream)	1.0×10^{-5}

The staff has carried out an independent analysis of the dispersion characteristics of the site. This analysis has verified the adequacy of the applicant's relative concentrations for the exclusion area boundary. For the non-circular LPZ however, the staff has not attempted to utilize the more sophisticated techniques used by the applicant.

The staff was only able to calculate the relative atmosphere dispersion factors at the LPZ assuming a straightline trajectory. The shortest straightline LPZ in the upwind sector is 1551 meters; the corresponding shortest straightline distance in the downwind sector is also 1551 meters.

Following the guidance in Regulatory Guide 1.145, the staff has calculated generally equivalent concentrations compared with those presented by the applicant, but at the smaller distances.

VI. CONCLUSION

The staff concludes that the X/Q valves presented in Section V are appropriate for estimating exposures from postulated accidents and should be used in all accident calculations. This conforms to current licensing practice, no additional SEP is required.

M Boyle

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

July 13, 1982

Docket No. 50-29
LS05-82-07-021

Mr. James A. Kay
Senior Engineer - Licensing
Yankee Atomic Electric Company
1671 Worcester Road
Framingham, Massachusetts 01701

Dear Mr. Kay:

SUBJECT: SEP TOPICS II-1.A, EXCLUSION AREA AUTHORITY AND CONTROL; II-1.B, POPULATION DISTRIBUTION; II-1.C, POTENTIAL HAZARDS DUE TO HEARBY TRANSPORTATION, INSTITUTIONAL, INDUSTRIAL AND MILITARY FACILITIES; AND III-4.D, SITE PROXIMITY MISSILES - YANKEE NUCLEAR POWER STATION

Enclosed are the staff's final evaluations of SEP Topics II-1.A, II-1.B, II-1.C and III-4.D for the Yankee Nuclear Power Station. These evaluations are based on the safety analyses provided by your letters of March 18, 1982, May 1, 1981, April 9, 1981, and June 30, 1981, respectively. These evaluations compare your facility as described in Docket No. 50-29 with the criteria currently used for licensing new facilities.

The staff has concluded that the Yankee plant meets the acceptance criteria for the four topics.

These evaluations will be a basic input to the integrated safety assessment for your facility unless you identify changes needed to reflect the as-built conditions at your facility. The assessments may be revised in the future if your facility design is changed or if NRC criteria relating to these subjects are modified before the integrated assessment is completed.

Sincerely,

Ralph Caruso

Ralph Caruso, Project Manager
Operating Reactors Branch #5
Division of Licensing

Enclosure:
As stated

cc w/enclosure:
See page 2

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Mr. James A. Kay

cc
Mr. James E. Tribble, President
Yankee Atomic Electric Company
25 Research Drive
Westborough, Massachusetts 01581

Chairman
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Rowe, Massachusetts 01367

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U. S. Environmental Protection
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JFK Federal Building
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Resident Inspector
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ENCLOSURE

SYSTEMATIC EVALUATION PROGRAM
TOPIC II-1.A

YANKEE NUCLEAR POWER STATION

TOPIC: II-1.A, Exclusion Area Authority and Control

I. INTRODUCTION

The safety objective of this topic is to assure that appropriate exclusion area authority and control are maintained by the licensee as required by 10 CFR Part 100.

II. REVIEW CRITERIA

Section 100.3(a) of 10 CFR Part 100 requires that a reactor licensee have the authority to determine all activities within the designated area, including the exlusion and removal of personnel and property.

III. RELATED SAFETY TOPICS

Topic XIII-1, "Conduct of Operations" will assure that the licensee can adequately specify proper operation in routine, accident and emergency conditions. The topic is being covered as part of the NRC TMI Task Action Plan.

IV. REVIEW GUIDELINES

The review was conducted in accordance with the guidance given in SRP 2.1.2. The capability of the plant to meet the dose criteria of 10 CFR Part 100 at the exclusion area boundary will be evaluated in the Design Basis Event phase of the SEP review.

V. EVALUATION

The Yankee Nuclear Power Station is located in a valley in the town of Rowe, Massachusetts on the east bank of the Deerfield River, three-quarter of a mile south of the Vermont-Massachusetts border. The site consists of approximately 2,000 acres straddling the Deerfield River in the towns of Rowe and Monroe, Massachusetts and is owned, in fee, by Yankee Atomic Electric Company (YAEC), or an affiliate, the New England Power Company. YAEC is a subsidiary of New England Power Company. The site is shown in Figure 1.

The exclusion area boundary is defined by a 3,100 ft. radius centered on the reactor vapor container, with the exception of a small segment in the southern sector where the minimum distance is approximately 2,700 ft. This indentation into the 3,100 ft. exclusion radius is formed by an undeveloped corner of the Monroe State Forest and is situated behind an 1,800 ft. (MSL) ridge line with respect to the plant. Plant grade is 1,127 ft. (MSL). Figure 1 shows the exclusion area boundary.

All the land in the exclusion area is owned, including mineral rights, by Yankee Atomic Electric Company or New England Power Company, with the exception of a small parcel situated across the river and southwest of the plant which is owned by the Deerfield Specialty Paper Company. All the area within the exclusion boundary is under the control of YAEC. Written permission has been obtained from the paper mill in Monroe Bridge to have that portion of their land, which is within the 3,100 ft. radius, under Yankee control in the event of an incident. This piece of land is used as a disposal area by the paper mill and contains no permanent buildings or residents.

Two public secondary roads traverse the exclusion area. The closest is across the river from the plant and is approximately 1,500 ft. away at its closest point. This road runs north-south along the river between Monroe Bridge and Readsboro, Vermont. The second road, which connects to the main access to the plant, is approximately 2,500 ft. away at its nearest point and runs between Monroe Bridge and Rowe, south of the plant. Provisions have been made in the station's Emergency Plan to protect the public along these roadways should it become necessary in an emergency.

One house is located within the exclusion area. It is across the pond NNW from the plant, approximately 1,500 ft. away, on the river road. It is owned by the New England Power Company and occupied by one of its employees. Provisions have been made in the station's Emergency Plan to evacuate this residence if necessary in the event of an emergency.

VI. CONCLUSION

Based on the above evaluation, we conclude that YAEC has the proper authority to determine all activities within the exclusion area, as required by 10 CFR Part 100.

This completes the evaluation of this topic.

Yankee Nuclear Power Station
Site Area
Figure 1

SYSTEMATIC EVALUATION PROGRAM
TOPIC II-1.B

YANKEE NUCLEAR POWER STATION

TOPIC: II-1.B, Population Distribution

I. INTRODUCTION

The safety objective of this topic is to ensure that the previously-established low population zone and population center distance specified for the site are compatible with the current population distribution, and are in accordance with the guidelines of 10 CFR Part 100.

II. REVIEW CRITERIA

Sections 100.10 and 100.11 of 10 CFR Part 100, "Reactor Site Criteria" provides the site evaluation factors which should be considered when evaluating sites for nuclear power reactors. These sections include guidelines for determining the exclusion area, low population zone and population center distance.

III. RELATED SAFETY TOPICS

Topic II-1.A, reviews the licensee's control over the exclusion area. Various other topics will evaluate the capability of the plant to meet the dose criteria of 10 CFR Part 100 at the exclusion area boundary and low population zone. The adequacy of emergency preparedness planning for the area surrounding the plant including the low population zone is being assessed by the Commission in a separate review effort.

IV. REVIEW GUIDELINES

The review has been conducted in accordance with Standard Review Plan (SRP) Section 2.1.3, "Population Distribution".

V. EVALUATION

The population distribution in the vicinity of Yankee has been reviewed several times since the construction permit was issued. The safety objective of these reviews is to assure that the low population zone (LPZ) and population center distance specified for Yankee continue to be compatible with the current population distribution, and are in conformance with the guidance of 10 CFR Part 100.

The low population zone for Yankee has been defined as an S-shape area approximately 2 miles wide and centered on the Deerfield River, extending 2 miles upstream and 6 miles downstream from the plant. This area was originally selected based on the terrain and meteorology at the site. The total 1980 population within the low population zone is estimated to be approximately 260 persons, and has remained essentially unchanged over the last decade.

The nearest population center, as defined in 10 CFR Part 100, has remained unchanged over the operating life of the plant. The city of Pittsfield, Massachusetts, with a 1980 population of 51,942 and located 21 miles southwest of the site remains the nearest population center. The population center distance of 21 miles is greater than one and one-third times the LPZ distance as required by 10 CFR Part 100. Table 1 presents the 1960, 1970 and 1980 populations for all major cities and large towns within 50 miles, and shows growth has only been minor in nature or negative over recent history.

The rural nature of the towns in the immediate vicinity of the site has remained basically unchanged over the last two decades. Table 2 indicates the change in resident population from 1960 through 1980 for those towns totally or partially 5 miles of the site. The current population density for the 5-mile radius is approximately 26 persons per square mile. There has been no significant additions of either new housing or commercial, milita y or institutional installations in the immediate vicinity of the plant since the initial population review.

Within a 50-mile radius of the site, the total population has increased from an estimate of 1,397,666 in 1970 to 1,444,489 in 1980. This represents a growth of only 3.4 percent over this ten-year interval and reflects the stable nature of the population distribution in the region. Table 3 shows both the 1970 and 1980 population distributions within 50 miles by radial rings. Current estimates of the population by sector out to 50 miles are contained in the Yankee Nuclear Power Station, Emergency Plan", Section 4.2, Yankee Atomic Electric Company, April 1, 1981.

VI. <u>CONCLUSIONS</u>

The LPZ and population center distance are in conformance with the criteria of 10 CFR 100 and SRP Section 2.1.3. Therefore, the Yankee Plant meets current licensing criteria and the review for this topic is complete.

TABLE 1

POPULATION OF MAJOR COMMUNITIES

WITHIN 50 MILES OF YANKEE

Community	Distance and Direction from Site	Population 1960(a)	Population 1970(a)	Population 1980(b)
(Massachusetts)				
Agawam	46 miles SSE	15,718	21,717	26,281
Amherst	30 miles SE	13,718	26,331	33,210
Chicopee	40 miles SSE	61,553	66,676	55,048
Holyoke	35 miles SSE	52,689	50,112	44,819
Northampton	28 miles SSE	30,058	29,664	29,128
Pittsfield	21 miles SW	57,879	57,020	51,942
Springfield	48 miles SSE	163,905	163,905	152,212
Westfield	38 miles S	26,302	31,433	36,356
West Springfield	42 miles SSE	24,924	28,461	26,960
(New Hampshire)				
Keene	32 miles ENE	17,562	20,467	21,385
(New York)				
Albany	45 miles W	129,726	115,781	101 770
Bethlehem	44 miles W	18,936	23,427	[illegible],301
Colonie	44 miles W	52,760	69,147	74,534
Troy	38 miles W	67,492	62,918	56,614

(a) 1970 Census of Population: Characteristics of the Population; Bureau of the Census, U.S. Department of Commerce, Vol. 1, 1973.

(b) 1980 Census of Population and Housing: Preliminary Reports; Bureau of Census, U.S. Department of Commerce, 1980.

TABLE 2

POPULATION OF TOWNS

WITHIN 5 MILES OF YANKEE

Town	Population 1960[a]	1970[a]	1980[b]
(Massachusetts)			
Rowe	231	277	325
Monroe	210	216	173
Florida	569	672	731
Charlemont	897	897	1,141
Heath	304	383	482
(Vermont)			
Readsboro	783	638	638
Whitingham	838	1,011	1,043

(a) 1970 Census of Population: Characteristics of the Population; Bureau of the Census, U.S. Department of Commerce, Vol. 1, 1973

(b) 1980 Census of Population and Housing: Preliminary Reports; Bureau of Census, U.S. Department of Commerce, 1980.

TABLE 3

POPULATION DISTRIBUTION

WITHIN 50 MILES OF YANKEE

Ring (Miles)	Cumulative Population 1970[(a)]	1980[(b)]
0 - 1	98	116
0 - 2	229	295
0 - 3	558	744
0 - 4	1,070	1,445
0 - 5	1,618	2,058
0 - 10	17,654	24,431
0 - 20	100,485	122,689
0 - 30	260,899	287,663
0 - 40	606,734	655,622
0 - 50	1,397,666	1,444,489

(a) 1970 Census of Population: Characteristics of the Population; Bureau of the Census, U.S. Department of Commerce, Vol. 1, 1973

(b) 1980 Census of Population and Housing: Preliminary Reports; Bureau of Census, U.S. Department of Commerce, 1980.

SYSTEMATIC EVALUATION PROGRAM
TOPIC II-1.C

YANKEE NUCLEAR POWER STATION

TOPIC: II-1.C, Potential Hazards Due to Nearby Transportation Institutional Industrial and Military Facilities

I. INTRODUCTION

The safety objective of this topic is to ensure that the integrity of the safety-related structures, systems and components would not be jeopardized due to the potential for hazards originating at hearby facilities.

II. REVIEW CRITERIA

General Design Criterion 4, "Environmental and Missile Design Basis," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Licensing of Production and Utilization Facilities," requires that nuclear power plant structures, systems and components important to safety be appropriately protected against events and conditions that may occur outside the nuclear power plant.

III. RELATED SAFETY TOPICS

Topic III-4.D, "Site Proximity Missiles" reviews the extent to which the facility is protected against missiles originating from offsite facilities.

IV. REVIEW GUIDELINES

The review was conducted in accordance with the guidance given in Standard Review Plan (SRP) Section 2.2.1-2.2.2, "Identification of Potential Hazards in Site Vicinity."

V. EVALUATION

Industrial activity in the vicinity of the Yankee station is minimal. The only manufacturing or storage facility within five miles is the Deerfield Paper Company. They do not store or use any hazardous chemicals that could result in toxic fumes upon release into the atmosphere. The closest industrial gas supplier is H.A. George, located approximately 10 miles southwest of Yankee. They have two large LPG storage tanks having maximum capacities of 12,000 and 20,000 gallons and three LPG delivery trucks having maximum capacities of 1,200, 2,400 and 2,600 gallons. The maximum quantity of fuel oil is 75,000 gallons located in above ground storage tanks. The separation distance between the H. A. George gas storage location and the nuclear plant is considered adequate to preclude accidents affecting the safe operation of Yankee.

No mining or quarry operations have been identified as being within 10 miles of the station. For this reason possible truck-size shipments of explosives on nearby roadways were not considered as potential hazards to safe plant operation.

The closest military facility is Westover Air Force Base located outside of Springfield, approximately 50 miles southeast of Yankee. A discussion of the effects of an on-site aircraft impact can be found in SEP Topic Section III-4.D, "Site Proximity Missiles."

Hazardous chemicals have been identified as being shipped on an east-west rail line of the Boston and Maine Railroad, located at its closest point approximately four miles from the south side of the site. The evaluation of possible control room infiltration of toxic fumes from potentially hazardous chemicals was done according to methodology in Reference (1) and criteria established in Reference (2).

An analysis was done for all shipments defined by Reference (2) as being frequent, i.e., 30 per year for rail traffic, and was based on the maximum concentration accident, in which the quantity of the hazardous chemical considered was the instantaneous release of the total contents of the largest tank car. This is the worst case compared with partial ruptures in which the contents leak out in a continuous steady flow. The concentration buildup on the control room was the sum of the concentration due to the immediate, highly concentrated puff for low boiling point chemicals and the continual vaporization of any remaining fluid.

Results indicate that hazardous chemical shipments via the rail line do not present a significant threat to the safe operation of Yankee.

The only major highway by Yankee is Route 2, an east-west road located at its closest point six miles from the station. There is not data presently available that indicates frequencies, quantities and types of hazardous material that are shipped via tank trucks using this roadway. To determine the possible consequence of a truck accident involving hazardous material, an analysis was done assuming a sixteen ton chlorine tank truck rupture. Chlorine was used because of its high toxicity when compared to other chemicals and its production (6th among all chemicals in 1972[3]) exceeds other highly toxic, volatile chemicals such as hydrogen cyanide and phosgene by nearly two orders of magnitude. Results indicate that a chemical truck accident on the highway would not affect the safe operation of the plant.

There are presently no pipelines, gas or oil production fields, underground storage facilities, refineries or major waterways in the town of Rowe or in the surrounding area towns.

VI. <u>CONCLUSIONS</u>

We have concluded that Yankee is adequately protected and can be operated with an acceptable degree of safety with regard to industrial, transportation and military activities in the vicinity of the plant.

This completes the evaluation of this topic.

References

1. NUREG-0370, "Toxic Vapor Concentrations in the Control Room Following a Postulated Accidental Release," June 1979.

2. Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," June 1974.

3. "The Risk of Catastrophic Spills of Toxic Chemicals," UCLA-ENG-7425, J.A. Simmons, R.C. Erdmann, B.N. Naft, May 1974.

SYSTEMATIC EVALUATION PROGRAM
TOPIC III-4.D

YANKEE NUCLEAR POWER STATION

TOPIC: III-4.D, Site Proximity Missiles (Including Aircraft)

I. INTRODUCTION

The safety objective of this topic is to ensure that the integrity of the safety-related structures, systems and components would not be jeopardized due to the potential for a site proximity missile.

II. REVIEW CRITERIA

General Design Criterion 4, "Environmental and Missile Design Basis," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Licensing of Production and Utilization Facilities," requires that nuclear power plant structures, systems and components important to safety be appropriately protected against events and conditions that may occur outside the nuclear power plant.

III. RELATED SAFETY TOPICS

Topic II-1.C, "Potential Hazards or Changes in Potential Hazards Due to Transportation, Institutional, Industrial and Military Facilities" provides a description of the potential missile hazards.

IV. REVIEW GUIDELINES

The review was conducted in accordance with the guidance given in Standard Review Plan (SRP) Section 2.2.3, "Evaluation of Potential Accidents," 3.5.1.5, "Site Proximity Missiles (except Aircraft)," and 3.5.1.6, "Aircraft Hazards."

V. EVALUATION

The scope of possible hazardous activities on the vicinity of the Yankee plant has been discussed in SEP Topic II-1.C, "Potential Hazards Due to Nearby Industrial, Transportation, Institutional and Military Facilities." As indicated, there is minimal industrial activity in the plant vicinity. The separation distance and valley terrain between the plant and any industrial facilities, highways, railroads, gas pipelines, or military facilities is such that the risk associated with potential missiles from these concerns are well within the SRP 2.2.3 guidelines.

In addition to the review of fixed facilities and ground transportation routes in the site area, the potential of aircraft accident generated missiles has also been evaluated in detail. The methodology employed in this analysis is the same as that outlined in SRP 3.5.1.6.

There are four airports within thirty miles of Yankee: (1) Harriman-and West (North Adams), (2) Bennington State, (3) Pittsfield, and (4) Turners Falls. Each can be described in general as being small airports typically handling light, single-engine, private aircraft.

Table 1 summarizes each airport's annual operations. As shown, none of the airports are within ten miles of Yankee, and all of the airport reported annual operations are well within the 1,000 times distance squared criteria of SRP 3.5.1.6 which, if exceeded, would indicate the possible need for further analysis of aircraft from these airports effecting plant operations.

Table 1

	Distance (d) Statute Miles	$1000 \times d^2$	Annual Number of Operations
North Adams	12.0	144000	32500
Bennington	19.0	361000	10950 (a)
Pittsfield	28.0	784000	50000
Turners Falls	22.0	484000	34873

(a) Annual operation information obtained directly from Bennington Airport Manager, April 1, 1981.

In addition to the above noted airports, there are two federal airways, V2-14 and J16-94, which could bring aircraft near the plant site. V2-14 is used by aircraft below 18,000 feet, whereas J16-94 is used by aircraft at altitudes of 18,000 feet and above. Both airways have a total width of 8 nautical miles (9.2 statute miles); i.e., 4 nautical miles each side of centerline. Yankee is located approximately 2.5 nautical miles north of the V2-16 centerline, and 5 nautical miles north of the J16-94 centerline. In estimating the annual number of aircraft that may pass near the site due to these airways, a count was made of the IFR traffic on both these airways for the "peak traffic day" of 1980 (August 22). Federal Aviation Administration radar records associated with these corridors indicate a total of 155 aircraft could have flown near the plant site during the peak day. Based upon the peak day traffic, an annual estimate of 56,575 aircraft passing near the site was calculated. Employing the analytical model given in SRP 3.5.1.6, it is calculated on a conservative basis that the overall probability of an aircraft associated with these air corridors striking the plant is approximately 1.4 x 10^{-7} per year. This is an acceptable level of risk in accordance with the acceptance criteria of SRP 2.2.3.

In calculating the risk probability, an effective plant area of 0.0075 square miles was used. This was determined by assuming an aircraft crash angle of 30 degrees relative to the principal plant structures, including non-safety related buildings attached to the plant. Since Yankee is located in a valley, the crash angle was based on the kinds of aircraft identified within the V2-14 and J16-94 airways, and the width of the valley and height of the mountains surrounding the plant. An inflight crash rate of 3 x 10^{-9} per aircraft mile was used in the calculation (SRP 3.5.1.6). No information was identified from the FAA on future growth of traffic in these corridors. However, since the calculated probability of 1.4 x 10^{-7} conservatively assumes that a single day peak traffic load in the corridors is maintained throughout the year, any future real growth in aircraft activities in these corridors over the remainder of plant life would not be expected to change significantly the calculated risk factor.

VI. CONCLUSIONS

Based on our review, we conclude that operation of the Yankee plant does not present an undue risk to the health and safety of the public as a result of aircraft and site proximity missile hazards.

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

September 30, 1982

Docket No. 50-29
LS05-82-09-093

Mr. James A. Kay
Senior Engineer - Licensing
Yankee Atomic Electric Company
1671 Worcester Road
Framingham, Massachusetts 01701

Dear Mr. Kay:

SUBJECT: SEP TOPICS II-3.A, HYDROLOGIC DESCRIPTION, II-3.B, FLOODING POTENTIAL AND PROTECTION REQUIREMENTS, II-3.B.1, CAPABILITY OF OPERATING PLANT TO COPE WITH DESIGN BASIS FLOODING CONDITIONS, II-3.C, SAFETY-RELATED WATER SUPPLY (ULTIMATE HEAT SINK) - YANKEE NUCLEAR POWER STATION

Enclosed is the staff's final evaluation of SEP Topics II-3.A, II-3.B, II-3.B.1, and II-3.C. The evaluation is based upon the safety analysis report which you supplied on June 16, 1982, technical reports supplied by you and our consultant, Franklin Research Center, and other information available on Docket No. 50-29. Appendix A to the evaluation is our consultants technical evaluation report on the hydrologic considerations at Yankee and Appendix B is a copy of our draft flood study for Yankee.

The safety evaluation concludes that Yankee does not meet current NRC acceptance criteria for the hydrology topics. Specifically, the major area of deviation from current criteria is that of the effects of flooding due to a probable maximum flood (PMF). The NRC and YAEC evaluations of flooding potential were in disagreement because of the differences in the estimates of probable maximum precipitation (PMP) and Harriman Dam spillway capacity. These estimates led the NRC and YAEC to different safety conclusions on the flooding potential due to a PMF. Appendix C to the evaluation is a discussion of the conservatisms involved in estimating a PMP and a discussion of the differences between the NRC and YAEC PMF estimations.

The deviations from the hydrologic topics acceptance criteria will be resolved during the Integrated Safety Assessment for Yankee. This resolution will be integrated into the findings for SEP Topics II-4.E, "Dam Integrity," and III-3.A, "Effects of High Water Level on Structures."

The need for any further analysis or plant modifications will be made during the Integrated Safety Assessment. The evaluation may be revised

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P

Mr. James A. Kay

cc
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HYDROLOGIC ENGINEERING SAFETY EVALUATION
FOR SYSTEMATIC EVALUATION PROGRAM

Topic II-3.A, Hydrologic Description
Topic II-3.B, Flooding Potential and Protection Requirements
Topic II-3.B.1, Capability of Operating Plants to Cope with Design Basis Flood Conditions
Topic II-3.C, Safety-Related Water Supply (Ultimate Heat Sink)

Plant Name: Yankee Rowe Nuclear Power Station
Licensing Stage: Operating
Docket Numbers: 50- 029
Prepared by: Gary B. Staley, Hydrologist, HGEB

I. INTRODUCTION

The Systematic Evaluation Program (SEP) was established by the Nuclear Regulatory Commission (NRC) to evaluate the safety of 10 older nuclear power plants. The program evaluates the plants against current licensing criteria with respect to 137 selected topics.

The hydrologic topics provide:

- a brief description of the hydrologic features of the site and surrounding area, plant facilities and the design bases used for construction. Additionally both surface and groundwater and their interfaces with plant safety-related buildings and systems are described.

- Design bases floods for the plant are developed, using current criteria. and compared to the design bases events used when the plant was built. Deviations and their safety significance are discussed. Acceptability of current features are noted where applicable.

- Where physical protection is used to prevent plant flooding, the design and design bases are reviewed and compared to current criteria. The variations, if any, and their safety significance with respect to structural and equipment distress are discussed.

- The design basis groundwater level for hydrostatic loading is determined in accordance with current criteria and compared to the value used for design.

- Existing emergency plans or procedures and technical specifications related to flooding or safety-related water supply are reviewed and compared to current criteria. Deficiencies are noted and, where possible, acceptable fixes are recommended. Where emergency plans or technical specifications do not exist but are a potential solution to a problem, they are discussed and recommendations made, if appropriate.

- As reviewed here, the Ultimate Heat Sink (UHS) consists of water sources for the cooling water system, necessary retaining structures (e.g., a pond with its dam or a cooling tower supply basin), and the canals or conduits connecting the sources with (but not including) the cooling water system intake structures. The existing UHS is compared to current criteria with respect to available supply and maximum temperature, and if deficiencies exist, they are discussed and acceptable solutions recommended, if possible.

The information used to perform the reviews was gathered from the licensee's files, NRC files, other agencies, and the site visit. In some cases, detailed information was not available. In such cases, the staff and its consultants conservatively estimated these parameters required for analysis. For this evaluation the staff consultant was the Franklin Research Center.

II. REVIEW CRITERIA

Current licensing criteria for nuclear power plants, related to the SEP topics addressed in this report, were developed from the Code of Federal Regulations: 10 CFR Part 50, "Licensing of Production and Utilization Facilities," and General Design Criterion 2, 4, 5 and 44 of Appendix A, "General Design Criteria"; 10 CFR Part 100, "Reactor Site Criteria" and Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants."

The criteria which are applicable are (1) Standard Review Plans 2.4.1, 2.4.2, 2.4.3, 2.4.4, 2.4.5, 2.4.6, 2.4.7, 2.4.8, 2.4.9, 2.4.10, 2.4.11, 2.4.12, 2.4.14, 3.4.1, and 9.2.5 (Ref. 1); (2) Regulatory Guides 1.27, 1.59, 1.70, 1.102, and 1.127 (Ref. 2); and (3) American National Standards Institute (ANSI) Standard N170-1976 (Ref. 3).

III. RELATED SAFETY TOPICS AND INTERFACES

The effects of high surface water and ground water (pertaining to structural strength of building walls, loss of important equipment and its effect on the plants' ability to safely shutdown, etc) are outside the scope of the hydrologic evaluation. However, the levels of flood and ground water are determined in this evaluation and given to the structural and system reviewers for their use.

SEP interface topics are:

II-4.D - Stability of Slopes

II-4.E - Dam Integrity

II-4.F - Settlement of Foundations and Buried Equipment

III-1 - Classification of Structures, Components and Systems

III-3.A - Effects of High Water Level on Structures

III-3.B - Structural and Other Consequences of Failure of Underdrain Systems

III-3.C - Inservice Inspection of Water Control Structures

III-6 - Seismic Design Considerations

VII-3 - Systems Required for Safe Shutdown

VIII-2 - On-Site Emergency Power Systems - Diesel Generator

IX-3 - Station Service and Cooling Water Systems

XVI - Technical Specifications

IV. REVIEW GUIDELINES

The hydrologic issues identified in the Introduction are developed from design information for the nuclear power plant and from many sources containing hydrologic information for the site. Design bases (elevation of floods, depths of precipitation flooding, elevation of ground water and amounts of available cooling water) are determined and their conformance with or degree of departure from the current criteria is assessed. The Standard Review Plans and Regulatory Guides identified in Section II direct a complete evaluation of all issues and suggest or reference appropriate technical evaluation methods.

Regulatory Guides 1.27, 1.59 and 1.102 have been specifically identified as needing consideration for backfit on operating reactors. These guides are used in determining whether the facility design complies with current criteria or has some equivalent alternatives acceptable to the staff. The acceptability or nonacceptability of any deviations identified in this evaluation and the need for further action will be judged during the integrated assessment for this facility.

ɪALUATION

ıff's consultant, Franklin Research Center (FRC), has reviewed ble background information and made independent analyses necessary ɜpare the report, "Technical Evaluation Report, Hydrological derations, Yankee Rowe Nuclear Power Station" dated August 6, 1982 C5257-426).(4) (Reference (4) is attached to this report as ɜndix A). This work was performed under NRC Contract No. 03-79-118 ı provides the assessment for Systematic Evaluation Program (SEP) pics: II-3.A, Hydrologic Description; II-3.B, Flooding Potential and ɾotection Requirements; II-3.B.1, Capability of Operating Plants to ːope with Design Basis Flood Conditions and, II-3.C, Safety Related Water Supply (Ultimate Heat Sink (UHS)). It is noted that the assessment of Deerfield River flooding was performed by the NRC staff at an earlier date and incorporated into the TER by reference 5 and attached as Appendix B.

Background

Early in the SEP program it was recognized by the staff that Harriman Dam might post a severe flood threat to the Yankee Rowe Nuclear Plant and that the flood potential had never been evaluated by the staff. Thus the evaluation of Deerfield River flooding was escalated and the study was initiated by the staff in early 1979. The Licensee completed a Draft Probable Maximum Flood Study in January 1980.(6) The staff completed a Draft Flood Study for the Upper Deerfield River Basin in January 1981.[5] The licensee submitted a final Design Basis Flood analysis in March 1981 and dated October 1980.[7]

In June 1981, the staff contracted with Franklin Research Center to complete the SEP hydrologic engineering topics (II-3.A, B, B.1 and c) and the evaluations are contained in their Technical Evaluation Report of August 6, 1981.[4]

The most significant flood potential for the Yankee Rowe Nuclear Power Plant is the Harriman Dam which is situated on the Deerfield River about six miles upstream of the plant site. The dam and hydroelectric plant are owned and operated by the New England Power Co. and licensed by the Federal Energy Regulatory Commission (FERC). Since FERC has regulatory authority for Harriman Dam, the staff has maintained a liaison with that agency and especially with regard to evaluations or conclusions that involve Harriman Dam.

Topic II-3.A Hydrologic Description

This topic provides a brief description of hydrologic features, related plant facilities and design bases used for plant construction.

The Yankee Rowe Nuclear Power Station is situated on the Deerfield River in northwestern Massachusetts near the town of Rowe, in Franklin County. The plant is on the left (south) bank of Sherman reservoir just upstream of Sherman dam and about six miles downstream of Harriman Dam.

The site is potentially subject to flooding from large floods (including dam failure floods) on the Deerfield River and from local intense precipitation on the site area and small tributaries south of the site.

Yard grade in the vicinity of the plant proper is 1127.7 feet msl and 1119.7 feet msl at the screenhouse. Entrance floor elevation for the service, Primary Auxiliary and turbine buildings is 1128.3 ft msl, and for the screenhouse is 1124.7.

Information pertaining to the groundwater regime at the site is insufficient to extrapolate design basis levels.

The Deerfield River Basin upstream of the site is heavily wooded with fairly steep slopes, both along the central channel and the valley side slopes. Both Harriman (upstream) and Sherman (downstream) dams are important basin features with respect to site flooding potential and safety related water supply. Both dams are semi-hydraulic fill. Harriman Dam is about 200 feet high with a morning glory type spillway[1/] and Sherman Dam is about 100 ft high with a weir overflow spillway.[2/]

[1/] Vertical circular crest narrowing to throut section, then 90° bend to horizontal exit conduit. Design allows fairly constant discharge over wide range of heads.

[2/] Horizontal ogee crest narrowing to chute spillway. Design typified by wide range in discharge for small range in heads.

Local tributaries in the vicinity of the site include Wheeler Brook and two unnamed tributaries which drain the steep hillsides south and east of the plant. Wheeler Brook drains to Sherman Pond just northeast of the plant and the two unnamed tributaries drain partially across the site and partially east and west of the site. These drainage sub-basins have very steep slopes and are heavily wooded. There would be virtually no warning time for potential floods from these sources.

Some areas of the plant yard are between 1 and 2 feet below the top of Sherman Dam, while near the screenhouse it is much lower at elevation 1119.7 ft msl. The average slope across the plant site is 1.2%. Two independent storm drain networks exist on the site: one drains from the northeast area of the site into Sherman Pond and the other drains from the southwest area into the valley below Sherman Dam. There are two flood protection dikes, with top elevations about 2 feet above the crest of Sherman Dam, to protect the plant from some intermediate Deerfield River floods. The dikes run from Sherman Dam to the turbine building and from the service building to the bluff.

The design bases used for plant construction are listed below.

- The design basis groundwater elevations are unknown. Where specific groundwater levels and fluctuations are unavailable, it is the staff's position that ground surface elevation be used for structural evaluations.

- The original design basis for flooding (local and Deerfield River) is unknown.

- Building roofs are designed for 40 pounds per square foot which is equivalent to 7.7 inches of ponded water.

- The flood wall and building walls that serves as part of the floodwall were not designed for any floodwater loading. Since this feature does not provide an acceptable level of flood protection under the current NRC criteria, its hydrostatic and hydrodynamic design basis may be a moot point. However, if this feature did provide an acceptable level of protection, the required hydrostatic and hydrodynamic design basis would be the most severe combination of static and dynamic conditions; i.e., maximum stillwater level with wind waves superimposed.

- The design low water level for the UHS is reservoir elevation 1090.7 ft msl in Sherman Pond which is the lowest elevation at which the service water pumps can draw water.(10)

- The licensee uses operating procedure OP-3006 as an "active" flood protection measure to protect the screenhouse from Deerfield River flooding.

Topic II-3.B Flooding Potential and Protection Requirements

The purpose of this topic is to identify the plant and site design basis flood levels resulting from all potential flood sources external to the plant and site, using current NRC licensing criteria. This topic also includes the identification of the design bases groundwater levels for use in structural analyses.

Groundwater

The licensee has submitted some fragmented information on groundwater(8) that indicates a normal level of about 1110 ft msl (less than 10 feet below ground surface) around the screenwell; levels of 1110 to 1125 near the service building and levels of 1124 to 1130 near containment. Fluctuations would appear to be from 2 to 15 feet. Based on the partial groundwater information submitted, it appears that if the available information were properly organized and documented by the licensee, it might be possible for the staff to determine upper limit groundwater levels for use in structural evaluations. However, based on information currently available, we conclude that current NRC criteria would require that groundwater must be assumed at grade elevation for all structural evaluations.

Deerfield River Flooding

There are several possible scenarios with potential to flood the Yankee Rowe site. Excessive rainfall in the upper basin could overtop and fail Harriman Dam and the resulting flood wave could inundate the site. Excessive rainfall below Harriman Dam could exceed the capacity of Sherman Reservoir and if the dam does not fail when overtopped, the plant could be flooded. A non-hydrologic failure (seismic, founction, piping, etc.) of Harriman Dam when the reservoir is near the spillway crest, could flood the Yankee Rowe site.

all Induced Floods

the staff[5] and the licensee[7] have developed detailed hydrologic ff models of the Upper Deerfield River Basin to evaluate the od potential. The staff and licensee are in close agreement on features of the model, except for the spillway capacity for rriman Dam and the value of Probable Maximum Precipitation (PMP) or the Upper Deerfield River Basin.

Spillway Capacity - In our calculation, the staff has limited the spillway capacity of Harriman Dam to 75% of the theoretical capacity due to possible debris blockage and flow reduction at reservoir levels greater than the design value. The details of the justification for this reduction are discussed in Appendix B.

The licensee has made a number of analytical studies for the spillway and concludes that the existing capacity is about 6% less than was predicted by early model studies. He also concludes that debris will not cause any reduction in spillway capacity. Although the licensee has provided some interesting and pertinent discussion to support their position, the staff is convinced that there is a reasonably good chance of reduced spillway flow during runoff from extreme flood events.

The 25% reduction in spillway capacity is about 8,500 cfs when the pool level approaches the dam crest elevation. This reduced discharge amounts to a storage volume of about 4,300 ac ft in a 6 hour period. Six hours is the approximate time the reservoir capacity flood (13 inch rainfall) stays near the dam crest. In terms of reservoir stage it would be about two feet of stage when the pool level is at the dam crest elevation.

Probable Maximum Precipitation - Various estimates of PMP have been made for the Upper Deerfield River Basin. The following table shows a comparison of the estimates.

Table 1
PMP Estimates

Source	NRC Staff			Licensee			FRC
Method[3/]	HR# 51	HR# 33	Trans[1/] position	Hersh- field	Trans[2/] position	Gumbel 10^{-7}	Transposition
Area (sq mi)	236	236	236	Point	200	Point	236
Duration (hrs)	24	24	24	24	24	24	24
Depth (inches)	21.9	18.9	16.5	17.9	14.3	15.6	14.7

1/ Lower Limit Site Specific

2/ Based on transposition of many storms

3/ HR# 51 and 33 refers to generalized PMP estimates from NOAA publications. Transposition refers to one or more storms maximized and moved to the Deerfield Basin. Hershfield and Gumbel are statistical methods.

The National Oceanic and Atmospheric Administration (NOAA) Hydrometeorological Report (HR) Numbers 33(12) and 51(13) provide generalized PMP estimates east of the 105th Meridian. HR #51 identifies appalachian mountain areas, which include the Deerfield River Basin, where the generalized estimates may be either too high or too low due to orographic and/or terrain effects. Since NOAA is recognized by the technical community as expert in the field, there is no reason to use estimates other than HR #51 or 33 unless the estimates consider orographic effects. None of the estimates in table 1 include comprehensive analyses of the appalachian region orographic effects. It is the staff's conclusion that the 236 square mile, 24 hour PMP of 18.9 inches from HR #33[1/] is the current design basis PMP for the Upper Deerfield River Basin to be used in evaluating the flood potential at the Yankee Rowe Nuclear Plant Site.

<u>Comparison of Staff and Licensee Study Results</u> - The details of the staff's flood study for the Upper Deerfield River Basin and Yankee Rowe site are contained in reference 5. Essentially the results of our studies showed that for basin rainfall in excess of about 13 inches in 24 hours, Harriman Dam will be overtopped and assumed to fail. The resultant flood wave would also overtop and fail Sherman Dam. Since our estimate of PMP is 18.9 inches in 24 hours, we conclude that Harriman Dam would fail as a result of this event. We estimate that the maximum flood level at the Yankee Rowe Site would be about elevation 1175 ft msl or about 45 feet over plant grade.

[1/] Since Hydrometeorological Report No. 51 was not fully reviewed by major federal agencies, such as NRC, when staff studies were initiated, it was not adopted for the study.

The licensee's most recent flood study for the Yankee Rowe site[7] develops what they term a "Design Maximum Rainfall ([illegible])" of 14.3 inches in 24 hours for a 200 square mile area. [illegible]terpret their discussion to imply that this value was [illegible] due to basin shape, to 13 inches in 24 hours for a 20[illegible] sq[illegible] area. The licensee states that this rainfall will [illegible]man pool level of 1520.9 or 0.3 feet below the [illegible] Runup from wind waves would be 1 to 2 feet for several hours. The licensee concludes the dam will not breach. The licensee further states that this DMR will produce a Sherman Reservoir level that will exceed the dam crest by 2.0 feet. The dam is assumed to fail with two feet of overtopping flow. Thus the stillwater level at the site would be 1131.7 ft msl and maximum wave runup would be to elevation 1136.6 ft msl. The plant flood dike is constructed to a crest elevation of 1132.0 ft msl. The licensee states that they can and will protect the plant against these maximum failures.

Although the licensee's study does not specifically conclude that the site meets current NRC criteria with respect to flooding, it does infer that the site should be acceptable with respect to flooding.

Listed below are what we consider to be the significant differences between the two studies.

- The most significant difference is the value for PMP or design basis rainfall. The staff contends that the current criteria design basis rainfall is 18.9 inches in 24 hours for a 236 square mile area. The licensee has calculated a Design Maximum Rainfall (DMR)[1/] of 14.3 inches in 24 hours for a 200 square mile area. We conclude that this was further reduced to 13 inches (91.2%) based on basin shape. The staff made this same calculation for a storm centered over the basin and only calculated a 1% reduction. The licensee (7) performed a storm transposition study and an orographic effects study. The staff has reviewed these studies and concludes that the storm transposition studies did not include some major storms and in fact is probably considerably less comprehensive than what was done by NOAA for the Generalized PMP studies (12) (13). We also conclude that the orographic effects study[(7)] does not support the licensee's conclusions. It shows a general decrease in rainfall toward the north or northwest or away from the coast in New England. The generalized PMP studies show the same pattern. Consequently the staff has concluded that no information has been provided to support either an increase or decrease from the generalized PMP for the Deerfield River Basin.

- Of somewhat less significance is the assumptions for Harriman Dam spillway capacity. The staff assumed a 25% reduction for possible debris blockage and/or undesirable flow characteristics at high heads. The licensee assumed a 6% reduction based on analytical studies of flow characteristics. The 19% difference

[1/] The staff is not aware of a definition for DMR in flow reduction is about 6500 cfs. We estimate that this would amount to about 2 or 3 feet of reservoir stage during the full storm period.

- There is a difference in staff and licensee approach regarding the assumed breaching of earth dams when they are overtopped by flood flows.

Our assumption of whether or not a dam fails and how fast a breach enlarges is based to a large extent upon which scenario produces the most conservative result. The licensee's dam failure assumptions do not, however, always lead to conservative results. For example, the licensee (7) assumed that Sherman Dam would fail when overtopping flows reach a depth of two feet. While this is a reasonable assumption, it has not been shown that the dam would fail when overtopped by 2 feet. It might survive even greater overtopping. The longer this dam is assumed to remain unfailed and the slower the breach rate wnen finally failed, the higher the calculated water level at the site will be. In such situations, the staff will assume that the dam does not fail or it fails at a very slow rate, unless evidence to the contrary is provided.

Non-precipitation Induced Dam Failure Floods

In addition to failure by overtopping, dams can fail for other reasons such as seismic, piping or foundation failures. These types of failure generally result in an almost instantaneous release. The licensee has analyzed this failure scenario[9] for Harriman and Sherman Dams. The staff has discussed both the unsteady flow model (DAMBRK) and assumptions made in its use with the licensee and we generally concur in the results. We have minor reservations with the assumption of a full width breach of Sherman Dam in 5 minutes which is somewhat non-conservative but probably reasonable, given the other assumptions in the study. The results of this study show the potential flood levels at the Yankee Rowe site to vary from 1153.1 to 1173.8 feet MSL or 23 to 44 feet above plant grade.

Local Flooding

Local flood evaluations were based on the PMP from Hydrometeorological Report No. 33. Figure 1 of Appendix A shows the location and extent of the three tributaries that drain through or near the site. Wheeler Brook (trib #3) enters Sherman Pond north of the site and is separated by a natural divide and will not flood the site. Tributary #2 flows to Wheeler Brook and also will not flood the site. Tributary #1 does flow through the site and would have a peak discharge of 1329 cfs for the PMP. This flow will produce a flood depth of about 3 feet at the Diesel Generator and turbine buildings. Due to lack of information pertaining to the limiting flood elevation of safety-related equipment within these buildings, it is not known what the effects of this flood level would be.

The turbine building roof is the only roof for safety-related structures which is not protected against the PMP. The roofs are designed for 40 psf (7.7 inches of ponded water). The parapets around the roof of the turbine building are 21 inches high. If roof drains were blocked during the PMP, water could pond to 21 inches or 173 percent of the design basis live load.

Topic II-3.B.1 - Capability of Operating Plants to Cope With Design Basis Flood Conditions

Protection against floods can be accomplished by implementing emergency procedures and technical specifications. This topic focuses on the adequacy of the Yankee Rowe Plant emergency procedures to provide for safe shutdown and cooldown of the reactor during and after a severe river flood. Further, this topic addresses the need for other emergency plans and technical specifications to limit conditions of operation.

The licensee has a procedure designed to protect the intake structure from flood levels that are about 2.0 feet over the crest of Sherman Dam. Appendix A provides a critique of this procedure, however since this procedure will not protect the plant from a PMF or seismic dam failure flood on the Deerfield River, it does not meet current NRC criteria. Furthermore emergency procedures are not an acceptable alternative to hardened flood protection if sufficient warning time is not available, which is the case for seismic dam failure floods and local floods.

The Yankee Rowe Plant also needs an emergency plan and technical specifications to cover the loss of Ultimate Heat Sink (UHS) (Sherman Pond).

The feasibility of emergency plans and technical specifications will have to be determined during the integrated assessment. If feasible, the staff will work with the licensee to develop acceptable plans and Technical Specifications.

Topic II-3.C Safety Related Water Supply

This topic reviews the acceptability (supply and temperature) of water sources with respect to providing safety-related water during emergency shutdown and maintenance of safe shutdown.

Description

Sherman Pond (Deerfield River) is the normal source of emergency cooling water for the Yankee Rowe Plant. The licensee states (8) that in the event of a loss of this source, makeup water is available via the emergency feedwater system from the demineralized water system (85,000 gallons) and from the 350,000 gallon fire tanks.

Circulating and service water suction is from the bottom of Sherman Pond at elevation 1030.7 ft msl through a vortex-eliminating intake located about 180 feet from shore and rises through a 120 inch internal diameter corrugated steel pipe to the entrance of the screenwell house at elevation 1082.3 ft msl. Discharge from the circulating and service water systems is directed to the discharge seal pit from which the water is discharged over a weir, with an adjustable crest and minimum elevation of 1107.7 ft msl. The discharge is returned to Sherman Pond near the hydroelectric plant intake to ensure that warmed discharge water passes at once downstream.

Loss of Supply

The normal supply of emergency cooling water (Sherman Pond) can be lost through failure of Sherman Dam. Presently, the licensee is not prepared to cope with this situation except through use of the demineralized water system or the fire tank which can provide a limited water supply. We understand that the licensee has committed to provide a seismically qualified tank supply and dedicated fire truck. The review of supplies other than the Deerfield River (Sherman Pond) are not covered under this topic. Additional discussion can be found in topics III-6, VII-3 and IX-3.

Since the Deerfield River supply can be lost during several scenarios evaluated under current NRC criteria, the licensee will need an emergency plan and technical specifications to address these situations. Details of the emergency plan and technical specifications will be resolved during the integrated assessment.

Limiting Temperature of Normal Supply

Regulatory Guide 1.27 requires that the UHS provide water at an acceptable temperature. This is generally not a problem for once through cooling plants except where the supply is a pond where recirculation may occur. Sherman Pond is in this category but only during drought conditions. Our consultant, FRC, has provided an evaluation of this scenario in Appendix A and they conclude that this will not be a problem at the Yankee Rowe site. We concur with our consultants conclusions. We have evaluated larger plants with smaller cooling ponds and it is our judgement that because of stratification and surface evaporation, intake water temperatures will not exceed design limits. In addition, if intake water temperature should exceed design limits, the heat sink function can be maintained by venting steam to the atmosphere from the steam generators as discussed in Safe Shutdown Systems Report (Topic V-10B).

VI. CONCLUSIONS

Topic II-3.A

Although the licensee has submitted some information on the groundwater required, the data on groundwater levels is incomplete and not sufficiently documented and organized for the staff to extrapolate design basis levels. We therefore had to use conservative estimates. The design bases for construction was not available for some features.

Topic II-3.B

Groundwater

The NRC current criteria design basis groundwater level is ground surface.

Deerfield River Flooding

Harriman Dam will be overtopped and thus fail if basin rainfall exceeds about 13 inches in 24 hours. Our estimate of the PMP, the current NRC design basis rainfall, is 18.9 inches in 24 hours. The dam could also fail for other reasons, such as: seismic, piping or foundation failure. The failure of Harriman Dam for any reason would result in a flood water elevation at the Yankee Rowe site of at least 1174 feet msl or more than 40 feet over plant grade. In addition, a PMP centered over the drainage area between Sherman and Harriman Dams would overtop (elevation 1133.3 ft msl) and presumably breach Sherman Dam. If Sherman Dam does not breach, these flood levels could also inundate the Yankee Rowe Site.

Local Site Flooding

A PMP centered over tributary number 1 south of the Yankee Rowe Plant could result in runoff across the plant area that would be up to three feet deep.

Roof Flooding

Only the turbine building roof is susceptible to rainfall accumulations that could exceed the design capacity. With roof drains blocked, the PMP could pond to the top of the parapets (21 inches) which would exceed the design basis by 173%.

Topic II-3.B.1

The licensee has an operating procedure to protect the intake structure from intermediate flood levels, but it does not meet current NRC criteria since it does not protect the plant for the PMF.

Emergency plans and technical specifications may be required for the ultimate heat sink, but that determination will be made during the integrated assessment.

The resolution of the potential river flooding problem will be made during the integrated assessment. The feasibility of emergency plans or technical specifications will be made at that time.

Topic II-3.C

The normal UHS water supply (Sherman Pond) does not meet current NRC criteria. The acceptability of other sources is reviewed under other topics.

VII. REFERENCES

1. Standard Review Plans, NUREG 0800 (formerly NUREG 75/087), U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation.

   a. 2.4.1 - Hydrologic Description

   b. 2.4.2 - Floods

   c. 2.4.3 - Probable Maximum Flood (PMF) on Streams and Rivers

   d. 2.4.4 - Potential Dam Failures

   e. 2.4.5 - Probable Maximum Surge and Seiche Flooding

   f. 2.4.6 - Probable Maximum Tsunami Flooding

   g. 2.4.7 - Ice Effects

   h. 2.4.8 - Cooling Water Canals and Reservoirs

   i. 2.4.9 - Channel Diversions

   j. 2.4.10 - Flooding Protection Requirements

   k. 2.4.11 - Low Water Considerations

   l. 2.4.12 - Groundwater

   m. 2.4.14 - Technical Specifications and Emergency Operation Requirements

   n. 3.4.1 - Flood Protection

   o. 9.2.5 - Ultimate Heat Sink

2. Regulatory Guides, U.S. Nuclear Regulatory Commission, Office of Standards Development.

   a. 1.27 - Ultimate Heat Sink for Nuclear Power Plants

   b. 1.59 - Design Basis Floods for Nuclear Power Plants

   c. 1.70 - Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, NUREG-75/094.

   d. 1.102 - Flood Protection for Nuclear Power Plants

   e. 1.127 - Inspection of Water Control Structures Associated with Nuclear Power Plants.

3. American National Standard N170-1976, "Standards for Determining Design Basis Flooding at Power Reactor Sites," Published by the American Nuclear Society (ANS-2.8).

4. Franklin Research Center Technical Evaluation Report, TER-C5257-426, "Hydrological Considerations (SEP, II-3.A, B, B.1, C) Yankee Rowe Nuclear Power Station," August 6, 1982. Attached as Appendix A to this SER.

5. D. M. Crutchfield (NRC) to J. A. Kay (YAEC), "Topic II-3.A, Hydrologic Description and Topic II-3.B, Flooding Potential and Protection Requirements (Yankee Rowe)," dated February 9, 1981. Attached as Appendix B to this SER.

6. Yankee Atomic Electric Co., Draft Report, "Hydrologic Probable Maximum Flood Analysis", January 1980.

7. Yankee Atomic Electric Co., "Design Basis Flood Analyses," YAEC-1207, October 1980 (Transmitted March 1981).

8. J. A. Kay (YAEC) Letter to D. M. Crutchfield (NRC), Subject: "Additional Information for SEP Topics III-3.C (Inservice Inspection of Water Control Structures), II-3.A (Hydrologic Description), II-3.B and II-3.B.1 (Flooding Potential and Protection Requirements), and II-3.C (Safety Related Water Supply-Ultimate Heat Sink), June 16, 1982.

9. Yankee Atomic Electric Co., "Harriman Dam Performance Evaluation," YAEC-1298, April 1982.

10. Telephone communication, Mike Boyle (NRC) to Jim Kay (YAEC) August 18, 1982.

11. Department of the Army, Corps of Engineers, "EM 1110-2-1411-Standard Project Flood Determinations," March 26, 1952 (rev. March 1965).

12. U.S. Weather Bureau (now U.S. Weather Service, National Oceanic and Atmospheric Administration), 1956, Hydrometeorological Report No. 33, "Probable Maximum Precipitation East of the 105th Meridian for Areas from 10 to 1000 Square Miles and Durations of 6, 12, 24 and 48 hours."

13. U.S. Department of Commerce, National Oceanic and Atmospheric Administration - U.S. Department of the Army Corps of Engineers, Hydrometeorological Report No. 51, June 1978, "Probable Maximum Precipitation Estimates; United States East of the 105th Meridian."

TECHNICAL EVALUATION REPORT

HYDROLOGICAL CONSIDERATIONS (SEP, II-3.A, B, B.1, C; III-3.B)

YANKEE ATOMIC ELECTRIC COMPANY
YANKEE ROWE NUCLEAR POWER STATION

NRC DOCKET NO. 50-255

NRC TAC NO. 41359, 41348, 41337, 41326

NRC CONTRACT NO. NRC-03-79-118

FRC PROJECT C5257

FRC ASSIGNMENT 16

FRC TASK 426

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FOREWORD

This Technical Evaluation Report was prepared by Franklin Research Center under a contract with the U.S. Nuclear Regulatory Commission (Office of Nuclear Reactor Regulation, Division of Operating Reactors) for technical assistance in support of NRC operating reactor licensing actions. The technical evaluation was conducted in accordance with criteria established by the NRC.

Mr. J. S. Scherrer, Ms. S. Roberts, Mr. J. Turner, Mr. G. J. Overbeck, Mr. C. Brockman, Mr. L. Crow, and Mr. A. Kahan contributed to the technical preparation of this report through a subcontract with WESTEC Services, Inc.

1. INTRODUCTION

1.1 PURPOSE OF REVIEW

The purpose of this review is to evaluate the assumptions, conclusions, and completeness of documentation in submittals by the Yankee Atomic Electric Company (YAEC) on systematic evaluation program (SEP) Topics II-3.A (Hydrologic Description), II-3.B (Flooding Potential and Protection Requirements), II-3.B.1 (Capability of Operating Plants to Cope with Design Basis Flooding Conditions), and II-3.C (Safety-Related Water Supply - Ultimate Heat Sink) for the Yankee Rowe Nuclear Power Generating Station. It includes independent analyses by the Franklin Research Center (FRC) needed to clarify and resolve several issues. The Nuclear Regulatory Commission (NRC) is reviewing other safety topics within the SEP and intends to coordinate an integrated assessment of plant safety after completion of the review of all applicable safety topics and design basis events (DBEs).

1.2 GENERIC BACKGROUND

The SEP was established to evaluate the safety of 11 of the older nuclear power plants. An important element of the evaluation is to judge the plants by current licensing criteria with respect to 137 selected topics, several of which relate to hydrologic assessments of the site.

In a letter dated January 14, 1981 [1], the NRC agreed to the SEP Owners Group's proposed redirection of the SEP whereby each licensee would select any 60% of the SEP topics and submit evaluations of these in time for a review by the NRC staff to be completed by June 1981. Evaluations of topics not selected by a licensee were the NRC's responsibility.

1.3 PLANT-SPECIFIC BACKGROUND

The present evaluation of the hydrologic influences at the Yankee Rowe plant site compares the provisions of the Yankee Rowe plant against the current NRC criteria for licensing new facilities. The Licensee, Yankee Atomic Power Company, will be instructed to inform the NRC whether the as-built facility differs from the information provided in this assessment. This report organizes all previously submitted Licensee-developed information under appropriate SEP Topics. Where the NRC has provided previous background information pertaining to hydrologic influences at the site, this information has been incorporated to define the sequence of information transfer between the Licensee and the NRC.

2. REVIEW CRITERIA

The criteria used for the hydrology topics were based on the Code of Federal Regulations, Volume 10, Section 50 (10CFR50), Appendix A, General Design Criteria, Overall Requirements, Criterion 2, entitled "Design Bases for Protection Against Natural Phenomena." Specific criteria were taken from the following documents:

Standard Review Plan (SRP) [2]

- 2.4.1 Hydrologic Description
- 2.4.2 Floods
- 2.4.3 Probable Maximum Flood (PMF) on Streams and Rivers
- 2.4.4 Potential Dam Failures
- 2.4.5 Probable Maximum Surge and Seiche Flooding
- 2.4.6 Probable Maximum Tsunami Flooding
- 2.4.7 Ice Effects
- 2.4.8 Cooling Water Canals and Reservoirs
- 2.4.9 Channel Diversions
- 2.4.10 Flooding Protection Requirements
- 2.4.11 Coolong Water Supply
- 2.4.12 Groundwater
- 2.4.14 Technical Specifications and Emergency Operation Requirements

Regulatory Guides

- 1.27 Ultimate Heat Sink for Nuclear Power Plants [3]
- 1.59 Design Basis Floods for Nuclear Power Plants [4]
- 1.102 Flood Protection for Nuclear Power Plants [5]
- 1.127 Inspection of Water Control Structures Associated with Nuclear Power Plants [6]
- 1.135 Normal Water Level and Discharge at Nuclear Power Plants [7].

American National Standards Institute

N170-1976 [8]

3. TECHNICAL EVALUATION

3.1 HYDROLOGIC DESCRIPTION (SEP TOPIC II-3.A)

3.1.1 Topic Background

This report is a review of Systematic Evaluation Program (SEP) Topic II-3.A, Hydrologic Description, for the Yankee Rowe Nuclear Power Station. The purpose of this review is to adequately describe the site hydrologic environment and identify plant hydrologic design bases where available.

The conclusions presented in this section were derived from several sources, including NRC docketed information, NRC staff files, the Licensee's responses [9, 10], and a plant site visit.

3.1.2 Topic Review Criteria

The review criteria used for this section are identified in American National Standards Institute N170-1976 [8] and Standard Review Plan Section 2.4.1, Hydrologic Description [4].

3.1.3 Evaluation

3.1.3.1 Site

Yankee Rowe Nuclear Power Station is situated on the Deerfield River in north- western Massachusetts in the town of Rowe, in Franklin County. The site is between a reservoir, called Sherman Pond, and a steep hill adjacent to the river.

3.1.3.2 The Deerfield River Basin

The Deerfield River flows from southern Vermont south into Massachusetts, where it turns eastward to join the Connecticut River in northwest Massachusetts. The Deerfield River has a watershed of 644 sq mi, and a fall of 2014 ft in 66 miles. The river valley has fairly steep slopes, forming a V-shaped cross-section [11].

The upper Deerfield River basin is divided into four sub-basins, each a tributary to a hydroelectric reservoir or pond on the river.

Somerset Reservoir is the uppermost of the hydroelectric projects on the Deerfield River basin. Its drainage area is 30 sq mi, its length approximately 5.6 miles, and its surface area is about 1623 acres at elevation 2133.6 ft msl [9].

Somerset Dam is a curved structure made of semi-hydraulic fill. Its upstream slope is protected by rip-rap. The spillway, at the west end of the dam, consists of a trapezoidal concrete section divided into eight bays, each 24 ft wide with crest at 2133.6 ft msl and two sections 10 ft wide with crest at 2130.6 ft msl and stop logs to elevation 2136.6 ft msl [13]. In 1964, new flashboard stanchions were installed to carry 3 ft of flashboards [9].

Searsburg Dam impounds only 283 acre-ft of water. Its drainage area is about 90 sq mi. The pond is approximately 1 mile long and has a surface area of approximately 28 acres at elevation 1754.7 ft msl [9]. Since 1967, this station has been operated by remote control from Harriman Dam [13].

The spillway at Searsburg Dam consists of a concrete ogee weir 137 ft long and with crest elevation 1749.7 ft msl. It is equipped with pin-type flashboards 5 ft high. A vertical concrete wall with a deck, situated at the north end of the spillway, blocks off the bypass channel used during construction [9].

Harriman Reservoir is approximately 9 miles long and has a surface area of 2050 acres and a storage volume of 103,375 acre-ft at 1492.7 ft msl, which is spillway crest elevation. The drainage area for the reservoir is 184 sq mi [9].

Harriman Dam is made of semi-hydraulic fill. Its original height was about 200 ft, to elevation 1506.7 ft msl. In 1939, the dam was raised 5 ft, to elevation 1511.7 ft msl, to increase flood storage [21]. In 1964, an addition of rolled earth fill increased top elevation 9.5 ft to 1521.2 ft msl [9]. The normal freeboard above the spillway crest is 29.5 ft [21].

The upstream surface of the dam is protected with riprap to elevation 1480.7 ft msl, sloping up to 1495.7 ft at the north end of the dam. The downstream slope of the dam has a grass surface [9]. In 1981, a compacted filter drain was installed and covered with an impervious blanket, raising the downstream face about 5 ft [21].

Harriman Reservoir is equipped with a morning glory spillway situated adjacent to the south end of the dam. The spillway is connected by a vertical shaft with a 90° bend to the concrete conduit used during construction for diversion. The spillway has a crest elevation of 1491.7 ft msl. It consists of 16 equally spaced piers and can accommodate 7 ft of flashboards [9].

Yankee Rowe Nuclear Power Station, situated on the south edge of Sherman Pond, is the fourth of a series of artificial lakes built for hydroelectric generation on the Deerfield River. The southeast end of Sherman Dam (top elevation 1129.66 ft msl) borders the Yankee Rowe plant site on the west. Sherman Dam is an earthfill dam with a riprap surface on its upstream face to elevation 1095.7 ft msl, and grass cover on its downstream face [9]. Construction of Sherman Dam was completed in 1927 [12]. The dam, owned by New England Power Company, lies 41.2 miles from the mouth of the Connecticut River and has a drainage area of approximately 236 square miles.

Sherman Pond is 2.2 miles long. Its surface area is approximately 194 acres when the water level is 1103.7 ft msl [9]. When Sherman Pond is full, its water level is approximately 1108 ft msl. At its deepest point, which is near the dam, bottom elevation is 1025 ft msl [12].

At the north end of the dam, the embankment is supported by a concrete retaining wall. The spillway is at the same end of the dam.

3.1.3.3 Flood History

There is no water level gage at the Yankee Rowe site. The maximum recorded volume of flow on the Deerfield River at Charlemont, 9 miles downstream, is 56,300 cfs, recorded in September 1938. At that time, Sherman Dam and all upstream dams had been completed. Floods have occurred in all seasons, from snowmelt and rainfall in winter and spring, and from tropical storms in summer and fall. Historical floods include October 1869; November 1927; March 1936; January 1949; August 1955; October 1955; September 1960; and August 1976.

Since 1920, the USGS has operated a flow gaging station at Charlemont, Massachusetts, 9 miles downstream from Sherman Dam. Water level has been measured at Harriman Dam since 1911, at which time construction began on the hydroelectric power project.

3.1.3.4 Local Drainage

The Yankee Rowe site is 2000 acres on both sides of the Deerfield River. It lies in a valley at elevation 1127.66 ft msl [9]. Most of the surrounding land is heavily forested. The ground rises to 2000 ft msl at distances of less than 4000 ft east of the plant structures and 4000 ft west of Sherman Dam [11]. There are two small, unnamed drainage basins in the hills southeast of the plant from which runoff flows through the site toward Sherman Reservoir. Wheeler Brook, with an 806-acre watershed northeast of the plant, passes adjacent to the site as it discharges into the reservoir [9].

Some areas of the plant yard are between 1 and 2 ft below the top of Sherman Dam, and the screenhouse is much lower, at 1119.66 ft msl [9]. The average grade across the plant site is 1.2%. Two independent storm drain networks exist on the site: one drains from the northeast area of the site into Sherman Pond, and the other drains from the southwest area into the valley below Sherman Dam [13].

Two flood protection dikes were constructed to prevent flooding of the plant site caused by flooding on the Deerfield River. Their impact on site drainage is unknown. One dike (labeled A on Figure 2) extends from the southeast corner of the service area 40 ft south into the hillside. The other dike (labeled B on Figure 2) extends north from the northeast corner of the turbine building and then turns northwest, a total length of 175 ft. The turbine building and service building walls form the central part of the continuous flood wall. The top elevation of both dikes is 1132 ft msl. The structural loading design basis of these flood walls and building walls is unknown.

Roof Drainage

The roofs of safety-related structures have a design basis live loading of 40 psf, with the exception of the vapor container, which is designed for an unspecified higher live load [13]. The equivalent of 40 psf is 7.7 inches of ponded water.

Parapet height is 21 inches on the turbine building, 6 inches on the primary auxiliary building, and 4 inches or less on all other safety-related structures. The vapor container is spherical in shape and drained by direct runoff. All other structures are equipped with interior roof drains, and roof surfaces slope toward the drain inlets. The turbine building has four 3-inch drains and four 4-inch drains. Other safety-related structures have 3-inch roof drains [13].

A table of design basis elevations is presented in Table 1. These design basis flood elevations were identified from various Licensee submittals.

3.1.4 Conclusion

Information pertaining to the groundwater regime at the site is not available. This information is necessary to determine a suitably conservative maximum groundwater elevation and normal high groundwater elevation for use in SEP Topics II-3.B (Flood Potential and Protection Requirements) III-3.A (Effects of High Water Level on Structures), and III-6 (Seismic Design Considerations).

A Licensee response to a NRC request for additional information pertaining to these topics has not been received. Information relating to flood design basis and limiting flood elevations of safety-related equipment is not available.

Table 1. Design Basis Elevations

Event	Elevation (ft msl) Original (1959)	Present (1982)	NRC Criteria (1982)
Flooding			
Deerfield River (PMF)*	Unknown	1132 ft msl	1167.16 ft msl
Local flooding (PMF)	Unknown	1127.66 ft msl	1131 ft msl
Groundwater			
Maximum level	Unknown	Unknown	Ground elevation
Normal high level	Unknown	Unknown	Ground elevation
Rainfall loading			
On roofs	40 psf (7.7 in of ponded water)	40 psf (7.7 in of ponded water)	PMP** (Variable depth on each safety-related building
Floodwater loading on the flood wall and walls of buildings which are part of the flood wall	Not considered	Top of floodwall Stillwater elev. of 1132 ft msl	Top of floodwall Stillwater elev. of 1132 msl
Low Water			
Sherman Pond	988 ft msl	988 ft msl	Sherman Dam failure (water confined to original stream channel)
Plant Grade	1127.66 ft msl	1127.66 ft msl	1127.66 ft msl

*Probable maximum flood.

**Probable maximum precipitation.

3.2 FLOODING POTENTIAL AND PROTECTION REQUIREMENTS (Topic II-3.B)

3.2.1 Topic Background

The purpose of this topic is to identify the design basis flood level for the plant and site, under current licensing criteria, resulting from all potential flood sources external to the plant and site. Significant differences between the levels or values used for design and construction of the plant and those derived under current licensing criteria are evaluated. This evaluation includes the flood effects on safety-related structures, systems, and equipment and the effects of changed hydrostatic and hydrodynamic loads on safety-related structures, systems, and equipment where information is available. Features of existing or proposed flood protection measures such as revetments, flood walls or doors, and emergency or administrative procedures are discussed.

3.2.2 Topic Review Criteria

This topic was reviewed against the following criteria:

- Regulatory Guide 1.59, Design Basis Floods for Nuclear Power Plants
- ANSI N170-1976
- Standard Review Plan

  2.4.2 Floods

  2.4.3 Probable Maximum Floods on Streams and Rivers

  2.4.4 Potential Dam Failures

  2.4.11 Low water considerations

  2.4.10 Flooding Protection Requirements

  2.4.12 Groundwater

3.2.3 Evaluation

3.2.3.1 Introduction

Figure 1 depicts the placement of the Yankee Rowe site in the hydrologic environment. Pertinent flooding mechanisms appropriate to the site include

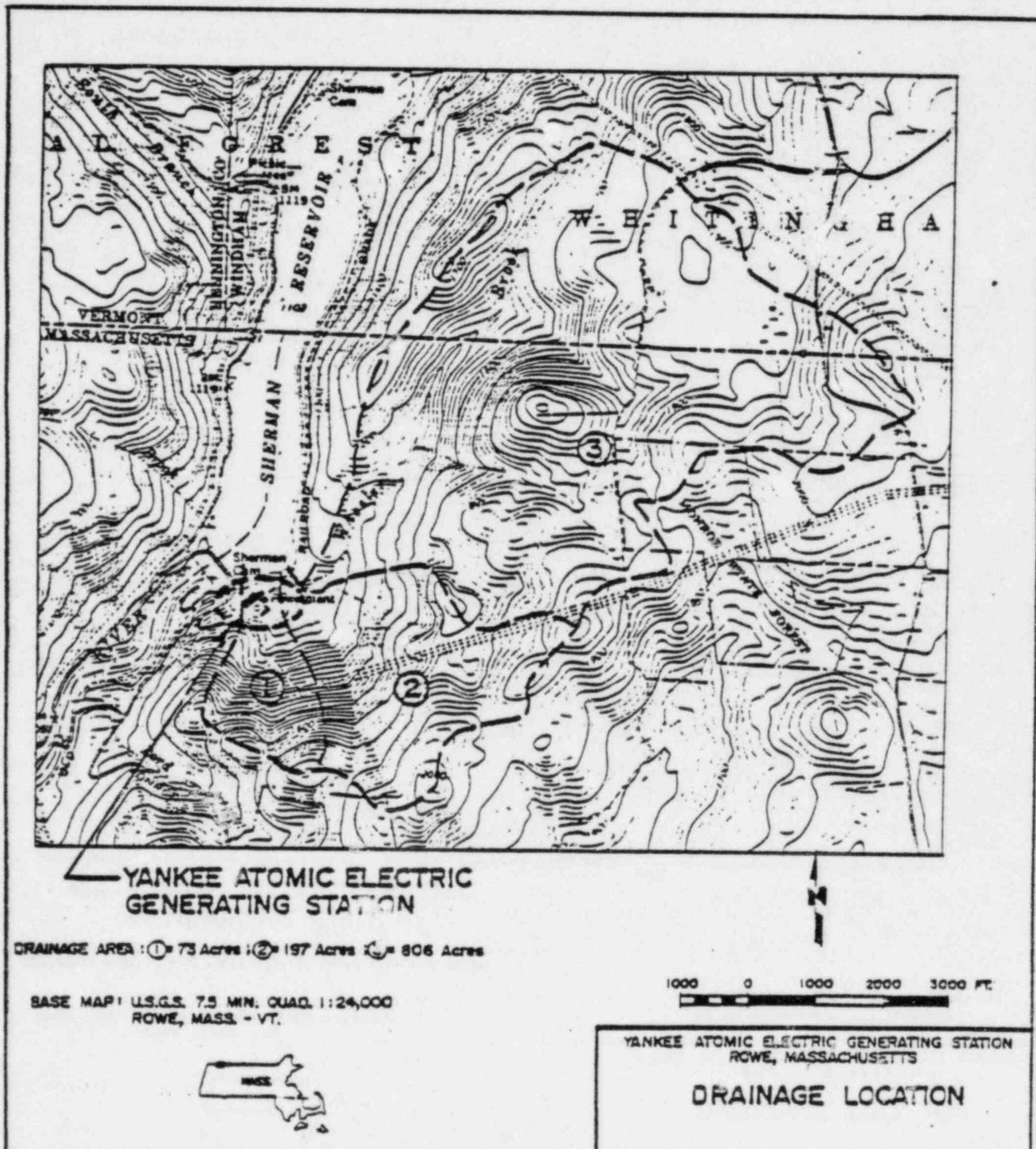

Figure 1. Site Location

flooding from the Upper Deerfield River, flooding due to failure of upstream dams, local runoff from small tributaries adjacent to the site, local flooding on-site from intense precipitation, and probable maximum precipitation (PMP) loading on roofs of safety-related structures. Hydrologic causal mechanisms not pertaining to the site are hurricane surge, tsunami, and lake seiche.

3.2.3.2 Flood History

History of flooding at the site has been identified by the Licensee as confined to the security guard house during occurrences of intense rainfall and local flooding conditions. No other flooding experiences have been identified.

3.2.3.3 Local Flooding

Probable Maximum Precipitation

At the NRC's request, a PMP for the Deerfield River basin has been developed for the purpose of defining the probable maximum flood (PMF) of the Deerfield River. Appendix A presents this independent evaluation of the magnitude and duration of a PMP event in the Deerfield River Basin. The PMP magnitude and duration presented in Appendix A was not used in the evaluation of local flooding potential presented below.

Local Plant Flooding

Introduction

This study consists of an independent evaluation of the depth of flooding adjacent to the turbine and ancillary buildings at the Yankee Rowe site. The flooding results from site runoff during rainfall equivalent to the PMP.

Drainage Location

The Yankee Rowe plant is situated at the confluence of three watersheds; two watersheds are unnamed and the third is drained by Wheeler Brook. Figure

1 contains a contour map of the region surrounding the power plant with the three drainage area boundaries delineated individually. The area within watersheds 1, 2, and 3 are 73 acres, 197 acres, and 806 acres, respectively; Wheeler Brook is represented by watershed 3. The watershed divide separating the three drainage basins are in part controlled by the location of powerline rights-of-way and access roads. Verification of the watershed divide locations was made during a site visit on June 1, 1982.

The watercourse for drainage area 1 originally passed through the power plant site until it was diverted at the time of plant construction. This watershed is the primary source of local runoff and is the dominant source for local plant flooding.

Runoff from drainage area 2 has been diverted due to the construction of a powerline and its right-of-way, seen in Figure 1. This right-of-way conveys runoff away from the original mid-watershed mainstream to a stream channel located adjacent to the western edge of the watershed. Evidence of erosion and channel headcutting were observed during the site visit in the right-of-way in drainage area 1. Continued erosion will ultimately result in the diversion of runoff from above or south of the powerline from drainage area 2 into that runoff of drainage area 1.

Wheeler Brook is the principal stream channel for draining watershed 3. At the mouth of Wheeler Brook is a 60-inch-diameter, corrugated pipe culvert passing through the embankment of an abandoned railroad grade. This culvert has an inlet invert elevation of 1109 ft msl (YAEC Drawing YR-D-10-013) and is located about 680 feet east of the containment sphere. Presently, runoff from drainage areas 1 and 2 is tributary to Wheeler Brook at its mouth.

A schematic drawing of the nuclear power plant along with flowlines can be seen in Figure 2. These flowlines represent the relative position of the stream channels draining watersheds 1, 2, and 3 as identified in Figure 1.

Typical vegetation covering the watersheds consists of coniferous and deciduous trees. The crown density of the natural vegetation is estimated to

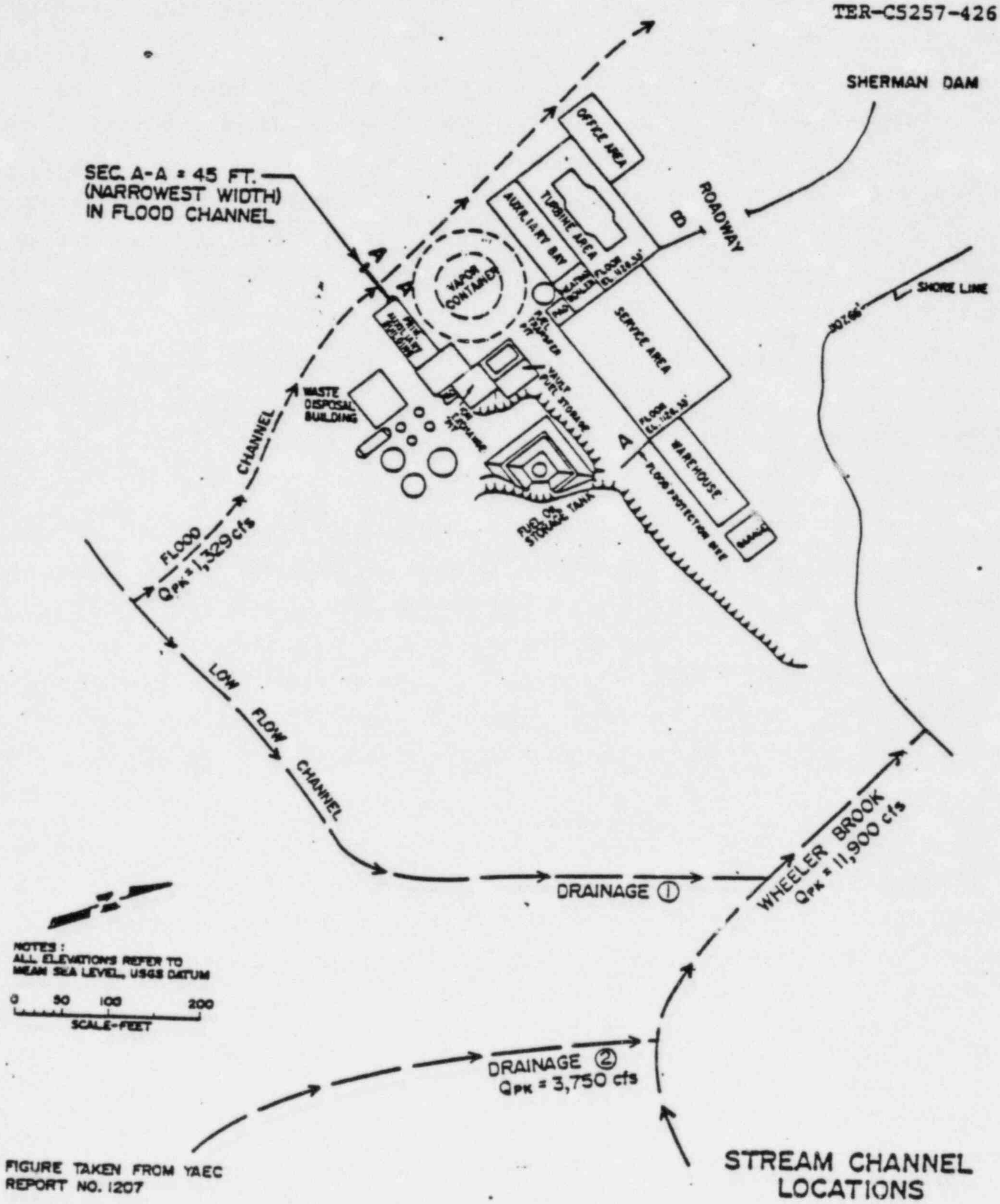

Figure 2. Local Flooding Flowlines

be approaching 90% except in the powerline right-of-way where the larger trees have been systematically removed.

Vegetation clearing for two powerline rights-of-way (seen in Figure 1) transect both watersheds 1 and 2. This reduce- -- retarding effect of heavy vegetation, resulting in an increase in runoff compared to the undisturbed areas of the watersheds, as well as providing a path or channel for the collection and conveyance of the runoff.

A critical assumption made during the site drainage analysis was that the watershed divide separating drainages areas 1 and 2 was intact and that no runoff from the larger eastern watershed entered watershed 1. However, continued erosion of the powerline right-of-way will render this assumption incorrect. The Licensee should develop a means to stop head cutting, or as an alternative, design the local flood protection structure with consideration to runoff from both drainage areas 1 and 2.

Probable Maximum Precipitation

The most severe hydrometeorological condition with respect to site drainage is the PMF resulting from rainfall equivalent to the PMP. The HEC-1 flood hydrograph program [26], in conjunction with the rainfall isohyetals in Hydrometeorological Report No. 33, was utilized in estimating the maximum discharge from the three watersheds identified in Figure 1 during the PMP. The PMP was distributed temporarilly according to the U.S. Corps of Engineers manual, "Standard Project Flood Determinations," EM-1102-2-1411 (1952) [23].

The PMP used in the site drainage study was 26.0 inches in 24 hours [22, 23] with the maximum intensity of 7.63 inches per hour. Figure 3 contains the 5-hour segment of the 24-hour PMP having the greatest cumulative rainfall.

Probable Maximum Flood Hydrographs

The PMF for the three forested watersheds was simulated by the HEC-1 flood hydrograph program. Runoff calculations were based on assumed rainfall and watershed response functions. The watershed response function was

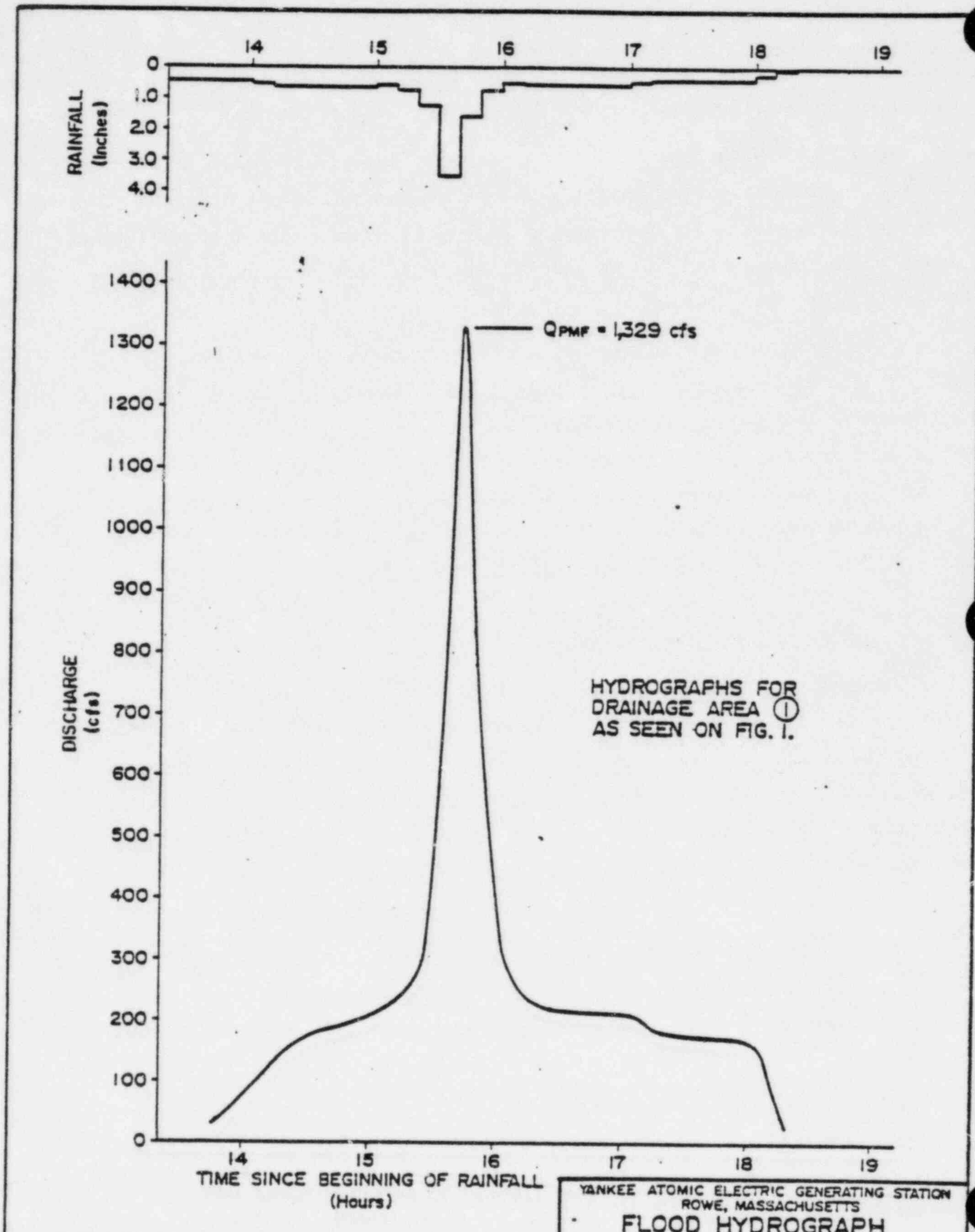

Figure 3. Flood Hydrograph

independently determined based on background information provided by the Licensee [21], a site visit, and analytical methods presented by the Soil Conservation Service [24].

The watershed response function contains a lumped parameter called a "curve number" that is dependent upon soil type, vegetative cover, and antecedent soil moisture conditions. The soil type found in watershed 1 is classified as the Berkshire Series [25], and the vegetative cover is a coniferous and deciduous forest. The antecedent moisture condition selected for use during this study was AMC III. A value of 78 was assigned to the curve number representing the runoff characteristics of the watershed adjusted for clearing of the right-of-way.

Peak discharge during the PMF from the 73-acre drainage basin 1 will be 1,329 cfs. This discharge does not include the base flow, estimated to be less than 50 cfs. The shape of the PMF hydrograph, seen in Figure 3, is generally sharp and is characteristic of a small, steep watershed with an intense localized rainfall centered over the basin.

The PMF for watersheds 2 and 3 was also calculated for the purpose of comparison. Their PMF values were 3,750 cfs and 11,900 cfs, respectively. However, because Wheeler Brook and watershed 2 presently do not discharge through the plant site, only the PMF from the unnamed drainage area 1 was used to determine the depth of flooding adjacent to the nuclear facility.

Runoff from watershed 1 is presently the only source of water that will flood the plant site during the probable maximum event, and an evaluation of the depth of flooding from this event is presented in the next section. Runoff from watershed 2 is presently isolated from the neighboring watershed 1 by a natural divide; however, continued erosion over an indefinite period of time will ultimately redirect floodwater from drainage area 2 to drainage area 1. Once the divide is eroded, the onsite PMF will increase significantly. Wheeler Brook (drainage area 3) poses no flood hazard to the site.

Site Drainage and Depth of Flooding

The analysis of site flooding by runoff from the three watersheds located above the nuclear power plant was restricted in its precision due to a lack of adequate topographic maps of the plant site. The most inadequate data were the location, size, and slope of the flood channels passing through or around the nuclear power facility. However, during the site visit, estimates were made of the general location, dimension, and slope of the principal drainage paths in the plant vicinity. This information was subsequently used in the site flooding analysis.

The first step in the site flooding analysis was to investigate the conveyance capabilities of Wheeler Brook and the low flow channel for drainage area 1, described in detail below. Normal depth calculations indicated that the bankfull discharge or maximum flood capacity of Wheeler Brook is about 34,000 cfs, which is greater than the combined PMF from drainage areas 1 and 2. The crest of the railroad embankment at the mouth of Wheeler Brook is lower than plant grade and is separated from the plant site by a natural steam divide. An assessment of site flooding potential indicated that the influence of the Wheeler Brook railroad embankment was insignificant.

The natural watercourse for drainage basin 1 was altered during the construction of the nuclear power plant such that the runoff is presently being diverted eastward around the plant by a small unlined ditch that joins Wheeler Brook at its mouth. This alternate route is shown graphically on Figure 2. A new drainage channel starts at the southwest edge of the plant site and drains in a northwesterly direction parallel to the outer security fence. The left-hand or northwest edge of this channel is composed of a small levee having a height of approximately 4 ft above the invert. The outer security fence is set on top of and parallel to this levee.

The shape of the existing hillside runoff drainage channel is highly irregular; however, the general shape can be described as trapezoidal with a base width of about 5 ft, a top width of about 15 ft, and a depth of about 4 ft. The channel slope and "n" value were estimated to be about 2% and 0.10, respectively. This channel is sufficient for low discharges, but for flows exceeding 150 cfs, the channel side will breach, resulting in a change in direction of the flow path back to the original stream channel that passes through the plant.

Once the channel levee is breached, the runoff will enter the plant site and flow along an asphalt roadbed located between the diesel generator building and a low retaining wall west of the containment sphere. The narrowest width of this roadbed, marked cross-section A-A in Figure 2, is 45 ft.

The depth of flooding adjacent to the diesel generating building was calculated based on 100% diversion of the peak flood discharge of 1,329 cfs and using a rectangular channel having a base width of 45 ft. It was assumed that the flow was at a hydraulic critical depth due to the severe roadbed slope of 3% measured from the Licensee's 2-ft contour map of the plant site (YAEC Drawing YR-D-10-013). A flood depth of 3.0 ft at the diesel generator building will result from runoff during a rainfall event equivalent to the PMP. This corresponds to a local elevation of 1131 ft msl, 3 ft above grade at the diesel generator building and turbine building.

The depth of flooding will be considerably greater if watershed 2, as seen in Figure 1, is allowed to spill over into the unnamed watershed 1 as a result of continued erosion of the powerline right-of-way.

Due to lack of information pertaining to the limiting flood elevation of safety-related equipment within these buildings, it is not known what effect floodwater 3 ft above grade would have on safety-related equipment.

Roof Flooding

The roofs of safety-related structures at the Yankee Rowe plant were designed to withstand a live loading of 40 psf, with the exception of the vapor container, which is designed for an unspecified higher live load [13]. The equivalent of 40 psf is 7.7 inches of ponded water.

Current NRC criteria require that the roofs of all safety-related structures be able to withstand the loading caused by the PMP if all roof drains are fully blocked and non-functional.

The vapor container is spherical in shape, and rain runs directly off of it. For this reason, the vapor container is not affected by the PMP.

The parapets around the roof of the turbine building are 21 inches high [13]. If roof drains were blocked during PMP, the heaviest single hour of precipitation would cause ponding that would exceed the design basis live loading by 45 percent. Continued precipitation would result in ponding up to 21 inches, exceeding the design basis live loading by 173 percent. For these reasons, the turbine building roof does not fulfill current NRC criteria.

The primary auxiliary building roof has parapets 6 inches high, and all other safety-related structures have parapets 4 inches high or lower [13]. These buildings could not pond water high enough to meet the design basis live loading on the rooftops, and therefore they fulfill NRC criteria for rooftop ponding.

In summary, the turbine building roof is the only roof on a safety-related structure which is not protected against PMP.

3.2.3.4 Flooding of Rivers

PMF Without Dam Failure

Information under this topic to be supplied by the NRC.

PMF With Dam Failure

Information under this topic to be supplied by the NRC.

3.2.3.5 Failure of Dams

The Licensee has presented an evaluation of the integrity of Harriman Dam under SEP Topic II-4.E [21]. The Licensee has concluded that the probability of failure of Harriman Dam is extremely remote; however, should the dam fail completely, stream flow would rise to a depth of between 25 and 45 ft above the turbine floor of the Yankee Rowe plant.

No evaluation of the integrity of the Sherman Dam has been presented in the Licensee's submittal of SEP Topic II-4.E [21], Dam Integrity. This information is necessary for review of SEP Topic II-3.C, Ultimate Heat Sink.

3.2.3.6 Groundwater

No information was available for review under this topic at the time of writing. Further information from the Licensee, in response to an NRC request for additional information, should be available in the near future.

3.2.4 Conclusion

Local Flooding

Local flooding of the plant site is shown to occur during a PMP falling in basins tributary to the site. ooding to 3 ft above grade occurs in the area of the diesel generator building and the turbine building. Flooding effects on safety-related equipment is unknown due to a lack of sufficient information.

Roof Flooding

The turbine building roof is the only roof on a safety-related structure which is not protected against the PMP.

Flooding of Rivers

Further study of this issue will be undertaken by the NRC. No final conclusions are presented in this evaluation.

Failure of Dams

Further study of this issue will be undertaken by the NRC. No final conclusions are presented in this evaluation.

3.3 CAPABILITY OF OPERATING PLANTS TO COPE WITH DESIGN BASIS FLOOD CONDITIONS (SEP TOPIC II-3.B.1)

3.3.1 Topic Background

Protection against postulated floods can sometimes be accomplished by implementing emergency procedures and technical specifications. The purpose of this evaluation is to focus on the adequacy and efficacy of the Yankee Rowe plant emergency procedures to preclude flooding of safety-related equipment necessary for maintaining the safe operation and cooldown of the reactor system. Further, this evaluation addresses the existence of technical specifications for flood control systems and procedures.

The following evaluation used information obtained during a Yankee Rowe plant site visit [20], Docket No. 50-29, the PMF hydrograph developed in Section 3.2 of this report, and Yankee Rowe's OP-3006, "Environmental Flooding Conditions" [26].

3.3.2 Topic Review Criteria

The following references were used as review criteria:

- ANSI N170-1976
- NRC Regulatory Guides 1.59 and 1.102
- Standard Review Plan, Sections 2.4.10 and 2.4.14.

3.3.3 Evaluation

Background

The PMF and certain other floods of higher frequency have been determined to reach elevations which jeopardize equipment used in the normal operation of the Yankee Rowe plant. Consequently, the Licensee has adopted flood emergency procedures which provide guidance for operating personnel in the event of the forecast of a flood elevation reaching the screenhouse.

This evaluation focuses on the acceptability and efficacy of the Licensee's procedures as mechanisms to protect the equipment needed for reactor cooldown and control.

The NRC's criteria for protection against flooding are as follows:

1. The conditions resulting from the worst site-related flood probable at a nuclear power plant (e.g, PMF, seismically induced flood, hurricane, seiche, surge, heavy local precipitation) with attendant wind-generated wave activity constitute the design basis flood conditions according to which safety-related structures, systems, and components identified in Regulatory Guide 1.29 must be designed to withstand and retain capability for cold shutdown and maintenance thereof.

2. As an alternative to designing hardened protection [hardened protection means structural provisions incorporated in the plant design that will protect safety-related structures, systems, and components from the static and dynamic effects of floods. In addition, each component of the protection must be passive and in place, as it is to be used for flood protection, during normal plant operation. Examples of the types of flood protection to be provided for nuclear power plants are contained in Regulatory Guide 1.102] for all safety-related structures, systems, and components as specified in Regulatory Position 1 above, it is permissible not to provide hardened protection for some of these features if:

   a. Sufficient warning time is shown to be available to shut the plant down and implement adequate emergency procedures;

   b. All safety-related structures, systems, and components identified in Regulatory Guide 1.29 are designed to withstand the flood conditions resulting from a Standard Project event with attendant wind-generated wave activity that may be produced by the worst winds of record and remain functional;

   c. In addition to paragraph 2.b above, at least those structures, systems, and components necessary for cold shutdown and maintenance thereof are designed with hardened protective features to remain functional while withstanding the entire range of flood conditions up to and including the worst site-related flood probable (e.g., PMF, seismically induced flood, hurricane, surge, seiche, heavy local precipitation) with coincident wind-generated wave action as discussed in Regulatory Position 1 above [4].

   Temporary flood barriers, such as sandbags, plastic sheeting, portable panels, etc., which must be installed prior to the advent of the DBFL, are not acceptable for issuance of a construction permit. However, unusual circumstances could arise after construction that would warrant consideration of such barriers. One example of unusual circumstances that might justify use of temporary barriers is a post-construction change in the flood-producing characteristics of the drainage area, as discussed in Regulatory Position 3 of Regulatory Guide 1.59 "Design Basis Floods for Nuclear Power Plants." In such circumstances, and with strong justification, the staff may accept temporary barriers [5].

It is the position of this review that, to fulfill NRC guidelines, the Licensee's procedures must protect against all PMF events with a high degree of assurance that safety-related equipment will be able to maintain a safe shutdown condition throughout the flood event. To provide this assurance, all methods of communication should be reliable and all methods of flooding should be considered. Adequate time to complete the emergency actions must be available between receipt of the instructions to initiate the procedure and flood waters reaching equipment to be protected. Actions to be performed should be as simple as possible, and described clearly. Finally, assurance should be provided that the proposed protective measures are adequate to protect equipment necessary to maintain safe shutdown against the PMF.

Emergency Procedures

Local Flooding

No emergency procedures exist to protect safety-related structures and equipment from flooding resulting from a local PMP. In general, it can be assumed that the time-dependent characteristics of the PMP that induce the PMF are such that all prescribed emergency actions would need to be completed within a short amount of time. Thus, on-the-spot sandbagging or similar emergency procedures are not considered technically feasible. The interval of time between warning of the local flood event and inundation of safety-related equipment is not sufficient to erect flood barriers. For these reasons, protection from local flooding should be hardened, and always in place.

Flood of Deerfield River

The Licensee has stated [20] that two emergency procedures exist at the Yankee Rowe plant to protect against flooding on Sherman Pond. One of these procedures, OP-3006, "Environmental Flooding Conditions," is designed to protect the screenhouse from flooding on Sherman Pond [19]. The other procedure is not available for review. The Licensee has stated that this missing procedure addresses the installation of stop logs in flood walls, and that no other procedures for flooding from the environment exist at the Yankee Rowe plant [20].

It is not clear whether an emergency procedure should exist for the intake structure for the following reasons:

- o The PMF resulting from an upper basin PMP may, according to the NRC's present position, cause water to flood over the entire intake structure.

- o Failure of Harriman Dam will also cause flooding over the entire intake structure.

Should the resolution of these issues indicate that OP-3006 is a necessary procedure, the following critique is provided.

The objective of OP-3006 is to "set forth steps necessary to assure the safety of the plant in the event of external flooding conditions." Further discussion presented in OP-3006 indicates that a secondary objective is "to provide proper cooling for the reactor and associated equipment" by protecting the screenhouse and the pumps within the screenhouse when there is flooding on Sherman Pond. Therefore, the objective of this emergency plan is not to prevent flooding of the plant site, but to ensure the continous flow of reactor coolant water.

OP-3006 is initiated by the following:

Symptoms

1. Heavy and prolonged precipitation

2. Steady increase of pond level

3. Seismic shock of any magnitude in the area

4. Screenwell house high sump level Panalarm, No. SH-4 [19].

Receipt of flood warnings should also be included as an initiating event. Further, appropriate actions, associated flood elevations, and required times should be included under this emergency procedure.

Further elaboration is needed under item 1 above. For example, a minimum level of precipitation should be identified which would act to initiate operator actions. Under item 2, further elaboration should be provided to enable a plant supervisor to assess water level changes. For example, item 2 should indicate the water level at which the emergency procedure would be initiated. Further excerpts from OP-3006 are presented below:

"Immediate Operator Action

1. Contact the Harriman Station as to pond level increase and rate of increase and contact the dispatcher for availability of Shermán Station.

2. Initiate OP-3300, "Classification of Emergencies."

3. Start the screenwell house water eductor if needed.

4. Place the travelling screens in operation.

5. Place the emergency flood control panels inside the screenhouse door casings.

6. Notify higher supervision [19].

Subsequent Operator Action

1. Review OP-Memo 2A-1 for possible NRC notification requirements.

2. Sandbag the screenhouse door flood control panels to make the doors as watertight as possible.

3. Maintain contact with Harriman Station by phone and radio for information on river conditions.

4. Frequently clean strainers in cooling water lines to running equipment.

5. Frequently over-supply cooling water to operating coolers to flush out sediment.

6. When available personnel are on site, sand-bag low areas of the yard to prevent flood waters from entering the plant as necessary. Install the yard barrier stop logs if extreme flood levels are expected and sandbag or cover unprotected building doors.

7. Constantly check screenhouse sump pump and eductor for proper operation."

It should be recognized that NRC notification requirements should not hinder the efficient and timely initiation of a flood emergency procedure. Further, burlap bags and sand should be readily available to operators. Sand, which is heavy, should be stored near to the place it will be used, with shovels nearby to fill bags. These precautions enable efficient implementation of flood protection measures.

The two methods of communication with Harriman Station constitute an appropriate level of redundancy. However, it is likely that there will be insufficient time available between failure of Harriman Dam and the resulting rise of water to be able to implement this procedure.

Final Conditions

1. The plant is operating normally with low areas of the yard and the screenwell house successfully barricaded against abnormally high pond conditions, or

2. The plant is shut down and borated to full shutdown concentration. Decay heat is being controlled by the atmospheric dump [19].

This flood emergency procedure does not specify the maximum flood water level. Since the screenwell and pumphouse walls cannot resist hydrostatic pressure resulting from a flood level of 1,131.7 ft msl [18], the procedure OP-3006 is useless during the occurrence of a PMF.

Technical Specifications

The Licensee has no technical specifications which specifically address flooding of the plant site. Technical specifications which limit plant operation during flood conditions may be warranted.

3.3.4 Conclusion

The final review of the adequacy of emergency flood procedures is dependent upon the resolution of the final PMF elevations due to both a failure of Harriman Dam and the PMP-induced PMF. Further, the need to protect the intake structure is dependent upon a resolution of SEP Topic II-3.B, Flooding Potential and Protection Requirements.

3.4 SAFETY-RELATED WATER SUPPLY (TOPIC II-3.C)

3.4.1 Topic Background

This topic reviews the acceptability of a particular feature of the cooling water system, namely, the ultimate heat sink (UHS). The review is based on current criteria contained in Regulatory Guide 1.27, Rev. 2, which is an interpretation of General Design Criterion (GDC) 44, "Cooling Water," and GDC 2, "Design Bases For Protection Against Natural Phenomena," of 10CFR50, Appendix A.

GDC 44 requires, in part, that suitable redundancy of features be provided for cooling water systems to ensure that they can perform their safety function. GDC 2 requires, in part, that structures, systems, and components important to safety be designed to withstand the effects of natural phenomena without loss of ability to perform their safety functions. Regulatory Guide 1.27 has been specifically cited by the NRC's Regulatory Requirements Review Committee for consideration in the backfitting of operating reactors. This guide is used in judging whether the facility design complies with current criteria.

The UHS, as reviewed under this topic, is the complex of cooling water sources, including necessary retaining structures (e.g., a pond with its dam or a cooling tower supply basin), and the canals or conduits connecting the sources to the cooling water system intake structures, but excludes the intake structures themselves. The UHS performs two principal safety functions: (1) dissipation of residual heat after reactor shutdown, and (2) dissipation of residual heat after an accident.

Availability of an adequate supply of water for the UHS is a basic requirement for any nuclear power plant. Since there are various methods of satisfying the requirement, UHS designs tend to be unique to each nuclear plant, depending upon its particular geographic location. Regulatory Guide 1.27 provides UHS examples that the NRC staff has found acceptable.

The UHS must also be able to dissipate the maximum possible total heat, including the effects of a loss-of-coolant accident (LOCA) under the worst combination of adverse environmental conditions. The maximum tolerable temperature of an UHS such as a cooling pond may significantly limit its ability to dissipate the heat load following a LOCA or plant shutdown, while maximum temperature may not be a significant concern for an UHS such as a large lake, river, or ocean.

Because of the importance of the UHS, it should be able to perform its safety function during and following the most severe natural phenomena or accidents postulated at the site. In addition, the UHS safety functions should be ensured during other applicable site-related events that may be caused by less severe natural phenomena and accidents in reasonable combination.

3.4.2 Topic Review Criteria

The criteria for evaluating the UHS were taken from Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants," and are as follows:

"1. The ultimate heat sink should be capable of providing sufficient cooling for at least 30 days (a) to permit simultaneous safe shutdown and cooldown of all nuclear reactor units that it serves and to maintain them in a safe shutdown condition, and (b) in the event of an accident in one unit, to limit the effects of that accident safely, to permit simultaneous and safe shutdown of the remaining units, and to maintain them in a safe shutdown condition. Procedures for ensuring a continued capability after 30 days should be available.

2. The ultimate heat sink complex, whether composed of single or multiple water sources, should be capable of withstanding, without loss of the sink safety functions specified in regulatory position 1, the following events:

   a. the most severe natural phenomena expected at the site, with appropriate ambient conditions, but with no two or more such phenomena occurring simultaneously,

   b. the site-related events (e.g., transportation accident, river diversion) that historically have occurred or that may occur during the plant lifetime,

   c. reasonably probable combinations of less severe natural phenomena and/or site-related events,

   d. a single failure of manmade structural features.

3. The ultimate heat sink should consist of at least two sources of water, including their retaining structures, each with the capability to perform the safety functions specified in regulatory position 1,

unless it can be demonstrated that there is an extremely low probability of losing the capability of a single source.

4. The technical specifications for the plant should include provisions for actions to be taken in the event that conditions threaten partial loss of the capability of the ultimate heat sink or the plant temporarily does not satisfy regulatory positions 1 and 3 during operation."

In addition to Regulatory Guide 1.27, clarifications are contained in Standard Review Plan (SRP), Sections 2.4.11, "Low Water Considerations," and 9.25, "Ultimate Heat Sink."

3.4.3 Evaluation

The UHS for Yankee Rowe Nuclear Power Station is the Sherman Pond formed by the Sherman Dam along the Deerfield River. Circulating and service water suction is from the bottom of Sherman Pond at elevation 925 ft through a vortex- eliminating intake located approximately 180 ft from shore and rises through a 120-in internal diameter corrugated steel pipe to the entrance of the screen well house at elevation 976 ft 8 in.

Discharge from the circulating and service water systems is directed to the discharge seal pit. From the discharge seal pit, the water is discharged over a weir whose crest is at 1002 ft. The weir is adjustable to higher levels by the addition of stop logs. The discharge seal pit has been designed to return plant cooling water to the pond near the Sherman hydroelectric plant intake to ensure that warmed discharge water passes at once downstream.

3.4.3.1 Vulnerability of the UHS to a Single Failure of Manmade Structural Features

The failure of the Sherman Dam or the intake conduit can be postulated to occur due to failure during intense runoff, catastrophic structural failure, or seismically induced structural failure. Although all of these events are considered to be low probability events, consideration of these events is consistent with topic review criteria. The effect of earthquakes on the intake conduit of the dam is being reviewed under Topic III-6, "Seismic Design Considerations."

A postulated failure of Harriman Dam and subsequent flooding [21] which causes overtopping and failure of Sherman Dam deserves evaluation under this SEP Topic. The failure of Sherman Dam in this scenario would leave the plant without an UHS following the passage of waters released from Harriman Reservoir. It should be recognized, however, that this scenario differs only from simple failure of Sherman Dam in that Harriman Resevoir water is no longer available to act as an UHS.

In Reference 14, YAEC was requested to describe how the heat sink function is maintained following the occurrence of severe natural phenomena or catastrophic failure of Sherman Dam or the intake canal. It was noted that a cooling capacity of less than 30 days may be acceptable if it can be demonstrated that replenishment or use of an alternate water supply can be effected to ensure the continuous capability of the sink to perform its safety function, taking into account the availability of replenishment equipment and limitations that may be imposed on freedom of movement following the occurrence of natural phenomena. YAEC was requested to provide a discussion of the emergency procedures to be followed.

At the time of preparation of this evaluation, YAEC's response has not been received. As a consequence, insufficient information is available to evaluate the vulnerability of the UHS to a single failure of manmade structural features.

3.4.3.2 Vulnerability of the UHS to Missiles

The UHS water source, Sherman Pond, is not susceptible to damage from missiles, from natural phenomena (tornado) or site-related events (turbine missile) that would result in a complete loss of heat sink availability. The intake conduit as well is not susceptible to damage from missiles, because the conduit is either buried or below the water surface over its entire length. The Sherman Dam may be susceptible to damage. The effect of missile damage would not be more severe than that postulated to occur from the catastrophic failure described in Section 3.4.3.1. Further information with respect to missiles is available in SEP Topic III-4.A, III-4.B, III-4.C, and III-4.D.

3.4.3.3 Vulnerability of the UHS to Sedimentation

In Reference 15, YAEC stated that "divers normally check for silt buildup and inspect the conduits and surrounding areas and perform any maintenance required during refueling outages. The accumulation of silt could affect the performance of the UHS; however, this review concludes that the periodic inspection of the inlet conduit and surrounding area by divers should give ample warning of sediment buildup to allow for early corrective action." In addition, inspection for sedimentation will be included in the future formal inspection program under Regulatory Guide 1.127 (see SEP Topic III-3.C).

3.4.3.4 Vulnerability of the UHS to Low Water Conditions

In Reference 16, YAEC described the circulating water system at the Yankee Rowe plant. In this description, YAEC addressed the potential need for a cooling tower and states the following:

> "Early in the development of the project it became apparent that during a prolonged dry period and under certain possible conditions of hydro-electric power operation of the Deerfield River system, circulation of Sherman Pond water through the condenser could produce higher temperature downstream than could be tolerated.

In Reference 17, YAEC provided a plan view drawing of the circulating water system plan which includes the intake conduit, screenwell and pump house, seal pit, and cooling tower. The drawing, 9699-FM-38F, indicates that the stipulated lowest water level for operation of the hydroelectric station is at elevation 988 ft (NEP datum). It further states that the "hydro station must operate to prevent recirculation [and] overheating of [the] pond when turbine generator operates without [the] cooling tower."

In Reference 14, a concern was expressed that the capacity of the ultimate heat sink should be sufficient to provide cooling for the period of time needed to evaluate the situation and to take corrective action. In addition, the UHS should be able to dissipate rejected heat to ensure that design temperatures of safety-related equipment are not exceeded. During a prolonged dry period and under certain possible conditions of hydroelectric power operation of the Deerfield River system, circulation of Sherman Pond water through the condenser could produce higher pond temperatures. The UHS must be able to dissipate the maximum possible total heat, including the effects of a LOCA under the worst combination of adverse environmental conditions.

YAEC was requested to verify that the operation of the hydroelectric system and severe natural phenomena do not adversely affect the capability of the UHS to dissipate sufficient heat following a LOCA or plant shutdown, to clarify whether the need to have the Sherman hydroelectric facility operating is an environmental or safety concern, and to identify those situations in which the amount of water flowing through the Sherman hydroelectric facility is critical to the cooling water temperature at Yankee Rowe's intake. In making this determination, YAEC was to give consideration to the environmental factors which would tend to increase pond temperature and to the potential for recirculation between cooling water discharge and intake. If the flow is critical, then YAEC was to provide any transient analyses of pond temperature performed which demonstrate the capability of the Yankee Rowe UHS. YAEC was also requested to describe any plant technical specification which includes provisions for actions to be taken in the event that conditions threaten partial loss of the UHS.

Although YAEC's response has not been received at the time of preparation of this evaluation, it can be concluded that the UHS is able to dissipate rejected heat following a LOCA or plant shutdown such that design temperatures of safety-related equipment are not exceeded. This conclusion is based upon the fact that the drainage area above the Sherman Dam is approximately 236 square miles and that the available volume of water is large. The pond extends upstream approximately 2.2 miles and has a surface area of 218 acres at an elevation of 1107.66 (USGS datum). Other statistics are as follows:

Normal Maximum Reservoir Elevation	1107.66 ft USGS
Normal Operating Reservoir Elevation	1106.66 "
Normal Tailwater Elevation	1026.66 "
Usable Storage (9 ft drawdown)	1,707 acre-ft
Earth Dam - Length	810 ft
Height (Maximum)	110 "

Spillway - Length		178.83 "
Crest Elevation		1103.66 "
Discharge Capacity	(W.S. elev. 1107.66)	4,800 cfs
	(W.S. elev. 1127.66)	67,800 cfs

The usable storage is the volume of water between the normal operating reservoir elevation and low normal level in the pond. Another 10 ft of drawdown is available before the lowest water level for operation of the hydroelectric station is reached. If additional makeup is not provided from Somerset Reservoir with 57,345 acre-ft and Harriman Reservoir with 116,075 acre-ft and hydroelectric operation is halted, then the remaining water would become a cooling pond with a submerged intake and a discharge at the surface. The available surface for evaporation would still be large and the water loss would be small compared to the water volume available. Examples of heat sinks which have been deemed acceptable by the NRC staff (Regulatory Guide 1.27) include a large lake. In this instance, Sherman Reservoir can be considered a large lake because the volume of water impounded vastly exceeds the quantity of water which may be needed for 30-day cooling during the worst meteorological conditions. In addition, the scenario of an isolated cooling reservoir is highly unlikely because "agreements with New England Electric System ensure sufficient river flow to prevent Sherman Pond temperatures from reaching objectionable limits" [17].

3.4.3.4 Vulnerability of the UHS to Probable Maximum Flood

An NRC-prepared flood study forwarded to YAEC in Reference 9 shows that Harriman Dam can safely pass about 13 inches of basin rainfall; however, the 18.9 inches (the NRC-determined design basis probable maximum precipitation) on the Upper Deerfield River Basin will overtop and fail Harriman and Sherman Dams and produce flood levels at the plant site that are 40 ft or more above plant grade. Based upon the severity of these results, it can be concluded that the UHS complex is not capable of withstanding the most severe natural phenomena expected at the site without loss of the heat sink safety function.

3.4.3.5 Vulnerability of the UHS to Reasonably Probable Combinations of Less Severe Natural Phenomena and/or Site-Related Events

A review of the UHS design compared to less severe natural phenomena and/or site-related events did not identify reasonably probable combinations which would produce an effect worse than those previously identified (e.g. loss of dam) or which would effect a significant loss of heat sink function.

3.4.4 Conclusion

Criterion 1 of Regulatory Guide 1.27 was established for heat sinks where the supply may be limited and/or the temperature of plant intake water from the heat sink may become critical. Similarly, Criterion 2 was established to ensure that the heat sink function would not be lost due to natural phenomena, site-related events, or a single failure of man-made structural features. Criterion 3 was established to provide a high level of assurance that a plant's UHS would be available when needed. The Regulatory Guide suggests that the UHS consist of at least two sources of water, unless it can be demonstrated that there is an extremely low probability of losing a single source. An UHS design that satisfies the intent of Criteria 2 and 3 then must also be capable of providing sufficient cooling for simultaneous safe shutdown and cooldown of all nuclear reactor units that it serves and to maintain them in a safe shutdown condition for 30 days as described in Criterion 1.

The Yankee Rowe UHS partially complies with Criterion 2. The UHS is capable of withstanding the following events without loss of the sink safety function:

- o Reasonably probable combination of less severe natural phenomena and/or site-related events.
- o The site-related events that historically have occurred or that may occur during the plant lifetime.

The UHS is not capable of withstanding the most severe natural phenomena expected at the site in which a PMF inundates the site and fails Sherman Dam. With respect to a single failure of manmade structural features insufficient information is available to evaluate the vulnerability of the UHS.

The Yankee Rowe UHS does not comply with Criterion 3. The UHS complex does not consist of at least two sources of water, each capable of providing cooling of the reactor and of maintaining it in a safe shutdown condition for 30 days. Insufficient information exists to support a reasonable assurance finding that there is an extremely low probability of losing the capability of the Sherman Pond. Specifically, the Yankee Rowe facility is susceptible to flooding from either a single failure of the Harriman Dam or from a PMP that overtops and fails the Harriman Dam. The UHS appears to be susceptible to a single failure of the Sherman Dam because the intake conduit does not extend to the Deerfield River bed and sufficient net positive suction head may not be available even if the intake conduit remains submerged.

Since the Yankee Rowe UHS partially complies with Criterion 2 and does not meet Criterion 3, it cannot be concluded that the UHS is capable of providing sufficient cooling for the safe shutdown and cooldown of the reactor that it serves and of maintaining a safe shutdown condition for 30 days as stipulated in Criterion 1.

Criterion 4 requires that the plant technical specifications include provisions for actions to be taken in the event that conditions threaten partial loss of the UHS. This criterion was established to ensure that the manner in which plant technical specifications were written was such that the plant would be placed in a safe condition or provisions would be implemented if a condition existed which threatened the availability of the UHS. An example of such a condition might be the prediction of a severe flood which would jeopardize a UHS dike or retaining structure, a severe drought with the potential to reduce the capacity of a cooling pond, or a prediction of severe river icing conditions which could preclude or inhibit water for a once-through cooling system.

In each of these situations, technical specifications requiring the plant to be placed in a safe shutdown condition or implementation of procedures to mitigate the consequences of a threatened partial loss of the UHS would be prudent. The Yankee Rowe Technical Specifications do not include provisions for actions to be taken in the event that the plant requires protection from nigh water during severe flooding or severe low water conditions. Therefore, the Yankee Rowe facility does not comply with Criterion 4.

4. CONCLUSIONS

4.1 HYDROLOGIC DESCRIPTION

Information pertaining to the groundwater regime at the site is not available. This information is necessary to determine a suitably conservative maximum groundwater elevation and normal high groundwater elevation for use in SEP Topics II-3.B and II-3.C, Effects of High Water Level on Structures.

A Licensee response to an NRC request for additional information pertaining to these topics has not been received. Information relating to flood design basis and limiting flood elevations of safety-related equipment is not available.

4.2 FLOOD POTENTIAL AND PROTECTION REQUIREMENTS

Local Flooding

Local flooding of the plant site is shown to occur during probable maximum precipitation (PMP) falling in basins tributary to the site. Flooding to 3 ft above grade occurs in the area of the diesel generator building and the turbine building. Flooding effects on safety-related equipment is unknown due to a lack of sufficient information.

Roof Flooding

The turbine building roof is the only roof on a safety-related structure which is not protected against the PMP.

Flooding of Rivers

Further study of this issue will be undertaken by the NRC. No final conclusions are presented in this elevation.

Failure of Dams

Further study of this issue will be undertaken by the NRC. No final conclusions are presented in this elevation.

4.3 EMERGENCY PROCEDURES AND TECHNICAL SPECIFICATIONS

An insufficient amount of background information (i.e., the Licensee response to FRC request for additional information) has affected the degree of completion of this review.

The final review of the adequacy of emergency flood procedures is dependent upon the resolution of final PMF elevations due to both a failure of Harriman Dam and the PMP-induced probable maximum flood (PMF). Further, the need to protect the intake structure is dependent upon a resolution of SEP Topic II-3.C, Ultimate Heat Sink.

4.4 ULTIMATE HEAT SINK

Criterion 1 of Regulatory Guide 1.27 was established for heat sinks where the supply may be limited and/or the temperature of plant intake water from the heat sink may become critical. Similarly, Criterion 2 was established to ensure that the heat sink function would not be lost due to natural phenomena, site-related events, or a single failure of man-made structural features. Criterion 3 was established to provide a high level of assurance that a plant's UHS would be available when needed. The Regulatory Guide suggests that the UHS consist of at least two sources of water, unless it can be demonstrated that there is an extremely low probability of losing a single source. An UHS design that satisfies the intent of Criteria 2 and 3 then must also be capable of providing sufficient cooling for simultaneous safe shutdown and cooldown of all nuclear reactor units that it serves and to maintain them in a safe shutdown condition for 30 days as described in Criterion 1.

The Yankee Rowe UHS partially complies with Criterion 2. The UHS is capable of withstanding the following events without loss of the sink safety function:

- o Reasonably probable combination of less severe natural phenomena and/or site-related events.
- o The site-related events that historically have occurred or that may occur during the plant lifetime.

The UHS is not capable of withstanding the most severe natural phenomena expected at the site in which a PMF inundates the site and fails Sherman Dam. With respect to a single failure of manmade structural features, insufficient information is available to evaluate the vulnerability of the UHS.

The Yankee Rowe UHS does not comply with Criterion 3. The UHS complex does not consist of at least two sources of water, each capable of providing cooling of the reactor and of maintaining it in a safe shutdown condition for 30 days. Insufficient information exists to support a reasonable assurance finding that there is an extremely low probability of losing the capability of the Sherman Pond. Specifically, the Yankee Rowe facility is susceptible to flooding from either a single failure of the Harriman Dam or from PMP that overtops and fails the Harriman Dam. The UHS appears to be susceptible to a single failure of the Sherman Dam because the intake conduit does not extend to the Deerfield River bed and sufficient net positive suction head may not be available even if the intake conduit remains submerged.

Since the Yankee Rowe UHS partially complies with Criterion 2 and does not meet Criterion 3, it cannot be concluded that the UHS is capable of providing sufficient cooling for the safe shutdown and cooldown of the reactor that it serves and of maintaining a safe shutdown condition for 30 days as stipulated in Criterion 1.

Criterion 4 requires that the plant technical specifications include provisions for actions to be taken in the event that conditions threaten partial loss of the UHS. This criterion was established to ensure that the manner in which plant technical specifications were written was such that the plant would be placed in a safe condition or provisions would be implemented if a condition existed which threatened the availability of the UHS. An example of such a condition might be the prediction of a severe flood which would jeopardize a UHS dike or retaining structure, a severe drought with the potential to reduce the capacity of a cooling pond, or a prediction of severe river icing conditions which could preclude or inhibit water for a once-through cooling system.

In each of these situations, technical specifications requiring the plant to be placed in a safe shutdown condition or implementation of procedures to mitigate the consequences of a threatened partial loss of the UHS would be prudent. The Yankee Rowe Technical Specifications do not include provisions for actions to be taken in the event that the plant requires protection from high water during severe flooding or severe low water conditions. Therefore, the Yankee Rowe facility does not comply with Criterion 4.

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   2.4.7 Ice Effects
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PROBABLE MAXIMUM PRECIPITATION ESTIMATES FOR THE UPPER DEERFIELD RIVER BASIN

APPENDIX A

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APPENDIX A - PROBABLE MAXIMUM PRECIPITATION ESTIMATES FOR THE UPPER DEERFIELD RIVER BASIN

This appendix presents the findings of a probable maximum precipitation (PMP) analysis of the 236-square-mile drainage basin of the Deerfield River above the Yankee Rowe Nuclear Power Generating Station [1].

Previous PMP studies conducted by the Nuclear Regulatory Commission (NRC) staff and Yankee Atomic Electric Company (the Licensee) have been assessed, with emphasis on the Licensee's compliance with Regulatory Guide 1.59 and the propriety of the meteorological judgments involved.

BACKGROUND

It is noted that Regulatory Guide 1.59 has its basis in the Corps of Engineers procedures which, prior to the publication of Hydrometeorological Report No. 51 (HR 51) [2], relied heavily on the isohyets on generalized maps of PMP variation contained in Hydrometeorological Report No. 33 (HR 33) [3].

Although the hydrometeorological experts who wrote HR 33 were aware of the unreliability of the isohyets for small basins in the rugged terrain of the Appalachians and Ozarks and included recommendations for detailed studies of terrain effects, their generalized maps show smooth isohyets crossing the Appalachian region. The authors of HR 51 further stressed the hazards of overlooking the complicating effects of rugged terrain by stippling the entire Appalachian area of their maps of PMP variation.

Terrain effects can either increase or decrease the PMP over rugged basins. A widely used approach to addressing this complex issue has been the transposition of actual extreme rainstorms from the locale where they occurred to the basin under consideration, applying empirical correction factors to modify the storm for its new locale. Usually, the maximization process assumes that moisture supply to the area under study is the only variable not already maximized by natural processes.

The studies performed for Yankee Atomic Electric Company (YAEC) [4-6] made use of the above-mentioned storm transposition approach. This is a much more reasonable approach than relying on the generalized isohyets of HR 33 [3], provided (1) the storms selected for transposition truly reflect the pertinent history and (2) the empirical adjustment factors reflect valid meteorological judgment.

This study consists of the development of four input assumptions necessary for determination of the estimate of PMP as follows:

1. Study of terrain in and around the Deerfield River Basin for its likely effect on the inflow of moist air
2. Examination of the storm transpositions covered in the reports provided for review
3. Consideration of storms which were not transposed in the reports provided but which logically could have been
4. The determination of a reasonable <u>range</u> of PMP estimates and storm durations that should be used by hydrologists to prepare the design basis flood for use in judging the adequacy of protective works at the Yankee Rowe Station.

TRAJECTORIES AND VELOCITY OF MOVEMENT OF HURRICANE AND TROPICAL STORMS

In recognition of the importance of hurricane and tropical storm rainfall to the production of near PMP accumulations over the Upper Deerfield River Drainage Basin, a study was performed to determine the trajectories of past hurricanes and tropical storms. Also, past storms which came closest to providing the conditions most favorable to the production of heavy rainfall over the area of interest were identified.

Reference 7 presents trajectories for 34 storms which passed within 300 nautical miles of the Yankee Rowe Generating Station. Ten of these storms came as close as 100 nautical miles and seven of those were 50 miles or less at their closest approach.

The September 1938 hurricane passed directly over the Upper Deerfield River Drainage Basin and continued a due north trajectory beyond the basin. The life cycle track of this storm is shown in Figure A-1. It produced 4.94

inches of rainfall at Harriman Dam in 24 hours and a record runoff from the basin as measured at Charlemont, Massachusetts. Hurricanes Diane and Agnes passed within 140 and 120 nautical miles, respectively, and both deposited heavy rainfall centers nearby (see Figures A-2 and A-3).

When a hurricane or tropical storm leaves the open ocean and comes ashore, increased friction affects the lower levels of the storm. What has previously been an essentially vertical axis of circulation acquires a tilt as the upper levels continue their march unaffected by the increased friction felt by the lower levels. The effect of this change of circulation is a rapid unloading of precipitation fairly near the coast. The rougher the underlying terrain, the more dramatic the unloading; but the process does not continue indefinitely. Shut off from the energy-supplying water surface, the storm continues to produce precipitation but at decreasing intensities. The unloading previously mentioned would primarily occur over the first rising terrain encountered. The approaches to the Upper Deerfield River Drainage Basin from the south and southeast provide sufficiently elevated terrain to offer protection against the occurrence of initial unloading over the basin.

The rate of movement of a precipitation-producing storm over a catchment area is a major factor in determining the resulting accumulation. In areas where a storm center decelerates, intense rainfall rates operate over an increased time span. Sometimes hurricane and tropical storm center trajectories recurve so sharply that a complete loop is traced. This looping trajectory permits the center to affect an area for a longer period of time. Trajectories for the period 1871-1977 were examined, showing that such looping trajectories were limited to latitudes far south of the Upper Deerfield River Basin. The closest looping trajectory for a storm still in the hurricane or extra-tropical stage occurred in the period September 11-16, 1961, at latitude 38 N. There is good reason to expect an absence of high latitude looping trajectories from the future record of storm behavior. In the lower latitudes, a weakening cyclonic circulation may have to wait for some time for a trough aloft to come by from the west and supply the energy for renewed transport to the northeast. Farther north, strong westerlies operate to keep

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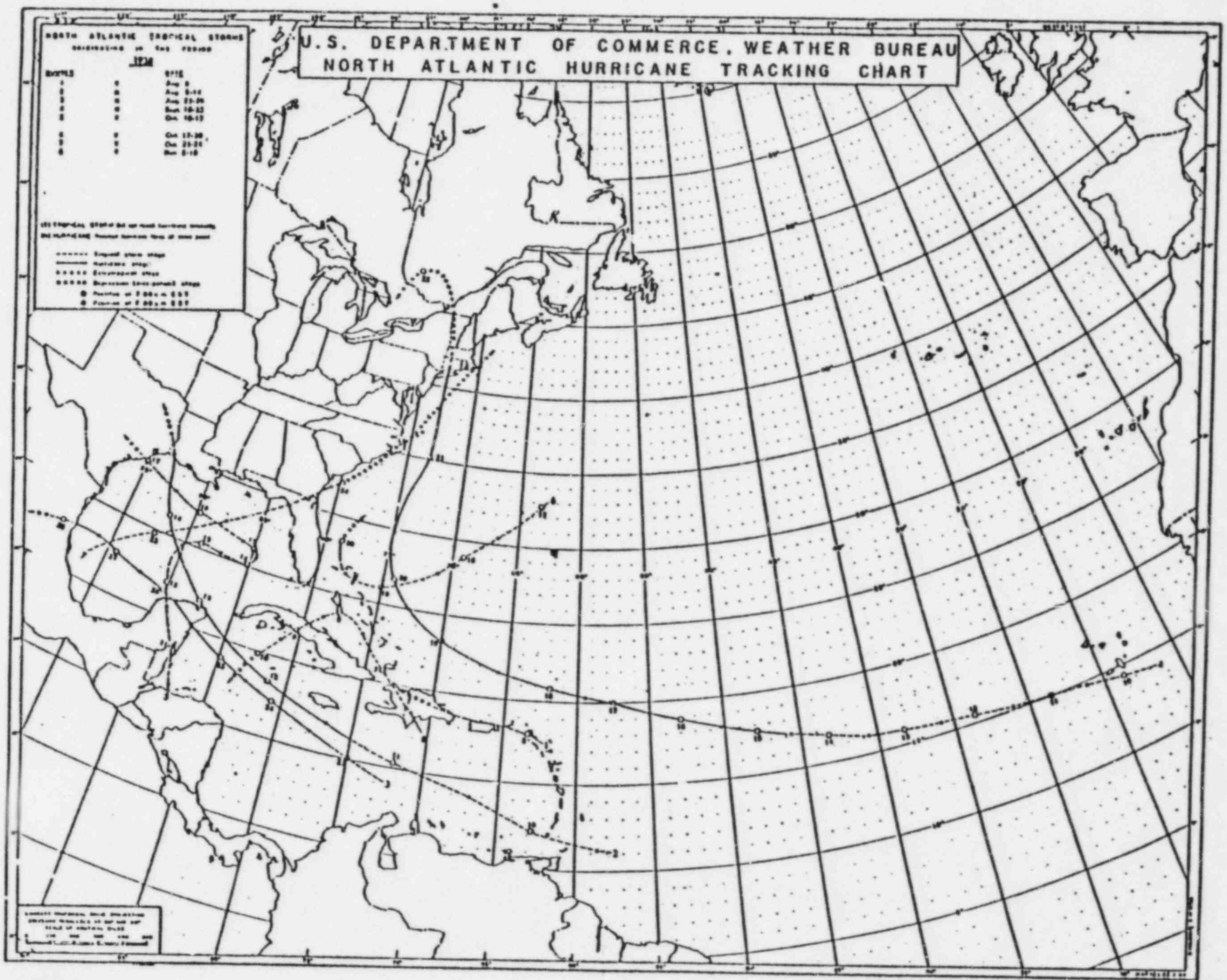

Figure A-1. Hurricane Trajectory for North Atlantic Tropical Storms - 1938

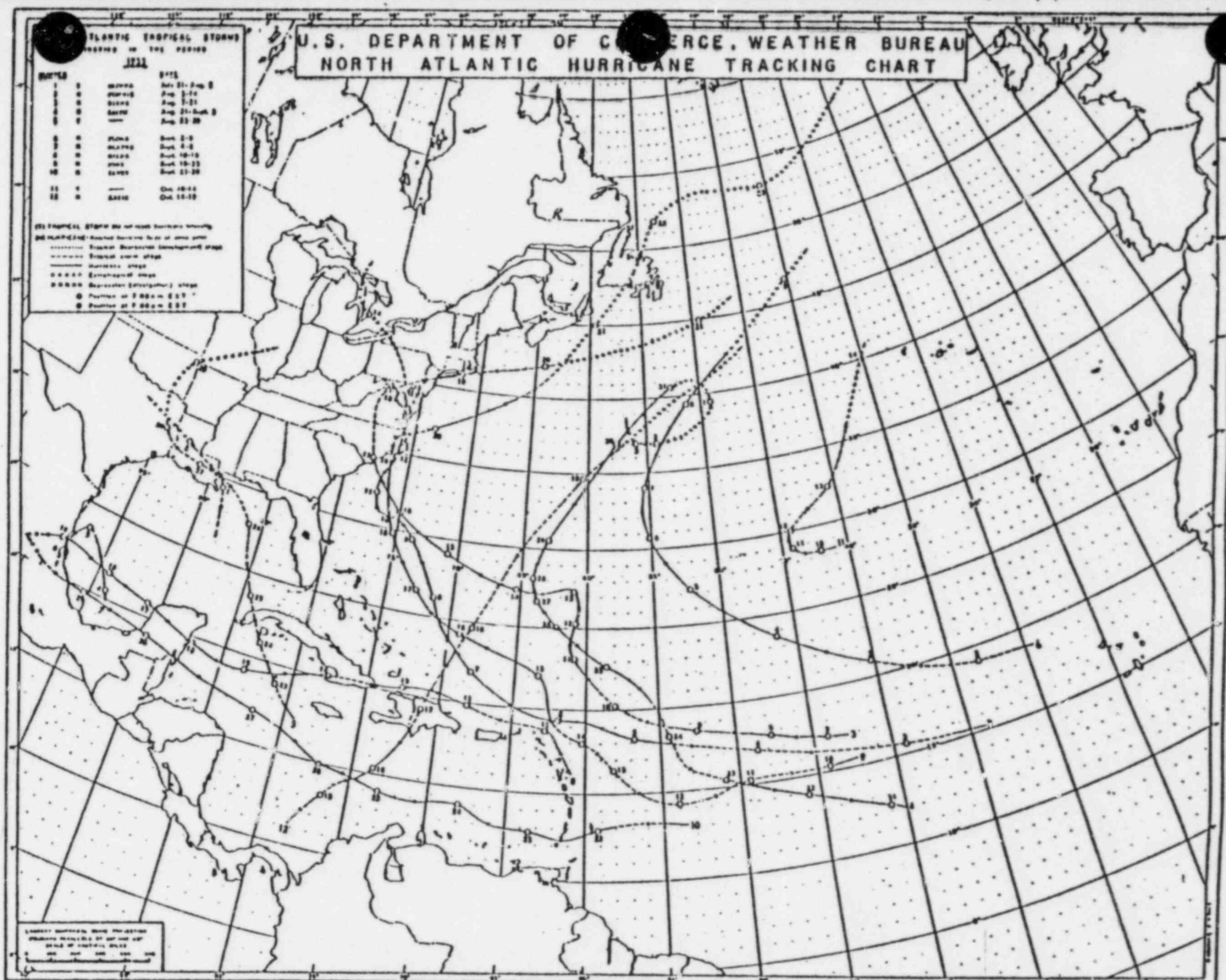

Figure A-2. Hurricane Trajectory Showing Hurricane Diane's Position Relative to the Yankee Rowe Site

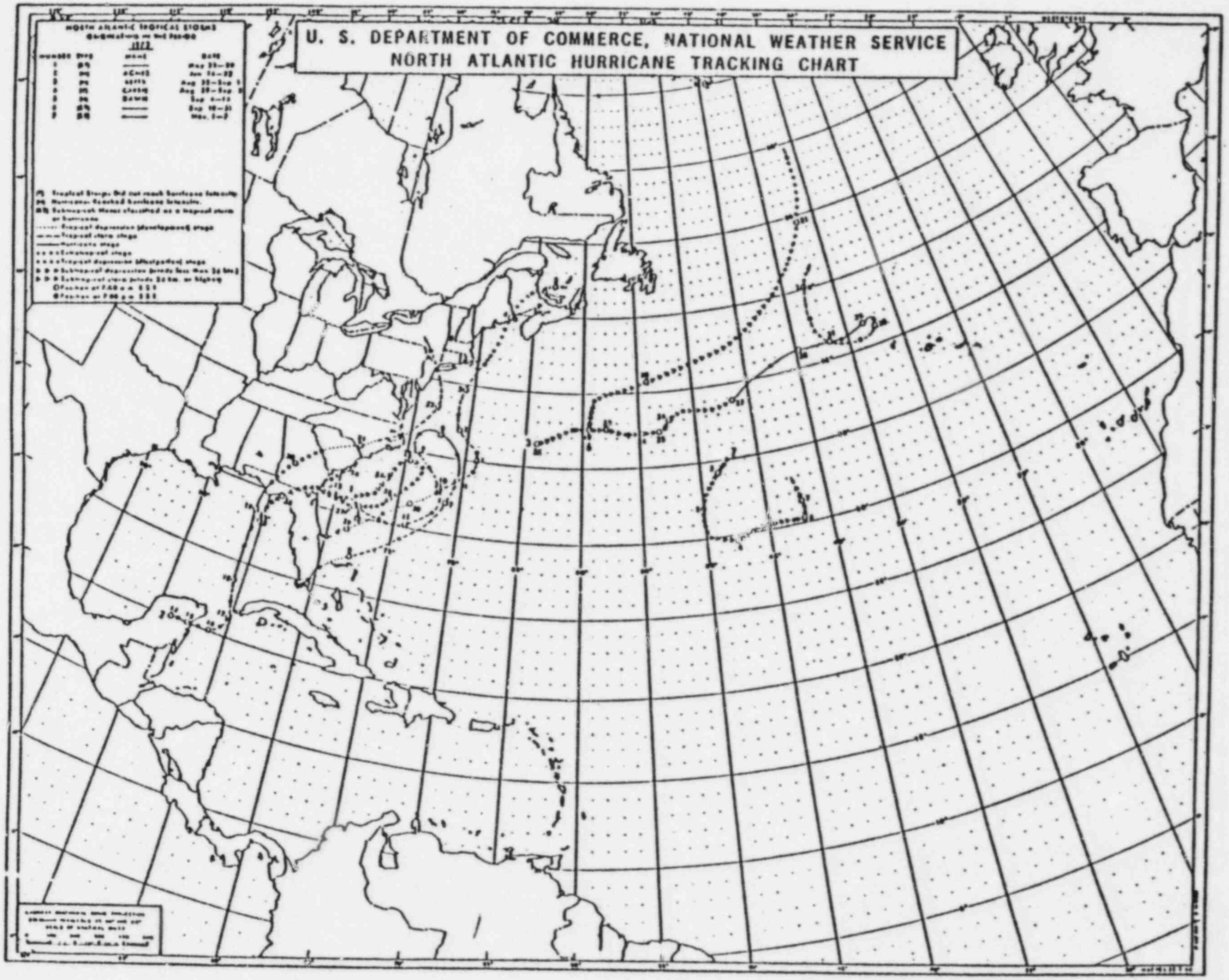

Figure A-3. Hurricane Trajec[illegible] Showing Hurricane Agnes' Path Relative to [illegible] Yankee Rowe Site weakening circulations from stalling in place. All 34 storms that passed within 300 miles of the Yankee Rowe Station showed 24-hour travel distances in excess of 200 miles.

Reference 8 presents maps showing the number of tropical cyclones passing through 2.5° latitude-longitude boxes over the Atlantic Ocean and nearby coastal areas, and their velocity of movement. While data available for the box in which the Yankee Rowe plant is located were too sparse to be analyzed, adjoining boxes show no storm movements slower than 20 knots.

Long residence time for a storm over the 236-square-mile Upper Deerfield Basin is not required for production of PMP rainfall. We have produced three maps which we hope will put in proper perspective the matter of required residence time over the basin for storms capable of producing PMP. The heaviest rainfall of record over the basin was produced by the September, 1938 storm which moved across Long Island with a velocity of approximately 56 miles per hour.

Figure A-4 shows the precipitation situation at 1600 EST on 21 September 1938. The rainfall distribution for this map was derived from Figure 86 on page 181 of Reference 9.* The rainfall at 1600 EST at reporting stations was classified as follows: one dot - light rain, double dot - moderate, triple dot - heavy. The definitions of light, moderate, and heavy rain as employed by Weather Bureau Observers are as follows [17]:

Light: Trace to 0.10 inch per hour; maximum 0.01 inch in 6 minutes

Moderate: 0.11 inch to 0.30 inch per hour; more than 0.01 inch to 0.03 inch in 6 minutes

Heavy: More than 0.30 inch per hour; more than 0.03 inch in 6 minutes.

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*Figure 86 originally appeared in an article by C. Y. Pierce entitled "The Meteorological History of the New England Hurricane of September 21, 1938," which appeared in Monthly Weather Review, LXVII (1939), pp. 237-285.

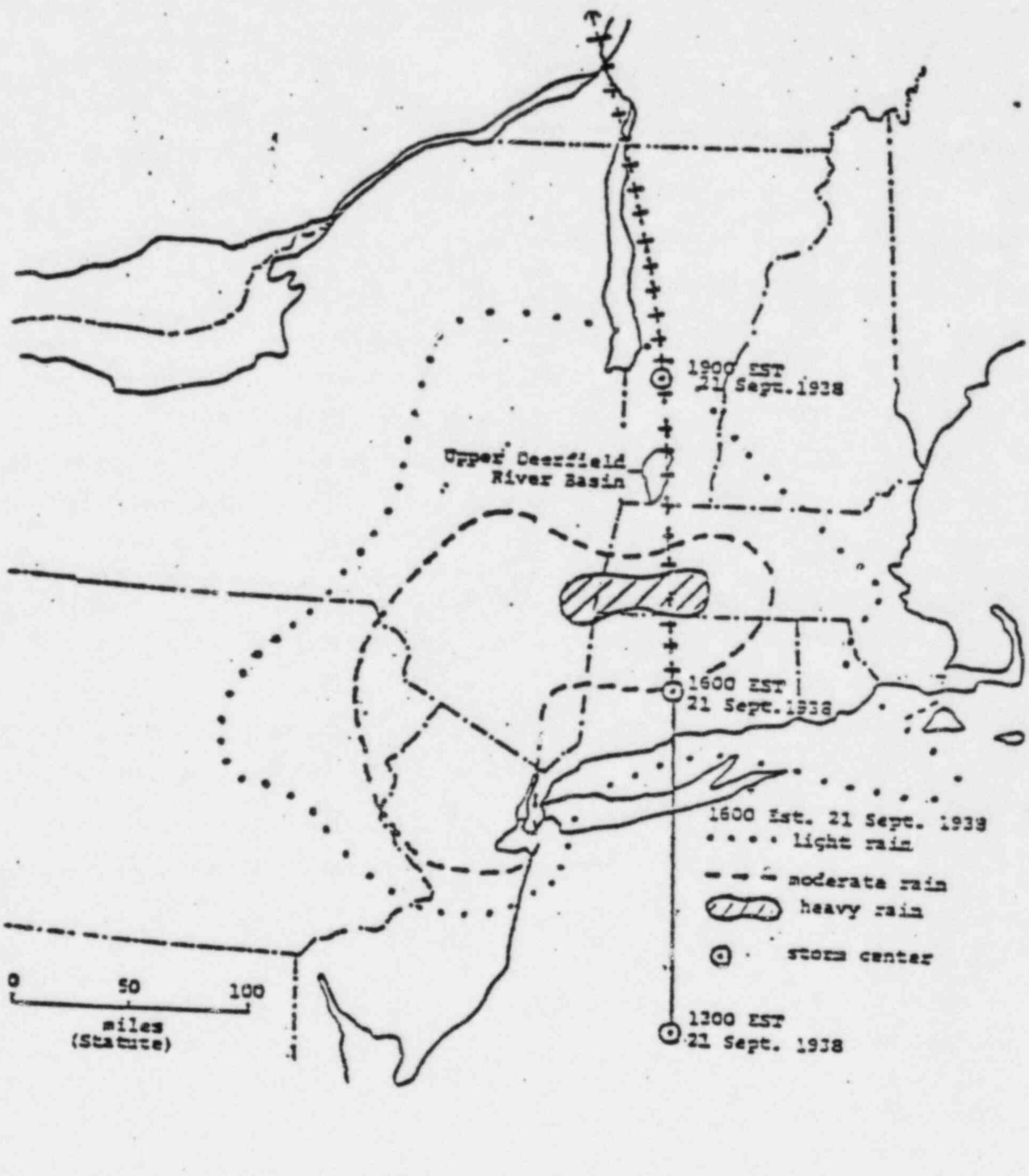

Figure A-4. Map Showing Rainfall Intensity Distribution on September 21, 1938 at 1600 EST

Clearly, "more than 0.30 inch per hour" can apply to very intense rainfall, such as 2 or 3 inches per hour.

Only two of the reporting stations were experiencing three-dot rain at 1600 EST, shortly after the storm center made landfall in south central Connecticut. Seven stations were experiencing two-dot rain and 12 reported one-dot rain. Stations 200 miles northeast and southwest of the storm center were not experiencing any rainfall at the time. At 1600 EST, the Upper Deerfield River Basin lay between the one-dot and two-dot isopleths.

The storm track plotted on the map in Figure A-4 is taken from Figure A-1, which is a photocopy of the hurricane trajectories for 1938 [7].

The centroid of the jelly-bean-shaped center of three-dot rain was 50 miles NNW of the storm center and 40 miles SSW of the Yankee Rowe site at 1600 EST. Three hours later, at 1900 EST, the storm center had moved approximately 145 miles to the NNW of its 1600 EST position. If the center of heavy rainfall maintained a fixed position relative to the storm center, it too would be found 145 miles NNW of the Yankee Rowe site. The center of three-dot rain had a south-to-north dimension comparable to the south-to-north dimension of the Upper Deerfield River Basin. Its estimated positions at 1600 EST, 1700 EST, 1800 EST, and 1900 EST are shown on Figure A-5. If the area of heavy intensity retained its initial shape and its initial position relative to the storm center, its residence time over the Upper Deerfield River Basin, from first entry at the south end of the basin to last effect at the extreme north, could not be much more than 1 hour. However, this storm produced 4.95 inches in a nominal 24-hour period at Harriman Dam. When observations are taken once a day, 24-hour durations are frequently assigned to rainfall that may have had a much shorter actual duration. The area that was experiencing rain of any intensity (even as small as a trace per hour) at 1600 EST had a south-to-north dimension of 270 miles. The observed velocity of over 45 mph would have carried the precipitation across the Upper Deerfield River Basin in a period of about 6-1/2 hours if the maximum south-to-north dimension is considered to have affected the basin.

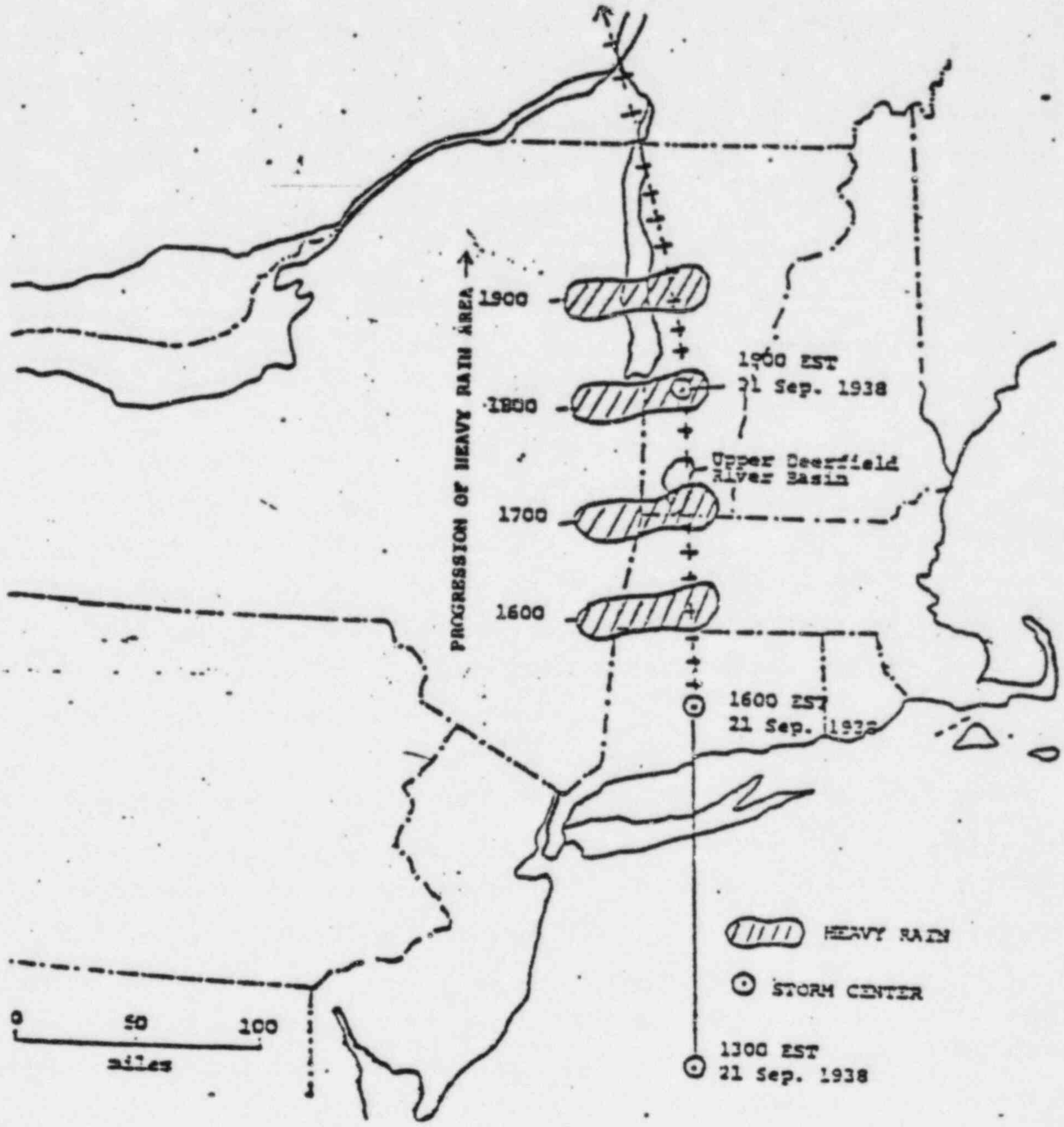

Figure A-5. Map Showing Heavy Rainfall Intensity Progression Northward

Figure A-6 depicts the idealized slow passage northward of a heavy rainfall area located to provide maximum impact over the Upper Deerfield River Basin. The track of the storm has been displaced westward to permit the maximum south-to-north dimension of the heavy precipitation shield to move along the south-to-north axis of the basin. The velocity of movement has been slowed to 16 mph, one-third of the velocity observed in connection with the storm of September 21, 1938. Figure A-7 [10] in this report shows the upper and lower limits of velocity for severe probable hurricanes as they tracked northward. By the time such storms reach the latitude of Boston, Massachusetts, they are expected to exhibit a minimum velocity of 20 knots. This would seem to indicate that the choice of 16 mph is quite conservative. Added confirmation of the 16 mph translational movement in New England is supported by Reference 11. To choose a much lower value for the velocity of the storm would come into conflict with the production of very heavy rain. In a discussion of forward speed [10], one finds the statement, "Very slow speeds weaken a hurricane," and the statement, "As the angle between the coastal orientation and θ decreases, the slower hurricane weakens more than the faster-moving hurricane." The weakening referred to in these statements is primarily concerning winds in the hurricane. While the relationship between wind velocity and production of heavy rain is by no means linear, there is a definite requirement for stro g convergence in the flow of moist air if heavy rains are to result. Suppose that the maximum south-to-north dimension of the precipitation shield in our idealized case is 45 miles. At a velocity of 16 mph, it would affect the Upper Deerfield River Basin for approximately 4 hours. If one assumes continuing precipitation intensities comparable to those experienced in the September 21, 1938 storm, the reduction in velocity to 16 mph and increased precipitation shield would produce up to 15 inches of rain over the Upper Deerfield River Basin. This would be clearly in the PMP range without long residence time over the basin.

STORMS SUITABLE FOR TRANSPOSITION TO THE UPPER DEERFIELD RIVER BASIN

Professional judgment suggests that the PMP for the Upper Deerfield River Basin would result from a hurricane or tropical storm whose center of low

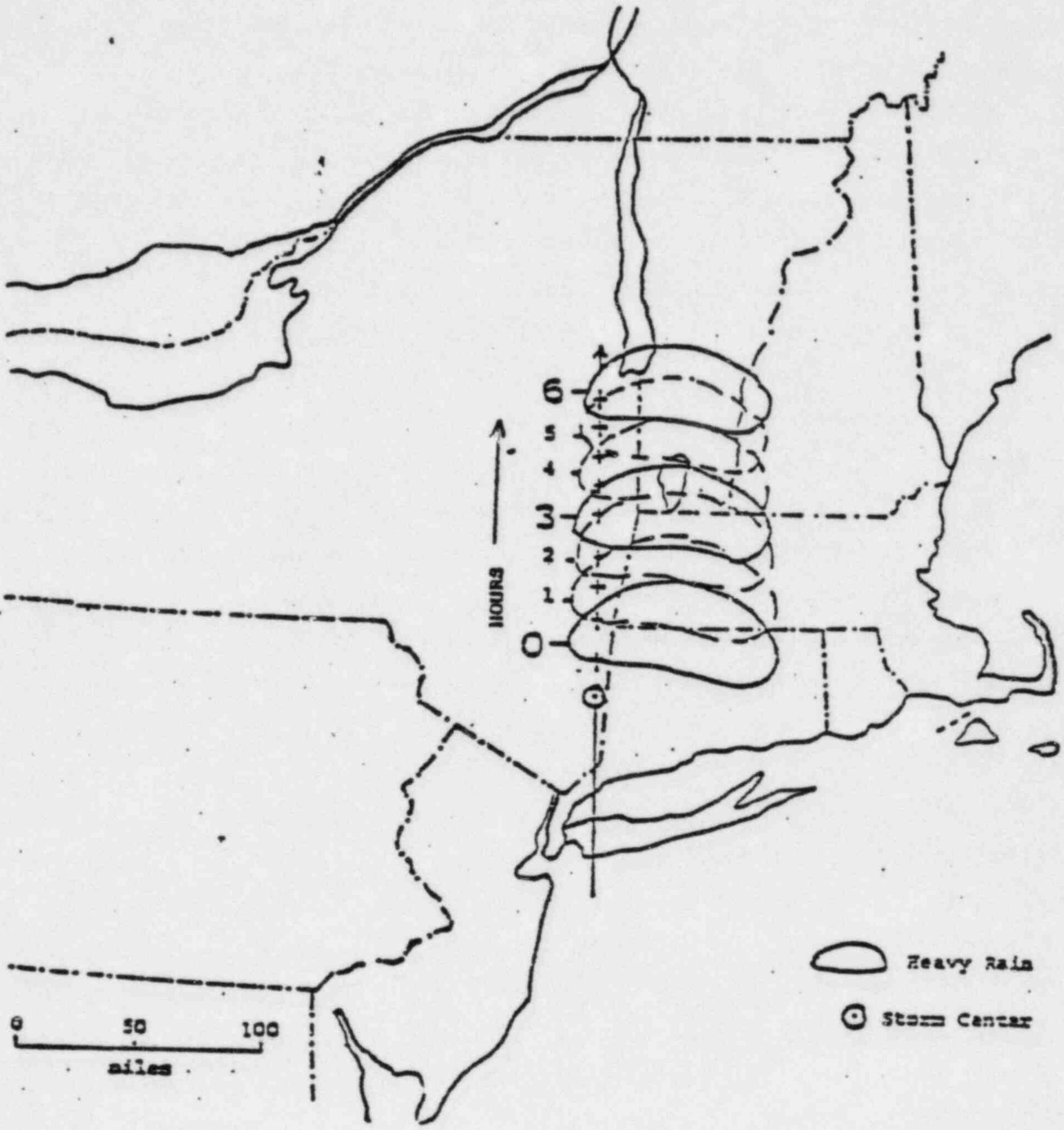

Figure A-6. Map Showing Idealized Slow Passage of Heavy Rain Northward

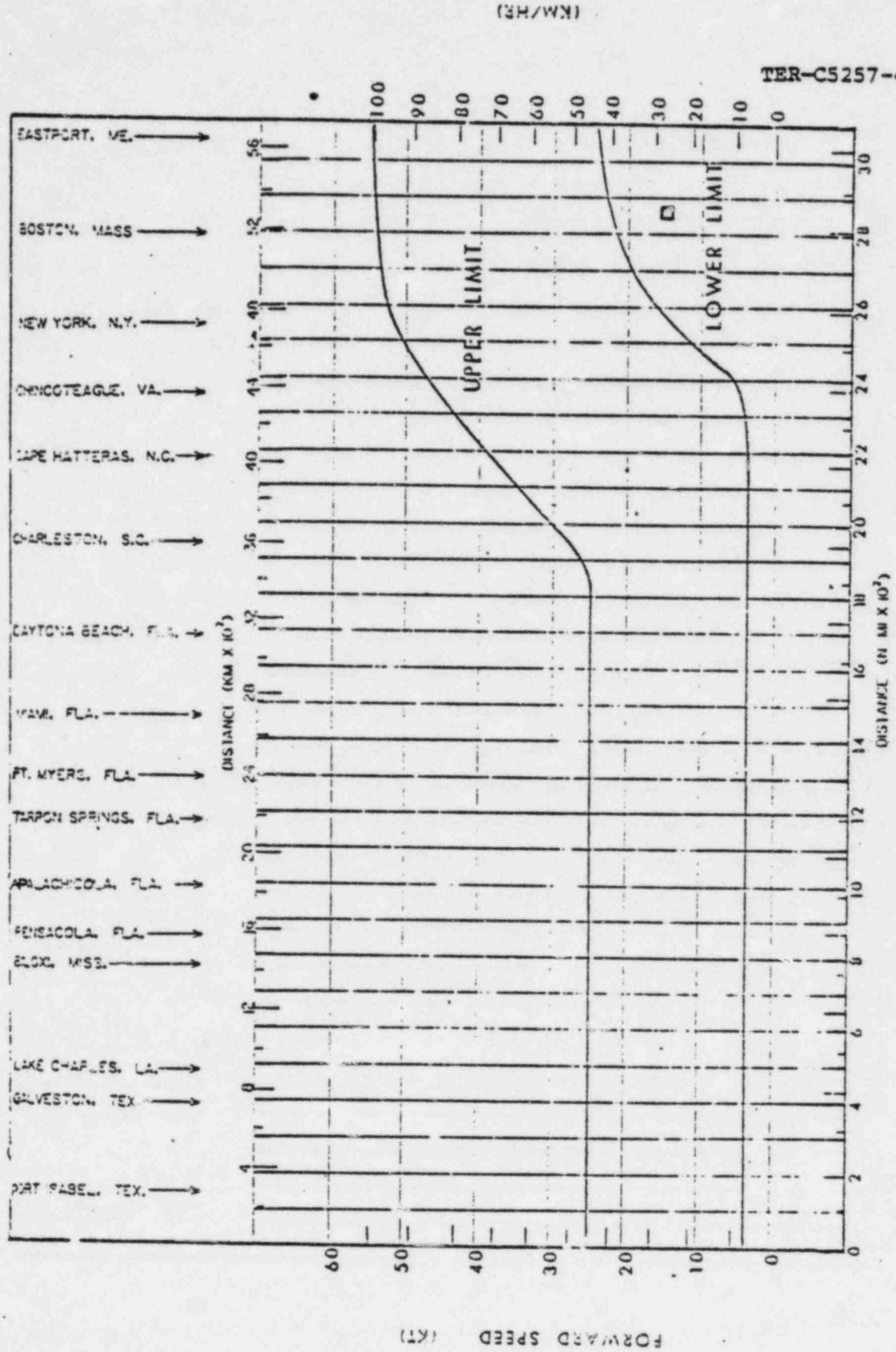

Figure A-7. Adopted Standard Project Hurricane Upper and Lower Limits of T [10] pressure decelerated west-southwest of the Yankee Rowe plant under the influence of a deep pressure trough aloft approaching from the west. The resulting quasi-stationary circulation would feed moist, unstable air northward and expose the drainage area to the heavy rainfall characteristics of the northeast quadrant of such storms. The hurricane of September 1938, Hurricane Diane in August 1955, and Hurricane Agnes in June 1972 are examples of storms which could have provided near-PMP rainfall in the Upper Deerfield River Basin had their trajectories taken them farther north before they recurved toward the northeast.

Both YAEC and the NRC included the Westfield, Massachusetts center, a product of Hurricane Diane, for transposition to the Upper Deerfield Basin - an acceptable procedure. Noted, however, was the elimination from consideration of the 18+ inch (storm total) center produced by Hurricane Agnes in northern Pennsylvania. YAEC apparently considered transposing this storm in its January 1980 report, but dropped it from the list of transposable storms in its October 1980 report. A study of this storm yielded the conclusion that its inclusion would not alter the enveloping curve developed by YAEC, but it certainly falls closer to the curve than some of the lesser storms which were included.

On September 16, 1932, a dying hurricane that had reached the extra-tropical stage passed 60 miles offshore from Rhode Island moving at 23 knots on a northeasterly course. Reference 2 indicates that the storm delivered maximum average depths of rainfall of 6.8 inches in 6 hours, 10.2 inches in 12 hours, 11.1 inches in 18 hours, and 11.6 inches in 24 hours over an area of 200 square miles near Scituate, Rhode Island.

The representative dew point for the Scituate storm was 63°F at a distance of 75 miles east of Westerly, Rhode Island, which is located at 41°22' N, 71°50'W. The maximum persisting 12-hour, 1000 mb dew point for mid-September is 71°F for a point 75 miles east of Westerly, Rhode Island, and 70°F for a point 75 miles east of the centroid of the Upper Deerfield River Basin [12].

The precipitable water depth for a saturated atmosphere having a 63°F temperature at 1000 mb is 1.61 inches. The depth for a 71°F temperature is 2.38 inches. The depth for a 70°F temperature is 2.27 inches. The mean elevation of the Upper Deerfield River Basin is 2100 feet. The precipitable water depth between 1000 mb and 2100 feet for a 63°F 1000 mb dew point is 0.34 inch [13]. Subtracting 0.34 inch from 1.61 inches gives 1.27 inches.

The adjustment factor or ratio to be applied for maximizing, transposing, and correcting for elevation change from the storm site to the Upper Deerfield River Basin takes the form:

$$r = (2.38/1.61) \times (2.27/2.38) \times (1.27/1.61)$$

$$= 1.48 \times 0.95 \times 0.79$$

$$= 1.11.$$

Applying this factor to the depth duration data for 200 square miles listed above yields PMP values of 7.55 inches in 6 hours, 11.3 inches in 12 hours, 12.3 inches in 18 hours, and 12.9 inches in 24 hours. None of these values exceed the enveloping curve, Figure A-8, derived by YAEC.

On September 1, 1940, a hurricane passed 120 nautical miles offshore from New Jersey moving at 19 knots on a northeasterly course. The storm delivered maximum average depths of rainfall of 15.0 inches in 6 hours and 16.5 inches in 12 hours over an area of 200 square miles near Ewan, New Jersey [2]. The representative dew point for the storm was 72°F at a point 50 miles SSW from Latitude 39°42' N, longitude 75°12' W.

The maximum persisting 12-hour 1000 mb dew point for mid-August for a point 50 miles SSW from Ewan is 76°F. The comparable value for a point 50 miles SSW from the centroid of the Upper Deerfield River Basin is 74°F [12].

The precipitable water depth for a saturated atmosphere having a 72° temperature at 1000 mb is 2.50 inches. The depth for 74°F is 2.76 inches, and for 76°F, 3.04 inches. The depth between 1000 mb and 2100 feet for a 72°F surface dew point is 0.46 inches. Subtracting 0.46 inch from 2.50 inches gives a remainder of 2.04 inches.

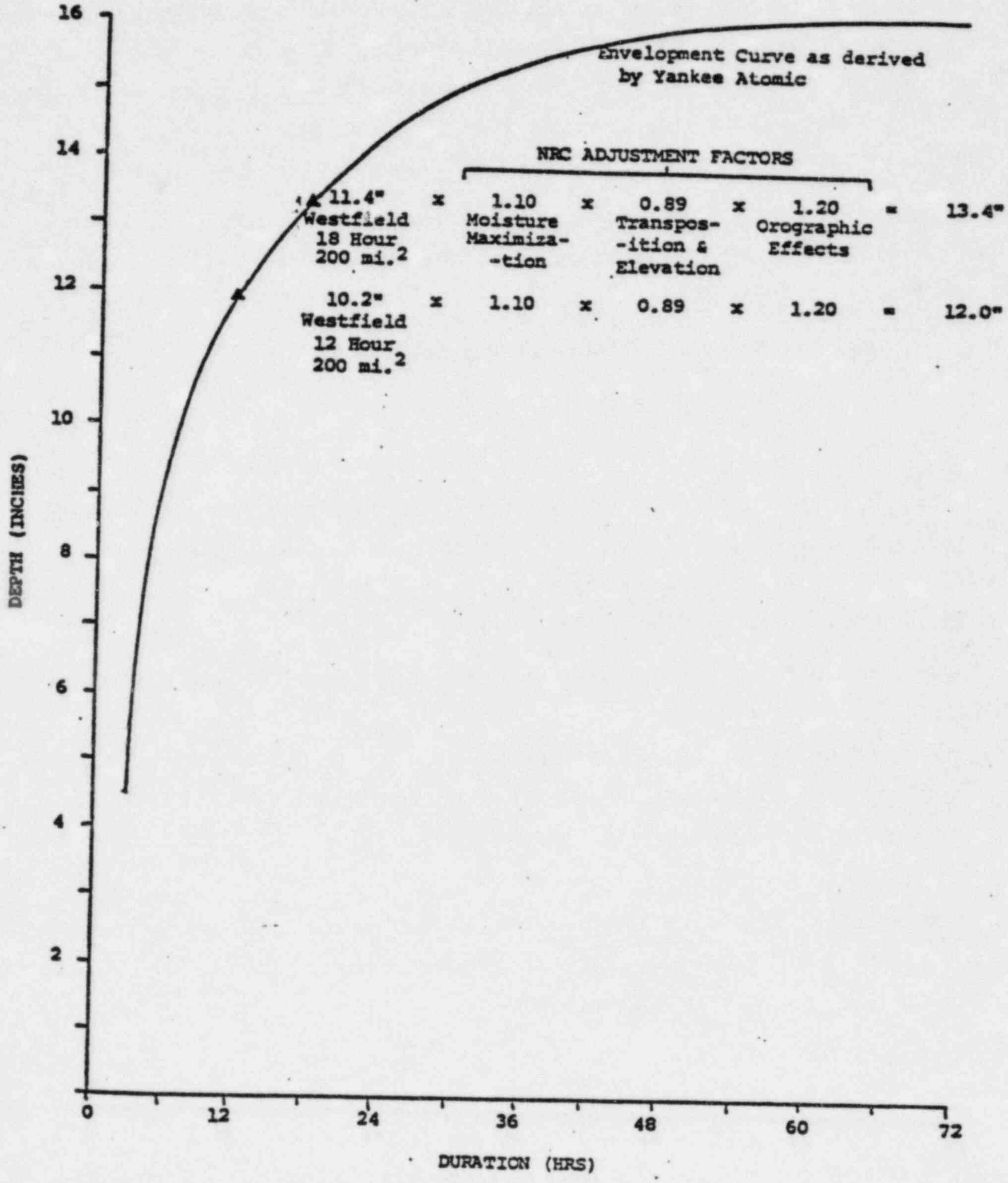

Figure A-8. Depth-Duration Curves of Deerfield Basin, Yankee Rowe Nuclear Plant

The adjustment factor for maximizing, transposing, and correcting for elevation change between the Ewan storm site and the Upper Deerfield River Basin takes the form:

$$r = (3.04/2.50) \times (2.76/3.04) \times (2.04/2.50)$$

$$= 1.22 \times 0.91 \times 0.82$$

$$= 0.91.$$

Applying this factor to the depth duration for 200 square miles yields PMP values of 13.65 inches in 6 hours, and 15.0 inches in 12 hours. These values plot well above the enveloping curve derived by YAEC and somewhat above the modified envelopment curve, Figure A-9, discussed later in this report.

The suitability of this storm for transposition is doubtful. Data from HMR 51 [2] are limited to 12 hours' duration and a 1000-square-mile area. A large thunderstorm can cover 1000 square miles. A hurricane center that never left the warm water of the Atlantic Ocean can be expected to produce much more rainfall along a nearby coast than one with an extended overland trajectory would produce over an inland basin.

A storm that produced record-breaking rainfalls occurred July 17-18, 1942, at Smethport, Pennsylvania. It was not hurricane-related, but featured a trough aloft near the Appalachians and a very moist current of air flowing from the southwest. Its 200-square-mile rainfall depths were 13.1 inches in 6 hours, 16.8 inches in 12 hours, 19.3 inches in 18 hours, and 19.9 inches in 24 hours. Its 10-square-mile value for 24 hours was an astounding 29.2 inches. Hydrometeorologists have generally declined to transpose storms across major mountain barriers. The Smethport storm is usually considered for transposition only in the area west of the Appalachian crest, since the westward slopes of the Appalachians are considered to have played a major role in the production of the heavy precipitation.

Present understanding of terrain effects on precipitation does not provide a precise basis for computing what fraction of the Smethport rainfall

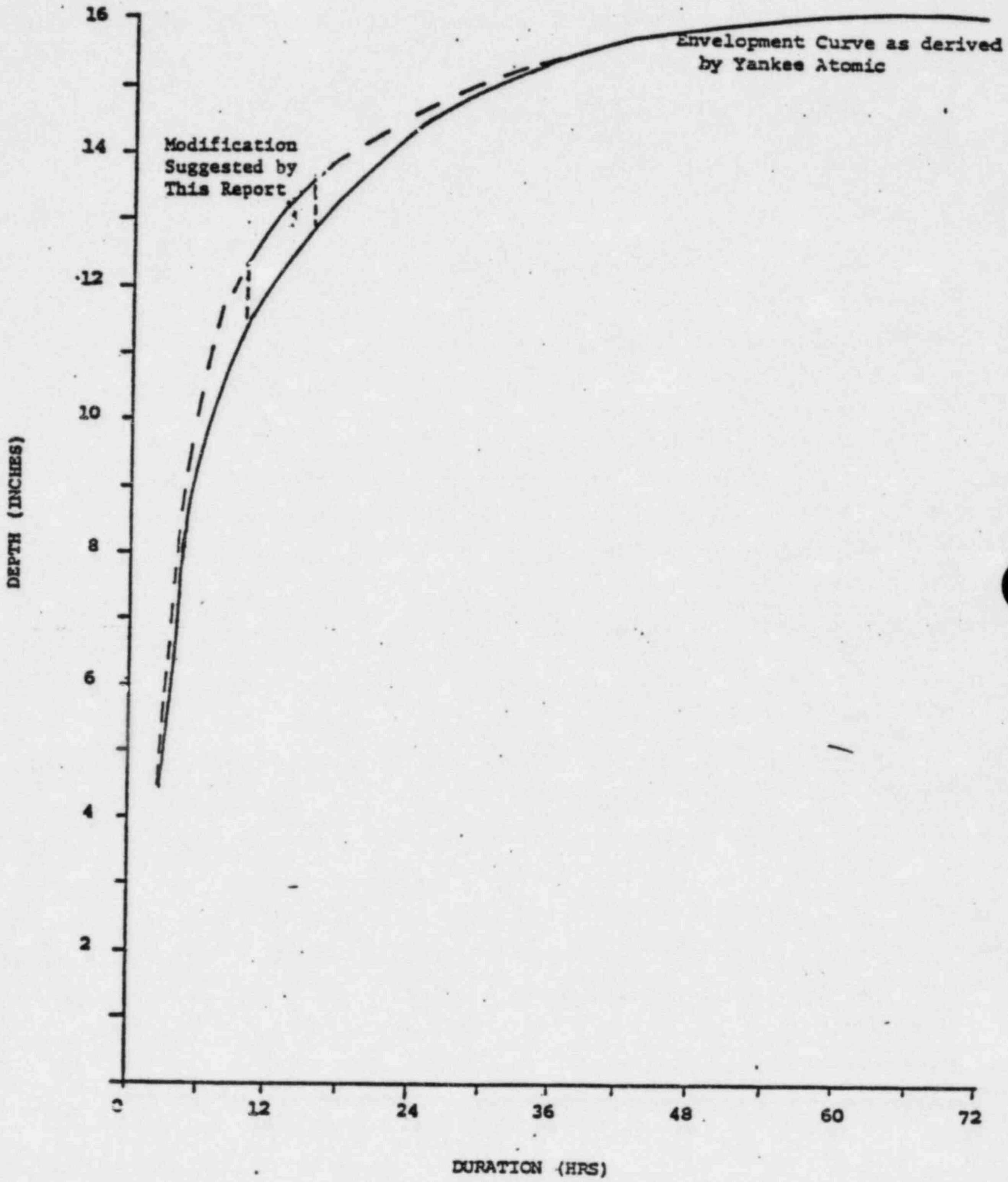

Figure A-9. Modified Depth-Duration Curve for Deerfield Basin.

could be reasonably expected to occur over the Upper Deerfield drainage, but the occurrence of such heavy rainfall so far north in the Appalachians should serve as a warning signal to designers of protective works.

While winter storms have produced heavy rainfall in the drainage area, it is noted that such storms are not part of the PMP concern. The high dewpoints required to produce the PMP do not occur in the winter months.

ESTIMATES OF PROBABLE MAXIMUM PRECIPITATION

Differing design maximum rainfall (DMR) estimates were developed by YAEC and NRC. Yankee Atomic proposed a DMR of 14.3 inches over 200 square miles in 24 hours. This value was reached by enveloping data obtained by storm transposition. The NRC points to a value of 18.9 inches over 200 square miles in 24 hours that generalized maps of PMP in HR 33 would suggest for the drainage basin, but developed a lesser estimate of 16.6 inches for 200 square miles in 24 hours, also by storm transposition. Both the 14.3-inch and 16.6-inch estimates depend to a major extent on data derived by transposing the Westfield, Massachusetts storm of August 17-20, 1955, an offshoot of Hurricane Diane.

HR 51 [2] shows data for the Westfield, Massachusetts storm. The maximum average depth of rainfall for 200 square miles was 7.4 inches in 6 hours, 10.2 inches in 12 hours, 11.4 inches in 18 hours, 14.2 inches in 24 hours, 17.1 inches in 30 hours, 17.6 inches in 36 hours, 18.2 inches in 48 hours, and 18.4 inches in 60 hours. HR 51 suggests a moisture adjustment factor of 110; this means that the observed values listed above can be converted to PMP values in place by increasing each rainfall depth by 10%.

After storm rainfall is maximized in place by applying the moisture adjustment discussed above, storm transposition practice calls for application of additional adjustment factors which are expected to compensate for differences between the area where the storm occurred and the area to which it is being transposed. A moisture adjustment is applied if the maximum dew point suited to the new area differs from that for the area where the storm occurred. An elevation adjustment factor is applied if an increase in surface elevation from the storm area to the new area would operate to reduce the depth of the moist columns of air which would be involved in the precipitation process. Finally, an adjustment factor for topographic effects on rainfall is applied where a topographic barrier can operate to affect the moisture available to the precipitation process.

Both YAEC and NRC applied the moisture adjustment factor of 1.10 to the 14.2 inches of rainfall over 200 square miles for 24 hours that occurred in the Westfield storm. The Licensee applied reduction factors of 0.95 and 0.84 for transposed moisture difference and elevation increase, respectively. The product of 1.10 x 0.95 x 0.84 is 0.88, which reduces the 14.2 inches observed value to 12.5 inches. Their enveloping curve of overall storm experience provided the basis for increasing this to 14.3 inches. NRC applied a factor of 0.89 for transposition and elevation effects and in addition applied a factor of 1.20 to reflect its evaluation of orographic barrier effects. The product of 1.10 x 0.89 x 1.20 is 1.17, which, when applied to the 14.2 inches of observed rainfall, gives the 16.6 inches as an estimate of DMR to be used in judging the adequacy of existing control structures.

The significant difference between the YAEC estimate and the NRC estimate arises primarily from the decision by NRC to apply the 1.20 orography factor. In the absence of that factor, the two estimates would have been 14.3 and 13.9 inches, a remarkably close agreement considering the subjectivity involved in the estimation process.

The decision by NRC to apply the 1.20 orography factor appears to have been influenced by the following paragraph contained in HR 45 [15]:

> "Precipitation increase of 10 percent per 1000 feet from sea level up to 2500 feet on <u>first</u> <u>upslopes</u> with no further increase above 2500 feet. Precipitation increase of 5 percent per 1000 feet from sea level on <u>secondary</u> <u>upslopes</u> at all elevations.
>
> Five percent decrease per 1000 feet of depression in sheltered areas."

To illuminate the reality of orographic barriers operating to increase extreme precipitation 20% as a result of transposing a storm from the Westfield, Massachusetts area to the Upper Deerfield River Basin above Sherman

Dam, an examination of the topography of the region was conducted. Four east-west cross sections were constructed taken at equidistant intervals over the 60 miles from Westfield, Massachusetts to Somerset Dam in Vermont. The USGS Albany Quadrangle, NK 18-6 revised 1974, scale 1:250,000 was used to pick off the elevations of terrain in the cross sections. The east-west orientation was chosen because of the south-to-north flow of moist air that would be required to produce the PMP in the Upper Deerfield Basin. These cross sections are shown in Figure A-10.

The increase in elevation along the 72°45' W longitude line from cross section D-D' to cross section A-A' is of the order of 300 feet in the 60-mile interval. The mean elevation of the western half of cross section A-A', which half transects the Upper Deerfield River Basin, is approximately 2400 feet. Comparable mean elevation for the western half of B-B', C-C', and D'D' are 1700, 1300, and 1100 feet, respectively. It is noted here that such a gradual upslope does not provide an orographic barrier capable of increasing the extreme rainfall observed at Westfield, Massachusetts by 20%.

Figure A-11, "Adjustments of Non-orographic PMP for Elevation and Slope, Hawaiian Islands," [16] is presented here, despite the fact that the Upper Deerfield River Basin is located in Vermont and Massachusetts, because the chart may help to explain the combined effect of elevation and slope on non-orographic precipitation. In the moisture-rich air of the Hawaiian Islands, the combined effect of increasing elevation and increasing slope is negative when elevations and slopes comparable to those in the Upper Deerfield River Basin apply.

To make the application of Figure A-11 to the PMP for the Upper Deerfield River Basin more understandable, we ask the reader to consider the following illustrative example:

Suppose a sea level PMP of 19 inches had been selected for the vicinity of the Upper Deerfield River Basin and it is desired to adjust this value for the elevation and slope the moist air would encounter if it flowed from south to north during the storm. The elevation at the Yankee Rowe site is approximately 1150 ft above mean sea level. The terrain rises sharply to over

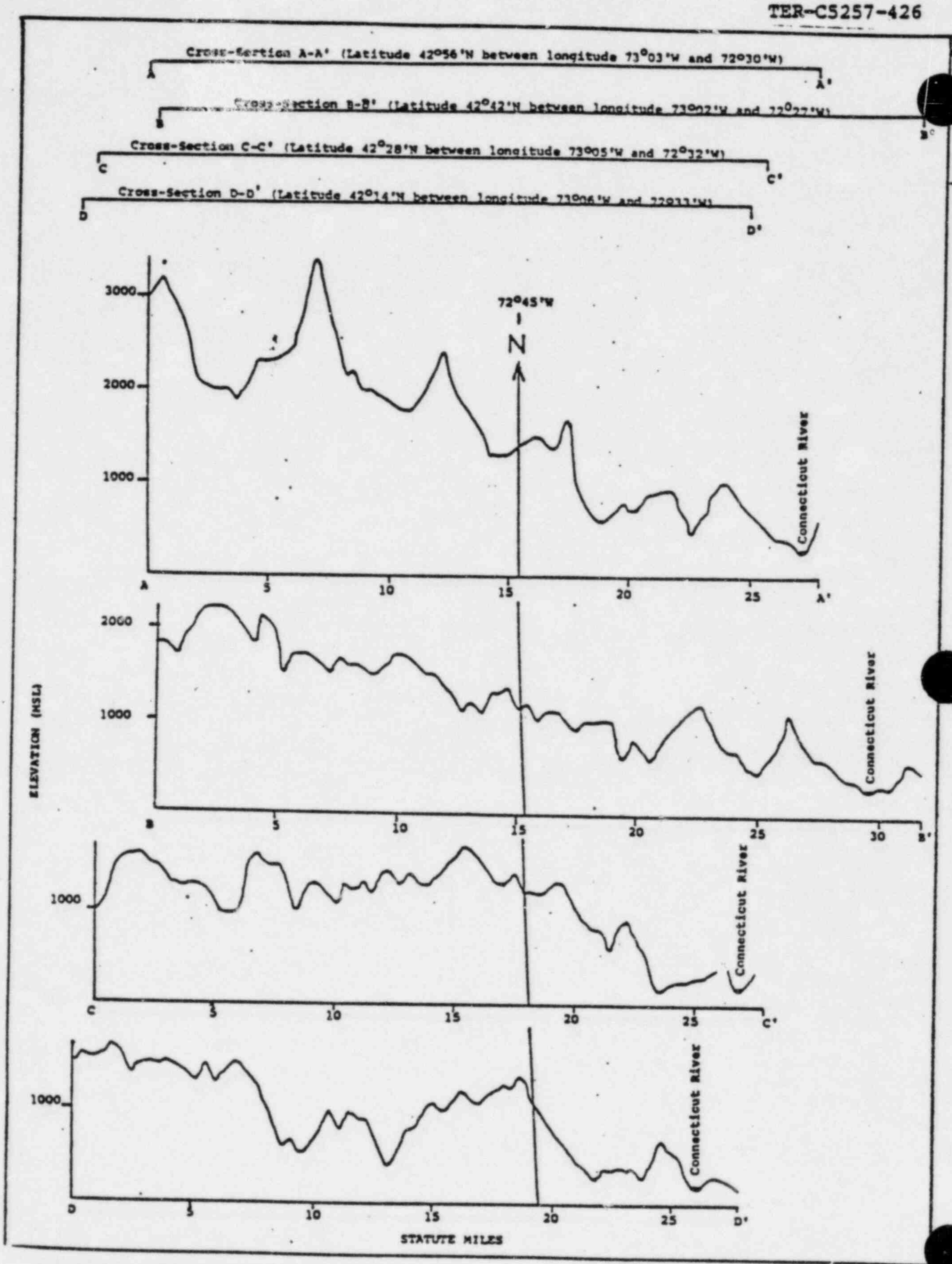

Figure A-10. Latitudinal Cross Sections of the Terrain Between Westfield, Massachusetts and the Upper Deerfield River Basin

GENERALIZED ESTIMATES

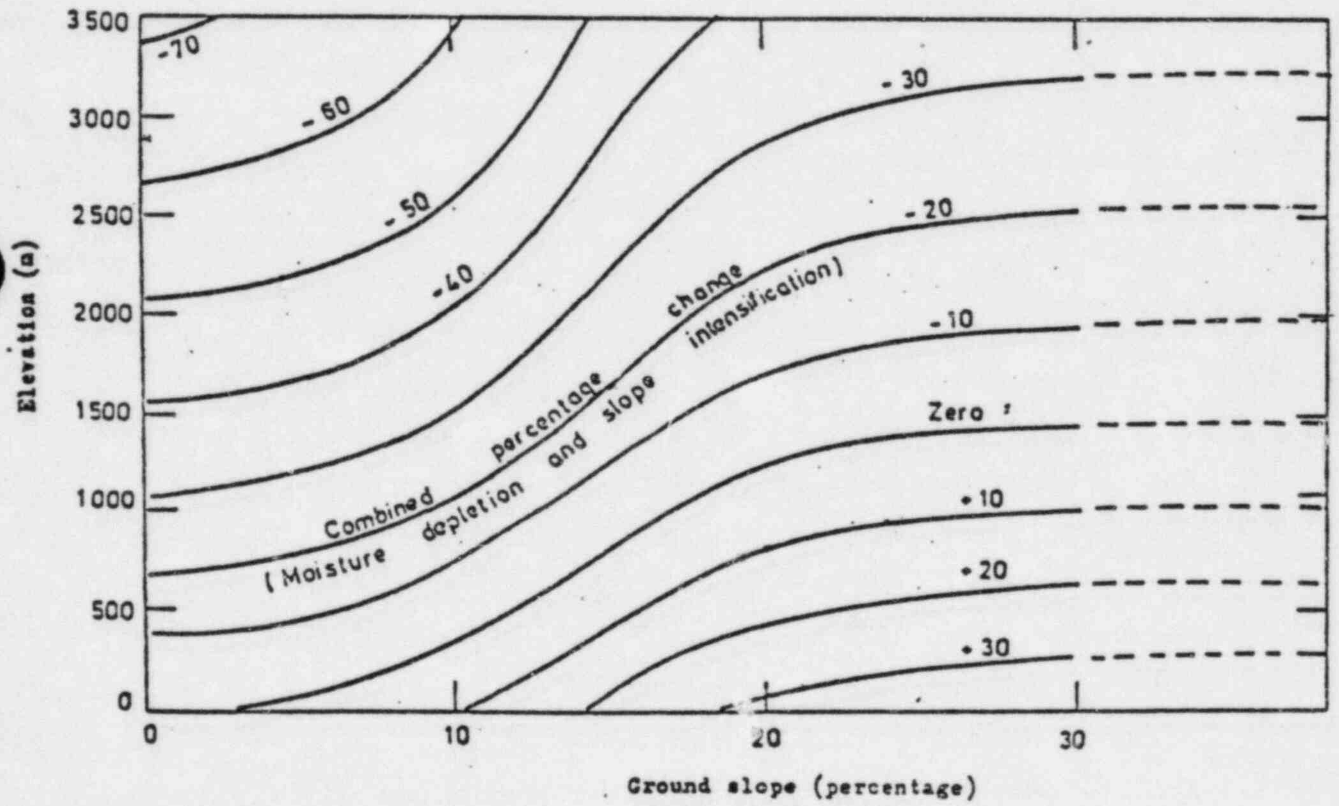

Figure A-11. Adjustments of Non-orographic PMP for Elevation and Slope, Hawaiian Islands

2000 ft within a mile of either bank of the Deerfield River forming a steep V-shaped valley which characterizes much of the basin, but the slope that would be encountered by air flowing from south to north within the basin is approximately 1%. Air flowing from south to north from the Atlantic Coast approximately 100 miles away from the Yankee Rowe site would encounter a terrain rise of 1150 ft in 528,000 ft at a slope of approximately 0.2%. The average elevation of the Upper Deerfield River Basin is approximately 2100 ft or 640 meters above mean sea level.

Enter the diagram (Figure A-11) at an elevation of 640 meters and proceed horizontally to the vertical line representing the locus of 1% slope values. The intersection of the 640-meter elevation line and the 1% slope line defines the point X which is located near, but not on, the curve labeled -20. Interpolation between the -20% and -10% curves would provide a -19% value for point X. This is the percent of decrease to be applied to the sea level PMP value of 19 inches. Reducing 19 inches by 19% gives 15.4 inches as the adjusted value of PMP.

Now explore the use of Figure A-11 in correcting the transposed Westfield, Massachusetts storm for moisture depletion and slope effect imposed by the terrain between Westfield, Massachusetts and the Upper Deerfield River Basin. The terrain rises 1300 ft (2400-1100) in the 60-mile interval between Westfield, Massachusetts and the Upper Deerfield River Basin near Somerset Dam. The 1300-ft rise in 60 miles represents a slope of 1300 divided by 316,800, or 0.4%. Since 1300 ft is approximately 400 meters, Figure A-11 should be entered at 400 meters. Proceeding horizontally to the vertical line that would pass through the slope value of 0.4 would locate a point lying just above the -10 change line. If the 14.2-inch, 24-hour total over 200 square miles were used, a 10% reduction would yield a 12.8-inch adjusted value.

In order for the Westfield storm to be increased by 20%, a terrain slope of approximately 25% would be required.

ENVELOPING CURVE FOR THE UPPER DEERFIELD RIVER BASIN ABOVE YANKEE ROWE

Envelopment is a means of passing from the array of disparate transposed maximized rainfall data derived from several storms to a reasonable estimate of PMP for a particular area and duration. No single storm is apt to provide the governing data for the entire range of durations of interest. The curve provides smooth transition from one duration to the next. The Licensee's October 1980 report [6] presents such an enveloping curve, on page 65, derived by transposing and maximizing data from seven major storms. This curve was studied at some length. It provides a reasonably conservative extrapolation from transposed and maximized storm data to PMP values. For durations of longer than 24 hours, the curve is controlled by data derived from the Westfield, Massachusetts storm. For durations of up to 24 hours, the curve is heavily dependent on data derived from a storm at Jewell, Maryland, July 26-29, 1897, and a storm at Kinsman Notch, New Hampshire, November 2-4, 1927.

If the adjustment factors proposed by NRC, including the questionable orographic factor of 1.20, are applied to the 12-hour and 18-hour observed 200-square-mile values for the Westfield, Massachusetts storm, PMP depths of 12.0 and 13.4 inches, respectively, result. The portion of the curve between 10 and 15 hours is considered in this study to be the portion most in keeping with the realities of basin characteristics and storm movement (Figure A-8). Only in nearly flat drainages such as those of the Texas Gulf Coast would an area of 200 square miles be compatible with a full 24-hour rainfall duration.

Recognizing the possibility that application of modern bucket-survey thoroughness to the 1897 storm might have resulted in identification of rainfall centers of greater depth, and keeping in mind the Smethport, Pennsylvania storm, it is prudent to consider only the consequences of PMP values that fall in the zone which has been placed above the Yankee Atomic envelopment curve between 10 and 15 hours duration (Figure A-9). This is a subjective judgment made with an understanding of the uncertainties established hydrometeorological practice.

DURATION OF THE PROBABLE MAXIMUM PRECIPITATION

Although both the Licensee and NRC have focused on 24-hour duration rainfall in arriving at their widely divergent estimates of PMP, the suitability of using the 24-hour duration for the steeply sloping 236-square-mile drainage basin of the Upper Deerfield River above the Yankee Rowe plant is in question. If the equilibrium time for the basin for continuous effective rainfall falling at a constant rate for an indefinite period is considerably less than 24 hours, the amount of rainfall required to produce the probable maximum flood (PMF) is reduced.

The equilibrium time for any basin can be found by constructing the S-hydrograph for the basin. This amounts to summing the ordinates of lagged unit hydrographs until the sum reaches its upper limit. Unit hydrographs for the four partial areas that comprise the 236-square-mile drainage above the Yankee Rowe plant were derived synthetically, using Snyder's method, by both YAEC and NRC, again with somewhat divergent results. NRC decided that the values derived by the Licensee were more conservative than those derived by NRC, and adopted the broader, flatter unit hydrographs developed by the Licensee.

The 15-minute unit hydrographs developed by YAEC are shown in Reference 6. The estimate of equilibrium time for the entire 236-square-mile basin would be largely determined by the 15-hour time base of the unit hydrograph developed for the 94-square-mile contributing area between the Searsburg and Harriman Dams. It is estimated here that the travel time for water to reach the Yankee Rowe plant from Harriman Dam is less than 2 hours (7 river miles divided by a velocity of 3.75 mph). Page 170 of Reference 14 presents discharge measurements taken during the flood of March 2?, 1948 on the Middle Branch Westfield River at Goss Heights, Massachusetts which were used as reasonable estimates of flow conditions for the Deerfield River. This travel time estimate is supported by NRC's estimate of 1.5 hours. (See Paragraph 4.3, Flood Routing, page 21, of Reference 5.)

The foregoing leads to an estimate of equilibrium time of under 17 hours. It is likely that the true value is less than 18 hours, possibly as little as 12 hours. A 10-hour time base for the unit hydrograph for the partial area between the Searsburg and Harriman Dams would be about as defensible as the 15-hour time base chosen. That shorter time base plus a less than 2-hour travel time would provide an estimated 12-hour equilibrium time.

The foregoing discussion of equilibrium time for the Upper Deerfield River Basin applies to the condition of zero available reservoir storage capacity. If the reservoirs in the basin are not full at the time of onset of the storm that produces the PMF, a longer duration of rainfall will be required before the equilibrium condition will be reached. How much longer will depend on how much reserve storage capacity remains at the onset of effective precipitation and the intensity of that precipitation. A conservative approach to deriving the PMF from the PMP involves postulating antecedent rainfall sufficiently heavy to completely saturate the basin. That rainfall could be expected to reduce available storage. Paragraph 4.3.3, Antecedent Floods, page 24, of the Draft Flood Study prepared by the NRC for the Yankee Rowe Nuclear plant and Upper Deerfield River Basin hypothesizes all reservoirs to be at the spillway crest elevation at the start of the PMF. The amount of storage remaining between the spillway crests and the crest of the dams would determine what part of the PMP could be stored before the basin began to behave in the zero storage mode.

The principal component of the storage available at the start of the PMF would be provided by Harriman Dam. Its spillway crest elevation is 1491.7 feet above mean sea level. The elevation of the top of the dam is 1521.2 feet above mean sea level. The surface area of the reservoir is about 2050 acres at elevation 1492.7 feet above mean sea level. A crude estimate of the storage capacity remaining between the onset of spilling and overtopping of the embankment would be 70,000 acre-feet. The drainage area above the dam is approximately 184 square miles or 117,760. Approximately 7 inches of runoff from the entire 184 square miles would have to arrive at Harriman Dam to fill the available storage capacity in view of spillway release from storage. The precise timing of such a condition is a problem in flood routing that is outside the province of the meteorological review, but it seems reasonable to consider that the equilibrium condition could be achieved in storm durations shorter than 24 hours.

It is concluded that attention should be focused on PMP depth-duration-area values between 12 and 18 hours.

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16. "Manual for Estimation of Probable Maximum Precipitation," World Meteorological Organization, p. 121, Figure 5.6

17. Intensity of Precipitation on Rate-of-Fall Basis of Manual Surface Observation, 7th Edition (Revised), Circular N (WBAN) Table 3-1, page 303

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

AFFENDIX B

February 09, 1981

Docket No. 50-29
LS05-81-02-01

Mr. James A. Kay
Senior Engineer - Licensing
Yankee Atomic Electric Company
1671 Worcester Street
Framingham, Mass. 01701

Dear Mr. Kay:

SUBJECT: TOPIC II-3.A, HYDROLOGIC DESCRIPTION AND TOPIC II-3.B, (PARTIAL), FLOODING POTENTIAL AND PROTECTION REQUIREMENTS (YANKEE ROWE)

Enclosed is a copy of our draft evaluation of Systematic Evaluation Program Topics II-3.A and II-3.B. You are requested to examine the facts upon which the staff has based its evaluation and respond either by confirming that the facts are correct, or by identifying errors and supplying the corrected information. We encourage you to supply any other material that might affect the staff's evaluation of these topics or be significant in the integrated assessment of your facility.

Your conclusions regarding the subject topics and your seismic evaluation of the dam should be considered together because of possible interrelationships between the subjects.

Your response is requested within 30 days of receipt of this letter. If no response is received within that time, we will assume that you have no comments or corrections.

In future correspondence regarding Systematic Evaluation Program topics, please refer to the topic numbers in your cover letter.

Sincerely,

Dennis M. Crutchfield

Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing

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1/1

Enclosure: As stated

cc w/enclosure:
See next page

DSU use EX (11)

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Mr. James A. Kay

YANKEE-ROWE ATOMIC
POWER STATION
DOCKET NO. 50-29

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SEP DRAFT SAFETY TOPIC EVALUATION
YANKEE ROWE NUCLEAR POWER STATION

TOPICS II-3.A, HYDROLOGIC DESCRIPTION
TOPIC II-3.B, (PARTIAL), FLOODING POTENTIAL AND PROTECTION REQUIREMENTS

I. Introduction

It must be assured that the designs of safety-related structures, systems and components have considered appropriate hydrologic conditions. Hydrologic considerations include the interface of the plant with the hydrosphere, the identification of hydrologic causal mechanisms that may require special plant design or operating limitations. The scope of these safety topic evaluations is to assure that appropriate hydrologic factors have been considered and to assess any hydrologic considerations which may have changed since being reviewed during the initial licensing of the plant. Should flooding potential exist, the impact of the flood on the plant will be examined. If flooding protection is required, it must be assured that the protection relied upon is available, appropriate, and that provisions have been made to implement the required protection. The protection will be reviewed to assure that safety-related structures, systems and components are protected against floods.

II. Current Review Criteria

The current NRC criteria applicable to these topics are (1) Standard Review Plans 2.4.1 through 2.4.14, 3.4.1 and 9.2.5; (2) Regulatory Guides 1.102, 1.127, 1.27, 1.59 which includes American National Standards Institute Standard N170-1976, and Regulatory Guide 1.70.

III. Related Safety Topics and Interfaces

The Topic identifies water levels and other hydrologic information that may be pertinent to other review areas for assessment of effects on safety-related buildings and equipment. The related interface Topics are: (1) III-3.A Effects of High Water Level on Structures; (2) II-4E Dam Integrity; (3) III-6 Seismic Design Considerations; (4) VII-3 Systems Required for Safe Shutdown; (5) VIII-2 On Site Emergency Power Systems - Diesel Generators; (6) XI-3 Station Service and Cooling Water Systems; and (7) XVI Technical Specifications.

The categories of "In Service Inspection of Water Control Structures" and "Structural and Other Consequences of Failures of Underdrain Systems" also require hydrologic review and input; however, the hydrologic aspects are addressed in Topics III-3.C and III-3.B, respectively.

IV. Review Guidelines

This report includes: a discussion of the potential flood related problems at the plant site as a result of severe precipitation up to and including the severity of the Probable Maximum Precipitation (PMP) on the Deerfield River Basin; a brief description of the hydrologic features of the site and related surrounding area; a description of the analysis procedures used to predict the flood levels at the site; and a discussion of the study results.

IV. Review Guidelines (cont)

As a result of the predicted possible severe flooding of the Yankee Rowe Nuclear Plant Site, this report was expedited and is therefore limited to the discussion of flooding potential for the Deerfield River Basin and Yankee Rowe Nuclear Plant Site. The review of groundwater, local flooding and safety related water supply will be deferred until the more severe problem of potential Deerfield River flood effects has been assessed.

Regulatory Guides 1.59 and 1.102 have been specifically identified by the NRC's Regulatory Requirements Review Committee as needing consideration for backfit on operating reactors. These guides are utilized in determining whether the facility design complies with current criteria or some equivalent alternatives acceptable to the staff.

This evaluation was performed under the auspices of the Systematic Evaluation Program and is prepared as input to the Integrated Assessment Report.

V. Evaluation

1.0 INTER-AGENCY COORDINATION

The Federal Energy Regulatory Commission (FERC) has the licensing responsibility for Hydroelectric Developments. The NRC has previously met with FERC to apprise them of our preliminary findings. Interagency coordination is also required between NRC and the Federal Emergency Management Agency (FEMA) in accordance with criteria set forth in Appendix E, 10 CFR Part 80, NUREG-0654 and the Inter Agency Steering Committee. FEMA has also been apprised of our preliminary findings, and both agencies will be included on distribution for the completed draft flood study and any subsequent correspondence relating to the potential flood problem at the Yankee Rowe site.

2.0 Discussion of Problem

The Yankee Rowe Nuclear Power Plant is located on the east bank of the Sherman Dam and Reservoir, the fourth dam in a chain of hydroelectric dams on the Upper Deerfield River Basin (See Figures 3.1.1 and 3.1.2). These dams were constructed in the 1920s. The first step in the NRC's SEP review is to compare the existing plant to current licensing criteria for new plants. Thus, in this study it is required to determine if, the Harriman Dam, the first upstream dam from the plant, can safely pass a Probable Maximum Flood (PMF) - the current design basis for new nuclear power plants. Since the failure of Harriman Dam could induce damaging flood levels at the plant site, it is also necessary to estimate the magnitude of these flood levels.

This flood study shows that although Harriman Dam can safely pass about 13 inches of basin rainfall, it does not meet current licensing criteria in that 18.9 inches (the design basis Probable Maximum Precipitation based on current criteria) on the Upper Deerfield River Basin will overtop and fail Harriman and Sherman Dams and produce flood levels at the plant site that are 40 or more feet above plant grade.

Tabulated below are several key evaluation areas that have a significant influence on the determination of flood level at the plant site.

1. Magnitude of precipitation in the basin.

2. Distribution of precipitation within the storm (% of rainfall in each 6 hour period).

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3. Magnitude and timing of antecedent storms.

4. Reduction in Harriman Spillway capacity due to debris or reduced hydraulic performance characteristics at heads greater than the design value.

5. The shape and duration of dam breaches due to overtopping.

6. Early failure of Sherman Dam.

The significance of these items will be discussed further in Section 4.0.

It should also be noted that failure of the Sherman Dam, which impounds the plant's normal and emergency water supply, could affect the plant's safe shutdown capability. This subject will be addressed at a later date under Topic II-3.6, "Safety Related Water Supply."

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3.0 HYDROLOGIC DESCRIPTION

3.1 Yankee Rowe Site and Facilities

3.1.1 Site Description

The Yankee Rowe facility is located in the town of Rowe, in Franklin County, Massachusetts, on the east side of the Deerfield River, three-quarters of a mile south of the Vermont-Massachusetts border. Figure 3.1.1 shows the site location on a general area map.

The site consists of approximately 2,000 acres straddling the Deerfield River in the towns of Rowe and Monroe, Massachusetts. The reactor facility is located on the eastern side of the Deerfield River next to Sherman Dam and adjacent to the Sherman Reservoir, which serves as a source of cooling water for the Yankee plant's once-through condenser and service water cooling system.

Most of the land in the immediate vicinity of the site is heavily forested. At the site, which is in a valley, the elevation is about 1130 feet above mean sea level (feet msl). Within a distance of one mile, however, the hills on both sides of the site rise to above elevation 2000 feet msl. This steep-slope character of the Deerfield River extends from Wilmington, Vermont, 12 miles north, to Charlemont, Massachusetts, 8 miles south-southeast.

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3.1.2 Station Description

The Yankee Nuclear Power Station is a single-unit pressurized water reactor nominally rated at 185 MWe gross generating capacity with a rated net capacity of 176 MWe. The containment, a steel sphere elevated 30 feet above the ground, encloses the entire primary system, including the steam generators. The station has operated since July 1960, under Atomic Energy Commission license number DPR-3, issued under section 104(b) of the Atomic Energy Act of 1954 (as amended).

A once-through open cycle system with an average flow of 310 cfs is used for condenser cooling.

There is an 82.6-acre watershed southeast of the plant that drains across the plant site to Sherman Pond. There is a 994-acre watershed located north and east of the main plant area. This watershed is drained by Wheeler Brook which empties into Sherman Pond just north of the main plant area. Flood potential from these two drainage areas and the plant site will be deferred to a later report.

Yard grade in the vicinity of the plant proper is 1127.7 feet msl and 1119.7 feet msl at the screenhouse. Floor slab elevation of the turbine building is 1128.3 feet msl.

Figure 3.1.2 illustrates the general layout of the station.

3.2 Hydrosphere - The Deerfield River Basin

3.2.1 General Description

The Deerfield River, a tributary of the Connecticut River, has a total drainage area of 664 square miles and extends from southern Vermont into the northwestern corner of Massachusetts. The Yankee Rowe power station, situated on the central portion of the Deerfield River Basin (Figure 3.2.1) is only affected by hydrologic events in the 236 square mile drainage area of the upper basin.

The upper Deerfield River Basin, in general, has fairly steep slopes and comprises four sub-basins - Somerset, Searsburg, Harriman, and Sherman - which are delineated in Figure 3.2.1 by the bold dotted lines. The average Deerfield River flow for the 61-year record at Charlemont, Massachusetts, is 887 cubic feet per second[1]. Charlemont is approximately 9 miles downstream from Sherman Dam and has a drainage area of 362 square miles. The maximum flow at Charlemont was 56,300 cubic feet per second recorded during the hurricane of 1938.

3.2.2 Description of Upper Deerfield River Reservoir Developments

The Upper Deerfield Project includes the Somerset, Searsburg, Harriman, and Sherman Developments.

The Deerfield River rises in Southern Vermont and flows generally in a south and east direction through a valley that is narrow at the headwaters but broader as it approaches the entrance to the Connecticut River. At the Somerset Reservoir Dam in the upper reaches of the river, the elevation is 2134 feet msl and at its confluence with the Connecticut River its elevation is 120 feet msl, a drop of 2014 feet.

Somerset Reservoir

The drainage area above the dam is approximately 30 square miles. The reservoir extends upstream for approximately 5.6 miles and has a surface area of about 1623 acres at elevation 2133.6 feet msl.

The spillway structure is located at the west end of the earth embankment. It consists of a trapezoidal concrete section divided into eight bays, each 24 feet wide with a crest at elevation 2133.6 feet msl, and two 10-foot wide sections with crest at elevation 2133.6 feet msl provided with stop logs to elevation 2136.6 feet msl. A creosoted timber bridge spans the spillway on concrete piers. New flashboard stanchions were installed in 1964 to carry three feet of flashboards. The spillway discharges into a channel excavated in ledge. The channel is about 800 feet long, 45 feet in average width, and from 6 to 30 feet deep.

The Somerset Dam is of the semi-hydraulic fill type and is constructed on a curved alignment. The entire upstream slope is protected by riprap. The roadway at the crest is of gravel construction.

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River flow leaving Somerset Reservoir flows south through the East Branch of the Deerfield River to the Searsburg Reservoir.

Searsburg Development

The drainage area above the development is approximately 90 square miles. The pond extends upstream for approximately 1 mile and has a surface area of about 28 acres at elevation 1754.7 feet msl.

The spillway consists of a concrete ogee weir 137 feet long with crest at elevation 1749.7 feet msl and provided with pin type flashboards 5 feet in height. A bypass channel used during construction is located at the northerly end of the spillway, and closure in this area consisted of a vertical concrete wall and deck.

The earth embankment at the northerly end of the dam is of the semi-hydraulic fill type, supported by a gravity concrete retaining wall. The roadway at the crest is of gravel construction.

Because of the limited storage capacity (283 acre-feet), the failure or non-failure of this dam will have no appreciable effect on downstream flood flows. Therefore, this dam and reservoir were ignored in subsequent flood analyses.

Figure 3.2.3 and Table 3.2.1 show some pertinent features of the dam and reservoir.

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The flow continues downstream in a south and east direction into Harriman Reservoir.

Harriman Development

The drainage area above the development is approximately 184 square miles. The reservoir extends upstream for approximately 9 miles and has a surface area of about 2050 acres at elevation 1492.7 feet msl.

The 200-foot high earth dam is of the semi-hydraulic fill type and was raised with rolled earth fill from elevation 1511.7 feet msl to elevation 1521.2 feet msl in 1964 to increase flood retention capability. The upstream slope has riprap protection to elevation 1480.7 feet msl for most of the length of the embankment, varying to elevation 1495.7 feet msl at the northerly end. The roadway at the crest is of gravel covered with a seal coat of asphalt. The downstream slope has a grass cover.

The spillway is of the morning-glory type and is located upstream of the southerly end of the dam. The spillway discharges through a 22.5-foot minimum diameter vertical shaft and 90 degree bend into the concrete bypass conduit that was used for diversion during construction. The bypass conduit is now plugged with concrete upstream of the vertical shaft. This conduit has a cross-sectional area equivalent to a 22.5-foot diameter circle. The spillway has a crest elevation of 1491.7 feet msl and 16 equally spaced piers that can accommodate 7 feet of flashboards. The spillway shaft and crest were resurfaced in 1954.

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Figures 3.2.4 and 3.2.5 and Table 3.2.1 show some pertinent features of the dam and reservoir.

Flow leaving the Harriman Dam continues downstream in a southerly direction into Sherman Reservoir.

Sherman Development

The drainage area above the development is approximately 236 square miles. The pond extends upstream approximately 2.2 miles and has a surface area of about 194 acres at elevation 1103.7 feet msl.

The dam is of the semi-hydraulic fill type and was raised 10 feet to elevation 1129.7 feet msl with rolled earth fill in 1964 in order to increase spillway capacity. The embankment is supported at its northerly end by a concrete retaining wall which also was raised in 1964. The upstream slope has riprap protection to elevation 1095.7 feet msl for the full length of the embankment. The downstream slope has grass cover over its entire area.

The spillway structure is located at the north end of the dam. It consists of a gravity concrete ogee weir section with a crest elevation of 1103.7 feet msl and is provided with pin type flashboards 4 feet in height. There is a spillway channel, excavated in ledge, about 360 feet long and 50 feet wide spanned by a plate girder bridge. The spillway channel was deepened in 1964 by the removal of 4600 cubic yards of material to increase discharge capability by reducing backwater effect. An eroding area downstream of

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the spillway bridge on the west side of the channel was graded and riprapped at the same time. A concrete bypass conduit, used during construction, runs through the earth dam. This has a cross-sectional area of 140 square feet and has been plugged at the upstream end with concrete.

Figures 3.2.6, 3.2.7, and 3.2.8 and Table 3.2.1 show some pertinent features of the dam and reservoir.

3.3 Floods

3.3.1 Flood History

The flood history of the Deerfield River during this century is readily available from records which extend back to 1911 when construction began on the hydroelectric facilities. These records and those from the U.S. Geological Survey gaging station at Charlemont clearly document major storm and discharge events which are summarized in Table 3.3.1.

As shown, the most significant event was during the "1938 hurricane," closely followed by the 1948-49 "New Year's Eve" storm, and the 1927 and 1936 events. Although undocumented, a local reference[2] states that the flood of October 1869 was similar in severity to the 1927 event. The rainfall which accompanies tropical storms (including hurricanes) often produces major floods during the summer and early fall. Extratropical storms and/or snow melt produce principal floods during the winter and spring months.

Maximum average depths of rainfall for selected historic storms of record for the region are shown in Table 3.3.2. These actual storm values are presented as an indication of what has occurred in the region historically. Also shown is the controlling storm for the northeast United States (OR 9-23), commonly known as the Smethport, Pa. storm. This storm occurred on the west side of the Appalachian Mountains and is generally not transposed across the mountains.

4.0. Analysis Procedure

4.1 General

In order to determine the Probable Maximum Flood (PMF) elevation at the Yankee Rowe Plant site, the Probable Maximum Precipitation (PMP) is applied to individual subbasins using a unit hydrograph to define the runoff characteristics of the subbasins. The hydrographs thus obtained are routed through the stream channels and reservoirs to account for attenuation due to channel and valley storage. Where dams are overtopped the erosional failures and resulting outflow hydrographs are simulated by synthetic methods.

The PMF is used by many federal, state and local agencies and architectural engineering firms to predict upper limit flood levels for planning and design purposes. The PMF is defined as "The hypothetical flood (peak discharge, volume and hydrograph shape) that is considered to be the most severe reasonably possible, based on comprehensive hydrometeorological application of probable maximum precipitation and other hydrologic factors

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favorable for maximum flood runoff such as sequential storms and snowmelt." The PMP is defined as "The estimated depth for a given duration, drainage area, and time of year for which there is virtually no risk of exceedance. The PMP for a given duration and drainage area approaches and approximates the maximum which is physically possible within the limits of comtemporary hydrometeorological knowledge and techniques."(3)

On a complex river basin such as the Deerfield where multiple water levels and sensitivity analyses are required, a computer model is used. In this case, the HEC-1 Flood Hydrograph Package for Dam Safety Investigations(4) was used.

The basic HEC-1 Flood Hydrograph Package was developed by the U.S. Army Corps of Engineers' Hydrologic Engineering Center for modeling basin stream networks. This computer program is used by many federal, state and local agencies as well as architectural and consultant engineering firms. The dam safety investigation program is a modification to the basic program that allows the estimation of the overtopping potential of a dam and the downstream hydrologic-hydraulic consequences resulting from assumed structural failures of a dam.

4.2 Rainfall and Runoff

4.2.1 Probable Maximum Precipitation

The staff considered several sources for the PMP for this study. Hydrometeorological Report Number 33 (H.R. #33), April 1956(5), is the source most commonly used for PMP estimates east of the 105th meridian. This report was revised and expanded and released as Hydrometeorological Report Number 51 (H.R. #51)[6] in June 1978. H.R. #51 predicts slightly larger rainfall values for the study area than H.R. #33. Since H.R. #51 has not been fully reviewed by some major federal agencies such as NRC, we have not used precipitation estimates from H.R. #51 for the study.

The licensee, Yankee Atomic Power Company, has recently completed a Draft Probable Maximum Flood Analysis for the Yankee Rowe Nuclear Generating Station.[7] This report attempts to derive a PMP for the Upper Deerfield River Basin by transposing and maximizing several maximum regional storms to the Deerfield Basin. The NRC staff has not accepted this analysis due to the lack of supporting data and apparent erroneous transposed storm rainfall values. The staff assertion of erroneous results is based in part on the staff's independent analysis of the Westfield storm which resulted in rainfall estimates considerably larger than the licensee's estimates.

NRC regulations do not specifically require a PMP for the Design Basis Flood for nuclear power plants. Title 10, Part 50, Appendix A of the Code of Federal Regulations states, "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 sourrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historic data have been accumulated." In order to be assured that the licensee is not being unduly penalized by the use of Generalized PMP values, the staff made an independent analysis of the August 1955 storm that was centered at Westfield, Mass. (about 50 miles south of the Rowe site). The maximization and transposition of this storm to the upper Deerfield River Basin provides a rainfall estimate that can be considered as a lower limit value for a design basis flood.

The storm was maximized and transposed to the Deerfield Basin using procedures suggested in H.R. #51[6] and the manual for Estimation of Probable Maximum Precipitation[15]. The transposed storm had an adjustment factor of 1.17 which includes factors of 110% for moisture maximization, 89% for transposition and elevation and 120% for orographic effects. The 200 square mile, 24-hour adjusted rainfall for the storm is 16.6 inches. Depth area-duration curves for the transposed Westfield Storm are shown on Figure 4.2.2. An idealized isohyetal storm pattern, with this rainfall, was centered on the upper Deerfield River Basin and planimetered to obtain an average 236 square mile basin rainfall of 16.5 inches for 24 hours. The resulting flood runoff would overtop Harriman dam and produce a flood level at the Yankee Rowe site of 1172.4 feet msl. It is noted that this lower limit rainfall value by itself does not qualify as a PMP. It follows that a comprehensive regional PMP study would predict at least 16.6 inches for the 200 square mile, 24-hour value and would probably predict a somewhat larger value but less than the 18.9 inches from H.R. #33. Since this lower limit rainfall does not alter the conclusions of this study, it was concluded that the rainfall from H.R. #33 would be used as the design basis rainfall for this study.

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The depth area-duration curves from H.R. #33 for the upper Deerfield River Basin are shown in Figure 4.2.1.

The distribution of rainfall in the worst 6-hour period has a significant influence on the potential flood levels at the Yankee Rowe site. H.R. #33 suggests that 72.8% of the 236 square mile, 24-hour rainfall occurs in the critical 6-hour period. Referring to Table 3.3.2, the maximum 6-hour values for these historic storms range from 43 to 66 percent of the 200 square mile, 24-hour value. This percentage ranges up to 90% for northeast U.S. storms shown in reference (6). For a sensitivity test, the model was run with 43% of the 24-hour PMP in the maximum 6-hour period. The results of this run show that Harriman Dam would still be overtopped and the level at the Yankee Rowe site would increase by 8.0 feet. The reason for the higher stage at the site is because the higher percentages of rainfall in the critical 6-hour period causes Sherman Dam to overtop and fail about 4 hours before the peak outflow from Harriman Dam reaches the Yankee Rowe site. Whereas with the 43% distribution, Sherman Dam does not fail until the peak outflow from Harriman Dam reaches the site. This higher initial Sherman reservoir level induces a higher peak flood stage at the Yankee Rowe site. The rainfall distribution suggested in H.R. #33 was selected for use in this flood study.

4.2.2 Rainfall Losses

Rainfall loss rates can be derived from historic storm and flood records. The loss rates thus computed would generally be directly applicable to

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maximized storms such as the PMF. The licensee derived loss rates in Reference (7) from two historic Deerfield River storms. The computed rates were 0.03 and 0.06 inches per hour. Other verification evaluations(7) indicated about 0.1 inch per hour. We discussed rainfall losses with personnel from the New England Division of the U.S. Army Corps of Engineers who have studied many storms in the region. They recommend a loss rate of 0.1 inch per hour and no initial loss for a PMF study. Based on our own experience and the information supplied by the licensee and the Corp of Engineers, we selected a loss rate of 0.1 inch per hour and no initial loss. The licensee used an initial loss of 0.5 inch and 0.1 inch per hour for his PMF study.(7) Since the PMF by definition optimizes and maximizes parameters, the use of no initial loss is justifiable and reasonable. Additionally, the losses are a small part of the total rainfall and do not have any significant effect on subsequent flood levels.

4.2.3 Unit Hydrograph Coefficients

The Unit Hydrograph is "the hydrograph of surface runoff (not including groundwater runoff) on a given basin, due to an effective rain falling for a unit of time. The term 'effective rain' means rain producing surface runoff. The unit of time may be one day or preferably a fraction of a day. It must be less than the time of concentration."(8) Unit hydrographs can be derived from actual storms or by synthetic methods using empirical equations. For this study, both methods were used to derive unit hydrograph coefficients. Several synthetic methods from the literature were used to derive coefficients for each of the four subbasins discussed in Section 3.2.2. The methods used were: (1) Snyders Equation (EM 1110.2-1405)[9], (2) Design of small Dams[10], (3) Linsley, Kohler and Paulhus[11], and (4) Standard Project Food Critaria, Southern California[12]. Another set of coefficients were derived from a historical hydrograph presented in the Licensee's studies[13]. Unit Hydrograph coefficients were also developed from information obtained in discussions with personnel of the New England Division of the U.S. Army Corps of Engineers who have done many similar studies in the region. Another set of coefficients were selected based on personal observation and experience.

Ultimately, a set of Clark Unit Hydrograph Coefficients were selected for each subbasin based on professional judgment, personal experience, and with due consideration of the values obtained from the above methods. Prior to the completion of this study, the licensee also furnished Snyder Unit Hydrograph Coefficients for each subbasin. The Snyder values were derived from actual storms and verified in other storm reconstitutions. The following table shows a comparison of Snyder and Clark coefficients for the NRC and licensee values:

UNIT HYDROGRAPH COEFFICIENTS
DEERFIELD RIVER BASIN
COMPARISON OF NRC AND YANKEE ATOMIC VALUES

Drainage Subbasin	Snyder Coefficients 1/ NRC		Snyder Coefficients 1/ Yankee Atomic		Clark Coefficients 1/ NRC		Clark Coefficients 1/ Yankee Atomic	
	TP	CP	TP	CP	Tc	R	Tc	R
Somerset	2.42	.56	2.68	.81	2.7	2.7	10.37	3.65
Searsburg	3.09	.57	2.98	.81	3.4	3.4	11.50	4.09
Harriman	4.09	.58	4.16	.81	4.4	4.4	16.79	5.14
Sherman	3.17	.57	3.23	.81	3.5	3.5	12.86	4.22

1/ Conversions from Snyder to Clark and visa versa were done with the HEC-1 program (4)

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The somewhat large Yankee Atomic Clark coefficients reflect the slower times of concentration that the licensee obtained in the case studies. The runoff hydrographs developed from the Yankee Atomic coefficients will be broader and flatter (lower peak discharge) than the hydrographs developed with the NRC coefficients.

Table 4.1 shows a comparison of output from sensitivity studies run with the HEC-1 model. This comparison shows that the choice of unit hydrograph coefficients has little effect on resulting flood levels. However, in the interest of conservatism, the staff's final results and conclusions are based on the model runs using Snyder coefficients furnished by the licensee. Additionally, unit hydrograph coefficients derived and verified with actual flood events are generally more acceptable to the technical community.

The HEC-1 model is used to develop runoff hydrographs for each subbasin from the unit hydrographs, the rainfall, and rainfall losses. These subbasin runoff hydrographs are then in turn routed through the channels and reservoirs in the river basin.

4.2.4 Maximum Regional Rainfall

It is common practice, when analyzing potential flood problems, to determine the maximum historic rainfall for the region.

Table 3.3.2 shows the maximum recorded rainfall values for storms that have occurred in the region. The Westfield, mass storm (No. NA 2-22A) is the largest storm in the region by a considerable margin.

This storm was transposed to the Deerfield River Basin for the purpose of determining the potential effect on Harriman Dam and the Yankee Rowe Nuclear Plant. The transposed storm would have a 24 hour average basin (236 square miles) rainfall of about 13 inches.

4.3 Flood Routing

The channel and reservoir routing with the HEC-1 model uses the modified Puls method. This method of routing accounts for hydrograph attenuation due to channel, valley, and reservoir storage, but it does not account for the time required to convey water from one subbasin to the next. Above Harriman Dam this is accounted for in the subbasin hydrographs;

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between Harriman and Sherman Dams it is not. We estimate this time to be about 1.5 hours between Harriman and Sherman Dams. Thus, in terms of predicted flood levels at the Yankee Rowe site the peak inflow hydrograph to Sherman Reservoir could lag the predicted arrival time by less than 1.5 hours. In most cases analyzed, Sherman Dam is predicted to fail prior to the arrival of the predicted failure hydrograph from Harriman Dam. Therefore, any delay in the arrival of the Harriman peak flow could result in somewhat lower flood levels at the nuclear plant site. However, any failure hydrograph from Harriman Reservoir is sufficient to cause significant flood levels at the site, regardless of when Sherman Dam fails.

4.3.1 Reservoirs Routing

Reservoir routing was done by the Modified Puls method. Reservoir storage curves were provided by the licensee. The storage curves for Somerset, Harriman, and Sherman Reservoirs are presented in Figures 4.3.1, 4.3.2, and 4.3.3, respectively. The curves have been conservatively extrapolated by the licensee above the top of dam levels using an incremental storage procedure.

Reservoir outflows were of three types: (1) discharge through the normal reservoir spillways or outlet works, (2) discharge over non-eroded portions of the dams, and (3) discharge through the eroded or breached dam sections. The spillway rating curves for Somerset, Harriman, and Sherman Reservoirs were furnished by the licensee and are shown in Figures 4.3.4, 4.3.5 and 4.3.6, respectively.

The staff has some reservations with respect to the spillway for Harriman Reservoir. This spillway is the "Morning Glory" type which is designed to operate with a small change in discharge for a large range in heads. The spillway is located in the corner of the reservoir and very close to the dam and southern bank of the reservoir. This location would be conducive to debris accumulation and potential blockage, especially for rare floods that will carry many large trees and other debris downstream. These spillways are also noted for undesirable discharge characteristics at reservoir levels above the design value. The Harriman spillway was designed for a reservoir elevation of 1498.6 feet msl. The dam has been raised subsequently, and predicted levels could be as high as elevation 1525 feet msl. When these type spillways are subjected to heads above the design value, there is possibility of shifting control between weir, orifice and full pipe (pressure flow) with the associated uncertain discharge capability, slug flow, vortices, cavitation, and vibration. There is another uncertainty with respect to the rather narrow discharge channel immediately downstream of the tunnel outlet. The unknown is whether the narrow channel could submerge the outlet at higher flows, thus forcing a hydraulic jump in the conduit and thus reducing capacity.

During past meetings with the licensee he has demonstrated a willingness to construct booms or other considerations to preclude debris from the spillway if the additional capacity would ensure non-overtopping of Harriman Dam. The spillway was model tested in 1925 prior to construction. The model test does mention some runs at higher heads and the associated vortex formation and some fluctuation in discharge. It is doubtful that the friction in the riser and barrel were properly modeled and even if

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they were it is very difficult to predict, with a model, the flow characteristics for this type of spillway, for greater than design conditions. In light of the many uncertainties with this spillway, we limited the assumed discharge to the capacity of the riser throat as an orifice. This is about a 25% reduction in capacity at the higher heads.

The spillway for Sherman Dam has a rather narrow discharge channel downstream of the crest. Downwater computations for this spillway indicate the possibility of hydraulic jumps or turbulent flow conditions at very high discharges. Depending on the resolutions of other more serious problems, this issue may be investigated more thoroughly at another time.

Discharge over the non-eroded dam sections is computed by the HEC-1 program using the standard weir equation.

The breach hydrograph is computed by a weir flow equation appropriate for the shape of breach selected.

These various possible outflows are summed by the program and used in the routing procedure.

4.3.2 Channel Routing

The only channel routing required in the model was from Harriman Dam to the Sherman Reservoir. The geometric elements for the routing cross

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section were obtained from a 1:62,000 topographic map. The cross section was used for a one step Modified Puls routing to attenuate the hydrograph for channel and valley storage between Harriman Dam and Sherman Reservoir.

4.3.3 Antecedant Floods

Our current criteria requires that when analyzing potential single or multiple dam failures during a PMF, that the PMF be preceded 3 to 5 days by a flood equivalent to 40% of the PMF. The purpose of this requirement is to allow for the estimation of the loss of reservoir flood storage capacity by antecedant floods. For this study the antecedant flood was routed separately. This routing indicated that all reservoirs would be at the spillway crest elevation at the start of the PMF. Therefore, all subsequent runs were started with the reservoirs at the spillway crest elevation, except Searsburg, which was not considered as a reservoir as discussed in Section 3.2.2.

4.4 Erosional Dam Failures

As discussed in Section 4.1, the Modified HEC-1 program has provisions to simulate erosional type dam failures. The program allows for user discretion in selecting the shape and duration of the breach. Two breach shapes (trapezoidal and triangular) and three durations (1, 2 and 3 hours) were modeled for both Harriman and Sherman Dams in order to determine which would be critical in terms of water level at the Yankee Rowe site. The side slopes of both triangular and trapezoidal breaches were assumed to be the angle of repose ($\emptyset$) of the embankment material which was assumed to be 35°. The bottom width for trapezoidal sections was assumed to be the width of the natural valley at the toe of the dam, and the lower limit of the eroded sections was assumed to be limited to the elevation of the natural valley at the dam site.

The results of these analyses showed that the trapezoidal shape for a duration of one hour would be the critical assumptions for both dams. Since the duration of the Harriman Dam breach has a significant influence on the depth of flooding at the Yankee Rowe site, the TVA Breach Model[13] was used as a method of quantifying the duration of breach.

The Harriman inflow hydrograph for the TVA model was obtained by routing the PMF with the HEC-1 model and assuming infinite dam heights. Other inputs to the model were a $\emptyset$ angle of 15 degrees and a breach width of 400 feet. The section was eroded to elevation 1320 feet msl which is the approximate natural valley floor elevation. The model results showed a time of about one hour to breach the section.

Unfortunately, there is only a limited amount of information available on methodology for simulating erosional failures of earthen embankments. To the best of our knowledge, the TVA model is the only one available that attempts to predict a rate of failure. There is some historic information available, but at best this only supports the unpredictability of this type of dam failure. Photographic documentation of the recent Teton Dam failure indicates that the rapid failure of a large portion of the embankment section occurred in about 30 minutes. Although this was not an

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overtopping breach, the rapid failure of a major portion of the embankment gives a good indication of the erosional potential of reservoir storage.

4.5 Recurrence Intervals of Natural Phenomena

Recurrence intervals or probabilities of natural phenomena (rainfall or flood events) are often used as a decision-making tool. The staff is reluctant to attempt to associate frequencies with rare natural phenomena due to the large confidence intervals and the tendency to place more reliance on the probability estimates than is justified by the basic data. However, there is some frequency information available for rainfall in the region, and it is included in the following paragraphs.

Technical Paper #40(16), which does not include the last 24 years of records nor the effects of Hurricane Diane, shows a regional 24 hour-100 year point rainfall for the Deerfield River Basin of about 6 inches. The Westfield gage (about 50 miles south of the Rowe site) recorded rainfall during Hurricane Diane; the 24 hour-100 year point rainfall for this gage is 12.4 inches. The Springfield gage, which is about 10 miles east of Westfield, has a 24 hour-100 year point rainfall of 8.5 inches. The licensee has provided some rainfall frequency information in Appendix A of Reference 7. They show a 24 hour-100 year point rainfall for Harriman Station of about 5.5 inches. The other gages they selected all have 24 hour-100 year point rainfalls of between 4 and 6 inches. However, they did not select any sites in the region that have recorded severe historical rainfall events, such as Westfield, Mass., Kinsman Notch, N.H., or Springfield, Mass.

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Based on consideration of the above 100-year rainfalls, it is the staff's judgment that the regional 24 hour-100 year point rainfall for the Upper Deerfield River Basin would be about 7 to 8 inches. NOAA Technical Report NWS 24[17] provides a generalized curve for converting point rainfalls to area rainfalls. The 8-inch point rainfall is equivalent to 7.4 inches on the 236 square mile basin. This then can be compared to the 13-inch basin rainfall that Harriman Dam can safely pass. The only guide we can suggest for putting the 13-inch rainfall into a frequency perspective is that the 13-inch rainfall is approximately a regional record value, and based on past experience of other record storms and their extrapolated frequencies, which introduce significant uncertainty, the recurrence intervals are generally in the 500 to 1000 year range.

It should also be noted that this probability information deals only with the rainfall event and makes no allowance for any conservatism in our methods of determining the PMF, such as locating the storms critically over a specific watershed.

The percent chance of a rainfall value being exceeded in the next 20 years can be determined mathematically. There is an 18% chance of the 100-year (7.4 inch) rainfall being exceeded in the next 20 years. For the 500-year and 1000-year recurrence interval rainfall (13 inches), there is a 4% and 2% chance, respectively, of being exceeded in the next 20 years. Again, it is noted that there is much uncertainty involved in trying to associate frequencies with rare natural phenomena. The above approximate analysis is only intended to provide "ball park" estimates of the likelihood of exceeding the existing capacity of Harriman Dam in the next 20 years.

5.0 Results

This flood study has analyzed floods for a range of rainfalls from 13.0 to 21.3 inches. The resulting flood levels at the Yankee Rowe Nuclear plant site and other outputs are shown in Table 4.1. The 18.9 inches is the 24-hour, 236 square mile PMP from H.R. #33. It is the staff's judgment that, under current criteria, the PMF based on this rainfall or rainfall derived from a detailed regional study as discussed in Section 4.2.1 is the flood that the Yankee Rowe plant should be protected against.

The flood resulting from 13 inches of rainfall on the basin is approximately what Harriman Dam can contain without overtopping. However, the reservoir level for 13 inches of rain would be at the top of the dam and does not include any allowances for coincident wind generated waves which would be about 3 feet high for a 30 to 40 mph wind and the runup would be about 8 feet above the pond level. Additionally, the upper 25 feet of the embankment does not have riprap for erosion protection.

The PMP (18.9 inches) used for the PMF would have a point rainfall of 25.6 inches which compares to a maximum 24-hour point rainfall on the Harriman subbasin of 5.5 inches (Table 3.3.1) that produced the record reservoir level of 1498.1 ft msl. Regionally, a 24-hour, 200 square mile value of 14.2 inches (Table 3.3.2) was recorded at Westfield, Massachusetts, just 48 miles south of the Harriman Dam. This storm when maximized and transposed to the Upper Deerfield River Basin would yield about 16.6 inches of rainfall in 24 hours for a 200 square mile area.

Hydrometeorological Report Number 33 is recognized by major water resources engineering groups throughout the country as the source for Probable Maximum Precipitation for the design of large dams whose failure could result in loss of life and major property damage. In lieu of rainfall values based on a comprehensive regional study, the generalized estimate from HR #33 is the value that should be used for the Probable Maximum Flood for the Yankee River site.

Table 5.1 compares pertinent values of other Deerfield River flood studies to the NRC Flood Study. The values for the C. T. Main (1977) and Yankee Atomic (1977) studies were taken from a report prepared by the Yankee Atomic Electric Company, YAEC-1139[14]. This report was prepared for the Federal Power Commission (FPC) (currently known as the Federal Energy Regulatory Commission - FERC) as part of their licensing requirements.

The C. T. Main values are similar to the NRC PMF values, except for the breach assumptions for the dams which have a significant influence on the resultant predicted maximum reservoir levels. Since the C. T. Main breach assumptions were not provided, we cannot discuss the comparison. The Yankee Atomic (FPC Report) values are considerably lower than ours, and we attribute the difference to the unit hydrographs, rainfall loss rates, and starting pool levels. In all cases, we consider the Yankee Atomic values to be nonconservative.

The values for the 1980 Yankee Atomic flood study were taken from reference (7). The significant difference between these values and the NRC PMF values is the PMP rainfall (see Section 4.2.1). Minor differences

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are in the Harriman spillway capacity (100%-licensee vs approximately 75%-NRC) (See Section 4.2.1).

The values for the U.S. Army Corps of Engineers (COE) 1963 study were taken from a letter report that is attached as Appendix A. The study resulted from interagency coordination on licensing requirements in 1962 and 1963. The COE study was for a Spillway Design Flood (SDF) for Somerset, Harriman, and Sherman Dams. The SDF is equivalent to the PMF except that the rainfall would be for the drainage area controlled by each dam. Their study also assumed infinite dam heights. The COE values indicate that the results are very close to the NRC PMF study.

In addition to rainfall-induced floods, the site may also be exposed to flood waves from a Harriman Dam failure induced by other causes such as seismic, piping, etc. For these type failures the staff generally assumes instantaneous removal of the dam section and modeling of the flood wave by unsteady flow techniques. Since only 100-foot topographic mapping is available for a portion of the reach and accurate cross sections cannot be prepared, the use of the unsteady flow model is not warranted. Therefore, the staff used approximate methods to estimate a water level of 1148 feet msl at the Yankee Rowe site assuming instantaneous failure of the Harriman Dam with the pool at elevation 1493.0 feet msl and 50% attenuation due to channel and valley storage.

The 13 inches of rainfall that Harriman Dam can contain without overtopping is about 70 percent of the 18.9 inch PMP used by NRC as the design bases PMP under current criteria. It is about 78 percent of the lower limit value of 16.6 inches. The 5.5-inch maximum 24-hour point rainfall at Harriman Station is about 22 percent of the 25.6-inch, 24-hour point rainfall from H.R. #33.

The significance of the difference in assumed Harriman Spillway capacities (100% vs 75%) can be quantified by converting the difference in discharge to equivalent reservoir storage. The difference between 100% and 75% spillway discharge at the top of dam elevation is about 10,000 cfs. This flow for about 6 hours is equivalent to about 2-1/2 feet of reservoir storage. The 13-inch rainfall above Harriman Dam would maintain the reservoir within 2 feet of the top of the dam for about 6 hours.

VI. Conclusions and Recommendations

1.0 Hydrologic Conclusions

Based on the results of the NRC Flood Study, it is the staff's judgment that the Harriman Dam can safely pass the runoff from a storm with an average upper basin rainfall of about 13 inches. It is further concluded that this storm would result in a maximum flood elevation at the Yankee Rowe site of no more than 1132.0 feet msl. This rainfall is about 70 percent of the design basis rainfall under current NRC licensing criteria for new plants.

The staff also concludes that if the upper Deerfield River Basin is subjected to a storm with average 24-hour basin rainfall of much more than 13 inches, that Harriman Dam will be overtopped and will fail. The failure of Harriman Dam for any reason when the pool is above elevation 1490 feet msl will produce a flood level at the Yankee Rowe site that is anywhere from 15 to 70 feet above plant grade.

Current practice by designers and constructors of major dams, especially where there is a potential for loss of life or major property damage, is to provide sufficient storage and spillway capacity to safely pass a Probable Maximum Flood or its equivalent. Harriman Dam is a large dam, and there is potential for loss of life and major property damage in the event of a failure of the dam. If the dam were being constructed today, it would probably be designed to safely pass the PMF.

The Federal Guidelines for Dam Safety (19) state:

> "C. Flood Selection for Design (or Evaluation) - The selection of the design flood should be based on an evaluation of the relative risks and consequences of flooding, under both present and future conditions. Higher risks may have to be accepted for some existing structures because of irreconcilable conditions.
>
> When flooding could cause significant hazards to life or major property damage, the flood selected for design should have virtually no chance of being exceeded. If lesser hazards are involved, a smaller flood may be selected for design. However, all dams should be designed to withstand a relatively large flood without failure even when there is apparently no downstream hazard under present conditions of development."

Therefore, based on the results of this NRC Flood Study and on the Federal Guidelines for Dam Safety, Harriman Dam should probably be considered for upgrading.

2.0 Recommendations

The staff has considered possible remedial measures for the potential flood problem at the Yankee Rowe site. The best "fix" would be either an emergency spillway in the west abutment of the Harriman Dam or a diversion to divert excess flows to an adjoining drainage basin. The emergency spillway would have to be sized to pass the difference in runoff volume between present capacity and the volume for a Spillway Design Flood. The Spillway Design Flood, SDF, is that flood discharge, regardless of other designation or method of computation, which is used to develop the hydrologic and hydraulic design of a spillway and dam. Other possible remedial measures would be to raise the existing dam (about 15 feet) to contain the PMF or remove the dam completely.

Since Harriman Dam can safely pass a flood that is about equivalent to a maximum regional event, which is a rare event (see Sections 4.2.4 and 4.5), it is recommended that continued operation of the Yankee Rowe Plant be allowed provided that the licensee initiate a program of analysis, design and installation or construction of engineered mitigation measures. Such activities should be scheduled for completion in coordination with other influential decisions affecting the plant and the dams; i.e., seismic effects, but in any event, to be completed by January 1, 1984.

7. References

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4. U.S. Army Corps of Engineers Hydrologic Engineering Center, September 1978, "Flood Hydrograph Package (HEC-1) Users Manual for Dam Safety Investigations."

5. U.S. Weather Bureau (now U.S. Weather Service, National Oceanic and Atmospheric Administration), 1956, Hydrometeorological Report No. 33, "Probable Maximum Precipitation East of the 105th Meridian for Areas from 10 to 1000 Square Miles and Durations of 6, 12, 24 and 48 hours."

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13. Yankee Atomic Electric Company (YAEC) Report No. YAEC-1139, July 1977, "Yankee Nuclear Station Probable Maximum Flood Analysis."

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DRAFT

TABLE 3.2.1

CHARACTERISTICS OF UPPER DEERFIELD RIVER DAMS

	Somerset	Searsburg	Harriman	Sherman
Construction Completed	1913	1922	1924	1926
Drainage Area (square miles)	30	90	184	236
Height (feet)	104	50	215.5	110
Length (feet)	2010	475	1250	810
Dam Crest Elevation - USGS (feet)	2146.58	1762.66	1521.16	1129.66
Spillway Crest Elevation USGS (feet)	2133.58	1749.66	1491.66	1103.66
Active Storage at Spillway Crest (Acre feet)	57345	282.5	103375	4561
Surface Area at Spillway Crest (Acres)	1623	25	2050	194
Discharge Capacity, 5 ft. over Spillway Crest (cfs)	4950	5850	15040	6800
Discharge Capacity, 10 ft. over Spillway Crest (cfs)	8930	14390	33520	20700
Generation Capacity (MW)	--	4	33.6	7.2

TABLE 3.3.1

Major Storm Events and Related Information
in the Upper Deerfield River Basin at Harriman Station Since 1924

Year	Harriman Max. 24-Hr. Rainfall (In.)	Storm Duration (Hr.)	Harriman Total Rainfall (In.)	Harriman Max. Water Level (Ft.)	Charlemont Max. Flow (
1927 (Nov.)	5.38	>72	5.77	1482.3	36,000
1936 (Mar.)	1.94	62	3.50	1486.5	32,200
1938 (Sep.)	4.74	82	8.89	1497.8	56,300
1948-49 (Jan.)	5.53	74	9.02	1498.1	42,600
1955 (Aug. 14)	4.27	82	8.08	1491.2	2,980
1955 (Aug. 19)	2.49	42	2.82	1489.1	5,240
1955 (Oct.)	3.87	66	7.76	1494.0	20,200
1960 (Sep.)	3.87	38	5.73	1487.8	12,800
1976 (Aug.)	4.11	86	6.06	1494.8	18,100

Harriman Spillway Crest (USGS Datum) - 1491.7 feet
Harriman Dam Crest (USGS Datum) - 1521.2 feet
Harriman Dam Drainage Area - 184 sq. miles
Charlemont Drainage Area - 362 sq. miles

TABLE 3.3.2

HISTORIC RAINFALL

No. NA 1-21; Storm Center: Elka Park, N.Y.; Oct. 4-6, 1932

MAXIMUM AVERAGE DEPTH OF RAINFALL IN INCHES

Area in Sq. Mi.	Duration of Rainfall in Hours										
	6	12	18	24	30	36	48	66			
Max. Station	4.7	7.5	10.0	10.0	10.0	11.5	11.7	11.7			
10	4.5	7.2	9.8	9.8	9.8	11.2	11.5	11.5			
100	3.9	6.3	8.9	9.1	9.1	10.1	10.6	10.6			
200	3.8	5.9	8.6	8.8	8.8	9.7	10.1	10.1			
500	3.6	5.5	7.8	8.1	8.1	8.8	9.1	9.1			
1,000	3.3	5.2	7.2	7.4	7.4	8.1	8.4	8.4			
2,000	3.1	4.8	6.6	6.8	6.8	7.4	7.7	7.7			
5,000	2.6	4.1	5.5	6.0	6.0	6.5	6.9	6.9			
10,000	2.2	3.5	4.8	5.4	5.4	5.9	6.3	6.3			
20,000	1.9	3.0	4.1	4.8	4.9	5.3	5.7	5.7			
50,000	1.5	2.4	3.3	4.1	4.2	4.6	5.0	5.0			
60,000	1.4	2.3	3.1	3.9	4.1	4.4	4.8	4.8			

No. NA 2-22A; Storm Center: Westfield, Mass.; Aug. 17-20, 1955

MAXIMUM AVERAGE DEPTH OF RAINFALL IN INCHES

Area in Sq. Mi.	Duration of Rainfall in Hours									
	6	12	18	24	30	36	48	60	72		
Max. Station	7.9	11.7	14.3	18.2	19.4	19.5	19.8	19.8	19.8		
10	7.8	11.1	13.0	18.4	18.5	18.9	19.4	19.6	19.6		
100	7.6	10.5	11.6	14.6	17.8	18.1	18.8	19.0	19.0		
200	7.4	10.2	11.4	14.2	17.1	17.6	18.2	18.4	18.4		
500	6.8	9.7	10.8	13.4	16.3	16.8	17.2	17.3	17.3		
1,000	6.2	9.2	10.2	12.4	15.4	15.9	16.2	16.4	16.4		
2,000	5.4	8.0	9.4	11.2	14.0	14.5	14.9	15.2	15.2		
5,000	4.0	6.3	7.9	9.5	11.7	12.1	12.6	13.0	13.0		
10,000	3.1	5.0	6.5	8.0	9.7	10.0	10.6	10.8	10.8		
20,000	2.1	3.6	4.9	6.3	7.6	7.9	8.3	8.5	8.5		
35,000	1.3	2.5	3.6	4.7	5.6	6.0	6.4	6.5	6.5		

No. NA1-3; Storm Center: Paterson, N.J.; Sept. 20-24, 1882

MAXIMUM AVERAGE DEPTH OF RAINFALL IN INCHES

Area in Sq. Mi.	Duration of Rainfall in Hours										
	6	12	18	24	30	36	48	60	72	96	108
10	5.6	9.7	11.3	11.4	11.4	14.3	16.9	16.9	17.6	17.9	17.9
100	4.9	8.7	10.3	10.6	10.6	13.2	15.7	15.9	16.1	16.6	16.6
200	4.6	8.3	9.8	10.1	10.1	12.6	15.0	15.2	15.5	15.9	15.9
500	4.1	7.6	8.8	9.0	9.1	11.6	13.7	14.0	14.3	14.6	14.6
1,000	3.6	6.7	7.7	7.9	8.1	10.6	12.2	12.6	12.9	13.2	13.2
2,000	[illegible]	5.8	6.6	6.8	6.9	9.2	10.6	11.0	11.3	11.7	[illegible]
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
[illegible]	2.2	[illegible]	[illegible]	[illegible]	[illegible]	4.2	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	3.1	3.7	3.9	4.2	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	2.3	2.6	2.9	3.5	4.2	[illegible]	5.2	5.5	[illegible]	[illegible]

TABLE 3.3.2 (Con't)

No. NA 1-17; Storm Center: Kinsman Notch, N.H.; Nov. 2-4. 1927

MAXIMUM AVERAGE DEPTH OF RAINFALL IN INCHES

Area in Sq. Mi.	Duration of Rainfall in Hours										
	6	12	18	24	30	36	48	60			
10	7.8	10.6	11.7	12.0	12.6	13.7	14.0	14.0			
100	5.8	8.3	8.8	9.2	9.5	10.1	10.3	10.3			
200	5.7	8.2	8.6	8.8	9.0	10.0	10.2	10.2			
500	5.5	7.9	8.2	8.3	8.5	9.0	9.2	9.2			
1,000	4.8	7.3	7.7	7.8	8.2	8.8	8.9	8.9			
2,000	4.0	6.4	7.0	7.3	7.9	8.1	8.2	8.2			
5,000	2.7	4.8	6.1	6.7	7.2	7.7	7.9	7.9			
10,000	2.3	4.0	5.5	6.3	6.7	7.0	7.3	7.3			
20,000	2.0	3.5	4.7	5.3	5.8	6.2	6.4	6.4			
50,000	1.6	2.8	3.6	4.1	4.5	4.9	5.1	5.1			
60,000	1.4	2.5	3.3	3.8	4.2	4.5	4.8	4.8			

No. NA 1-27; Storm Center: Hector, N.Y.; Jul. 6-10, 1935

MAXIMUM AVERAGE DEPTH OF RAINFALL IN INCHES

Area in Sq. Mi.	Duration of Rainfall in Hours									
	6	12	18	24	30	36	48	60	72	90
10	5.2	10.2	11.4	11.8	12.0	13.4	14.2	14.2	14.2	14.2
20	5.1	9.7	11.1	11.5	11.6	12.9	13.9	13.9	14.0	14.1
100	4.9	8.6	10.1	10.5	10.7	11.5	13.0	13.1	13.4	13.6
200	4.7	8.0	9.6	10.0	10.3	10.9	12.5	12.6	12.9	13.2
500	4.3	7.3	8.8	9.3	9.5	9.8	11.6	11.8	12.0	12.4
1,000	4.0	6.7	8.2	8.6	8.8	9.0	10.6	10.8	11.1	11.5
2,000	3.5	6.0	7.3	7.8	8.0	8.2	9.5	9.8	10.0	10.4
5,000	2.7	4.8	5.9	6.4	6.6	6.8	7.7	8.2	8.5	8.7
10,000	2.1	3.7	4.6	5.1	5.4	5.7	6.4	7.0	7.2	7.5
20,000	1.3	2.6	3.2	3.7	4.1	4.5	5.1	5.6	5.9	6.2
50,000	0.9	1.9	2.4	2.9	3.4	3.8	4.3	4.8	5.2	5.5
58,500	0.7	1.4	1.9	2.4	2.9	3.3	3.8	4.3	4.8	5.1

No. NA 2-2; Storm Center: Barre, Mass; Sept. 17-22, 1938

MAXIMUM AVERAGE DEPTH OF RAINFALL IN INCHES

Area in Sq. Mi.	Duration of Rainfall in Hours										
	6	12	18	24	30	36	48	60	72	96	120
10	6.4	8.2	9.6	11.3	12.2	13.2	14.3	15.0	15.8	16.9	17.1
100	5.0	6.8	8.3	9.5	10.4	11.4	13.0	14.0	15.1	16.5	16.8
200	4.6	6.3	7.8	9.0	9.8	10.9	12.4	13.4	14.8	16.1	16.4
500	4.1	5.6	7.1	8.3	9.0	10.2	11.6	12.6	14.2	15.4	15.7
1000	3.7	5.1	6.6	7.7	8.4	9.6	11.0	12.0	13.8	14.6	15.0
2000	3.3	4.6	6.0	7.2	7.8	9.0	10.4	11.3	13.2	13.8	14.2
5000	2.7	3.9	5.1	6.3	6.9	8.2	9.5	10.3	12.0	12.4	12.8
10000	2.3	3.3	4.4	5.7	6.2	7.4	8.6	9.6	10.9	11.3	11.6
20000	1.9	2.8	3.5	4.9	5.5	6.4	7.5	[illegible]	[illegible]	[illegible]	[illegible]
50000	1.4	2.1	2.8	3.7	4.2	4.8	[illegible]	6.6	7.3	7.7	[illegible]
[illegible]	1.2	1.9	2.5	3.3	3.7	4.2	5.1	5.9	6.9	6.8	7.1

TABLE 3.3.2(Con't)

No. OR9-23; Storm Center: Port Alleghany, Pa.; Jul. 17-18, 1942

"Smethport"

MAXIMUM AVERAGE DEPTH OF RAINFALL IN INCHES

Area in Sq. Mi.	Duration of Rainfall in Hours										
	6	12	18	24							
Max. Station	30.7	34.3	35.5	35.5							
1	29.3	32.0	33.8	34.2							
5	26.4	28.6	30.5	31.0							
10	24.7	26.7	28.7	29.2							
20	22.8	24.8	26.8	27.4							
50	19.7	21.9	24.1	24.6							
100	16.4	19.4	21.8	22.4							
200	13.1	16.8	19.3	19.9							
500	9.1	13.2	15.7	16.3							
1,000	6.4	10.3	12.6	13.3							
2,000	3.9	7.2	9.2	10.2							
4,300	2.5	4.6	6.1	7.1							

TABLE 4.1

SENSITIVITY ANALYSIS

	HARRIMAN SPILLWAY ≅ 75% CAPACITY 4/										HARRIMAN SPILLWAY = 100% CAPACITY							
	Licensee Snyder Coefficients 1/						NRC Clark Coefficients 2/				Licensee Snyder Coefficients 2/				NRC Clark Coefficients 2/			
Index Rainfall (inches)	13.0	16.0	16.5	18.9 3/	18.9	21.3	16.0	16.5	18.9	21.3	16.0	16.5	18.9	21.3	16.0	16.5	18.9	21.
HARRIMAN RESERVOIR																		
Max. Pool (elev)	1520.7	1524.6	1524.0	1524.3	1525.7	1526.4	1523.9	1523.6	1524.4	1525.2	1524.1	1523.9	1524.9	1525.6	1522.6	1523.7	1524.5	1575.
Peak Q (1000 cfs)	2/	2,463	2,500	2,454	2,547	2,596	2,413	2,448	2,474	2,522	2,433	2,479	2,526	2,570	41	2,408	2,462	2,5
Time of Failure	-	69.33	68.67	69.33	68.0	67.33	73.0	71.67	69.67	68.67	69.67	69.00	68.0	67.33	-	73.67	70.33	69
Time of Qp (hrs)	72.33	70.33	69.67	70.33	69.0	68.33	74.0	72.67	70.67	69.67	70.67	70.00	69.0	68.33	75.0	74.67	71.33	70
Duration over top (hrs)	0	1.55	1.24	1.53	1.29	1.33	2.49	2.18	1.55	1.27	1.51	1.55	1.25	1.29	6.0	3.47	1.88	1.
SHUGIAN RESERVOIR																		
Max. Pool (elev)	1131.6	1171.3	1172.4	1182.5	1174.6	1176.5	1196.6	1197.4	1172.5	1174.1	1170.9	1171.7	1174.0	1175.9	1132.4	1169.7	1172.0	117
Peak Q (1000 cfs)	80	2,151	2,193	2,146	2,273	2,352	2,065	2,096	2,183	2,250	2,117	2,164	2,248	2,329	325	2,078	2,161	2,7
Time of Failure (hrs)	-	66.00	66.33	69.67	65.67	65.33	73.67	72.33	66.33	66.0	65.67	65.67	65.0	64.67	66.67	66.67	66.0	65.
Time of Qp (hrs)	67.33	70.67	70.00	70.65	69.33	68.67	74.35	73.02	71.00	70.0	71.00	70.33	69.33	68.67	67.65	75.00	71.67	70.
Duration over top (hrs)	2.33	2.43	2.76	3.00	2.82	2.51	3.67	3.33	2.41	2.45	2.47	2.45	2.47	2.47	1.41	2.41	2.47	2.

1/ NRC Flood Study Using Adopted Values

2/ Sensitivity Study for Comparison Purposes

3/ 43% of 24-Hour Rainfall in Worst 6-Hour Period

4/ Spillway Capacity Limited to Orifice Control in the Riser - Elev. 1414

TABLE 5.1

COMPARISON OF OTHER FLOOD STUDIES TO THE NRC FLOOD STUDY

	C. T. MAIN (FPC Report)[12] (1977)	YANKEE ROWE (FPC Report) (1977)	YANKEE ROWE (1980 Report)	NUCLEAR REGULATORY COMM. PMF	NUCLEAR REGULATORY COMM. Lower Limit	Corps of Engineers 1963 4/
SOMERSET RESERVOIR						
Peak Inflow (cfs)	40,000	30 300	33,000	54,489	39,000	55,900
Peak Outflow (cfs)	7,600	7.[illegible]00	7,500	8960	6,600	10,800
Max. Pool Elev. (ft msl)	2143.7	2141.4	2141.9	2145.4	2142.5	2144.7
Top of Dam (ft msl)	2146.58					
Starting Pool Elev. (ft msl)		2133.50	2133.58	2133.6	2133.6	
SubBasin Rainfall (24 hr-200 S.M.)	HR#33	19.5	14.1	18.9 2/	14.0 2/	25.1 3/
HARRIMAN RESERVOIR						
Peak Inflow (cfs)	210,600	161,500	149,900	240,570	172,100	170,000
Peak Outflow (cfs)	216,300	40,700	37,000	2,547,000	37,100	38,600
Max. Pool Elev. (ft msl)	1523.2 1/	1521.J	1519.2	1525.7 1/	1522.9	1527±
Top of Dam (ft msl)	1521.16					
Starting Pool (ft msl)		1488.96	1491.66	1491.6	1491.6	
Sub Basin Rainfall (24 hr-200 S.M.)	HR#33	19.5	14.1	18.9 2/	14.0 2/	20.6 3/
SHERMAN RESERVOIR						
Peak Inflow (cfs)	265,300	74,140	87,300	2,121,600	86,700	113,000
Peak Outflow (cfs)	241,800	71,000	207,000	2,273,300	312,600	110,000±
Max. Pool Elev. (ft msl)	1131.66 1/9	1128.6	1131.7	1174.6 1/	1132.2 1/	1135±
Top of Dam (ft msl)	1129.66					
Starting Pool (ft msl)		1103.66	1103.66	1103.6	1103.6	
Sub Basin Rainfall (24 hr-200 S.M.)	HR#33	19.5	14.1	18.9 2/	14.0 2/	25.1 3/

1/ Dam Assumed to fall when overtopped by 2.0 feet.

2/ 24 hour average rainfall for 236 square mile drainage area

3/ The Corps study was for a spillway design flood, so their rainfall is probably based on the reservoir subbasin drainage area.

4/ No flow over dams.

APPENDIX A

[illegible]

8 February 1963

Chairman
Federal Power Commission
Washington 25, D. C.

Dear Mr. Chairman:

Reference is made to the Commission's letter dated 15 November 1962 concerning the application filed by New England Power Company for license for constructed hydroelectric project (No. 2323) located on the Deerfield River in Massachusetts and Vermont.

The applicant's project consists of seven dams with hydroelectric power stations located on the Deerfield River and one storage dam and reservoir on the East Branch of the Deerfield River. The projects are located between points 11 miles and 59 miles of the Deerfield River upstream of its junction with the Connecticut River.

There is an existing Federal navigation project on the Connecticut River from its mouth to Hartford, Connecticut, [illegible] miles below the mouth of the Deerfield River. The Deerfield River itself is not currently used for commercial navigation nor is there any indication that improvements of the river for navigation are desired or warranted at this time. The plans of the structures affecting navigation are satisfactory and the project would not affect the interests of navigation. Therefore, special terms and conditions for insertion in the license, if issued, are not considered necessary insofar as the interests of navigation are concerned.

The Somerset, Harriman and Sherman dams are considered major earth structures with the Somerset and Harriman dams having significant effect on reducing flood flows of record. The Sherman dam is a run of river project and would not reduce flood discharges. The applicant has studied the adequacy of existing spillways of all dams of the project and selected a design flood that is considered a rare event. However, computations made by the Corps of Engineers based on recent criteria indicate that the Harriman and Sherman dams would not pass the Corps' spillway design flood with spillway capacities and heights of dams as now proposed by the applicant. Computations indicate that the Somerset dam would pass the spillway design flood using either the Corps' or the Company's criteria but there would be a reduction in freeboard associated with the Corps' criteria. The Commission may wish to review the spillway design features of two of the major structures and data used by the Corps of Engineers will be furnished upon request.

One copy of the application is returned as requested.

Sincerely yours,

1 Incl
Application

ROBERT C. [illegible]
Colonel, Corps of Engineers
Assistant Director of Civil Works
for Eastern Divisions

CC: U. S. Army Engineer Division, New England

Mr. Coh ✓Mr. Childs/mbe/546

Mr. Cohen

NEDGW (15 Nov 62) 2nd Ind
SUBJECT: New England Power Company, Project No. 2323

U. S. Army Engr Div, New England, Waltham, Mass. 30 January 1963

TO: Chief of Engineers, ATTN: ENGCW-ZP, DA, Washington, D. C.

1. The application of the New England Power Company (NEPCO) of Boston, Massachusetts to the Federal Power Commission for a license for the constructed hydroelectric project (No. 2323) located on the Deerfield River in Massachusetts and Vermont, together with supplementary data submitted with your letter of 18 January 1963, has been reviewed and the following report is submitted herewith.

2. Project. - The project consists of seven dams with hydroelectric power stations located on the Deerfield River between points 11 miles and 59 miles upstream of its junction with the Connecticut River and one dam and reservoir on the East Branch of the Deerfield River used to regulate flows. The seven dams with power stations are:

Deerfield No. 2 near Conway and Shelburne, Mass. with three generating units totalling 4,800 KW.

Deerfield No. 3 near Shelburne and Buckland, Mass. with three generating units totalling 4,800 KW.

Deerfield No. 4 near Shelburne and Buckland, Mass. with three generating units totalling 4,800 KW.

Deerfield No. 5 near Monroe and Florida, Mass. with three generating units totalling 15,000 KW.

Sherman near Monroe, Mass. with one generating unit of 7,200 KW.

Harriman near Whitingham and Wilmington, Vt., with three generating units totalling 33,600 KW.

Searsburg near Searsburg, Vermont with one generating unit of 4,000 KW.

These seven power stations contain a total of 17 hydroelectric generating units with a combined capacity of 74,200 KW. One of the dams (Harriman) also provides a storage reservoir to regulate river flows and one (Somerset) provides storage only with no hydroelectric generating facilities. The remaining dams are used for pondage only so that water use is essentially run of river flow.

DRAFT

NEDED (15 Nov 62) 2nd Ind 30 January 1963
SUBJECT: New England Power Company, Project No. 2323

3. Effect of Project on Flood Flows. -

a. Of the eight NEPCO dams on the Deerfield River and its East Branch, three are considered major earth structures. Two of these dams have a significant effect on reducing flood flows. Somerset dam, located in the upper portion of the Deerfield River, controls the runoff from 30 square miles of drainage area and contains a usable capacity of 57,300 acre-feet or almost 36 inches of runoff. Harriman dam, located downstream of Somerset dam has a net drainage area of 154 square miles and contains 116,000 acre-feet of usable storage which is equivalent to about 14 inches of runoff. Sherman dam, a run of river plant is a high earth dam and is located about two miles downstream of Harriman dam.

b. Somerset Reservoir is a storage reservoir and in general is drawn down during the summer, fall and winter months to augment the natural river flow. The reservoir refills during the spring snow melt season. Harriman Reservoir is a storage reservoir which also contains a penstock for power development. The reservoir draw down and filling in general follows the same pattern as Somerset Reservoir. The remaining dams on the Deerfield River are run of the river plants with no appreciable storage.

c. During the March 1936 flood, both Somerset and Harriman dams stored almost the entire storm runoff. During the September 1938 flood, the record flood in this area, Somerset dam stored almost the entire inflow and reduced the computed peak inflow of 7,500 cfs to a discharge of about 1,000 cfs. At Harriman dam, the computed peak inflow of 44,400 cfs was reduced to about 15,000 cfs.

d. Somerset and Harriman dams are considered major earth structures and both have been designed to pass a major flood. Following the August 1955 flood in southern New England, the Power Company performed studies to determine the adequacy of existing spillways at all dams on the Deerfield River. The design storm was patterned after the August 1955 storm, which at its center near Westfield, Massachusetts, produced 18.2 inches of rainfall in 48 hours. The runoff from the design storm is 14 inches. Unit hydrographs were developed from the 1948 - 1949 year-end flood.

e. The design flood, routed through the reservoirs, indicated that remedial measures would be required at Harriman and Sherman dams. Somerset dam is capable of handling the design flood with about four feet of freeboard. At Harriman dam, the Power Company proposes to raise the top of the earth embankment seven feet to elevation 1518.66 feet, m.s.l. This would allow 5.2 feet of freeboard during the design flood. At Sherman dam, the Power Company plans to raise the dam 10 feet to elevation 1129.66 feet, msl and to enlarge the spillway discharge channel. No major alterations are planned for the remaining dams.

f. The New England Division has built several flood control dams in the Connecticut River basin just north and south of the Deerfield River. The more recent reservoirs have been designed to pass the latest Corps spillway design flood with five feet of freeboard. Some NED dams built prior to 1955 do not have adequate spillway capacity to pass a spillway design flood computed from the latest criteria. A spillway design flood was computed for Somerset, Harriman and Sherman dams from data used to derive the recent spillway design floods at Ball Mountain Dam on the West River and Littleville Dam on the Middle Branch, Westfield River. This analysis indicated that Somerset dam could pass the spillway design flood with about two feet of freeboard but additional spillway capacity would be required at both Harriman and Sherman dams.

g. At Harriman dam, if additional spillway capacity were not provided to supplement the limited capacity of the existing morning glory, the dam would have to be raised an additional 13 feet more than the present proposal in order to store the excess volume of runoff.

h. At Sherman dam, the additional spillway capacity required to pass the spillway design flood would be at least double the proposed capacity and could be even greater if additional spillway capacity were provided at Harriman dam. The following table compares the results derived from the Power Company studies and the Corps criteria for Somerset, Harriman and Sherman dams.

Description	Somerset Dam		Harriman Dam		Sherman Dam	
	Corps Criteria	Power Co. Criteria	Corps Criteria	Power Co. Criteria	Corps Criteria	Power Co. Criteria
Drainage Area - sq. mi.	30.0	30.0	154 net	154 net	52 net	52 net
Design Storm Rainfall-inches	25.1	18.2	20.6	18.2	25.1	18.2
Design Storm Runoff-inches	22.4	14.0	18.1	14.0	22.4	14.0
Net Peak Inflow - cfs	55,900	21,000	159,000	77,000	95,000	32,500
Total Peak Inflow - cfs	55,900	21,000	170,000	80,000	113,000	63,000

(Table continued)

Description	Somerset Dam Corps Criteria	Somerset Dam Power Co. Criteria	Harriman Dam Corps Criteria	Harriman Dam Power Co. Criteria	Sherman Dam Corps Criteria	Sherman Dam Power Co. Criteria
Peak Outflow - cfs	10,500	5,700	39,600	36,000	110,000±	61,500
Maximum Pool Stage	2144.7	2142.68	1,527±	1513.16	1,135±	1125.16
Power Company - Proposed Top of Dam		2146.58		1516.66		1129.66

i. It is recognized that the design flood selected by the Power Company is a rare event and will provide a much higher degree of protection against failure than the present design. Somerset dam can pass the spillway design flood with two feet of freeboard and is considered satisfactory. Harriman dam, with a fixed spillway capacity of about 36,000 cfs would be vulnerable to a flood approaching the magnitude of a Corps spillway design flood. Similar conditions occur at Sherman dam.

j. Inspection of the remaining five dams indicates that they are run of river plants with no appreciable storage. The lesser design criteria used for these dams is reasonable since failure of any of them should not release the tremendous volume of water required to cause a catastrophe.

4. Effect of Project on Navigation. - There is an existing Federal navigation project on the Connecticut River from its mouth to Hartford, Connecticut, 68 miles below the mouth of the Deerfield River. The Deerfield River itself is not currently used for commercial navigation, nor is there any indication that improvements of the river for navigation are desired or warranted at this time.

5. Conclusion. - It is concluded that the two existing storage dams, Somerset and Harriman, have a significant effect in reducing flood flows. Harriman and Sherman dams, as modified by proposals of the Power Company, will be safe against failure in a large flood. However, these two high earth dams have spillways which will not pass a Corps spillway design flood, even after proposed modifications. The project would not

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affect present or anticipated commercial navigation and special terms and conditions for insertion in the license, if issued, are not considered necessary.

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1. nc
wd 1 incl-2

P. C. [illegible]
Colonel, Corps of Engineers
Division Engineer

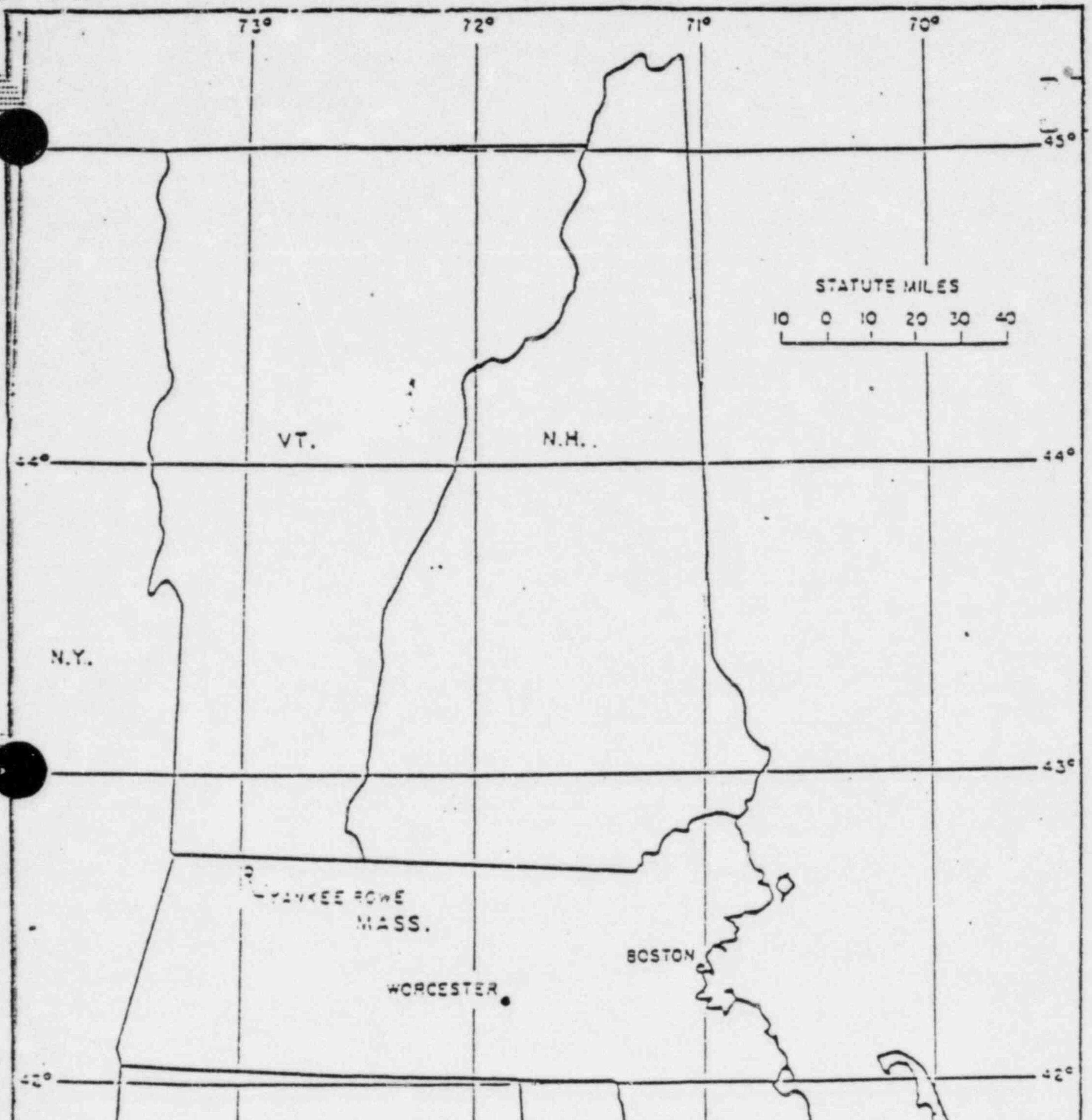

Figure 3.1.1 — General Area and Site Location Map

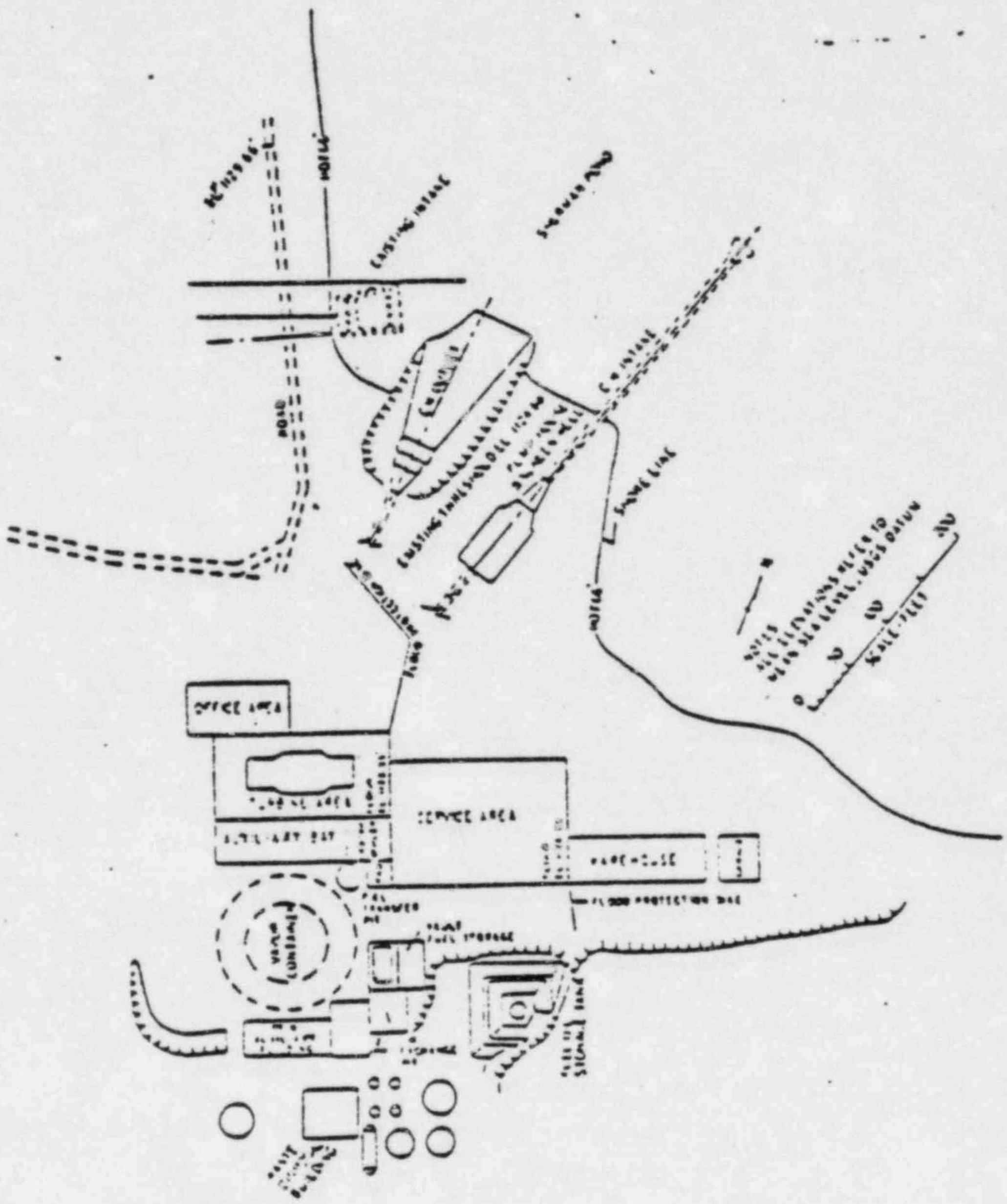

Figure 3.1.2 — General Station Layout

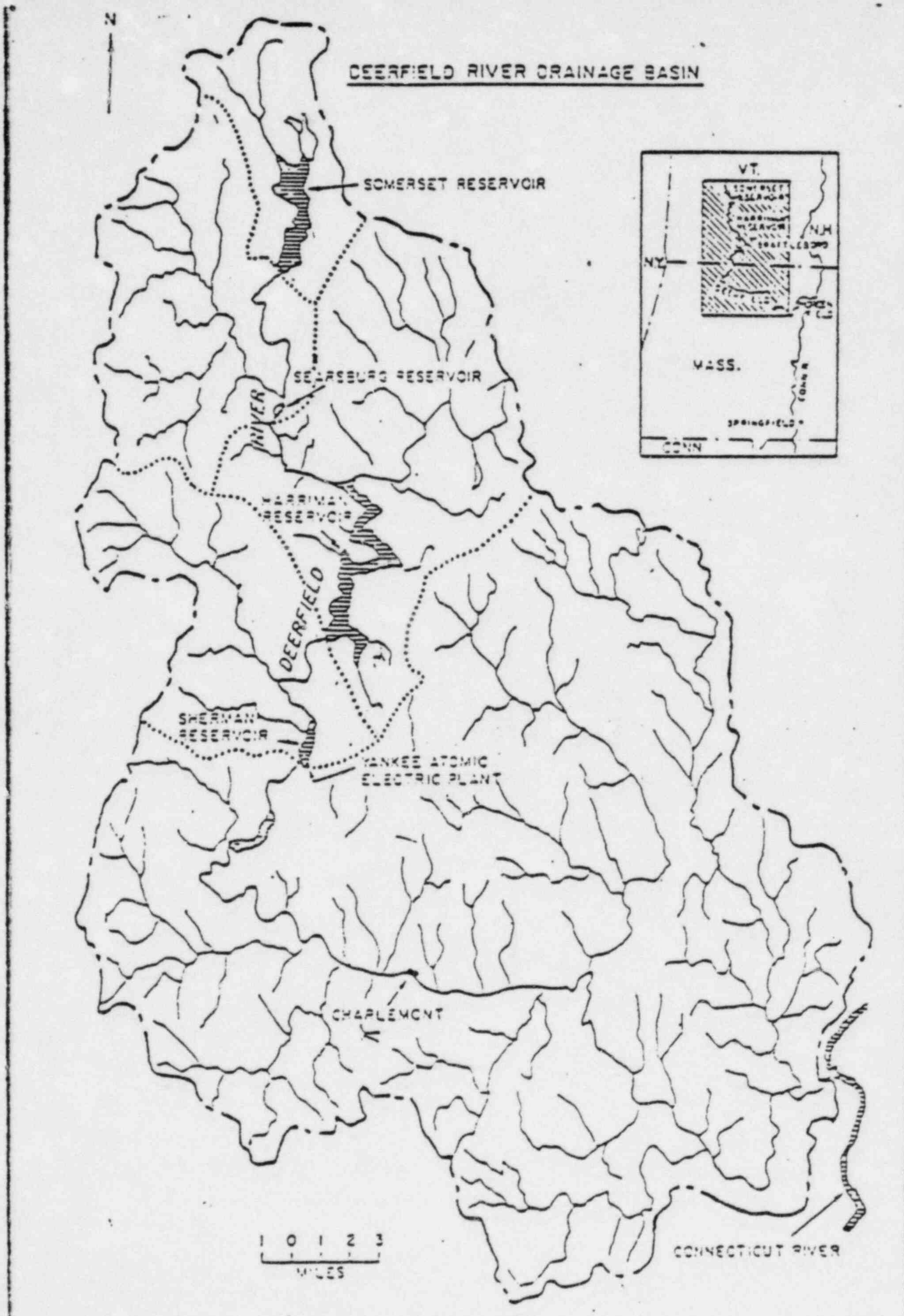

Figure 3.2.1 — Deerfield River Drainage Basin Map

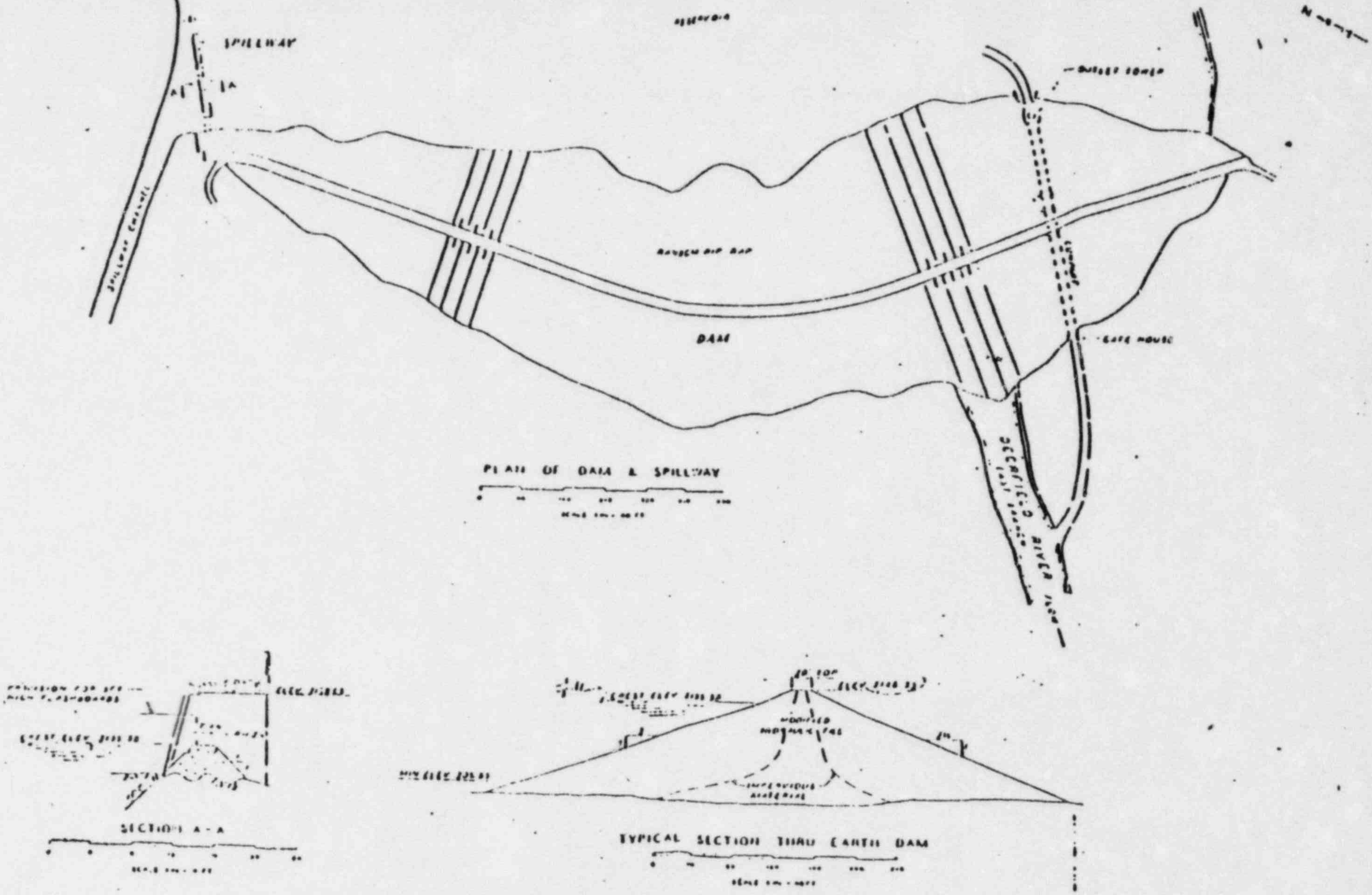

Figure 3.2.2 Somerset Dam and Spillway – Plan and Sections

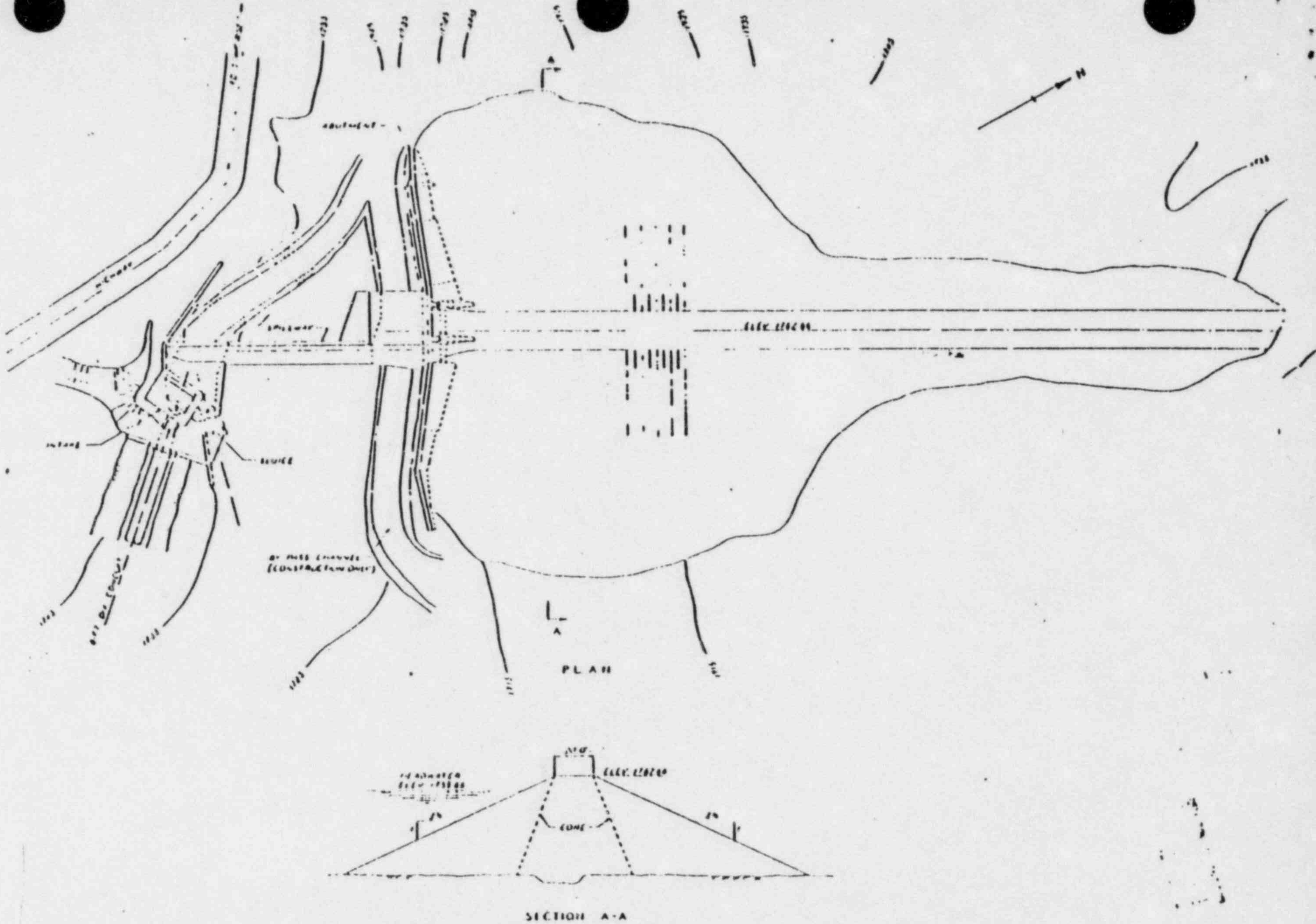

Figure 3.2.3 Searsbury Dam — Plan and Section

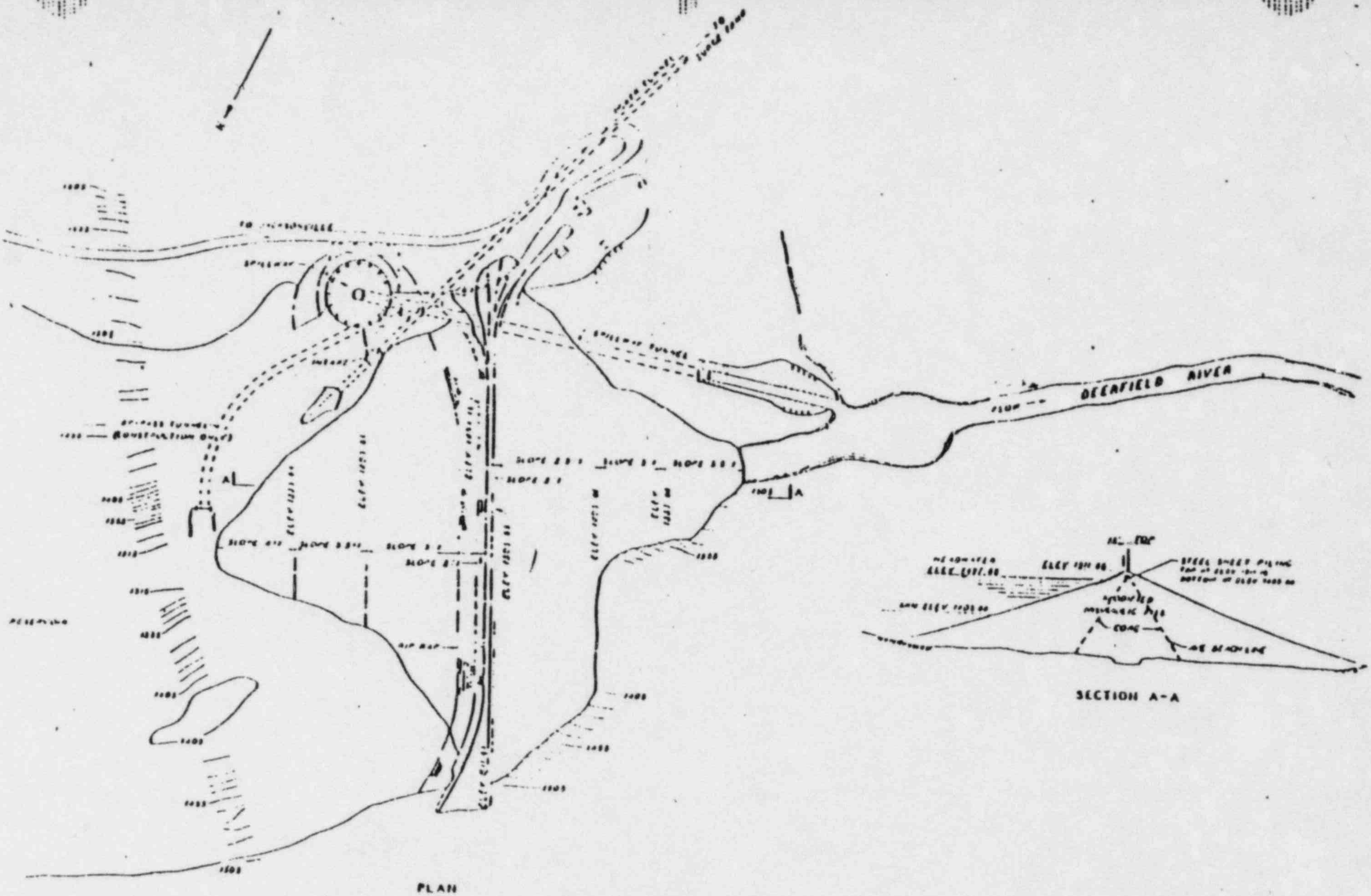

Figure 3.2.4 Harriman Dam — Plan and Section

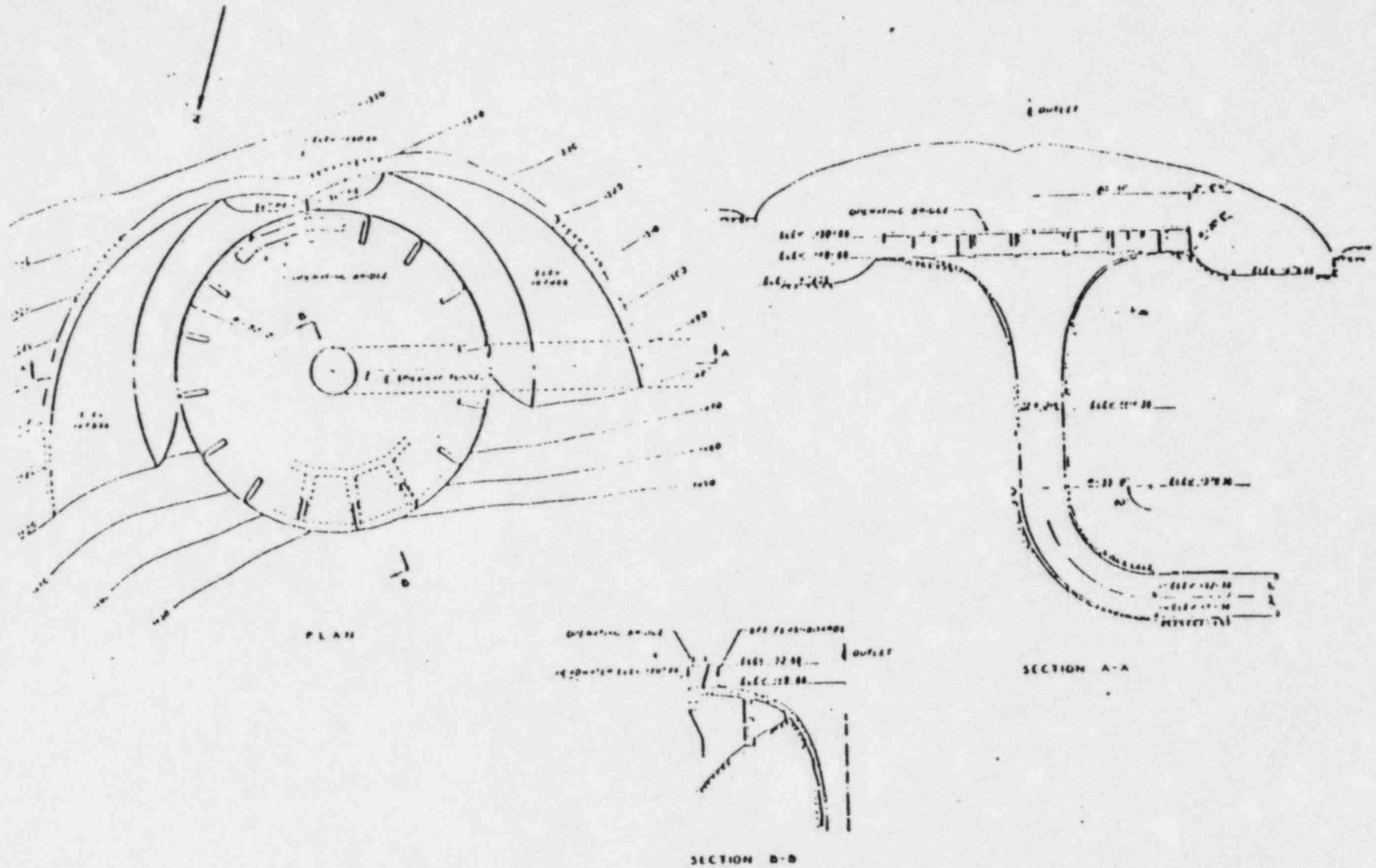

Figure 3.2.5 Harriman Spillway -- Plan and Sections

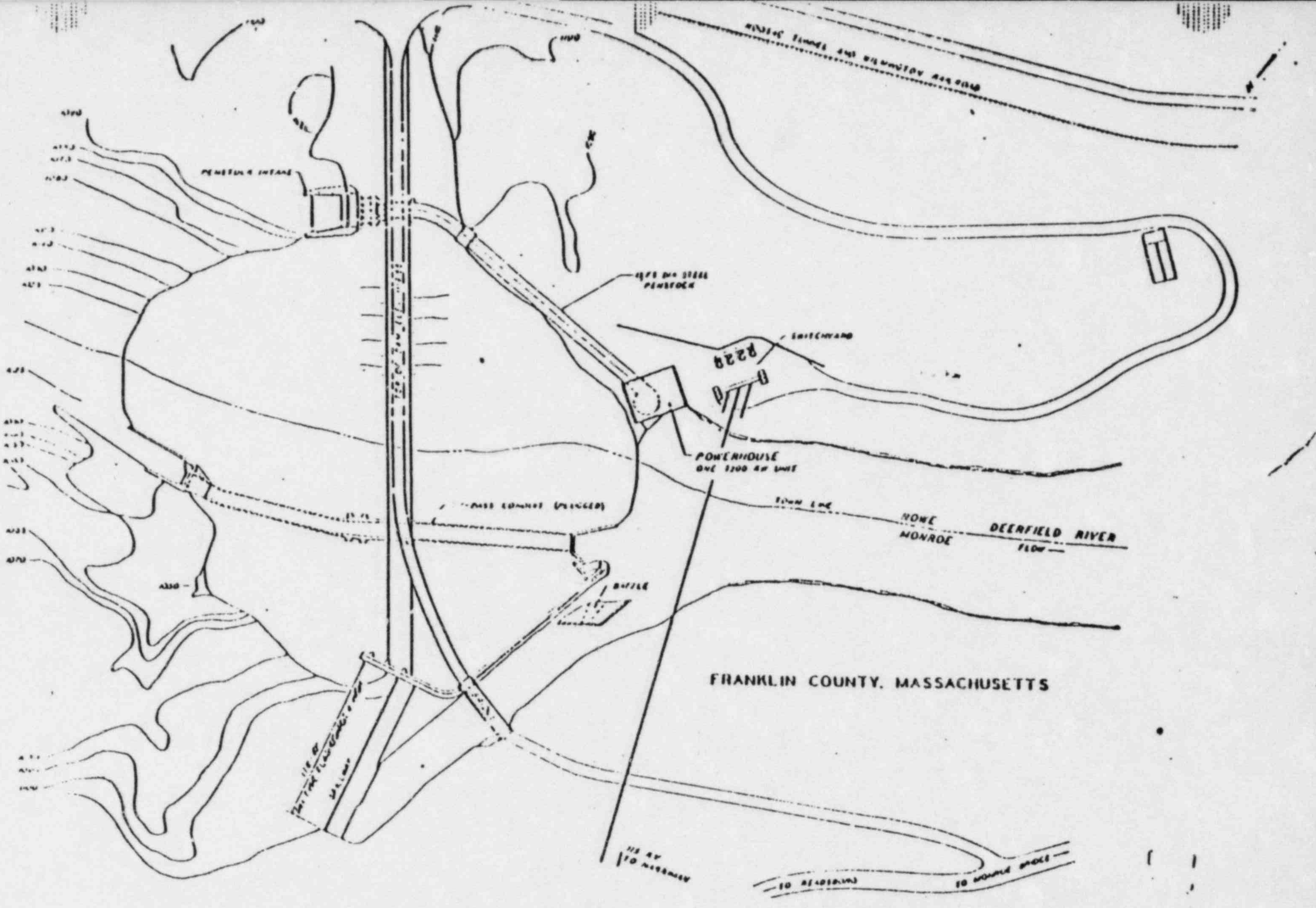

Figure 3.2.6 Sherman Dam — General Plan

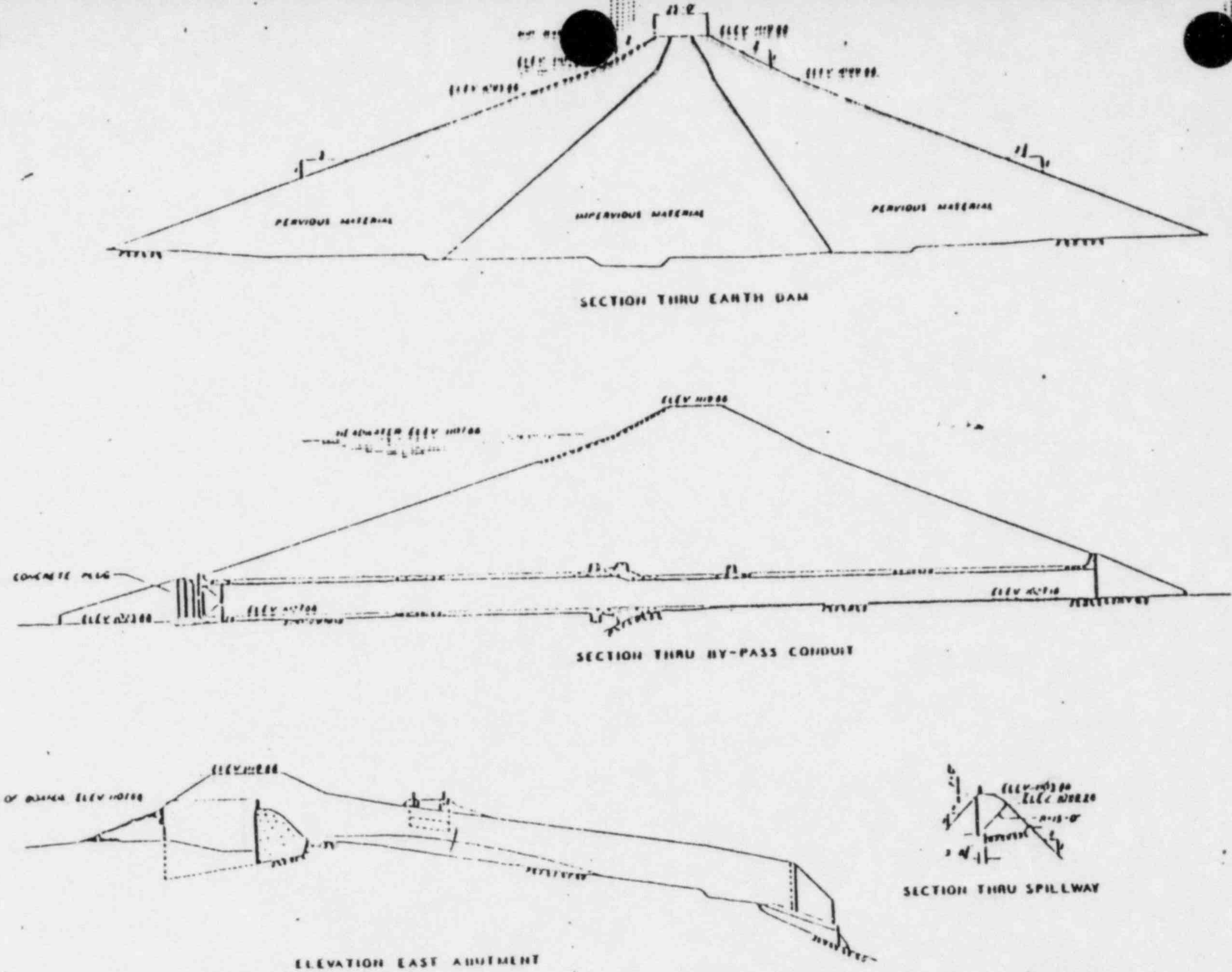

Figure 3.2.7 Sherman Dam — Sections

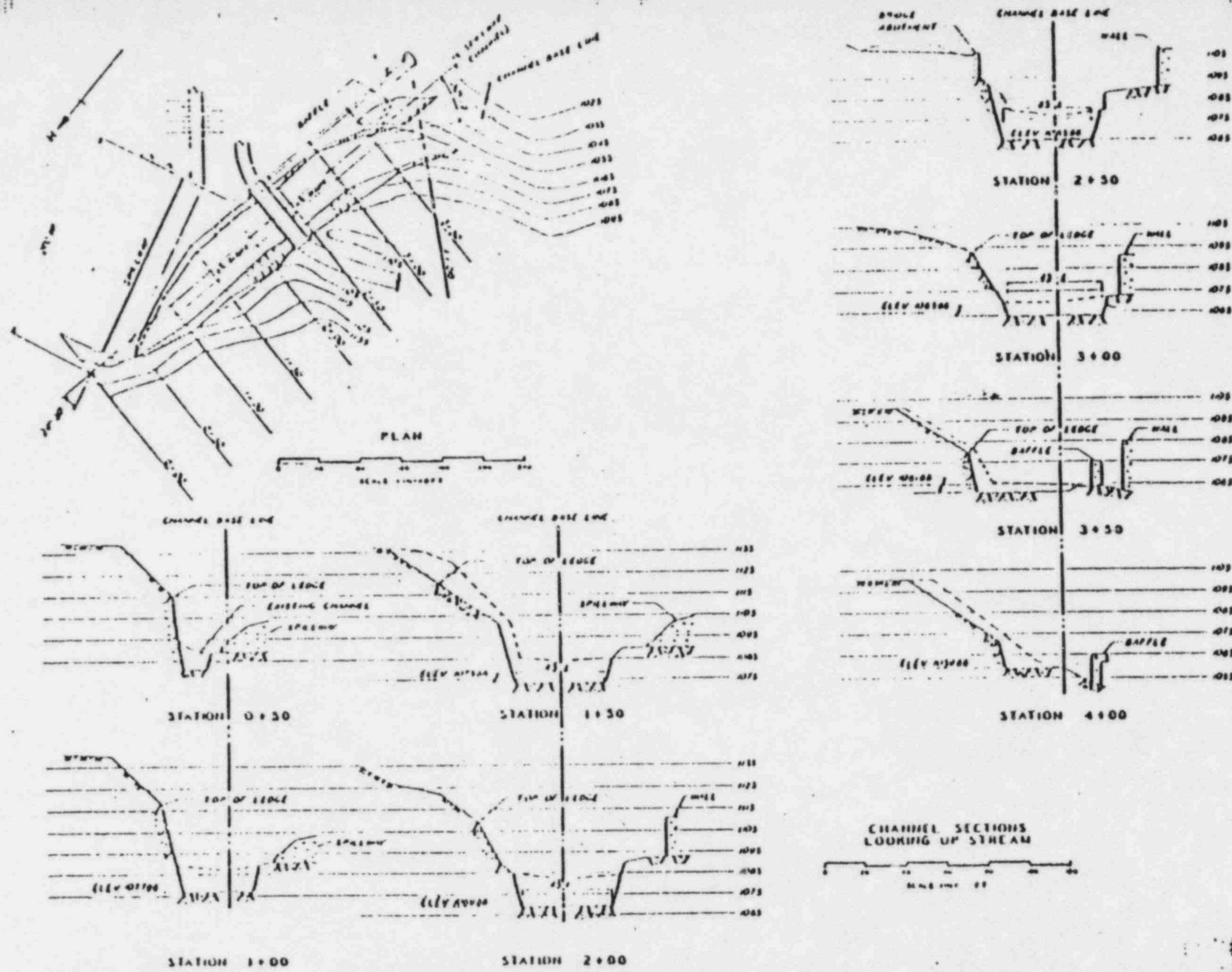

Figure 3.2.8 Sherman Spillway – Plan and Sections

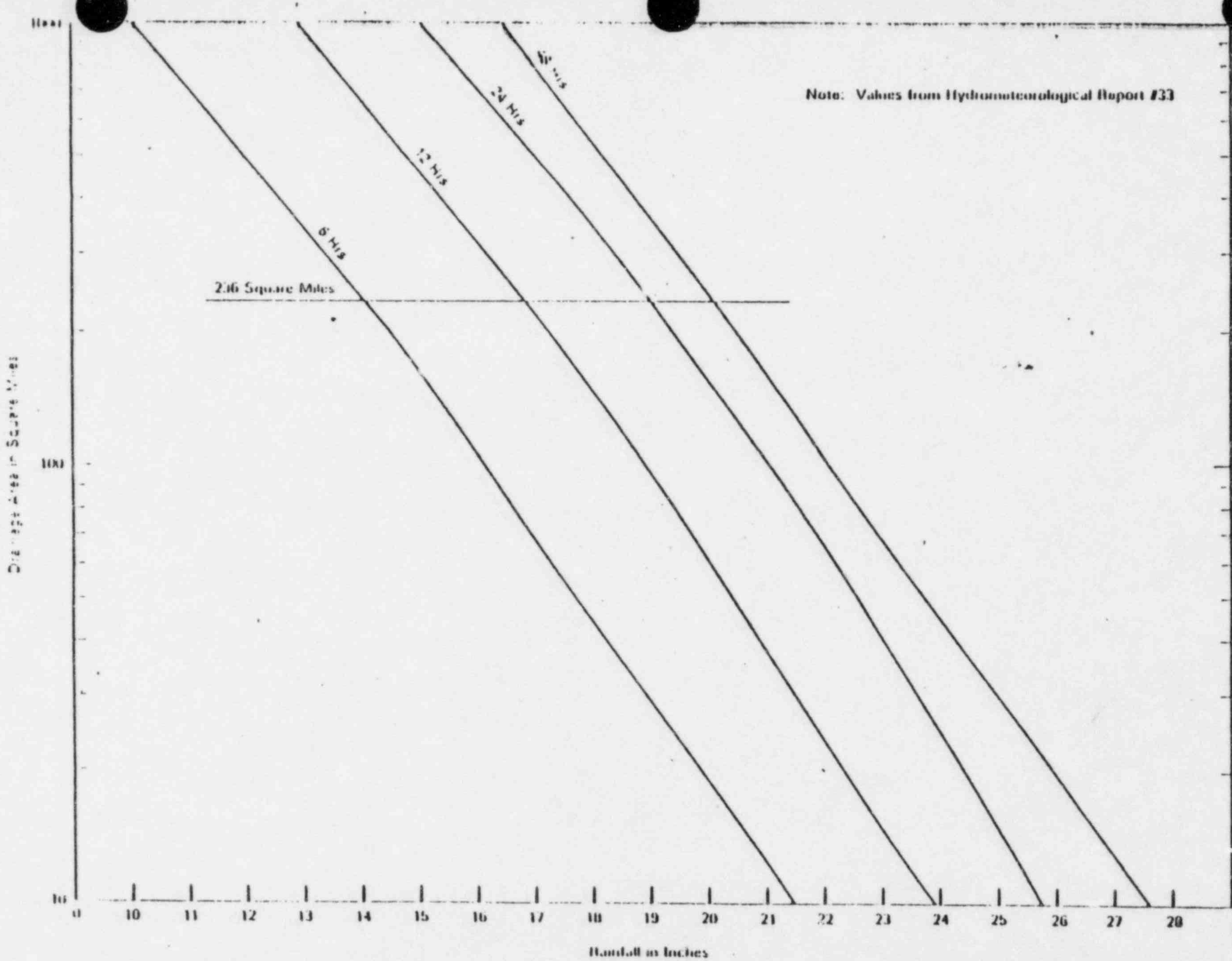

Figure 4.2.1 Rainfall Depth — Area — Duration

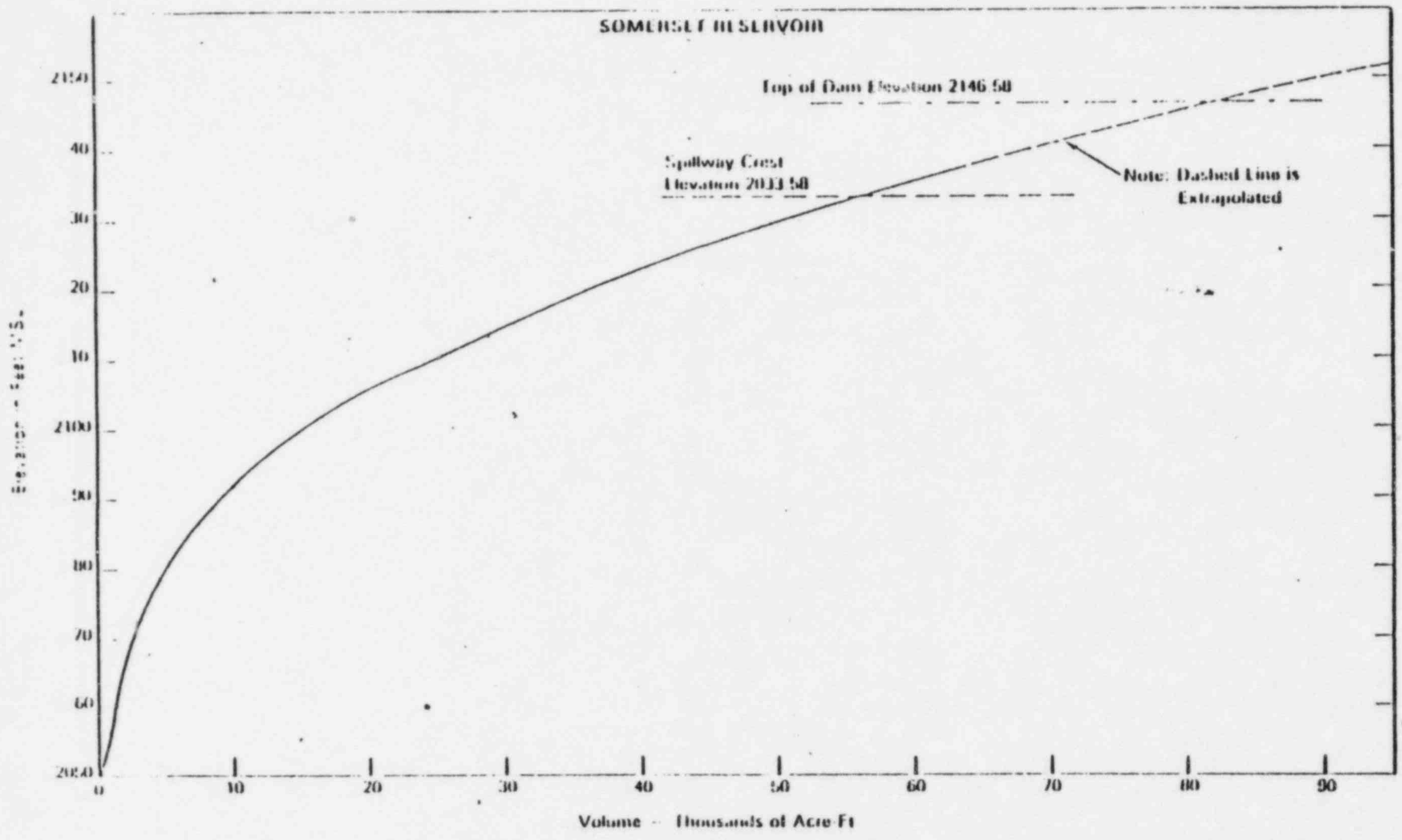

Figure 4.3.1 Somerset Reservoir Capacity Curve

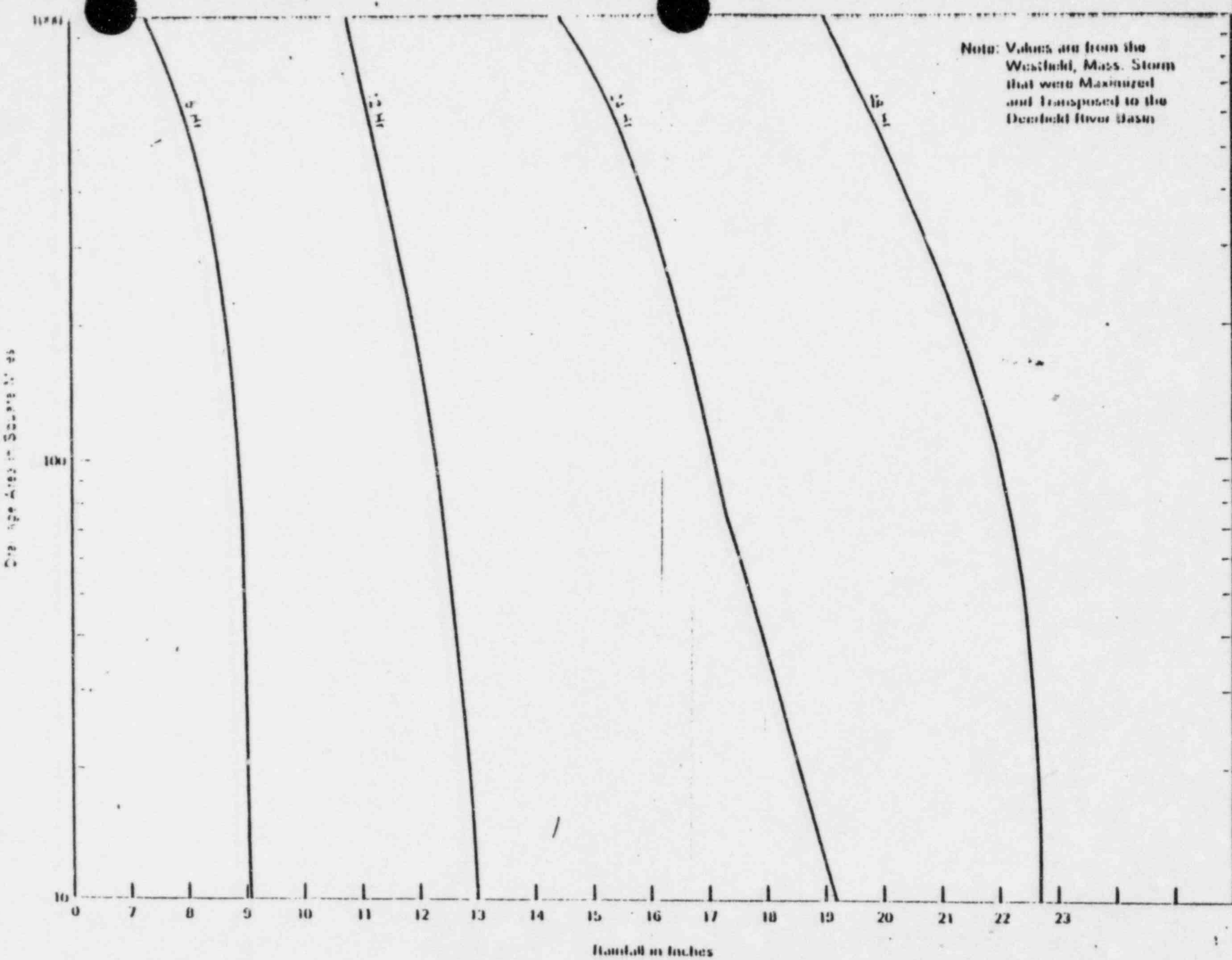

Figure 4.2.2 Rainfall Depth -- Area -- Duration

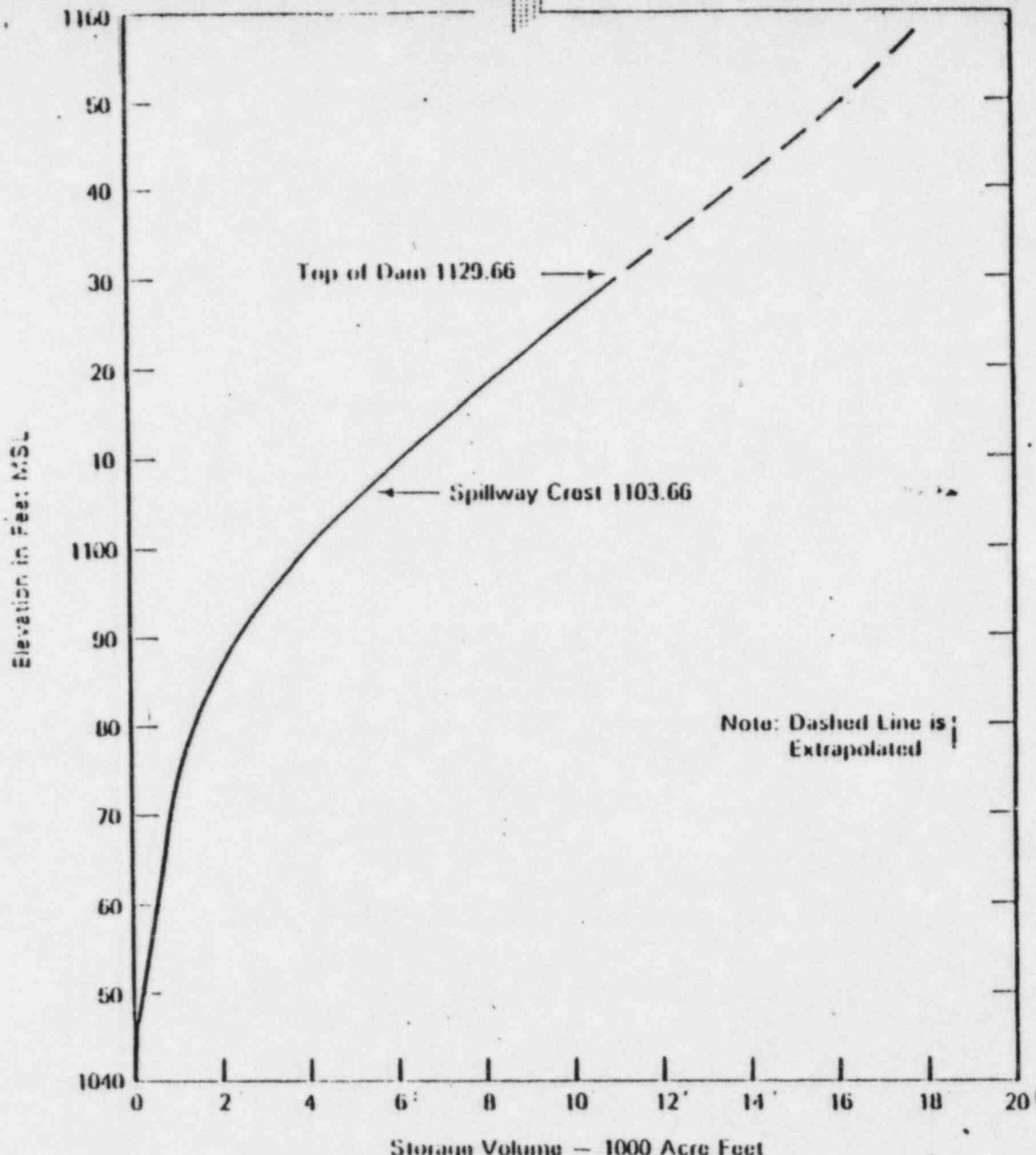

Figure 4.3.3 Sherman Reservoir Capacity Curve

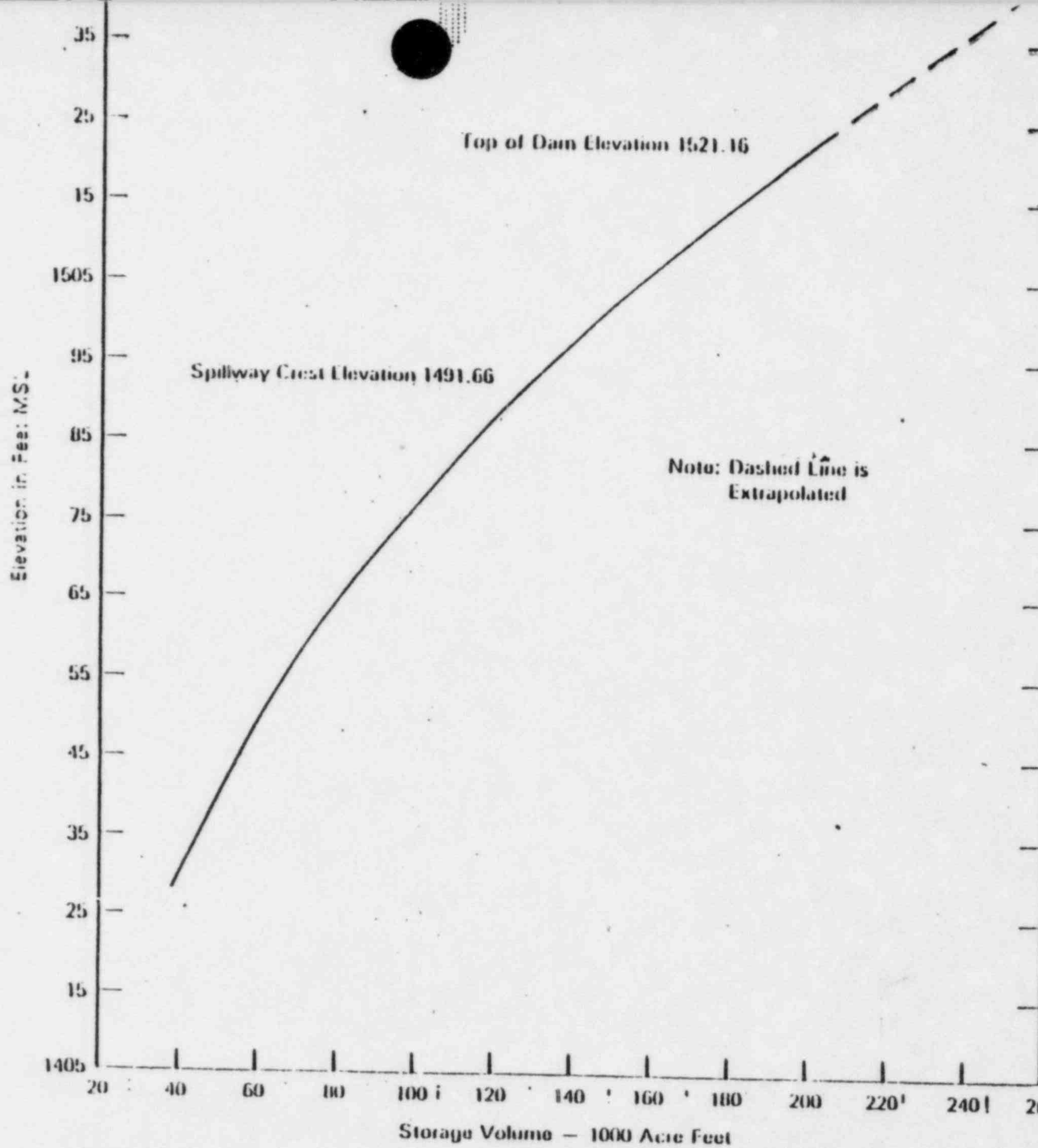

Figure 4.3.2 Harriman Reservoir Capacity Curve

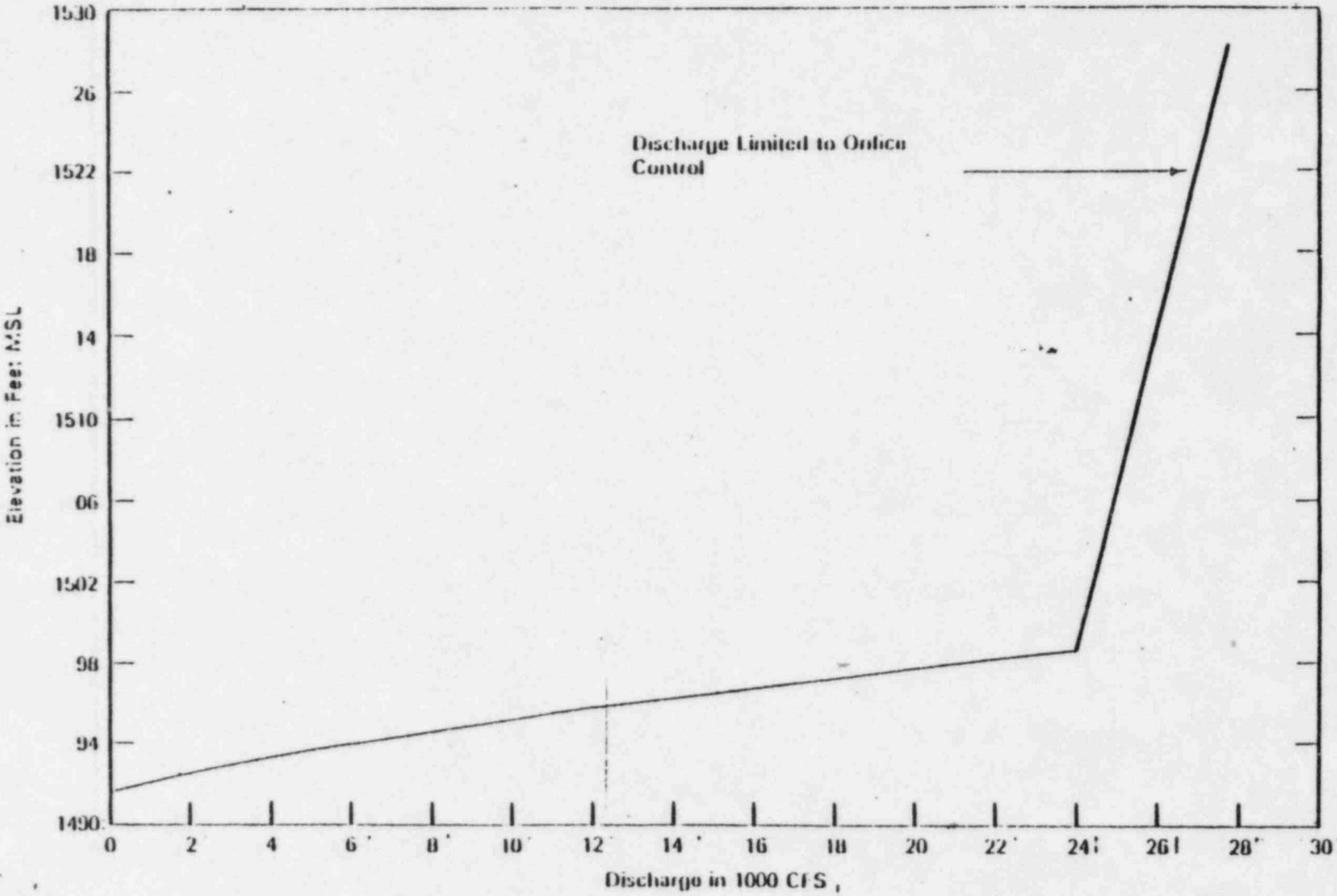

Figure 4.3.5 Harriman Dam Spillway Rating Curve

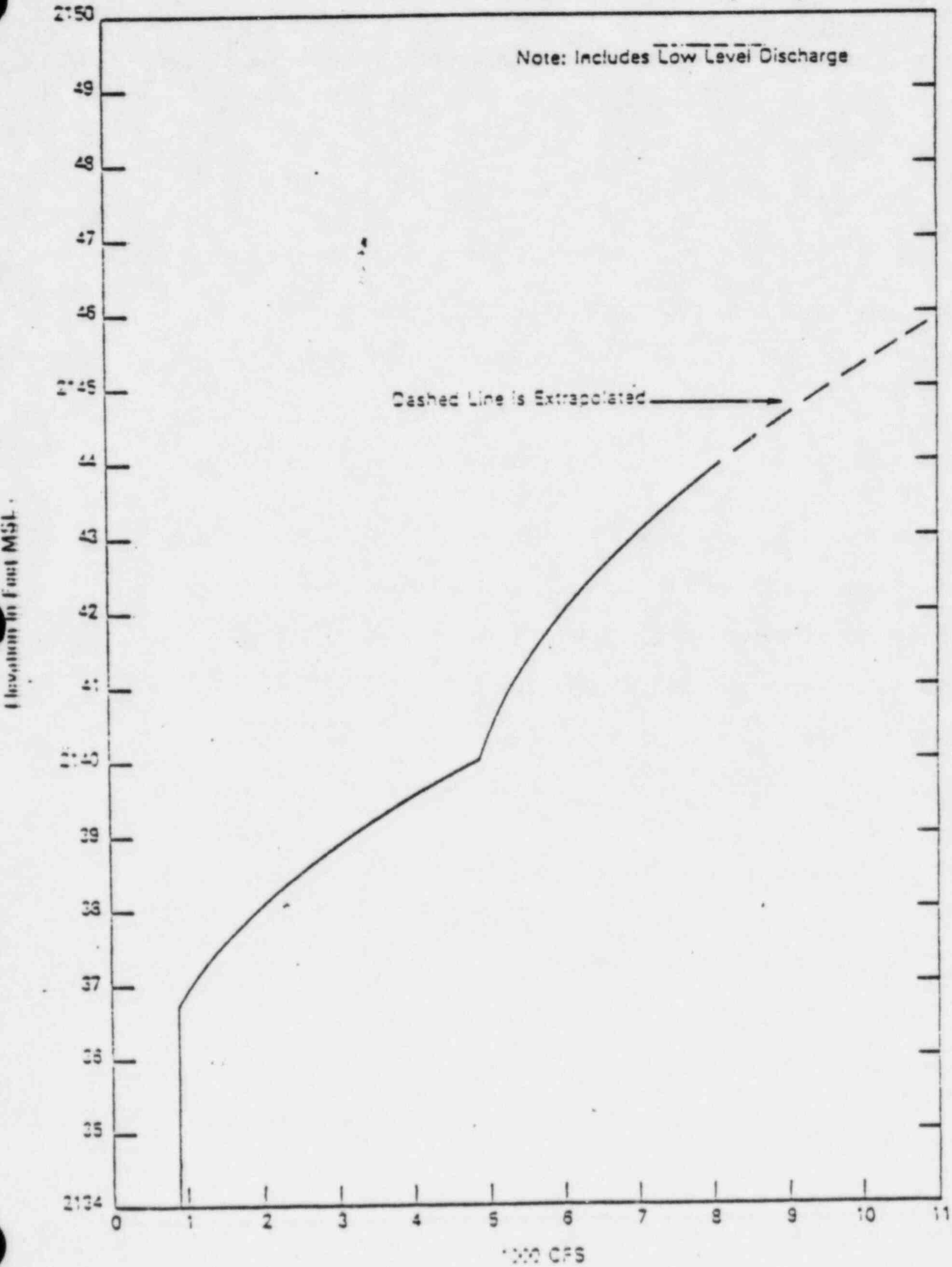

Figure 4.3.4 Somerset Dam Spillway Rating Curve

Conservatisms in a PMF Estimation

The purpose of this appendix is to present some of the conservatism involved in determining a Probable Maximum Flood.

Definitions

Probable Maximum Flood (PMF) - The PMF is defined as the hypothetical flood (peak discharge, volume and hydrograph shape) that is considered to be the most severe reasonably possible, based on comprehensive hydrometeorological application of Probable Maximum Precipitation and other hydrologic factors favorable for maximum flood runoff such as sequential storms and snow melt."[1/]

Probable Maximum Precipitation (PMP) - The PMP is defined as "the estimated depth for a given duration, drainage area, and time of year for which there is virtually no risk of exceedance. The Probable Maximum Precipitation for a given duration and drainage area approaches and approximates the maximum which is physically possible within the limits of contemporary hydrometeorological knowledge and techniques."[1/] The following are elements involved in determination of the PMF:

1. The primary factor contributing to a PMF is the PMP. Estimates of PMP are generally obtained from National Weather Service reports of generalized PMP estimates for different regions of the country. In some cases where generalized PMP is not available or the generalized values are not pertinent to a specific site, site specific studies can be undertaken to develop site

[1/] Definitions from "Standards for Determining Design Basis Flooding at Power Reactor Sites", ANSI N170-1976, ANS-2.8.

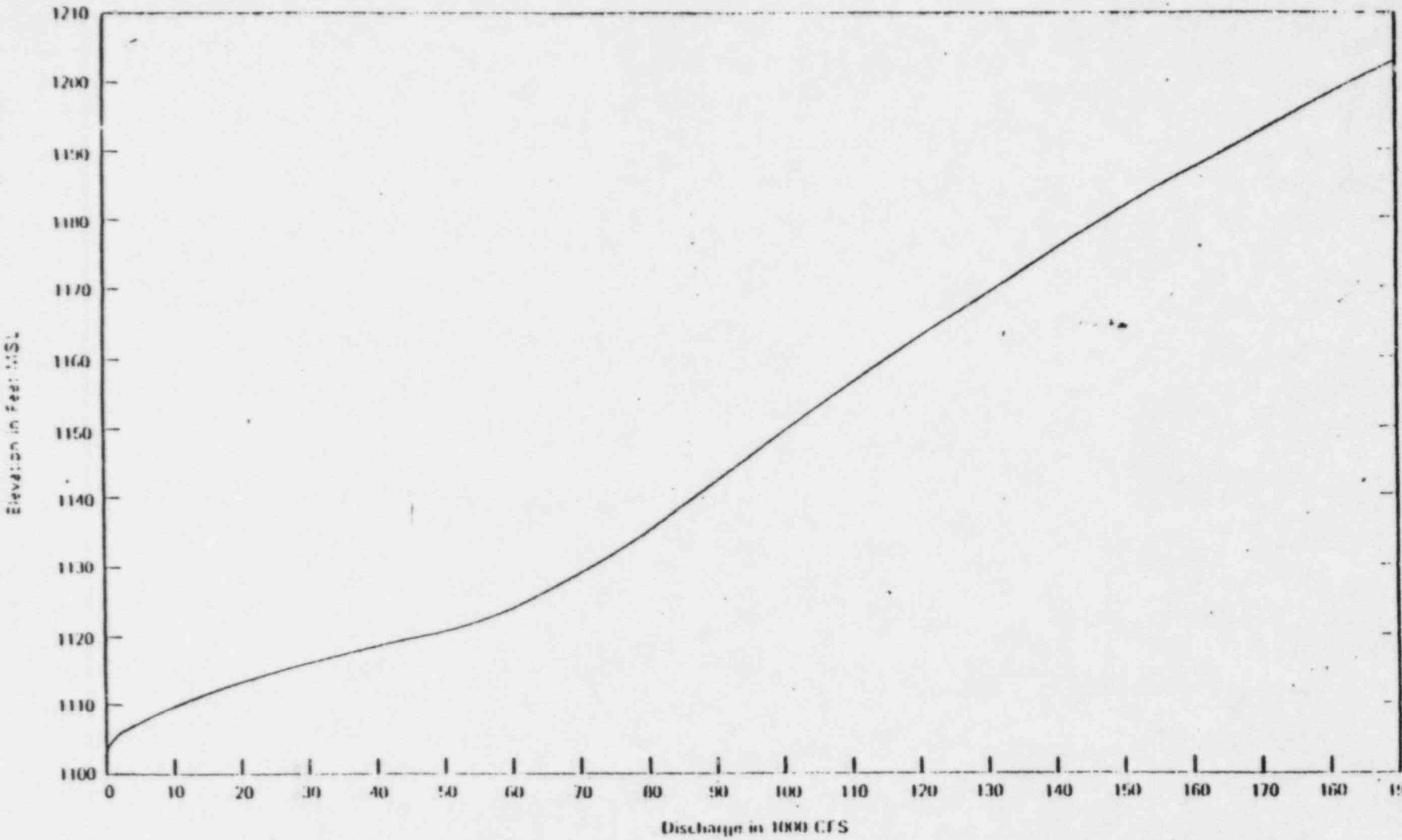

Figure 4.3.6 Sherman Dam Spillway Rating Curve

specific PMP values. Criteria for the derivation of PMP estimates has been developed by the World Meteorological Organization and the Office of Hydrology, National Weather Service.

The significant conservatisms in generalized PMP estimates are (1) Only the most severe storms of record are considered; (2) the moisture content of these storms are adjusted to obtain the maximum percipitable water and (3) the process of enveloping or smoothing will introduce some additional conservatisms.

2. It is assumed that the storm producing rainfall is critically centered over the basin to produce maximum runoff.

3. An antecedent storm is assumed to have occurred 3 to 5 days before the PMF. This assumption minimizes rainfall losses and maximizes runoff in that it would saturate the soil and fill depression storage. Where storage reservoirs are involved, this assumption will generally insure that reservoirs are full at the start of the PMF. This then reduces attenuation due to storage and maximizes peak discharge.

4. The unit hydrograph coefficients are derived from storms of record in the basin whenever possible. Where storm data are not available, then conservative synthetic coefficients are derived to produce a maximum peak discharge for the basin. This can be a significant conservatism for open river floods. Where streams are controlled by reservoir storage, hydrograph coefficients are usually not a significant factor since they do not effect runoff volume.

5. For open river conditions, backwater computations are used to predict water surface elevations from peak discharge. Computed water surfaces are sensitive to the roughness coefficient (manning "n" value). These can be derived from historic storm data, but since roughness coefficients decrease with increasing stage or discharge, this approach would be conservative when used with the much larger PMF discharges. When these coefficients can not be derived, conservatively high values are assumed.

6. Where reservoirs and dams are involved, unsteady flow and routing models may be used to conservatively estimate reservoir elevations and elevations of downstream flood waves when dams fail. An important feature that may allow for conservatism in these models is spillway capacity. There are two broad classifications of overflow spillways generally used for dams; (1) a wide rectangular spillway which usually incorporates an vyee type weir and may be gated or ungated and (2) a vertical circular riser connected by a 90° elbow to a horizontal outlet conduit through the dam or abutment, generally referred to as a "Morning Glory" type of spillway.

   The capacity of the ungated wide rectangular spillways can be determined accurately. The gated structures may require conservatism if gate operability cannot be assured or if debris or ice is a potential problem. Generally where gate operability cannot be assured, one or more gates are assumed to be failed in the closed position. Similarly, if debris or ice accumulation is a potential problem, the staff assumes partial blockage and the associated reduced capacity.

The morning-glory type spillways can also experience blockage due to ice or debris accumulations. The large diameter spillways on large dams normally can pass large logs and other debris. However, there is no experience on the type and size of debris that can be generated during a flood as large as the PMF and, therefore, partial blockage for these spillways are assumed. These spillways can also experience large fluctuations in discharge capacity and associated vibrations and stresses if they are subjected to water levels significantly in excess of the design values.

7. When dams are overtoped and assumed to fail, an important factor is the size of the breach and the time required to fully erode the breached section. Since there is very little methodology available for predicting these factors, they are generally conservatively assumed. These conservative values would trend toward worst case observed values. The resulting downstream flood stages could be significantly higher than what would be predicted using average observed values.

The above discussion only addresses some of the more significant factors or the ones that can result in the largest differences in estimated PMF values.

Differences in NRC and YAEC PMF Estimation

The previous section of this appendix addressed the major parameters and assumptions used to estimate a PMF. This section will address the major differences between the results of the NRC and YAEC PMF estimations.

Of all the items discussed in the previous section, only two, the estimation of PMP and the assumed spillway capacity, are areas of disagreement between the NRC and YAEC.PMF estimations.

The NRC estimated PMP by using National Weather Service reports of regional PMP estimates. The result of the NRC estimation is that the PMP will be 18.9 inches in 24 hours. YAEC estimated a design maximum rainfall (defined to be a site specific PMP) by performing site specific studies. The result of the YAEC estimation is that the PMP will be 14.3 inches of rain in 24 hours.

The NRC estimated the Harriman Dam spillway capacity to be 75% of its design value during the probable maximum flood. This value was chosen to conservatively estimate spillway capacity loss due to debris blockage or flow variations caused by water levels in excess of design values. YAEC estimated the spillway capacity to be 94% of its design value. This value was based on a number of analytical studies of the Harriman Dam spillway.

Details of the NRC and YAEC estimations of PMP and spillway capacity are presented in the staff safety evaluation, in the licensee's safety analysis (YAEC-1207, "Design Basis Flood Analysis," October 1980), and in Appendices A and B of the SER. Appendix A of the SER also contains Franklin Research Center's estimation of the Yankee site PMP.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555
July 15, 1982

Docket No. 50-29
LS05-82-07-040

Mr. James A. Kay
Senior Engineer - Licensing
Yankee Atomic Electric Company
1671 Worcester Road
Framingham, Massachusetts 01701

Dear Mr. Kay:

SUBJECT: SEP REVIEW TOPICS II-4, GEOLOGY AND SEISMOLOGY AND II-4.B, PROXIMITY OF CAPABLE TECTONIC STRUCTURES IN PLANT VICINITY - YANKEE NUCLEAR POWER STATION AT ROWE MASSACHUSETTS

Enclosed is a copy of our evaluation for Systematic Evaluation Program Topics II-4. "Geology and Seismology," and II-4.B, "Proximity of Capable Tectonic Structures in Plant Vicinity." These assessments compare your site condition, as described in the docket and references, with the criteria currently used by the staff for licensing new facilities. Please inform us if your site condition differs from the licensing basis assumed in our assessments.

Our review of these topics is complete and this evaluation will be a basic input to the integrated safety assessment for your facility unless you identify changes needed to reflect the existing site condition at your facility. These topic assessments may be revised in the future if NRC criteria relating to these topics are modified before the integrated assessment is completed.

Sincerely,

Ralph Caruso

Ralph Caruso, Project Manager
Operating Reactors Branch No. 5
Division of Licensing

Enclosure:
As stated

cc w/enclosure:
See next page

8207190323

cc
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Board of Selectmen
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Resident Inspector
Yankee Rowe Nuclear Power Station
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Ronald C. Haynes, Regional Administrator
Nuclear Regulatory Commission, Region I
631 Park Avenue
King of Prussia, Pennsylvania 19406

SEP - YANKEE ROWE NUCLEAR POWER PLANT
TOPICS II-4 - GEOLOGY AND SEISMOLOGY, AND
II-4B - CAPABILITY OF FAULTS IN THE SITE REGION

1 INTRODUCTION

1.1 Identification of Safety Issues

The SEP topics addressed in this chapter are the geology portion of Topic II-4, Geology and Seismology, and Topic II-4B, Capability of Faults in the Site Region. The seismology section of Topic II-4, Topics II-4A, and II-4C are addressed in "Final Review and Recommendations for Site Specific Spectra at SEP Sites" (memorandum from R. E. Jackson to W. T. Russell, 20 May, 1981).

1.2 Scope of the Review

Geologic and seismic evaluations were not made by the Atomic Energy Commission (AEC) during CP and OL activites of the Yankee Rowe site. Since that time, however, detailed investigations have been carried out in the region as part of licensing activities for Pilgrim 2, Seabrook, Units 1 and 2, Montague 1 and 2, Millstone 2 and 3, Indian Point 3, and New England 1 and 2 Nuclear sites. Data included in the SAR's and SER's of these facilities were considered in this review. Other investigations that have been done include: (1) geologic mapping of the area around the site by Weston Geophysical Corporation (WGR) reported in "Geology and Seismology Yankee Rowe Nuclear Power Plant", January 29, 1979; and (2) the New England Seismotectonic Study. This research is funded by the Nuclear Regulatory Commission and is under the direction of the Weston Observatory of Boston College. These investigations are ongoing and the current effort is expected to be completed in 1983. In conducting this review, the NRC staff has relied heavily on the results of all of these investigations as well as the published literature.

Several significant earthquakes have occurred in the site region recently, which are being assessed by the NRC to determine whether or not they could have an impact on New England sites: a series of earthquakes in New Brunswick, Canada, and a moderate size earthquake in central New Hampshire, which caused relatively high accelograph recordings near the epicenter. These occurrences are described below.

On January 9-11, 1982 a series of earthquakes occurred in New Brunswick Canada. The largest of these events was a magnitude 5.7 which occurred on January 9, 1982 and resulted in only minor damage (Modified Mercalli Intensity VI) in this sparsely populated area. In the past, however, events of such size in the eastern U.S. have typically resulted in Modified Mercalli Intensity (MMI) VIII. Many aftershocks of that event have occurred since the initial swarm, including a magnitude 4.8 on 31 March. This earthquake was felt through New Brunswick and northwestern Maine, but no damage was reported. These earthquakes are located in geologic terrain that appears to be similar to that found throughout the New England-Piedmont tectonic province, but the geology of the epicentral area is not well known. There are major faults mapped in the vicinity as well as a large pluton, but a relationship to structure is not known at this time. Geological investigations to attempt to identify a source mechanism was anticipated to begin this spring when the snow cover had melted. We do not know the status of this investigation at this time. Pending results of investigations in the epicentral area, there is no reason at this time to alter conclusions regarding the adequacy of the Yankee Rowe SSE. The NRC staff is actively monitoring the investigations of these earthquakes and will report on findings as they become available.

On January 18, 1982 a magnitude 4.5 earthquake occurred in Gaza, New Hampshire. The strong motion instrument at the Seabrook site some 50 miles to the southeast triggered indicating that the ground motion at Seabrook exceeded 0.01g. We have been notified that the U.S. Army Corps of Engineers had several instruments near the earthquake epicenter that also triggered.

Uncorrected copies of 3 sets of accelerograms recorded on and near a dam about 5 kilometers from the earthquake epicenter indicated a maximum peak acceleration of 0.56g. The short duration, high-frequency, ground motion associated with the highest peak acceleration (0.56g) is similar to that recorded from small nearby earthquakes at other locations which have also produced little or no damage. Since the instrument recording this motion was located on the dam abutment there is also a strong possibility that topography and/or the dam configuration contributed to the high peak. Based on our present information we do not believe these records indicate inadequacy of the Yankee Rowe seismic design. The geologic source of this event is not known. Investigations of this event are still ongoing, and, like the New Brunswick studies, the NRC staff is following them closely. We will report on the findings when they are available.

2 REVIEW CRITERIA

Current listing criteria which governed our review of the safety issues addressed in this chapter include Appendix A to 10 CFR Part 100, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," and NUREG-0800, Standard Review Plan, Sections 2.5.1 and 2.5.3.

3 RELATED SAFETY TOPICS AND INTERFACES

The geotechnical engineering aspects of the site are closely related to the topics covered in this chapter. They are addressed under Topics II-4D, II-4E, and II-4F. Topic II-4F is dependent on information from this chapter.

4 REVIEW GUIDELINES

Elements of Appendix A to 10 CFR Part 100, "Seismic and Geologic Siting Criteria for Nuclear Power Plants" were used in this review to provide guidance in identifying and evaluating tectonic structures in the site region to determine whether or not any of them are capable.

Chapter 2.5.1 of NUREG-0800, Standard Review Plan,guided the staff in its assessment of geologic features in the site area related to the potential for faulting, subsidence or collapse, landslides, weathering, or other foundation instabilities.

Chapter 2.5.3 of the SRP was utilized for guidance in considering the following subjects: The structural and stratigraphic conditions of the site and vicinity (Subsection 2.5.3.2), any evidence of fault offset or evidence demonstrating the absence of faulting (Subsection 2.5.2.2), earthquakes associated with faults (Subsection 2.5.3.3), determination of age of most recent movement on faults (Subsection 2.5.3.4), determination of structural relationships of site area faults to regional faults (Subsection 2.5.3.5), identification and description of capable faults (Subsection 2.5.3.6), and zones requiring detailed fault investigations (Subsection 2.5.3.7).

5 EVALUATION

5.1 Geology - Topic II-4

5.1.1 Regional Geology

The Yankee Rowe site is located in the Green Mountain Section of the New England Physiographic Province (Ferneman, 1938 and Thornbury, 1965). The Green Mountain section is a belt of highland ranging in elevation from +1000 to +3000 feet mean sea level (ft msl.). The belt is about 25 miles wide and more than 250 miles long. The New England Physiographic Province is a northern extension of the Appalachian Mountains that has been modified by glaciation.

The site is within the New England-Piedmont tectonic province. The New England part of the province, as considered here, coincides with the northern Appalachians as described by King (1969), Eardley (1973), Rodgers (1970) and Hadley and Devine (1974). The New England part is underlain by folded and faulted Precambrian to Mesozoic metamorphic, igneous and sedimentary rocks overlain by a few feet to several tens of feet of Pleistocene glacial deposits.

Although we accept the larger tectonic province, the New England-Piedmont Province in New England can be further subdivided based on geology into the Southeastern New England Platform, the White Mountain Plutonic Series, and the New England fold belt. The Southeastern New England Platform is separated from the rest of the New England-Piedmont Province in the site region by the Honey Hill-Lake Char thrust fault complexes. The boundary farther to the north and east is the Clinton-Newbury and Bloody Bluff thrust fault systems. It has been suggested (Rodgers 1972) that these generally northerly dipping thrust faults and associated rocks of high grade metamorphism represent a Paleozoic collision zone between a plate containing the Southeastern New England platform and a plate containing the New England fold belt.

The Southeastern New England Platform is composed of Precambrian granitic basement rocks, Silurian and Devonian volcanic and intrusive rocks, Cambro-Permian basins, an area of late Paleozoic intrusive and metamorphic rocks, and the zone of mid-Paleozoic, post-metamorphic thrust faulting represented in south-central and southeastern Connecticut by the Honey Hill-Lake Char fault zones. The Southeastern New England Platform has undergone relatively little structural deformation or metamorphic alteration since the Paleozoic (240 million years before present (mybp)). Known faulting is related to basin development during the Cambrian-Permian (570 mybp to 240 mybp). These basins include the Narragansett, Boston, North Scituate, Woonsocket, and Norfolk.

The White Mountain Plutonic series is an elongate, north-northwest oriented group of alkaline intrusives that extend from northeastern Massachusetts through New Hampshire. They were emplaced from Permian to Cretaceous. As a result of reviews of the Indian Point 3, Seabrook, Montague, Pilgrim 2, and New England sites, the staff concluded that there was a spatial relationship between the zone defined by these intrusives, which represent the youngest significant deformation features in New England, and historic seismicity. The largest New England earthquakes occurred within this zone.

The New England fold belt consists of major northeast southwest striking anticlinoria and synclinoria composed of metamorphic rocks and plutonic bodies. From the west in Vermont and western Massachusetts to the Atlantic Coast these major folds are: the Green Mountain - Sutton Mountain anticlinorium, the Connecticut Valley - Gaspe synclinorium, the Bronson Hill - Boundary Mountain anticlinorium, the Merrimack synclinorium, and the Coastal anticlinorium. The site lies on the Green Mountain - Sutton Mountain anticlinorium.

Several major geologic structures have been mapped in the site sub-region, including faults. These structures are described and evaluated with respect to the Yankee Rowe site in Topic II-4B, Section 5.2 of this report.

5.1.2 Site Geology

In addition to investigations relative to site validation and construction permit activities in 1956, the site was investigated in 1978 to obtain information for the Systematic Evaluation Program (SEP). These studies included core borings, geologic mapping, and seismic refraction surveys. The results of that exploration are reported in "Geology and Seismology Yankee Rowe Nuclear Power Plant," 19 January, 1979, by Weston Geophysical Corporation. The site information reported here is based on that report, several meetings with the utility and its consultant, a study of maps and aerial photographs, and two site reconaissances by NRC geologists and geotechnical engineers.

The site is located on dense till on the eastern bank of the Deerfield River at an elevation of 1,120 to 1,140 feet mean sea level (ft msl.). East of the plant the slope steepens to about 35° and increases in elevation to 1800 to 2000 ft msl.

Prior to construction, the site was underlain by up to 30 feet of sand and gravel glacial outwash. This material was removed from the site area. The plant is founded on a wedge of till which thickens from 0 east of the site to 80 feet beneath the containment and up to 200 feet south of the site. The till is a compact lodgement till consisting of gravel and cobbles in a silty sand to sandy silt matrix. In some places the till overlies a dense varved clay and silt lakebed deposit. The bedrock beneath the site is a crystalline gneiss of the Cambrian Hoosac formation. The surface of rock slopes from 17° to 32° to the west.

Structurally, the site is situated above the southeast flank of a northeast striking, south plunging anticline. Bedding dips uniformly to the southeast. The rock in outcrop around the site is relatively unfractured. Five small faults were mapped during the 1977-78 study. These faults are discussed in Section 5.2

Based on this review, geologic conditions at the site are such that the till on which the site is founded is adequate to support the plant under static loading conditions, and there are no geologic hazards in the vicinity to endanger the plant.

5.2 Capability of Faults in the Site Region - Topic II-4B

Recent regional seismic reflection studies in New England by COCORP show considerable horizontal westward transport of off-shelf sediments over relatively undeformed, lower Paleozoic miogeosynclinal rocks (Ando et al, 1982). These studies also indicate that the Precambrian basement rocks exposed in the Green Mountains of Vermont are allochthonous and underlain by shelf sediments or detachment horizons (Ando et al, 1982). There is no evidence, however, that these detachment surfaces, or thrust faults, if they exist, have been active since the Paleozoic Era (more than 240 mybp).

Several major faults or fault systems have been recognized in the site region. One of the most significant of these is the border fault bounding the eastern margin of the Connecticut Valley Triassic-Jurassic Basin located about 15 miles east-southeast of the site.

The classical structural interpretation of the Connecticut Valley Triassic-Jurassic Basin evolved from the work by Emerson, (1898) and as summarized by Rodgers, (1970) indicates that a major normal fault existed at the eastern contact between the Triassic sediments and the Pre-Triassic crystalline rocks to the east. The area of this border fault was investigated in detail during geologic studies in regard to the Montague site. These investigations suggested that a classical Triassic-Jurassic border fault didn't exist, but its presence could not be ruled out.

Three alternative hypotheses to the classical border fault concept were presented by the Montague applicants to describe the development of the basin:
(1) downwarping, (2) development of one or more high angle faults beneath the basin ("basin-forming faults"), or (3) erosional control of the basins formation.

No large displacement fault has been found to date within the Triassic-Jurassic Basin itself. The apparent absence of major faulting and the relatively undisturbed stratigraphic relationship would indicate that last movement on the inferred "basin-forming" fault(s) would have occurred prior to late Triassic-Jurassic time (more than 138 million years before present).

A major fault zone has been located approximately one mile east of the proposed Montague site in the general vicinity of the mapped location of the classical Triassic border fault. Some disagreement remains as to the extent, sense of movement, age, and the total amount of displacement on this fault. However, based on K-Ar radiometric dating of fault zone material, no evidence of movements younger than Jurassic (138 mybp - 136 mybp) has been found along the mapped fault zone. The applicant's studies indicate that this fault is a major thrust on which movement is within predominantly Paleozoic crystalline terrain. It is contended that the development of this fault accompanied compressional activity during the Paleozoic followed by an episode of normal faulting.

In spite of these investigations many experts maintain a strong view that this structure represents a fault with extensive normal displacement. It is clear, however, that the Montague applicants conducted an extensive investigation program to delineate and to date the time of last movement of all faulting in the vicinity of that site.

It has been suggested that there is a genetic relationship between the border fault and faults in the Paleozoic rocks north of the Connecticut Triassic-Jurassic Basin.

Rodgers (1970) in his section on the Connecticut Valley Synclinorium discusses the Ammonoosuc Fault as a postmetamorphic fault that can be traced by the offset of metamorphic isograds possibly thrusting lower grade rocks over older, higher grade rocks. This fault dips to the west at fairly low angles (averages 40 degrees). He also states: "It is also possible, however, that it is a normal fault, indeed, a large, late normal fault with similar attitude (although the dip is steeper) and similar silicification forms the eastern boundary of the Triassic basin in northern Massachusetts and can be traced into southern New Hampshire to within a few kilometers (miles) of the south end of the Ammonoosuc fault, and other late faults of the same kind are known in the intervening area and may interconnect them." The Yankee Rowe site lies about 20 miles west-southwest of projections of these faults at their closest approach.

In addition to the existence of a "border fault," several other faults are mapped in the Triassic-Jurassic Basin within five miles of the Montague site. These faults include the Falls River, Temple Woods, and other unnamed faults north of the general site area.

These faults are recognized because of offsets in the Deerfield diabase. A maximum offset of 800 feet for these faults occurs on the Falls River fault, the most prominent among them. The local faults are believed to be related to formation of the basin or fracture development in a flow basalt caused by compaction beneath it.

Evidence gathered during the Montague site investigations indicated that these faults are probably related to the last major tectonic activity associated with the development of the Connecticut Triassic-Jurassic Basin (240-138 million years before present). They are limited in extent and they are considered not capable.

Five major faults were mapped in the site vicinity by Weston geologists during investigations for the SEP program. Two of the faults show left-lateral displacement, and the other three appear to define normal offsets. None of the five faults displays fracturing, brecciation, or gouge indicative of significant fault movement (Weston Geophysical Corp., 1979).

The faults are described as follows (Weston Geophysical Corp., 1979).

(1) (East of Reservoir) The strike of this fault is N25°W, and the dip is about 80°NE, with slickensides which plunge 26° to the southeast, apparently reflecting left-lateral displacement. The single exposed fault surface is moderately curved and scalloped, with no associated fracturing in either wall, and cannot be traced laterally beneath the soil cover;

(2) (East of Reservoir) This feature which strikes N28°E, and dips 76°NW, may simply be a prominent joint. The single surface is not polished or slickensided, and no offset can be ascertained. There is no associated fracturing in adjacent walls, and the feature cannot be traced laterally.

(3) (East of Reservoir) The fault is defined by a quartz filling 1 to 2 inches thick. It strikes N10°E, and dips 60°E. Displacement appears to be normal, about 6 inches down on the east. The quartz filling is tight, welded to the enclosing walls. The fault cannot be traced on strike.

(4) (West of Reservoir) This fault strikes N50°E, and dips 75°SE. A quartz-feldspar pegmatite zone in thinly-layered gneiss appears to have about 12-15 inches of normal displacement, down on the southeast. The fracture surface is weathered, displays no slickensides, and cannot be traced on strike.

(5) (West of Reservoir) The fault strikes N30°W and dips 88°NE. A N65°W, 85°NE quartz vein is offset on the fault by 1 to 2 inches of apparant left-lateral displacement. Exposure is on a smooth bedrock surface in a stream bed, and the true sense of movement (whether left-lateral or normal) cannot be ascertained. The fault is tight with no brecciation or adjacent fracturing.

Based on regional relationships, these faults are interpreted to be no younger than Jurassic (138 mybp). Analysis of fractures of the site shows no anomalously-preferred orientation, nor any throughgoing zones of post-metamorphic faulting or shearing.

5. CONCLUSION

Based on studies performed by the utility, information in the published literature, and data obtained during the NRC-funded New Eng.and Seismotectonic Study, the staff concludes that there are no capable faults in the site vicinity.

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15. U.S. Atomic Energy Commission, 1974, Supplement No. 1 to the Safety Evaluation Report on the Matter of Public Service Company of New Hampshire Seabrook Station, Units 1 and 2; Docket Nos. 50-443 and 444, August 20, 1974.

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T Cheng

II-4
-4.A
-4.C

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

JUN 0 8 1981

LETTER TO ALL SEP OWNERS
(EXCEPT SAN ONOFRE)

Gentlemen:

SUBJECT: SITE SPECIFIC GROUND RESPONSE SPECTRA FOR SEP PLANTS LOCATED IN THE EASTERN UNITED STATES

Reference: Letter to SEP Group II Plant (Big Rock Point, Dresden 1, Haddam Neck, La Crosse, Yankee Rowe) Licensees from D.G. Eisenhut, NRC dated August 4, 1980

Our letter dated August 4, 1980 (reference) issued the preliminary version of site specific ground response spectra for the eastern United States SEP plants. Recently, these spectra have been finalized by the staff. Enclosure 1 includes the recommended ground response spectra (5% damping) for the eastern SEP sites. The bases of our final decision regarding the spectra and the digitized spectral acceleration values (5% damping) for these spectra are documented in Enclosure 2.

The site specific spectra (SSS) included in Enclosure 1 establish the ground motion acceleration values to be input into the structural reevaluation analyses to determine the resultant seismic loads. The geology reviews for Palisades, Ginna and Dresden 2 have been completed by the staff. The results of the review did not identify any geologic features that would affect the site specific spectra for those facilities. Based on our review to date for the remainder of the SEP facilities located in the eastern United States, we do not expect the SSS to be changed due to local geologic considerations.

Sincerely,

Dennis M. Crutchfield

Dennis M. Crutchfield, Chief
Operating Reactors Branch No. 5
Division of Licensing

Enclosure:
As stated

cc:
D. Eisenhut
J. Knight
G. Lainas
R. Jackson
G. Lear
W. Russell
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T. Cheng
P.Y. Chen

8110040494

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

SEP 0 2 1982

Docket No. 50-29
LS05-82-09-009

Mr. James A. Kay
Senior Engineer - Licensing
Yankee Atomic Electric Company
1671 Worcester Road
Framingham, Massachusetts C1701

Dear Mr. Kay:

SUBJECT: SEP TOPIC II-4.D, SLOPE STABILITY
YANKEE NUCLEAR POWER STATION

We have completed our review of the subject topic for Yankee Nuclear Power Station at Rowe, Massachusetts. Enclosed is a copy of our evaluation report for this topic.

You are requested to examine the facts upon which the staff has based its evaluation and respond either by confirming that the facts are correct, or by identifying errors and supplying the corrected information. We encourage you to supply any other material that might affect the staff's evaluation of this topic or be significant in the integrated assessment of your facility.

Your response is requested within 30 days of receipt of this letter. If no response is received within that time, we will assume that you have no comments or corrections.

Sincerely,

Ralph Caruso

Ralph Caruso, Project Manager
Operating Reactors Branch No. 5
Division of Licensing

Enclosure:
As stated

cc w/enclosure:
See next page

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Mr. James E. Tribble, President
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Chairman
Board of Selectmen
Town of Rowe
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U. S. Environmental Protection
Agency
Region I Office
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Resident Inspector
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Ronald C. Haynes, Regional Administrator
Nuclear Regulatory Commission, Region I
631 Park Avenue
King of Prussia, Pennsylvania 19406

Systematic Evaluation Program Topic Assessment

Topic: II-4.D - Stability of Slopes
Plant Name: Yankee Nuclear Power Station, Rowe, MA
Docket Number: 50-029
Prepared By: Dinesh Gupta, Geotechnical Engineer, HGEB

I. INTRODUCTION

This topic pertains to the Geotechnical Engineering Review of the stability of slopes, whose failure could adversely affect the safety of the plant. The scope of the review embraces the following subjects which are evaluated using data developed by the licensee and information available from all sources:

1. slope characteristics;
2. design criteria and analyses;
3. results of field and laboratory tests;
4. excavation, backfill, and earthwork in slopes;
5. liquefaction potential affecting slopes; and
6. proposed instrumentation and performance monitoring.

II. REVIEW CRITERIA

The applicable rules and basic acceptance criteria pertinent to the review of this topic are:

1. 10 CFR Part 50, Appendix A: General Design Criteria 1, 2 and 4

2. 10 CFR Part 100, Appendix A

3. Regulatory Guides

   (a) Regulatory Guide 1.132, "Site Investigations for Foundations of Nuclear Power Plants".

   (b) Regulatory Guide 1.138, "Laboratory Investigations of Soils for Engineering Analysis and Design of Nuclear Power Plants".

III. RELATED SAFETY TOPICS AND INTERFACES

1. SEP Topic II-4.F "Settlement of Structures and Buried Equipment"

2. SEP Topic II-4, "Geology and Seismology"

3. SEP Topic III-1, "Classification of Structures, Components and Systems"

4. SEP Topic III-6 "Seismic Design Considerations"

IV. REVIEW GUIDELINES

In general, the review process was conducted in accordance with the procedures described in Standard Review Plan Section 2.5.5. The geotechnical engineering aspects of the design and as-constructed condition of slopes were reviewed and compared to current procedures and criteria and the safety significance of any differences was evaluated.

Pertinent reference documents not cited in SRP Section 2.5.5 are included in part V "Topic Evaluation".

V. TOPIC EVALUATION

The Yankee Nuclear Power Station at Rowe, MA is located in a small valley adjacent to the Deerfield River, east of the pond formed by Sherman Dam. The Nuclear Power Station is bordered on the north, east and south by Berkshire Mountains and by Sherman Dam on the west. The natural slopes of these mountains rise to heights of about 1,000 feet above the site to either side and immediately behind it. This staff assessment deals with the static and seismic stability of those slopes.

1. Slope Characteristics

The licensee has selected two cross-sections of the slopes to the southwest and southeast of the plant to be representative of all the on-site slopes. The slopes begin to rise up at a distance of about 150 feet from plant structures, and rise from about elevation 1030 feet to 2000 feet. The licensee has stated that the slopes on the east side of the plant structures are likely to be better from a stability point of view than slopes on the south side because of the presence of a wooded knoll at the toe of the slope with frequent rock outcrops. The staff agrees with the licensee that the analyses of two selected cross-sections would therefore, lead to conservative results.

The licensee has also stated that the slope topography along the above two cross-sections was surveyed in 1981 by New England Power Company. The profiles of the two selected cross-sections of the slope were determined using this survey data. The licensee checked the accuracy of the profiles by supe imposing data from an enlarged USGS 7.5 minute quadrangle sheet for Rowe, Mass.-Vt., and found good correlation. We find this procedure to be acceptable for determining suitable profiles for stability analyses.

2. Results of Field and Laboratory Tests

To meet the current criteria, a comprehensive program of site investigations including borings, sampling, geophysical surveys, test pits, trenches, and laboratory and field testing is usually carried out to define the physical characteristics of all safety-related soil and rock slopes. Also, a summary and description of static and dynamic properties of the soil and rock comprising all slopes whose stability would directly or indirectly affect safety-related facilities should be provided. The text should include a complete discussion of procedures used to estimate, from the available field and laboratory data, conservative soil properties and profiles to be used in the analysis. This information is needed for the staff to ascertain that the program of field and laboratory tests has been adequate to define the in situ soil and rock characteristics to be used in slope stability evaluations.

The licensee has conducted seismic refraction surveys, excavated five 10 ft. deep test pits, and conducted laboratory tests on samples taken from these pits. We find that the scope and applicability of the results meets current licensing criteria, as explained in the following paragraphs.

The soil exploration related to the slopes at Yankee site consisted of seismic velocity measurements in 1978 by Weston Geophysical along seven lines using seismic refraction survey. These survey lines extend through part of the face of the slope at various locations. The results of the survey indicate that the surface of the bedrock is very irregular. Generally, the soil cover over the bedrock decreases at higher elevations and ranges from a few tens to about 200 feet in thickness along the slope. The shear wave velocity for soil cover ranges from 1500 to 2000 fps.

Results of five 10-feet deep field exploratory test pit-logs are available. These logs indicate the presence of lodgement till at shallow depths along the slope in the area of the test pits location. Lenses of both gravel and clay were observed in the till. The licensee, however, has concluded that these lenses are generally less than 50 feet in length and are rarely more than 100 ft long; so he ignored them in the stability analysis of the slopes. The staff finds this approach to be reasonable.

From triaxial test results on till samples obtained from the test pits, the licensee has assigned the following undrained shear strength parameters to the soils forming the slopes:

Depth	Angle of Internal Friction, Ø
0 to 30 feet	46°
30 to 90 feet	40°
Greater than 90 feet	35 °

The bedrock was assigned value of angle of internal friction of 70°. The groundwater along the slope was assumed at the surface.

Based on a review of the above information, the staff concludes that the scope of field and laboratory tests and the resulting values of shear strength parameters used in the analyses are reasonable and acceptable.

3. Design Criteria and Analyses

To meet current regulatory requirements, the discussion of design criteria and analyses is considered acceptable if

(a) Appropriate state-of-the-art methods have been employed.

(b) Conservative assumptions regarding soil and rock properties have been used in the analysis of slopes.

(c) Appropriately conservative margins of safety have been incorporated in the analysis.

To be acceptable, the static analyses should include calculations with different assumptions and methods of analysis to assess the following factors:

The uncertainties with regard to the shape of the slope, boundaries of the several types of soil within the slope and their properties, the forces acting on the slope, and pore pressures acting within the slope; failure surfaces corresponding to the lowest factor of safety; the effect of the assumptions inherent in the method of analysis used.

To be acceptable, the dynamic analyses must account for the effect of cyclic motion of the earthquake on soil strength properties. Actual test data are needed for in situ soils. The various parameters, such as geometry, soil strength, modeling method and hydrodynamic and pore pressure forces, should be varied to show that there is an adequate margin of safety. The results of stability analyses must be presented in tables identifying design cases analysed, strength assumptions for materials, and type of failure surface. Assumed failure surfaces should be graphically shown on cross sections and appropriately identified on both the tables and sections. The computer analyses should be explained and justified and an abstract of computer programs used should be provided. If the safety factors resulting from the analysis are not appropriate to the hazards posed by a slope failure and other than clearly conservative soil properties and profiles were used, additional data should be obtained to verify assumptions, or to show that, even if the worst possible conditions are assumed, there is an adequate margin of safety.

The licensee used a computer program, SSTAB 1 (August 1974) written by Prof. Wright of University of Texas, Austin, TX to assess the static and seismic stability of the slopes. The staff finds that the method used in this computer program utilizes state-of-the-art technique. For static stability, the licensee calculated minimum factors of safety for the two analyzed cross-sections to be 1.46 and 1.62, respectively; these are acceptable static results.

The licensee also analyzed the two selected cross-sections for seismic stability using a pseudo-static approach; this is acceptable. The results indicate that the seismic coefficients for the two cross-sections, which reduce the minimum factors of safety to 1.0, are 0.15g and 0.19g respectively. Since the peak ground acceleration, for which all components essential for equivalent safe shutdown of the plant must be analyzed, is 0.2g for Yankee site (see topic III-6, Seismic Design Considerations), the licensee's seismic analyses indicate slopes to be unstable.

In view of the above results from pseudostatic analysis of the stability of slopes, the licensee performed an analysis to evaluate the effect of potential slope failure on the plant safety. The two cross-sections previously selected for pseudo-static analysis were analyzed for predicting permanent displacements due to postulated seismic loads using Newmark's Method (Ref. 4). The results from the Newmark sliding block analysis show that the permanent displacements in the event of a slide due to seismic loads would be negligible. The licensee has concluded that these small displacements will not have any adverse impact on the plant safety. The staff finds this analysis procedure to be appropriate and the results to be reasonable.

VI. CONCLUSIONS

On the basis of the analyses performed and the available site data, the staff concludes that the stability of natural slopes at the Yankee Nuclear Power Station, Rowe, MA is adequate. Sufficient margins of safety exist against slope failure under postulated static loading conditions. For seismic loading, the predicted negligible permanent displacements of the slopes during earthquake should have no adverse impact on the plant safety systems.

VII. REFERENCES

1. Letter from J. A. Kay, Yankee Atomic Electric Company, to D. Crutchfield, NRC, Subject, "SEP Topic Assessment Completion (Topic II-4.D) February 26, 1982.

2. Letter from D. Crutchfield, NRC to J. A. Kay, Yankee Atomic Electric Company, Subject "Request for Additional Information for SEP Topic II-4.D, Stability of Slopes - Yankee Nuclear Power Station", March 31, 1982.

3. Letter from J. A. Kay, Yankee Atomic Electric Company to D. Crutchfield, NRC, Subject "Additional Information on SEP Topic II-4.D, Stability of Slopes", May 17, 1982.

4. "Effects of Earthquakes on Dams and Embankments", by N. M. Newmark, the Fifth Rankine Lecture, Geotechnique, 1965.

5. Undocketed, Information Package received from T. Cheng on August 19, 1982, containing discussion on (i) Information on Rock Knoll, (ii) Geologic Profile, (iii) Dynamic Stability of Natural Slopes, (iv) Slope Failure into Sherman Reservoir, and (v) Specification for Site Cleaning and Rough Grading for Yankee Atomic Electric Plant.

mBoyle

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

January 27, 1983

Docket No. 50-029
LS05-83-01-038

Mr. James A. Kay
Senior Engineer - Licensing
Yankee Atomic Electric Company
1671 Worcester Road
Framingham, Massachusetts 01701

Dear Mr. Kay:

SUBJECT: STATUS OF SEP TOPICS II-4.E, DAM INTEGRITY; III-7.B, DESIGN CODES, DESIGN CRITERIA, LOAD COMBINATIONS AND REACTOR CAVITY DESIGN CRITERIA; VI-1, ORGANIC MATERIALS AND POST-ACCIDENT CHEMISTRY; IX-6, FIRE PROTECTION; AND XV-2, SPECTRUM OF STEAM SYSTEM PIPING FAILURES INSIDE AND OUTSIDE OF CONTAINMENT (PWR)

The purpose of this letter is to identify those topics in the Systematic Evaluation Program for which the staff has concluded that the topic evaluation is complete. The differences from current licensing criteria, for the topics described below, will be evaluated in the integrated assessment for your facility.

Topic II-4.E is concerned with the integrity of Harriman Dam, which is owned by New England Power Company and licensed by the Federal Energy Regulatory Commission (FERC). FERC is presently conducting an evaluation of the integrity of this dam. Under an interagency agreement, the staff will use the results of the FERC evaluation to complete the integrated assessment. Also, the open items of this topic will be addressed in the integrated assessment of the related hydrology topics (Topics II-3.B, II-3.B.1 and II-3.C). Therefore, this topic is considered complete for the purpose of conducting the integrated assessment.

On September 1, 1982, the staff issued a draft SER for Topic III-7.B in which the staff identified areas where Yankee did not meet current acceptance criteria and where additional analysis was necessary. In the Yankee SEP Integrated Assessment, submitted on December 23, 1982, you did not identify any factual errors in the draft SER. However, you did supply additional information such that the staff could continue its review of the topic differences. The topic differences will be resolved during the integrated assessment. Therefore, the staff considers the topic SER to be final.

On September 9, 1982, the staff issued a draft SER for Topic VI-1, in which the staff identified differences from current licensing criteria. These differences were addressed in your Yankee SEP Integrated Assessment. Also in that document, plant modifications were proposed to resolve the differences. The need to implement those modifications will be addressed during the integrated assessment. Therefore, the staff considers the topic SER to be final.

Topic IX-6, "Fire Protection," is a topic being addressed generically by the Appendix R review and by Multi-Plant Action Item B-02. Therefore, since this topic is being reviewed outside of SEP, the staff considers Topic IX-6 complete.

On September 1, 1982, the staff issued a draft SER for Topic XV-2 in which the staff requested that you confirm the assumptions made in the staff's evaluation. In the Yankee SEP Integrated Assessment, you confirmed that the staff's assumptions were correct. Therefore, the staff considers this topic to be complete.

Sincerely,

Dennis M. Crutchfield

Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing

cc: See next page

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

M. Boyle

September 16, 1982

Docket No. 50-29
LS05-82-09-055

Mr. James A. Kay
Senior Engineer - Licensing
Yankee Atomic Electric Company
1671 Worcester Road
Framingham, Massachusetts 01701

Dear Mr. Kay:

SUBJECT: SEP TOPIC II-4.F, SETTLEMENT OF STRUCTURES AND BURIED EQUIPMENT - YANKEE NUCLEAR POWER STATION

Enclosed is our final evaluation of SEP Topic II-4.F, "Settlement of Structures and Buried Equipment." The evaluation is based upon a Safety Analysis Report which you supplied on August 31, 1981, additional information supplied on April 8, 1982, and other information available in Docket No. 50-29.

The evaluation concludes that settlement will not be a safety problem at Yankee. However, the staff requests that you investigate the liquifaction potential of submerged backfill and its potential effects, and the cause of cracking in the Spent Fuel Pool Building.

The evaluation will be a basic input to the Integrated Safety Assessment of Yankee. The need for any plant modifications will be evaluated in the Integrated Assessment. The evaluation may be revised in the future if the as-built conditions at Yankee are not accurately reflected in the evaluation or if NRC criteria relating to this topic are modified before the completion of the Integrated Assessment.

Sincerely,

Ralph Caruso

Ralph Caruso, Project Manager
Operating Reactors Branch No. 5
Division of Licensing

Enclosure:
As stated

cc w/enclosure:
See next page

~~8209240165~~

Mr. James A. Kay

Yankee
Docket No. 50-29
Revised 3/30/82

cc
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Chairman
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Systematic Evaluation Program (SEP) Topic Assessment
Topic: II-4.F - Settlement of Structures and Buried Equipment
Plant Name: Yankee Nuclear Power Station, Rowe, MA
Docket Number: 50-029
Prepared by: Dr. Dinesh C. Gupta, HGEB, GES

I. INTRODUCTION

This topic pertains to the review of plant Geotechnical Engineering aspects related to the properties and stability of subsurface materials and foundations as they influence the static and seismically induced settlement of Category I structures and buried equipment. The scope of the review includes:

(a) geologic features of the site;

(b) the static and dynamic engineering properties of soil and rock strata underlying the site;

(c) the results of field and laboratory tests, including data and discussions to support the established static and dynamic engineering properties, characteristics, and stratigraphy of soil and rock underlying the site;

(d) details of excavations, backfill, and earthwork illustrated on plot plans and profiles supported by laboratory testing and field compaction test results,

(e) groundwater conditions and piezometric pressures in all critical strata as they affect the loading and settlement and stability of foundation materials,

(f) liquefaction potential of all subsurface soils;

(g) results of static and dynamic analyses including bearing capacity, rebound, settlement, and differential settlement of supporting soil under loads, and

(h) results of confirmatory tests and performance monitoring of safety-related foundations and earthworks and buried equipment.

The information provided by the licensee is listed in Section VIII of this report. Reference 4, the Final Hazard Summary Report (in current terminology called PSAR) and the FSAR (reference 5) contain only brief narratives of the design and construction of foundations and buried equipment at Yankee Station. The licensee's safety assessment report (reference 1) and response (reference 3) to staff requests for additional information (reference 2) did not provide sufficient bases or detail to enable us to evaluate the settlement of foundations and buried equipment. The staff made a site visit and met with the licensee at the licensee's office in Framingham, MA during July 27-29, 1982. At this meeting, the licensee reiterated that most of the information requested by the staff (Reference 2) is not available. The staff gathered whatever additional information was available, including some drawings showing foundation details. Based on observations at the site and a review of all the information available to date, the staff has prepared the following topic evaluation.

II. REVIEW CRITERIA

The applicable rules and basic acceptance criteria pertinent to the review of this topic are:

1. 10 CFR Part 50, Appendix A:

   a. General Design Criterion 1 - "Quality Standards and Records." This criterion requires that 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. It also requires that appropriate records of the design, fabrication, erection, and testing of structure 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.

   b. General Design Criterion 2 - "Design Bases for Protection Against Natural Phenomena." This criterion requires that safety-related portions of the system shall be designed to withstand the effects of earthquakes, tornadoes, hurricanes, floods, tsunami and seiches without loss of capability to perform their safety functions.

   c. General Design Criterion 44 - "Cooling Water". This criterion requires that a system shall be provided with the safety function of transferring the combined heat load from structures, systems, and components important to safety to an ultimate heat sink under normal operating and accidental conditions.

2. 10 CFR Part 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants"-- These criteria describe the nature of the investigations required to obtain the geologic and seismic data necessary to determine site suitability and identify geologic and seismic factors required to be taken into account in the siting and design of nuclear power plants.

The following Regulatory Guides provide information, recommendations, and guidance and, in general, describe a basis acceptable to the staff that may be used to implement the requirements of the above described criteria.

(a) Regulatory Guide 1.127, "Inspection of Water Control Structures Associated with Nuclear Power Plants."
This guide describes a basis acceptable to the NRC staff for complying with the commission's regulation of 10 CFR Part 50 §50-36 with regard to developing an appropriate in-service inspection and surveillance program for dams, slopes, channels and other water control structures associated with emergency cooling water systems or flood protection of nuclear power plants.

(b) <u>Regulatory Guide 1.132, "Site Investigations for Foundations of Nuclear Power Plants."</u> This guide describes programs of site investigations related to geotechnical engineering aspects that would normally meet the needs for evaluating the safety of the site from the standpoint of the performance of foundation and earthworks under anticipated loading conditions including earthquake in complying with 10 CFR, Part 100 and 10 CFR, Part 100, Appendix A. It provides general guidance and recommendations for developing site-specific investigation programs as well as specific guidance for conducting subsurface investigations, the spacing and depth of borings, and sampling.

(c) <u>Regulatory Guide 1.138, "Laboratory Investigations of Soils for Engineering Analysis and Design of Nuclear Power Plants."</u> This guide describes laboratory investigations and testing practices acceptable for determining soil and rock properties and characteristics needed for engineering analysis and design for foundations and earthwork for nuclear power plants in complying with 10 CFR, Part 100 and 10 CFR Part 100, Appendix A.

III. RELATED SAFETY TOPICS AND INTERFACES

Geotechnical engineering aspects of slope stability are reviewed under Topic II-4.D. Other interface topics include:

II-3.B, "Flooding Potential and Protective Requirements";

II-3.C, "Safety-Related Water Supply (Ultimate Heat Sink)";

II-4.E, "Dam Integrity,"

III-3.A, "Effects of High Water Level on Structures;"

III-3.C, "In-Service Inspection of Water Control Structures;"

III-6, "Seismic Design Considerations;"

IX-3, "Station Service and Cooling Water Systems;" and

XVI, "Technical Specifications."

IV. REVIEW GUIDELINES

In general the review process was conducted in accordance with the procedures described in Standard Review Plan (NUREG-0800) Section 2.5.4. The Geotechnical Engineering aspects of the design, the design bases, and the as-constructed conditions of structures were reviewed and compared to current criteria, and the safety significance of any differences was evaluated. Our Topic Evaluation is provided below in accordance with the guidance provided in a memo from D. Eisenhut to H. Denton, dated April 2, 1982 (Ref. 15).

V. TOPIC EVALUATION

General Plant Description

The site is located on the eastern edge of the Deerfield River Valley in Rowe, Massachusetts. The present site grade ranges from about elevation 1128 feet MSL around the main plant structures to about 1140 feet MSL in the southern part of the site. The site is surrounded by the Berkshire Mountains, which rise to heights of about 1000 feet above the site grade on three sides of the plant site; Sherman reservoir is located on the northwest side of the plant site.

The foundations of safety-related structures, systems and components considered in our evaluation are:

(a) Main Plant Area

- Vapor container
- Primary auxiliary building
- Auxiliary bay portion of turbine building
- Spent fuel pool building
- Diesel generator building

(b) Separate Structures

- ECCS (Boron) Tank
- Fire-water tank

The licensee, in a meeting with the staff on July 28-29, 1982 indicated that the water intake structure on Sherman Reservoir is not safety related and that there are no safety-related buried pipes or equipment at the Yankee site; the SEP Branch staff requested the licensee to provide documentation of its position on the intake structure and buried piping and equipment. In view of licensee's verbal statement, we have not considered the settlement aspects of the intake structure or buried pipes and equipment in this evaluation.

The vapor container is a steel sphere, 125-feet in diameter, supported by sixteen 42-inch diameter steel columns. These columns are supported on individual concrete footings approximately 2-1/2 feet thick with their bases approximately 6 to 15 feet below grade level. The reactor vessel and reactor coolant system are housed in a concrete structure which is supported by two central columns and six outer columns. The central columns rest on an independent mat foundation and the outer columns are supported by a ring mat foundation. It appears that these mats are approximately 10 feet thick and that their bases are approximately 10 to 15 feet below grade level.

Details of the other main plant foundations could not be obtained. It appears that these foundations are 10 to 15 ft below the existing plant grade (around elevation 1110 feet, MSL).

The ECCS Tank is located on the order of 100 ft south of the vapor containment, and the Fire Water Tank is about 100 ft southeast of ECCS Tank. The details of foundations for the Fire Water Tank are provided in Reference 6.

Geologic Features

In meeting the requirement of the criteria, the presentation of geologic site data and discussion of site geologic features are acceptable if the maps, profiles, and discussion present a complete and inambiguous representation of the site geology. Exploratory techniques used in the site investigation are reviewed to determine if they are representative of the current state of the art and that samples extracted are representative of the in situ conditions. The areal extent of the investigations are reviewed to assure that all areas or zones of actual or potential surface or subsurface subsidence, uplift or collapse, deformation, alteration, solution cavities, structural weakness, unrelieved stresses in bedrock, or physically or chemically unstable soils or rocks have been identified and evaluated in detail.

The geologic features at the site were initially investigated by the licensee to obtain subsoil information at the Construction Permit stage in 1956 (Reference 6). Additional core borings, geologic mapping and seismic refraction surveys were performed in 1978 (Ref. 7). Based on a review of these investigations, the staff concludes that the extent and the type of exploratory techniques used for site investigation are acceptable. The results of the investigation are summarized as follows.

The licensee reported that about 30 ft of alluvial deposits was removed from the site prior to plant construction, and that the plant is founded on a wedge of lodgement till overlying bedrock. Bedrock is exposed on the east side of the site and the field investigations show that the till thickness increases to about 80 feet beneath the containment and to approximately 200 ft on the south side of the site.

In the southwestern part of the site, the lodgement till is underlain by an interbedded sequence of compact lacustrine deposit and very compact sand. The combined thickness of lodgement till, lacustrine deposit and sand in this area ranges up to about 80 feet.

The bedrock underneath the soil deposits is comprised of Lower Cambrian Gneiss, Schist, and dolomite marble, at a depth of about 100 ft below the plant foundation. The bedrock is hard, internally welded and the licensee has not detected any cavernous lithologies or throughgoing fault structures. Based on a review of these features, the staff agrees with the licensee that the site geologic features will not adversely affect the safe operation of the plant.

Field and Laboratory Tests

In meeting the requirements of the criteria, the discussion of the results of field and laboratory tests and the data and discussions to support the established static and dynamic engineering properties and stratigraphy underlying the site are acceptable if: (a) the site investigations and testing programs required to evaluate geotechnical engineering parameters related to site safety such as those described in Regulatory Guide 1.132 - "Site Investigations for Foundations of Nuclear Power Plants" and Regulatory Guide 1.138 - "Laboratory Investigations of Soils for Engineering Analysis and Design of Nuclear Power Plants" have been conducted and the results clearly reported; (b) the test parameters have been selected to conform to site conditions; (c) tests conducted are appropriate for the particular functions of facilities being evaluated; and (d) results among complementary tests are consistent.

The field exploration work at the site consisted of seismic refraction profiles obtained in 1956 (Reference 4), eight borings at the site in 1956 (Reference 4) and 1977 (Reference 7) and six borings performed in 1978 (Reference 10). The borings included standard penetration tests (SPT) conducted in accordance with ASTM D-1586. The licensee has submitted the boring logs and refraction profiles from these field investigations, but the staff could not find any additional field or laboratory test results that may have been obtained during these investigations.

The licensee provided five boring logs including SPT results obtained in 1979 (Reference 6) in the area of the Fire-Water Tank. Grain size distributions obtained from laboratory tests on the site soils were also provided for staff review. Also in 1979 (Reference 9) five 10 ft deep trenches were dug on the adjacent slopes. Triaxial tests were performed on soil samples from these test pits.

Based on a review of field and laboratory investigations including the SPT results, the staff agrees with the licensee that on site investigations have been adequate to conclude that, in general, the underlying lodgement till and bedrock are very stiff. However, the staff feels that the properties of the till probably vary and the extent of site investigation performed by the licensee may not have revealed local soil variations under individual structures.

Engineering Properties of Soil and Rock Strata

In meeting the requirements of the criteria, the discussion of the static and dynamic engineering properties of soil and rock strata underlying the site is acceptable if: (a) information provided is adequate to enable an independent evaluation of the static and seismically induced settlement characteristics of the foundation materials; and (b) assumptions made in assigning design soil parameters are reasonable, sufficiently explained, and conservative.

The applicant has obtained the engineering properties of the soil and rock strata from on-site shear wave velocity measurements, triaxial test results from surface samples obtained in test pits on the adjacent slopes, and field SPT results. We find that the scope and applicability of these results is adequate to define the general subsurface conditions at the site. The SPT data show that the lodgement till has consistently high SPT values of 35 blows/ft or more. The measured shear wave velocity of the till strata was 1,700 fps to 2,200 fps, and in the underlying bedrock, the shear wave velocity ranged from 6,200 fps to 8,200 fps.

The licensee also assigned the following undrained shear strength parameters to the lodgement tills and bedrock that support structures and forms the adjacent slopes:

Depth	Angle of Internal Friction, Ø
till at 0 to 30 feet	46°
till at 30 to 90 feet	40°
till at greater than 90 feet	35°
bedrock	70°

We find that the engineering properties assigned to the soils and bedrock samples on the basis of field and laboratory tests are reasonable and acceptable. However, as mentioned previously, there may be some variation in these properties. Also, we find that the SPT values observed in the six borings around the vapor containment building shows low values (from 1 to 20 blows/ft) in the upper backfill material. The staff concludes that these low values are indicative of loose material around the foundation and columns supporting the vapor container. The liquefaction potential of these loose soils is discussed later in the Evaluation Report.

Backfill and Earthwork

In meeting the requirements of the criteria, excavation backfill and earthwork elements of the projects are evaluated to assure that construction specifications and quality control procedures within state of the art conservative standards were applied and met. Results of field and laboratory investigations to establish properties of borrow materials are reviewed to determine their adequacy.

The licensee has submitted the 1957 specifications for the original site clearing and rough grading. These specifications required the backfilling within the project area to be done with suitable excavated material selected or approved for the purpose by the engineers. This was to be placed in uniform layers of not more than 12 inches of thickness. Each layer was to be compacted by at least ten passes of "heavy construction equipment".

Since no field or laboratory records of the backfill compaction are available, the staff cannot make a positive assessment of their adequacy. Based on a review of the SPT values recorded in the six 1979 borings (Ref. 7), we find that within the backfill around the foundation of the vapor container structure, the SPT values are very low and range from a low value of 1 to a maximum of 20 blows/ft. These low blow counts in this area indicates that the backfill is loose and possibly, the quality control procedures at the time of backfilling were inadequate. The staff could not find sufficient information to show that the backfill meets the current licensing criteria. Therefore, the staff cannot conclude that the backfill around the vapor container foundation is adequate.

Groundwater Conditions

In meeting the requirements of the criteria, groundwater conditions as they affect foundation stability are evaluated by analysis of piezometer and permeability data from tests and evaluations conducted at the site. Dewatering activities during and following construction are reviewed in conjunction with the impact of dewatering on soil properties.

Based on readings in a groundwater observation well installed in a boring near the fire water tank, the licensee found that the groundwater in that well varied from 3.6 feet to 4.3 feet below the ground surface between October 9 and October 23, 1979. This shows that water table was at approximately grade elevation (1139 feet MSL) at that location. The normal pool elevation of the adjoining Sherman Reservoir is 1105.66 ft (MSL). We find the licensee's description of ground water conditions to be reasonable and acceptable.

Liquefaction Potential

In meeting the requirements of the criteria, the liquefaction potential of subsurface materials is evaluated where safety-related structures are founded on potentially saturated soils. As detailed in SRP Section 2.5.4, Acceptance Criteria, Subsection 2.5.4.8, undisturbed samples obtained from the site may be required to show that the soils are not likely to liquefy.

The licensee has concluded that the lodgement till and underlying bedrock are not susceptible to liquefaction. The staff agrees with this assessment and finds it acceptable. However, the licensee has not addressed the liquefaction potential of the submerged backfill material that provides lateral support to the foundations and the buried columns that support the vapor container. The licensee should evaluate the liquefaction potential of these backfill materials, and the significance of the findings on the safety of the vapor container.

Static and Dynamic Analyses

In meeting the requirements of the criteria, the discussions of static and dynamic analyses are acceptable if the stability of all safety-related facilities has been analyzed taking into account bearing capacity, rebound, settlement, and differential settlements under: (a) dead loads of fills; (b) plant facilities; (c) lateral loading conditions; and (4) seismic loading. Soil and rock properties used in the analyses must be documented with field and laboratory test procedures and results. An assessment must be made of the dynamic volume change characteristics of foundation materials. The methods of analyses used must be appropriate for site-specific conditions and the function of the facility.

The licensee has indicated that all structures are supported on stiff lodgement till and therefore settlement of the safety related structures should not be a concern.

The shear wave velocity values that should be used in the dynamic analysis are given in the previous section on "Engineering Properties of Soil and Rock Strata." These are appropriate.

As indicated in previous sections, there may be a potential for liquefaction of the backfill soils surrounding the foundation of vapor container.

As mentioned in the following section, cracks in the Spent Fuel Pool Building walls may have been caused by differential settlements.

The staff was not able to find sufficient documentation of the licensee's static and dynamic analyses to meet current licensing criteria. However, other than the concerns mentioned above, we do not expect any problems with the static stability of structures founded on lodgement till. Dynamic stability is reviewed under SEP Topic III-6, "Seismic Design Considerations."

Performance Monitoring

In meeting the requirements of the criteria, the discussion of the results of confirmatory test and performance monitoring is acceptable if: (a) the purposes and locations of tests to confirm foundation and equipment settlement predictions are thoroughly detailed and explained; (b) the test methods used were appropriate for site conditions; (c) the overall instrumentation, purpose for each set of instruments, and reasons for their location are discussed and related to the types of data needed to confirm design assumptions and performance criteria; (d) the different kinds of instruments, special instruments and significant details for installation are discussed and are based on acceptable practices to assure reliability of measurements for the necessary time during or after construction; and (e) a program is described for periodic monitoring of instrumentation and inspection of foundation or settlement monument displacements, to assess both total displacements of singular foundations and the displacements of individual foundations with respect to adjacent facilities, to confirm design assumptions and to detect occurrences which could adversely affect operation of safety-related facilities.

The licensee submittals (References 1 and 3) indicate that the settlement was recorded for the vapor containment foundation for about one year after completion of construction. The observed settlement showed approximately 0.5 inch of total settlement during this period. The settlement monitoring was stopped after the first one year of construction completion. The results show that the settlements had attenuated, and therefore, we concur in licensee's assessment that settlement for vapor container is not a safety concern.

The licensee also mentioned in its topic assessment that there has been no evidence of any cracking caused by differential settlement in last 23 years. During our site visit, the staff, however, observed cracks in the walls of the spent fuel pool building, adjacent to the vapor containment structure. The licensee stated that those cracks are thermal cracks and are not due to differential settlement. The licensee further mentioned that they have a report on the cracks of this building. This report, dated February 25, 1977 (Reference 16) was later submitted for staff review. Based on this report, the licensee concluded that the cracks in the walls of the spent fuel pool building have occurred primarily from a combination of the hydrostatic head of water inside the fuel pit and the thermal gradient between the temperature of the water in the pit and the temperature on the external surface. The licensee further concluded that the cracking does not constitute an unsafe condition. Five cracks on the North, East and South walls of the building were patched by the licensee in late 1977 using epoxy. The licensee's report indicates that the 12 foot long crack on the north wall had been patched at least once before the patching work done in late 1977. During the staff site visit on July 1982, the cracks were again visibly open.

The staff has reviewed the licensee's report on the cracks in the walls of the spent fuel pool building. The licensee has not given adequate bases or substantiation for the reported conclusions. The staff does not find sufficient basis to conclude that the observed cracks are not a result of differential settlement of the building.

The applicant should further investigate the reasons for cracking in the walls of the Spent Fuel Pool Building in order to assure that recurrent cracking is not a result of differential settlements of foundations. As mentioned under the "General Plant Description" of this evaluation, the water intake structure is not considered a safety related structure by the licensee. However, during the site visit on July 27, 1982, cracks were observed in the walls of this structure also. If the safety classification of these structures is later changed, the licensee should investigate the reasons for cracking to assure that cracking is not a result of differential settlements of foundations.

VI. CONCLUSIONS

Based on a review of all the information submitted by the licensee and a visit to the plant site, the staff concludes that the licensee's assessment of the subject topic is generally acceptable to us except for the following items:

a) The licensee should investigate the liquefaction potential of submerged backfill material between the underlying lodgement till and the ground surface, and its potential effect on safety related structures.

b) The licensee should further investigate the reasons for the observed cracks in the walls of the Spent Fuel Pool Building in order to assure that the cracks are not caused by differential settlement of foundations and that these cracks do not pose any safety hazard.

VII. REFERENCES

1. Letter from J. A. Kay, Yankee Atomic Electric Company to D. Crutchfield, NRC, Subject "SEP Topic Assessment Completion (Topics XV-12, II-4.F, and IX-4)", August 31, 1981.

2. Letter from D. Crutchfield, NRC to J. A. Kay, Yankee Atomic Electric Company, Subject "Request for additional Information for SEP Topic II-4.F, Settlement of Foundations and Buried Equipment-Yankee", March 4, 1982.

3. Letter from J. A. Kay, Yankee Atomic Electric Company to D. Crutchfield, NRC, Subject, "Additional Information in Support of SEP Topic II-4.F, Settlement of Foundations and Buried Equipment", April 8, 1982.

4. Yankee Nuclear Power Station, "Final Hazard Summary Report".

5. Yankee Nuclear Power Station, "Final Safety Analysis Report", Vol. 1, January 8, 1974.

6. "Report on Foundation Investigation Fire Water Tank, Rowe Atomic Plant, Rowe, MA", Geotechnical Engineers, Inc. February 22, 1980.

7. "Geology and Seismology-Yankee Rowe Nuclear Power Plant", Weston Geophysical Corporation, January 29, 1979.

8. Test Pit Logs, 1978, R. G. Gerber.

9. "In-Situ Velocity Measurements, Yankee Nuclear Power Station, Rowe, MA", Weston Geophysical Corporation, March 30, 1979.

10. Boring Logs, 1979, Guild Drilling Company.

11. Letter from H. T. Evans, Stone & Webster Engineering Corporation, to R. J. Coe, Yankee Atomic Electric Company, Subject "Report by Hansen, Holley and Biggs-Design of Concrete Structure Supporting Reactor, Yankee Atomic Electric Plant", July 3, 1958.

12. Letter from J. J. Holley, Hansen Holley and Biggs, to H. T. Evans, Stone & Webster Engineering Corporation, Subject "Yankee Atomic Electric Company Power Plant at Rowe, MA", June 28, 1958.

13. Letter from W. F. Swiger to A. S. Lucks, Stone & Webster Engineering Corporation, Subject, "Foundation Studies - Yankee Rowe Atomic Power Plant", August 2, 1978.

14. Undocketed Information Package received from Tom Cheng on August 19, 1982 containing discussion on "(i) Information on the Rock Knoll (ii) Geologic Profile (iii) Dynamic Stability of Natural Slopes (iv) Slope Failure into Sherman Reservoir, and (v) Specification for Site Clearing and Rough Grading for Yankee Atomic Electric Plant.

15. Memorandum from D. G. Eisenhut to H. R. Denton, NRC, Subject, "SEP Topic Review and Integrated Assessment Schedule," April 2, 1982.

16. Letter from W. L. Klehm, Stone and Webster Engineering Corporation to R. P. Pizzuti, Yankee Atomic Electric Company, subject, "Survey of Fuel Transfer Pit, Yankee Atomic Electric Plant, Rowe, Massachusetts," February 25, 1977.