NRC RESPONSE TO THE SEP OWNERS GROUP SUGGESTIONS FOR DOCUMENTATION PROCEDURES FOR THE NRC SYSTEMATIC EVALUATION PROGRAM

The suggestions for documentation procedures presented by the SEP Owners Group at the May 31, 1978 meeting closely parallels the staff views on the subject. We further believe that because of the nature of the program, the magnitude, and duration, it is imperative that all parties clearly understand the method by which the reviews will be performed and the process by which results will be documented.

It is just as important to realize that the underlying principle of our regulations holds the licensee ultimately responsible for the safety of its facility. However, the SEP is unlike normal staff safety assessments in that the staff will be initiating unilateral and in some cases de nova reviews which are not requested by the licensee. This approach places further emphasis on the importance for establishing, with common understanding, sound rules by which such assessments and reviews will be documented.

Regarding the evolution of the SEP topics, at the onset of the Systematic Evaluation Program, a list totaling more than eight hundred safety topics was compiled from several lists of outstanding concerns. Members of the staff were asked to submit any safety issues that they thought were germane and should be covered in the program. The Systematic Evaluation Program group studied the list and reduced it to only those safety-related appropriate topics. Many topics were deleted because of duplication, some because of non-applicability to light water reactors and others on the basis of being research and development. Topics relating to work the staff is involved in for the purpose of refining its techniques were also deleted. Areas that the NRC is considering but not yet implementing on new facilities were culled from the list. Additional deletions were items periodically reviewed or which have already been reviewed and implemented. Topics which were similar or related were combined in the preparation of topic definitions. Topics culled for "lesser safety significance" were defined and a justification written. At the completion of the culling process the remaining topics were divided into two groups, generic and non-generic; the generic topics being denoted by an asterisk in the final list. The scheduling and review of generic topics will be performed essentially independent of the SEP. However, implementation of resolved generic issues will be integrated into the program to ensure appropriate consideration of the impact of any changes or modifications. Throughout the entire culling phase, the staff maintained a tracable record of the disposition of every topic and refers to this whenever items surface that may impact the program, to ensure that the status of any previously considered topic has not changed. As stated throughout the program new issues of safety significance may be added to the program as they are identified or they may be resolved on an expeditious basis apart from any SEP schedule. An item of this nature and its resolution would require management approval prior to its inclusion in the SEP.

The staff has completed the review of several topics, the assessments for which are enclosed as Attachment 1. It is important to note that the staff's definition of "complete" for this phase does not necessarily mean that the topic has been closed out, but only that the staff has come to a decision point in that either the topic is satisfactorily resolved or cannot be resolved until consideration in the Design Basis Event review or other related topics. Once significant deviations from criteria are identified, the topics will be evaluated to determine their impact on related design basis events. It is necessary to follow this approach since some topic definitions are written such that extensive review would be required to complete the evaluation independent of specific DBE evaluations. By reviewing each topic only to the extent necessary to evaluate its effect on the DBE, for a given plant the complete review and evaluation can occur as part of the DBE evaluations in a more efficient manner with more balanced decisions.

SEP staff members are assigned areas of review responsibility. Docketed and related background material will be carefully reviewed to obtain as much information about a specific area as possible with minimum impact on a licensee. It is expected that the review of a large number of topics on the NRC final topic list for each plant can be completed based on the presently available information on the docket. For this first category of topics, no information will be required from licensees. The NRC will send to the licensees its interim evaluation of each such topic as they are completed in order that the licensee can carefully examine the facts upon which the staff based its evaluation. The licensee should respond either that the bases (facts defining the plant) are correct, or are in error. If in error, correcting information should be supplied to the docket. The licensee is not required to agree with the NRC evaluation and is also encouraged to supply any other material for the docket on these topics as he may choose. Topics in Attachment 1 fall in this group.

For the second category of topics, the NRC may require information regarding the plants from the licensee. To expedite the review process, information required from the licensee will be obtained at working meetings or by conference calls. Information obtained by this process must, however at some later date, be placed on the official docket or formally supplied to the NRC in accordance with standard practice.

Analyses performed by the staff utilizing input from the licensee need only be reviewed and verified by the licensee for those facts describing the plant design unless for some reason the licensee chooses to use the staff's calculations for subsequent licensing justification. In that case, the licensee must be prepared to completely support the correctness of the staff information, bases, assumptions and calculations.

The third category of topics will be those that require licensee analyses. These will be treated in the same manner as those in routine licensing actions; licensee analyses will be placed on the official docket in accordance with standard practice. The staff will make every attempt to identify such topics to the licensees as early in the program as possible.

The content of staff assessments documenting completion of individual topics or areas of concern will be informally discussed with the licensees to ensure that the information used is factual and current and accurately protrays the facility. Initial assessments of individual safety topics or design basis events will be placed in the Public Document Room and forwarded to the licensee for comment. The initial assessments will be supplemented as needed to include correction or additional comments. At the completion of the program all initial assessments will be consolidated and a final assessment will be issued. NRC meeting minutes will typically be forwarded to the licensees for their review. Comments received will be placed in the Public Document Room.

ATTACHMENT 1

ASSESSMENTS OF ESSENTIALLY COMPLETE TOPICS

TOPIC III - 10C - Surveillance Requirements on BWR Recirculation Pumps and Discharge Valves

SEP Plants Affected - Millstone 1, Dresden 2

DBEs Affected - Loss-of-Coolant Accident

Discussion

This topic applies to the Low Pressure Coolant Injection System (LPCIS) at Boiling Water Reactors and specifically only to those systems which have undergone the LPCIS modification to remove the LPCIS loop selection logic. This logic network, which is still installed on two of the three applicable SEP Boiling Water Reactors (Millstone Unit No. 1 and Dresden Unit No. 2), is designed to direct LPCIS flow to the intact recirculation loop in the event of a Loss-of-Coolant Accident (LOCA). Oyster Creek has no LPCIS.

The logic network also was designed to close the suction and discharge valves of the intact loop to prevent LPCIS flow from bypassing the core and flowing out the break in the event of a LOCA. This modification was performed on all BWR-3 units (including Millstone Unit No. 1 and Dresden Unit No. 2) to allow closure of only the discharge valve. This is because in the unlikely event of a LOCA occurring between the suction and discharge valves of a recirculation loop with concurrent failure of the loop selection logic, rapid break isolation prior to sufficient reactor depressurization which would allow influx of low pressure, high volume cooling water could result in increased peak clad temperatures.

On BWR-4 facilities the loop selection logic has been disabled and LPCIS flow is now directed to both recirculation loops, with discharge valves on both loops directed to shut automatically. This topic is directed toward these facilities and concerns surveillance requirements for the discharge valves and recirculation pumps bypass valves.

Conclusion

This topic does not apply to Phase II SEP facilities.

TOPIC IV-1A - Operation with less than all loops in service

SEP Plants Affected - PWR's and BWR's

DBEs Affected - Loss-of-Coolant Accident

Discussion

The majority of the presently operating BWRs and PWRs are designed to operate with less than full reactor coolant flow. If a PWR reactor coolant pump or a BWR recirculation pump becomes inoperative, the flow provided by the remaining loops is sufficient for steady state operation at a power level less than full power.

Plants authorized for long term operation with one reactor coolant pump out of service have submitted, and the staff has approved, the necessary ECCS, steady state, and transient calculations. The remaining PWR and BWR licensees have Technical Specifications which require a reactor shutdown within a fairly short time if one of the operating loops becomes inoperable (with the exception of two which are discussed below).

SEP APPLICABILITY

The docketed material for the 11 systematic evaluation program plants has been reviewed with respect to operation with less than all loops in service. One licensee (Dresden 2) has requested authorization to operate with less than all loops in service, the staff is reviewing the analyses submitted with the request and approval will be granted when the staff approves the analysis. Five facilities (Yankee Rowe, Millstone 1, Ginna, Palisades, and San Onofre) are not authorized to operate with less than all loops in service, Technical Specifications restrict this mode to a period of 24 hours at which time the facility must have the idle loop restored to service or shutdown. Three facilities (Connecticut Yankee, Oyster Creek, and Dresden 1) have had an analysis reviewed and approved by the staff which authorizes N-1 loop operation. Two facilities LACBWR and Big Rock Point) have had authorization to operate in the N-1 loop mode since they were licensed, however there is no supporting ECCS analysis to justify operation.

Conclusion

This topic is complete for all the SEP facilities with the exception of LACBWR and Big Rock Point, for the latter two if continued authorization is to be permitted an analysis will have to be submitted which describes the thermal-hydraulic conditions of N-1 loop operation during ECCS, steady state, and transient conditions. Until such an analysis is performed and approved. Operation with less than all loops in service should be restricted to a 24 hour period at which time the plant should be shutdown unless the idle loop has been made operable.

TOPIC IV-3 - BWR Jet Pump Operating Indications

SEP Plants Affected - Millstone 1, Dresden 2

DBEs Affected - Loss-of-Coolant Accident

Discussion

The capability to reflood the core may be precluded in the event of a LOCA if all jet pumps are not operable. A jet pump instrument sensing line failure could result in inaccurate core flow measurements or the inability to detect a jet pump failure.

This topic applies only to Dresden Unit 2 and Millstone Unit 1; therefore, it should be removed from the review list for the nine remaining SEP plants.

The review of BWR Jet Pump operating indications has not begun for the two applicable facilities. The SEP staff cannot proceed any further until additional information is obtained from the licensee.[1] I&E and NRR are working closely to determine the adequacy of present jet pump operability technical specifications. If resolution cannot be made prior to the start of the Design Basis Events (DBE's) assessments the topic will be reviewed considering the potential effects on related DBEs.

---

[1] Dresden 2 has not replied to request for information.

TOPIC V-9 - Reactor Core Isolation Cooling System

SEP Plants Affected - None

DBEs Affected - None

Discussion

This topic applies to the RCIC system, a BWR system consisting of a steam-driven turbine/pump combination, piping, valves, and controls. RCIC was designed to inject water into the vessel in the case of vessel isolation upon loss of both on-site and off-site A-C power. In the General Electric Standard Safety Analysis Report (GESSAR), GE took credit for RCIC as a backup for the High Pressure Coolant Injection System in Loss-of-Coolant Accident (LOCA) analyses for certain small breaks. The NRC concern is that the RCIC system may not have been classified as a safety system, although credit was assumed in the safety analyses.

Conclusion

This topic does not apply to the SEP BWRs (Oyster Creek, Millsone Unit No. 1, Dresden Unit Nos. 1 and 2, La Crosse and Big Rock Point) since none of these facilities has an RCIC system.

TOPIC VI-7.A.2 - Upper Plenum Injection

SEP Plants Affected - Ginna

DBEs Affected - Loss-of-Coolant Accident

Discussion

On May 1, 1978, NRC issued Amendment No. 19 to operating license No. DPR-18 The staff Safety Evaluation Report which supported the license amendment addressed the upper plenum injection topic.

Ginna submitted ECCS performance analyses for the Westinghouse and new Exxon Nuclear Company (ENC) fuels. The Westinghuse analysis was performed for Cycle 7 fuel which the staff believes is a conservative evaluation for the Westinghouse fuel during Cycle 8. The ENC analysis was performed for Cycle 8 using the ENC WREM-II ECCS evaluation model. The ENC evaluation model has been reviewed and approved conditionally by the NRC.

The staff has recently considered whether the Westinghouse generic evaluation adequately represented the flow characteristics of Westinghouse two loop units. The generic evaluation model assumes that all safety injection water is introduced directly into the lower plenum. For the two loop units, the safety injection water is injected into the upper plenum. Thus, the staff was concerned that the Westinghosue model did not consider interaction between UPI water and steam flow. After plant specific submittals by licensees operating two loop plants were reviewed, the staff concluded that the calculations provided by the licensees (with certain modifications to the staff's model) are acceptable on an interim basis for continued safe operation of Westinghouse two loop plants, while long-term efforts continue for developing a model specifically treating UPI. For the Ginna plant the calculations which specifically considered UPI using the modified version of the staff model, resulted in a change of only 15°F from those using the generic model in which the UPI-core interaction was not specifically considered. In the interim, before these models are developed, Ginna has provided a modification to the current Westinghouse model which accounts for UPI-core interaction. It was demonstrated that the modification resulted in the increase of peak clad temperature by 15°F. Since for the Ginna plant both ENC WREM-II and Westinghouse models predict similar PCT's (1922°F for ENC WREM-II and 1957°F for Westinghouse) it can be expected that the UPI modification, when applied to the ENC WREM-II model, would allow about the same increase in PCT. The licensee has drawn a similar conclusion.

Conclusion

The staff has concluded that although the Westinghouse and Exxon two-loop generic-evaluation models should be changed to consider upper plenum injection (unless the plant is modified), analyses at the specific operating conditions applicable to the Ginna plant demonstrate that the effect of disregarding upper plenum injection interaction on refill and reflood conditions will not be significant (less than 20°F PCT). Therefore, the staff believes that, for the limited range to which the models do not deviate from the requirements of 10 CFR 50 Appendix K item 1.D.3, and the calculations are acceptable.

TOPIC VI-7D - Long Term Cooling Pressure Failures

SEP Plants Affected - All PWRs

DBEs Affected - Loss-of-Coolant Accidents

Discussion

This issue was raised by Mr. Ronald M. Fluegge in an October 24, 1976 letter to then Chairman Rowden. It was later defined in the Office of Nuclear Reactor Regulation as follows:

> "The General Design Criteria require that the Emergency Core Cooling Systems (ECCS) shall be capable of providing adequate core cooling following a Loss of Coolant Accident, assuming a single failure in Emergency Core Cooling Systems. The staff assumes the single failure to be either an active failure during the injection phase, or an active or passive failure during the long-term recirculation phase. The physical layouts of engineered safety feature pumps and components on some pressurized water reactors makes them vulnerable to flooding that might result from large passive failures in system piping, although they are protected for more likely events, such as sudden seal failure. Large pipe ruptures are not required to be protected against because of their low probability during the ECCS recirculation mode."

As stated in the "NRR Reports on Allegations Made by Mr. Ronald M. Fluegge" (11/76):

"The General Design Criteria (Appendix A to 10 CFR 50) include the following footnote regarding single failures:"

'single failures of passive components in electrical systems should be assumed in designing against a single failure. The conditions under which a single failure of a passive component in a fluid system should be considered in designing the system against a single failure are under development.'

"Thus, the General Design Criteria do not provide an explicit requirement for the treatment of failures of passive components. Appendix K to 10 CFR 50 pertains to ECCS performance requirements and also does not provide explicit guidelines on the treatment of failures of passive components after a loss-of-coolant accident (LOCA). Present plants are reviewed, however, to assure that the plant arrangement and design features provide the necessary protection of essential systems and components (such as shutdown cooling and pressurized portions of emergency core cooling systems) due to potential piping failures as an initiating event (not concurrent with or consecutive to a LOCA).

Piping failures outside containment are postulated in accordance with Branch Technical Positions MEB 3-1 and APCSB 3-1 in the USNRC Standard Review Plan, Section 3.6. Longitudinal or circumferential breaks in high energy fluid system piping or leakage-cracks in a moderate energy* fluid system piping are considered separately as a single postulated event occurring during normal plant conditions. The crack size assumed for a moderate energy pipe is equivalent to a slot of dimensions (1/2 x pipe thickness) x (1/2 x diameter). The plant must be designed such that the effects of such a postulated piping failure, including the environmental conditions resulting from the escape of container fluids, do not affect function of equipment essential to safe shutdown of the reactor.

With regard to postulation of failures in emergency core cooling systems subsequent to a loss-of-coolant accident, the USNRC Standard Review Plan on Emergency Core Cooling System (Section 6.3) provides additional guidance with the statement that: 'The ECCS should retain its capability to

*Subsequent to a LOCA, all pipes of relevance are moderate energy pipes defined as a piping system carrying fluid at a temperature below 200°F and at a pressure below 275 psig.

cool the core in the event of a failure of any single active or passive failure during the long-term recirculation cooling phase following an accident.' Based on this guidance, the staff assures the ECCS design and layout satisfies the requirement for redundancy in such systems. The implementation of the passive failure statement does not require significant ruptures of moderate-energy piping subsequent to LOCA, as this combined event would be extremely unlikely. The more credible passive failure is at pump or valve seals, or measurement devices. The staff review of the effects of such a postulated leak rate includes consideration of: (1) the flow paths of the radioactive fluid through floor drains, sump pump discharge piping, and the auxiliary building; (2) the operation of the auxiliary systems that would receive this radioactive fluid; (3) the ability of the leakage detection system to detect the passive failure; and (4) the ability of the operator to isolate the ECCS passive failure.

Therefore, the ECCS passive failure criterion being implemented by the staff requires the consideration of additional leakage but not pipe breaks beyond the initiating LOCA. The basis for this is the staff's judgment that the probability of serious multiple pipe failures is sufficiently low that they need not be considered a design basis event, since when operating in the long-term recirculation mode, the ECCS is subjected to temperatures and pressures much less than those for which the system is designed. In addition, after long-term cooling has been initiated, the need for recirculation diminishes due to the decrease in available core decay heat. For example, for a 3500 MWt reactor, the amount of core decay heat which is being produced at the beginning of a normal shutdown is 203 MWt; after one week it has decreased to 13 MWt; and after eight weeks it is only 5.7 MWt. This means that significantly less coolant recirculation would be necessary after several weeks. The needed cooling water to prevent core overheating can be provided by the RHR system even considering leakage in the suction or discharge side of the piping. In addition, should recirculation cooling be temporarily interrupted at the end of one week, the core would be adequately cooled by the heat transfer effected by vessel boiloff. To maintain vessel level, a makeup of only about 100 gpm would be necessary."

CONCLUSIONS

We consider this issue to be closed. The effect of ECCS leakage will be assessed on the SEP plants during the DBE evaluation of LOCAs.

TOPIC VII-1.B - Trip Uncertainty and Setpoint Analysis Review of Operating Data Base

SEP Plants Affected - All SEP Plants

DBEs Affected - All transients

Discussion

This issue was identified in September 1976 by the Electrical, Instrumentation, and Control System Branch of the Division of Systems Safety, Office of Nuclear Reactor Regulation. The issue was defined as follows:

> "Inclusion is needed in Technical Specifications of instrument errors in determining instrument trip setpoints in relation to allowable values of th- measured variable. Operating and under review LWRs are likely to have trip setpoints set at unsafe levels. The margin between trip setpoints and "allowable values" has not been reviewed. Standard Technical Specifications for BWRs for instrance do not even define "allowable values." Numerical values listed in the Standard Technical Specifications for trip setpoints and "allowable values" are identical."

Staff consideration of instrument errors in the evaluation and approval of trip setpoints for safety related instrumentation has been performed by either of two methods. Operating licenses issued on plants after the Spring of 1977 contain trip setpoints in their technical specifications whose values have been evaluated and approved based upon consideration of the individual factors used to assure an adequate margin of safety for each safety related channel. The information upon which our evaluations are made is contained in the detailed Regulatory Positions of Regulatory

Guide 1.105, Revision 1, "Instrument Setpoints," reissued in November 1976, and in the NRC Standard Review Plan.

Most operating licenses issued prior to this were evaluated in the more generalized manner. In this approach, the discrete components of each of the margins to safety in trip setpoint values are not evaluated on an individual basis but are included in an overall safety margin. Each set point value is based upon the most limiting transient or postulated accident condition associated with the bases for that set point. The magnitude of this safety margin and the resulting set points are established to ensure that there is a low probability of the margin being removed by an adverse combination of instrument calibration error, instrument error and instrument drift. The staff believes that this method is acceptable.

The staff has,however, changed from a generalized method of trip setpoint evaluation to a method that considers each of the discrete factors that make up the margins of safety for each safety related instrumentation channel. Either method contains conservatism; however, the newer method allows the safety margin in the trip setpoints to be quantified in a more detailed manner. In addition, consideration of instrument error is explicit in the newer method, whereas previously it was an implicit assumption presumed to be considered as part of the overall margin.

As new operating license reviews are completed, additional information will be included in FSARs relating to instrument drift and error because of the guidance now provided int eh NRC's Standard Review Plan and in Regulatory Guide 1.105. Accordingly, all Technical Specifications that are issued with new operating licenses after the Spring of 1977 will have the instrument drift allowance factored into the trip setpoint specifications. The staff is reviewing this more detailed information on instrument errors and draft to evaluate its impact, if any, upon the safety margins of the trip setpoints being used in older plants. Independent of the SEP, appropriate action will be taken to assure that the setpoitns in use retain an adequate degree of conservatism in maintaining safety margins as a result of this staff effort.

Conclusions

Adequate safety margins have been provided by the trip setpoints now in use for SEP plants, and this Topic does not warrant additional review apart from that for Topic XVI, Technical Specifications.

TOPIC XVII - Operational QA Program

SEP Plants Affected - All

DBEs Affected - All

Discussion

Since 1973 new guidance for operational quality assurance programs have been issued in the form of Regulatory Guides and WASH documents describing methods to comply with criteria of 10 CFR 50 Appendix B. The objective of this guidance is to assure that operation, maintenance, modifications and test activities do not degrade the capability of safety-related equipment to perform their intended function.

This topic has been completed for all SEP plants. Attached is a listing of the dates and specific reports containing the basis for their acceptance. Ten of the facilities were reviewed by the Quality Assurance Branch; the last (LACBWR) was reviewed by the Plant Systems Branch of DOR.

ATTACHMENT

DOCKET NO.	SEP PLANT	DOCUMENT
50-155	Big Rock Point	Topical Report Evaluation, 4/21/76
50-213	Connecticut Yankee	Letter, Switzer to Purple, 2/28/75
50-10	Dresden 1	Topical Report Evaluation, 4/78
50-237	Dresden 2	Topical Report Evaluation, 4/78
50-244	Ginna	Safety Evaluation Report, 9/30/74
50-409	LaCrosse	Memorandum, Eisenhut to Stello, 2/2/78
50-245	Millstone 1	Amendment 35 to SAR, 7/16/76
50-219	Oyster Creek	Safety Evaluation Report, 11/22/76
50-255	Palisades	Topical Report Evaluation, 4/21/76
50-206	San Onofre	Safety Evaluation Report, 4/8/75
50-29	Yankee Rowe	Topical Report Evaluation, 4/4/77

DBC-2192

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

August 3, 1978

Docket No. 50-29

Yankee Atomic Electric Company
ATTN: Mr. Robert H. Groce
Licensing Engineer
20 Turnpike Road
Westboro, Massachusetts 01581

Gentlemen:

This is in response to your application for license amendment dated June 6, 1978. You proposed changes to the Technical Specifications for the Yankee-Rowe reactor to permit ECCS recirculation system modifications for operation with Core No. 14.

We have reviewed your request and find that the additional information identified in the enclosure is required to continue our review. The items in the enclosure were discussed with representatives of your staff in a telephone conversation on July 27, 1978.

To maintain our review schedule your response is required by September 6, 1978. Please provide your schedule for submittal of this information.

Sincerely,

Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Enclosure:
Request for Additional Information

cc w/enclosure:
See next page

7822210260

cc w/enclosure:
Mr. Lawrence E. Minnick, President
Yankee Atomic Electric Company
20 Turnpike Road
Westboro, Massachusetts 01581

Greefield Community College
1 College Drive
Greenfield, Massachusetts 01301

REQUEST FOR ADDITIONAL INFORMATION
PROPOSED ECCS RECIRCULATION SYSTEM MODIFICATIONS
YANKEE-ROWE
DOCKET NO. 50-29

1. Provide additional information on the Low Head safety injection pumps and the High Head safety injection pumps as follows:

   a. Sectional Assemblies and parts lists with associated dimensions.

   b. The NPSH available and the NPSH required for each pump after completing this modification.

   c. Operating history of each pump defining the length of time each pump has run, any inspection results and previous repairs done.

   d. The water quality in a post LOCA environment under which these pumps will be operating.

   e. Documentation from pump manufacturers providing data on all qualification tests performed on these pumps and an indication that these pumps will be acceptable for the proposed service in the ECCS recirculation system.

2. Indicate to what extent Yankee-Rowe will comply with the requirements of 10 CFR 50.55a when performing pump and valve testing and inspection for the components in this system.

6/6/78 re mod. to ECCS recirculation Systems

50-29
"P"

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-29

YANKEE ATOMIC ELECTRIC COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 50 to Facility Operating License No. DPR-3, issued to Yankee Atomic Electric Company (the licensee), which revised the Technical Specifications for operation of the Yankee Nuclear Power Station (Yankee-Rowe) (the facility) located in Rowe, Franklin County, Massachusetts. The amendment is effective as of its date of issuance.

The amendment revises the Technical Specifications by eliminating specific pressurizer surveillance requirements and adding the requirements of 10 CFR 50.55a(g).

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated June 7, 1978, (2) Amendment No. 50 to License No. DPR-3, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C., and at the Greenfield Community College, 1 College Drive, Greenfield, Massachusetts 01301. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 31st day of July, 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

DR 50-29

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

July 31, 1978

Docket No. 50-29

Yankee Atomic Electric Company
ATTN: Mr. Robert H. Groce
Licensing Engineer
20 Turnpike Road
Westboro, Massachusetts 01581

Gentlemen:

The Commission has issued the enclosed Amendment No. 50 to Facility Operating License No. DPR-3 for the Yankee Nuclear Power Station (Yankee-Rowe). The amendment consists of changes to the Technical Specifications in response to your application dated June 7, 1978.

The amendment revises the Technical Specifications by eliminating specific pressurizer surveillance requirements and adding the requirements of 10 CFR 50.55a(g).

As discussed with your representative, you have agreed to retain the present wording in the basis to the Pressurizer Specification 3/4.4.4.

A copy of the related Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Enclosures:
1. Amendment No. 50 to DPR-3
2. Safety Evaluation
3. Notice of Issuance

cc w/enclosures:
See next page

7822102 63

cc
Mr. Donald G. Allen, President
Yankee Atomic Electric Company
20 Turnpike Road
Westboro, Massachusetts 01581

Greenfield Community College
1 College Drive
Greenfield, Massachusetts 01301

Chairman
Board of Selectmen
Town of Rowe
Rowe, Massachusetts 01367

Massachusetts Department of Public Health (w/filing dated 6/7/78)
ATTN: Commissioner of Public Health
600 Washington Street
Boston, Massachusetts 02111

Chief, Energy Systems Analyses Branch (AW-459)
Office of Radiation Programs
U. S. Environmental Protection Agency
Room 645, East Tower
401 M Street, S. W.
Washington, D. C. 20460

U. S. Environmental Protection Agency
Region I Office
ATTN: EIS COORDINATOR
JFK Federal Building
Boston, Massachusetts 02203

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

YANKEE ATOMIC ELECTRIC COMPANY

DOCKET NO. 50-29

YANKEE NUCLEAR POWER STATION (YANKEE-ROWE)

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 50
License No. DPR-3

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Yankee Atomic Electric Company (the licensee) dated June 7, 1978, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public;

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-3 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 50 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effeetive as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 31, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 50

FACILITY OPERATING LICENSE NO. DPR-3

DOCKET NO. 50-29

Revise Appendix A Technical Specifications by removing the following pages and inserting the enclosed pages. The revised pages contain the captioned amendment number and vertical lines indicating the area of change. Overleaf pages are included for document completeness.

REMOVE	INSERT
3/4 4-7	3/4 4-7
3/4 10-3	3/4 10-3*
6-21	6-21*

*These pages are included for the purposes of correcting clerical and administrative errors which occurred inadvertently during the issuance of Amendment No. 49, dated May 30, 1978.

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.4 The pressurizer shall be OPERABLE with a steam bubble.

APPLICABILITY: MODES 1 and 2

ACTION:

With the pressurizer inoperable, be in at least HOT STANDBY with the reactor trip breakers open within 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.4 No additional requirements other than those specified in accordance with 1C CFR 50.55a(g).

3/4.4.5 MAIN COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.5.1 The following Main Coolant System leakage detection systems shall be OPERABLE:

a. The containment atmosphere particulate radioactivity monitoring system,

b. The containment drain tank level monitoring system.

c. The incore detection system thimble leak alarm system.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the above required radioactivity monitoring leakage detection system inoperable, operation may continue for up to 7 days provided:

 1. Main Coolant System water inventory balance is performed at least once per 24 hours.

 2. The other above required leakage detection systems are OPERABLE and

 3. Appropriate grab samples are obtained and analyzed at least once per hour:

 otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the containment drain tank level monitoring system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With the incore detection system thimble leak alarm system inoperable, restore the leak alarm system to OPERABLE status within 7 days or close all thimble isolation valves; restore the leak alarm system to OPERABLE status within 31 days or be in at least HOT STANDBY within the the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SPECIAL TEST EXCEPTIONS

PRESSURE/TEMPERATURE LIMITATION - REACTOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.10.3 The minimum temperature and pressure conditions for reactor criticality of Specification 3.4.8.1 may be suspended during low temperature PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 2 percent of RATED THERMAL POWER,

b. The reactor low setpoints trips on the three OPERABLE Power Range Nuclear Channels are set at $\leq$25% of RATED THERMAL POWER, and

c. The Main Coolant System temperature and pressure are maintained $\geq$ 250°F and $\geq$ 300 psig, respectively.

APPLICABILITY: MODE 2.

ACTION:

a. With the THERMAL POWER > 2 percent of RATED THERMAL POWER, immediately open the reactor trip breakers.

b. With the Main Coolant System temperature and pressure < 250°F or < 300 psig, immediately open the reactor trip breakers and restore the temperature-pressure to within its limit within 30 minutes; perform the analysis required by Specification 3.4.8.1 prior to the next reactor criticality.

SURVEILLANCE REQUIREMENTS

4.10.3.1 The Main Coolant System temperature and pressure shall be verified to be $\geq$ 250°F and $\geq$ 300 psig at least once per hour.

4.10.3.2 The THERMAL POWER shall be detemined to be $\leq$ 2% of RATED THERMAL POWER at least once per hour.

4.10.3.3 Each Power Range Nuclear Channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating low temperature PHYSICS TESTS.

LIMITING CONDITION FOR OPERATION

3.10.4 The limitations of Specification 3.1.1.4, 3.1.3.1, 3.1.3.4, and 3.1.3.5, may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 2% of RATED THERMAL POWER, and

b. The reactor low setpoint trips on the three OPERABLE Power Range Nuclear Channels are set at $\leq$ 25% of RATED THERMAL POWER.

APPLICABILITY: MODE 2.

ACTION:

With the THERMAL POWER > 2% of RATED THERMAL POWER, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined to be $\leq$ 2% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.4.2 Each Power Range Nuclear Channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating PHYSICS TESTS.

(d) Total dissolved gas radioactivity (in curies) and average concentration released to the unrestricted area.

(e) Total volume (in liters) of liquid waste released.

(f) Total volume (in liters) of dilution water used prior to release from the restricted area.

(g) Total gross radioactivity (in curies) by nuclide released based on representative isotopic analyses performed.

(h) Percent of Technical Specification limit for total radioactivity.

(3) Solid Wastes

(a) The total amount of solid waste shipped (in cubic feet).

(b) The total estimated radioactivity (in curies) involved.

(c) Disposition including date and destination.

6.9.6 Special reports shall be submitted to the Director of the Office of Inspection and Enforcement Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. Inservice Inspection Program Reviews, Specification 4.4.9.1.

b. ECCS Actuation, Specifications 3.5.2 and 3.5.3.

c. Inoperable Meteorological Monitoring Instrumentation, Specification 3.3.3.3.

d. Sealed Source leakage in excess of limits, Specification 4.7.6.3.

e. Radioactive Solid Waste Disposal, Specification 3.7.7.1.

f. Fire Detection Instrumentation, Specification 3.3.3.4.

g. Fire Suppression Systems, Specifications 3.7.10.1, 3.7.10.2 and 3.7.10.3.

h. Environmental Monitoring Program, Specifications 3.7.8.

6.10 RECORD RETENTION

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of facility operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspection, repair and replacement of principla items of equipment related to nuclear safety.

c. All REPORTABLE OCCURRENCE reports submitted to the COMMISSION.

d. Records of surviellance activities, inspections and calibrations required by these Technical Specifications.

e. Records of reactor tests and experiments.

f. Records of changes made to Operating Procedures.

g. Records of radioactive shipments.

h. Records of sealed source leak tests and results.

i. Records of annual physical inventory of all sealed source material of record.

6.10.2 The following records shall be retained for the duration of the Facility Operating License:

a. Records and drawing changes reflecting facility design modifications made to systems and equipment described in the Final Hazards Summary Report.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of facility radiation and contamination surveys.

d. Records of radiation exposure for all individuals entering radiation control areas.

e. Records of gaseous and liquid radioactive material released to the environs.

f. Records of transient or operational cycles for those facility components identified in Table 5.7-1.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 50

FACILITY OPERATING LICENSE NO. DPR-3

YANKEE ATOMIC ELECTRIC COMPANY

YANKEE NUCLEAR POWER STATION (YANKEE-ROWE)

DOCKET NO. 50-29

Introduction

By application dated June 7, 1978, Yankee Atomic Electric Company (the licensee) requested an amendment to Facility Operating License No. DPR-3 for the Yankee Nuclear Power Station (Yankee-Rowe). The amendment would revise technical specification 4.4.4 to eliminate specific pressurizer surveillance and add the requirements of 10 CFR 50.55a(g).

Discussion and Evaluation

The present limiting conditions for operations (LCO's) in the Technical Specifications require the pressurizer to be operable with a steam bubble. To assure that this LCO is met, the Technical Specifications presently include surveillance requirements to verify every 18 months: (1) satisfactory performance of the pressurizer functions specified in Sections 4.4.4.a.1 and 4.4.4.a.2 and (2) that the pressurizer relief valve opens and closes at the pressure setpoint settings specified in 4.4.4.b. In order to achieve consistency with currently accepted pressurizer surveillance practice, the licensee proposed to substitute surveillance requirements in accordance with 10 CFR 50.55a(g) for the existing pressurizer surveillance requirements in Section 4.4.4 of the Technical Specifications.

Achievement of a steam bubble in the pressurizer is a prerequisite condition for plant startup and power operation (Modes 1 and 2). The present requirements for the pressurizer to be operable with a steam bubble would be satisfied by testing and inspections of the pressurizer and associated relief and code safety valves in accordance with the provisions in 10 CFR 50.55a(g). Such surveillance is presently required for Yankee-Rowe and provides an acceptable degree of confidence that the integrity of the pressurizer pressure boundaries will be maintained. Thus, the detailed surveillance of pressurizer functions as presently required by Sections 4.4.4.a.1 and 4.4.4.a.2

is not necessary and deletion of these provisions is acceptable. Furthermore, this proposed change is consistent with the Westinghouse Pressurized Water Reactor Standard Technical Specifications (W-STS) which are applicable to Yankee-Rowe.

The reactor coolant system, including the pressurizer, is protected against overpressurization by two pressurizer code safety valves. Each safety valve has sufficient capacity to relieve any potential overpressure condition during normal operation and reactor shutdown. The combined relief capacity of both safety valves is greater than the maximum surge rate from an assumed loss of load, with no credit for a reactor trip on loss of load, and assuming that the pressurizer relief valve or the steam dump valves do not perform their intended functions. Thus, the pressurizer relief valve does not perform a protective function. The purpose is to minimize undesirable opening of the code safety valves by relieving pressure surges below the lift setting of the code safety valves. No credit is taken in the safety analysis for operation of the pressurizer relief valve. Therefore, deletion of the surveillance requirement (4.4.4.b) for this valve is acceptable. This proposed deletion is also consistent with the W-STS which do not require the pressurizer relief valve to be operable or verification of the pressure setpoint settings of such valves. However, the W-STS do require the code safety valve to be inspected in accordance with Section XI of the Boiler and Pressure Vessel Code as required by 10 CFR 50.55a(g), including verification of its lift settings. Such surveillance is also required for the Yankee-Rowe code safety valves.

Based on our review of the licensee's June 7, 1978 application, as discussed above, we concluded that substitution of pressurizer surveillance provisions in Technical Specifications 4.4.4 as proposed does not decrease the level of safety of the facility, and is acceptable.

Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: July 31, 1978

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-29

YANKEE ATOMIC ELECTRIC COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 50 to Facility Operating License No. DPR-3, issued to Yankee Atomic Electric Company (the licensee), which revised the Technical Specifications for operation of the Yankee Nuclear Power Station (Yankee-Rowe) (the facility) located in Rowe, Franklin County, Massachusetts. The amendment is effective as of its date of issuance.

The amendment revises the Technical Specifications by eliminating specific pressurizer surveillance requirements and adding the requirements of 10 CFR 50.55a(g).

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated June 7, 1978, (2) Amendment No. 50 to License No. DPR-3, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C., and at the Greenfield Community College, 1 College Drive, Greenfield, Massachusetts 01301. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 31st day of July, 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

50-29
P

Telephone 617 366-9011

TWX
710-390-0739

WYR 78-70

YANKEE ATOMIC ELECTRIC COMPANY

YANKEE

20 Turnpike Road Westborough, Massachusetts 01581

July 28, 1978

United States Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Office of Nuclear Reactor Regulation
Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Reference: (1) License No. DPR-3 (Docket No. 50-29)
(2) USNRC Letter to YAEC dated June 26, 1978

Dear Sir:

Subject: Reference Drawings for SEP Reviews

In accordance with your request, Reference (2), we are forwarding one set of aperature cards of drawings for safety related systems and structures for the Yankee Rowe Plant. Also included is an index sheet for the drawings contained in this submittal. It is our belief that all of these drawings have been previously docketed either as part of the original FHSR or subsequent licensing submittals. If we can be of further assistance please don't hesitate to contact us.

Very truly yours,

YANKEE ATOMIC ELECTRIC COMPANY

Robert H. Groce
Robert H. Groce
Licensing Engineer

JKT/kg
Enclosure

7821600 75

A001 S 1/1 CARDS TO B.C.

~~9699~~	~~FM-1A~~	~~8MACH LOC PLANS VAPOR CONTNR SH 1~~	~~29X~~
9699	FM-1B	8MACH LOC PLANS VAPOR CONTNR SH 2	29X
9699	FM-1C	8MACH LOC SECT VAPOR CONTAINER	29X
~~9699~~	~~FM-1Y~~	~~1VAPOR CONT PRT SYS ELEV~~	~~29X~~
96990000FM-2A		10FUNDAMENTAL FLOW DIAG	
9699	FM-3A	18MAIN+AUX STM+STM GEN BD	
~~9699~~	~~FM-4A~~	~~FLOW DIA COND+ FDWTR LINES~~	
9699	FM-6A	22MAIN COOL PRESS BLW-DRN,VENT LIN	
9699	FM-7A	14EXTR STM FW HTR VNT+DRN LNS	
~~9699~~	~~FM-8A~~	~~17FL-CHRG+BLEED+CHEM SHTDN~~	
9699	FM-8B	19FL DIA-PRIM PLNT PURIFCTN	
	YM-C-7	4CHEM INJ + LOW SRG TANK	
~~9699~~	~~FM-9A~~	~~21MAIN COOL COMP SHIELD FUEL PIT C~~	
9699	FM-18A	7MACH LOC PLN GR FL TUR RM	
9699	FM-18B	12MACH LOC PLN GR FL TUR RM	
~~9699~~	~~FM-18C~~	~~9MACH LOCA-SECT TURBINE RM~~	
9699	FM-18D	8MACH LOC PLN MEZZ FL TURB RM	
96 00000FM-21A		13FUEL TRANSFER PIT	
~~9699~~	~~FM-25A~~	~~8MAIN COOLANT SYSTEM SH 1~~	~~29X~~
9699	FM-25B	3MAIN COOLANT SYSTEM SH 2	29X
9699	FM-25C	5MAIN COOLANT SYSTEM SH 3	29X
~~9699~~	~~FM-27B~~	~~2ED SERV WTR LINES PRIM PL 1~~	
9699	FM-27C	2ED SERV WTR LINES SEC PLT 2	
9699	FM-57A	7MACH LOC PLN PRIM AUX BLDG SH 1	29X
~~9699~~	~~FM-57B~~	~~7 MACH LOC PRI AUX BLDG SH2~~	
9699-	FM-61A	4 MACH LOC DIESEL GEN BLDG	
9699	FM-83A	9LOW PRES SFTY INJ HI PRES SFTY I	
~~9699~~	~~FE-1C~~	~~11 2400V ONELINE DIAG~~	~~29X~~
9699	FE-1D	20 480V ONE LINE DIAGRAM 1	
9699	FE-1G	16 480V ONE LINE DIAG SH 3	
~~9699~~	~~FE-1F~~	~~15 480V ONE LINE DIAGRAM 2~~	
9699	FE-1H	13 125V DC ONE LINE DIAG	
9699	FE-1J	8 480V ONE LINE DIAGRAM 4	
~~96 -~~	~~FSK-1A~~	~~3LIST OF SWITCHES + LEGEND~~	
11986	FSK-3A	1CNTL SWITCH CONTACT DIAGRAMS	16X
9699-	FSK-4A	4OUTLINE SAFETY INJ CONT PNL	
~~9699-~~	~~FSK-4AA~~	~~2SAFETY INJECTION CNTL PNL 1 4~~	
9699-	FSK-4AA	3SAFETY INJECTION CNTL PNL 2 4	
11986	FSK-4AA	2SAFETY INJECTION CNTL PNL SH NO3	16X
~~9699-~~	~~FSK-4AA~~	~~3SAFETY INJECTION CNTL PNL 4 4~~	
9699	FSK-4B	3SAFETY INJ CONTRL PNL SH2	
11986	FSK-6A	5ELEM DIAG 480V SWGR EMER GEN	16X
~~11986~~	~~FSK-6B~~	~~4ELEM DIAG 480V SWGR BUS TIES~~	~~16X~~
11986	FSK-6C	3ELEM DIAG 480V SWGR SFTY INJ PP	16X
96990000FSK-6AA		5 ELEM DIAG SAFETY INJ. VALVES SH1	
~~9699~~	~~FSK-6AB~~	~~4ED SAFETY INJ VALVES 2~~	
9699	FSK-6AC	4 480V MC CIRCUITS	
9699-	FSK-6AD	0ELEM DIA ACCUM ISO VL MOV1	
~~11986~~	~~FSK-8A~~	~~4AC ELEM DIAG EMER GENERATORS~~	~~16X~~
9699	FSK-8B	5DC ELEM DIAG EMER GEN	
9699	FSK-10A	4 B/D SAF INJ CNTL PNL ANN 1 2	
~~9699~~	~~FSK-10B~~	~~1SAFE INJ CON PNL ANNUN 2 2~~	
9699-	FSK-11A	8EL DIA AUTO STARNG RESTART	

50-29
"P"

Telephone 617 366-9011
TWX
710-390-0739

WYR 78-68

YANKEE ATOMIC ELECTRIC COMPANY

YANKEE
20 Turnpike Road Westborough, Massachusetts 01581

July 27, 1978

United States Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Office of Nuclear Reactor Regulation
Victor J. Stello, Jr., Director
Division of Operating Reactors

Reference: (a) License No. DPR-3 (Docket No. 50-29)
(b) YAEC Submittal Letter WYC-77-3, December 12, 1977

Dear Sir:

Subject: Yankee Topical Power Spike Model

The purpose of this letter is twofold. (1) To forward additional information which was requested by your staff concerning the percent contributions to power peaking from an infinite gap as calculated by PDQ and DOT, and (2) document the input parameters to the Yankee Power Spike Model (Reference b) which make it applicable to Yankee Rowe Core XIV. This data is enclosed on Attachments A and B, respectively.

We trust this information will be satisfactory to you; however, should you have any questions relative to this matter, please contact us.

Very truly yours,

YANKEE ATOMIC ELECTRIC COMPANY

W B Johnson
W. B. Johnson
Vice President

PTA/kg

7821 20045

A001
s
3/40

ATTACHMENT A

Figure 1 shows the PDQ and DOT calculated power distribution of a section of an assembly.

Figure 2 and 3 show the PDQ and DOT calculated percent contributions to the power peaking from an infinite gap.

FIGURE 1

Power Distribution

1.085 1.077	1.014 1.014	.975 .978	.973 .975	— PDQ — DOT
0.0 0.0	1.086 1.079	.971 .972	.938 .942	
0.0 0.0	1.084 1.077	.963 .965	.926 .931	
1.089 1.089	1.011 1.010	.948 .951	.923 .928	

FIGURE 2

Percent Contribution to Power Peaking from Infinite Gap

.09 0.0	.89 .59	2.87 2.04	4.73 3.38	— PDQ — DOT
Water Hole	1.19 .65	4.74 3.40	Gap	
Water Hole	.92 .46	3.01 2.07	5.18 3.54	
0.0 0.0	.40 0.0	1.06 .63	1.52 .86	

(Attachment A, cont'd)

FIGURE 3

Percent Contribution to Power Peaking from Infinite Gap

3.41 2.59	5.72 [illegible].2[illegible]	3.90 3.07	1.38 1.13	— PDQ — DOT
Water Hole	Gap	6.39 4.84	2.03 1.49	
Water Hole	5.72 4.36	4.05 3.01	1.54 1.18	
1.36 .83	1.88 1.29	1.52 1.05	.67 .43	

The PDQ percent contributions to power peaking from an infinite gap are presently used in the Yankee Power Spike Model thus, making the model conservative.

ATTACHMENT B

The Yankee Power Spike Model has been used to calculate the power spike for Yankee Rowe Core XIV using the PDQ calculated percent contributions to power peaking of Figure 4 and the relative effect of a finite size gap versus gap size of Figure 5. The results for Yankee Rowe Core XIV are shown in Table 1.

FIGURE 4

Yankee Rowe Core XIV

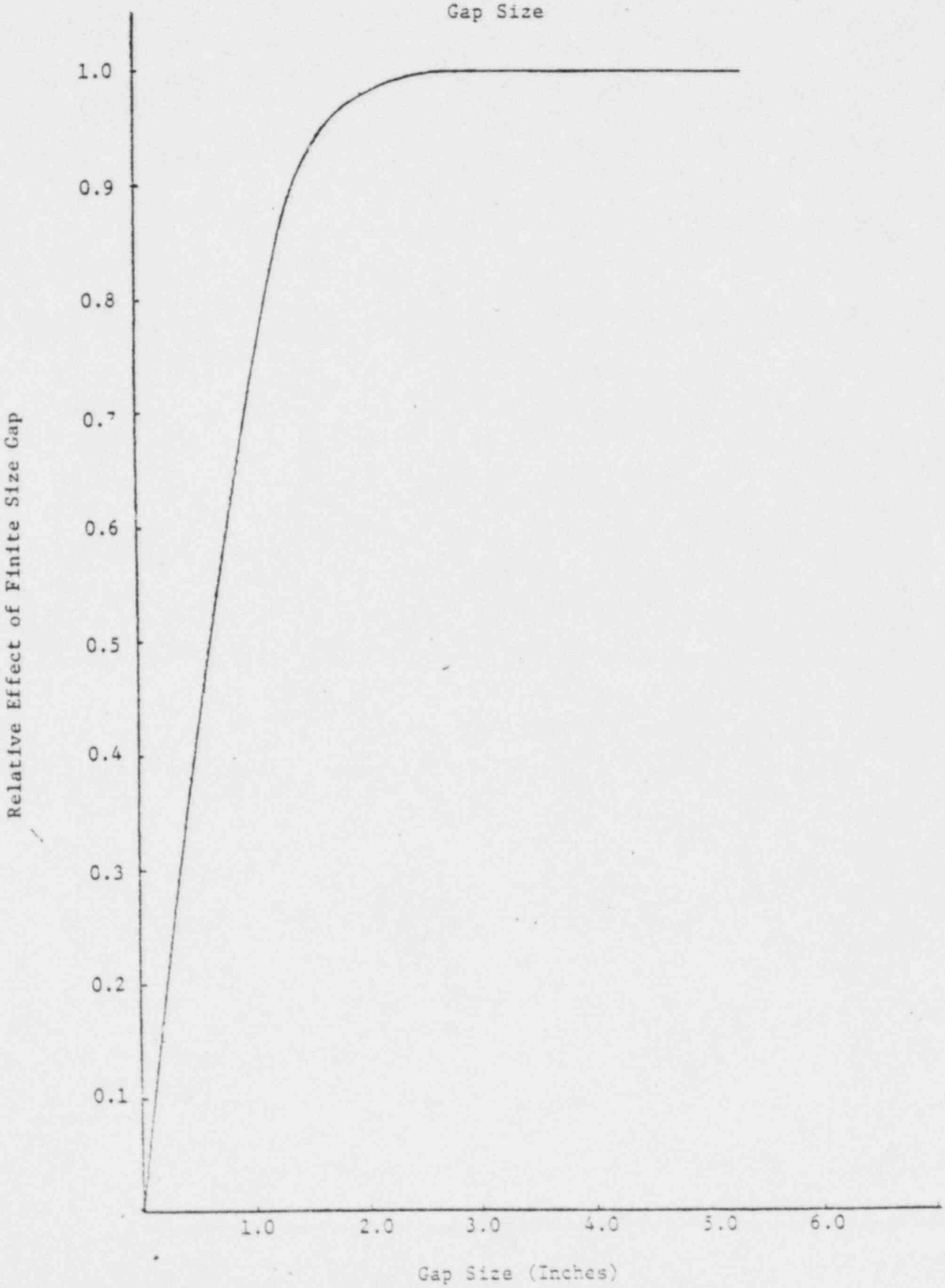
Percent Contribution to Power Peaking From Infinite Gap

Gap	6.4	2.3	.84
6.4	4.6	2.3	.84
2.3	2.3	.96	
.84	.84		

ATTACHMENT B (cont'd)

FIGURE 5

Relative Effect of Finite Size Gap
versus
Gap Size



ATTACHMENT B (cont'd)

TABLE 1

YANKEE ROWE CORE XIV

HEIGHT (INCHES)	POWER SPIKE (PERCENT)
8.00	.10
17.50	.23
20.50	.31
23.50	.37
26.50	.54
29.50	.59
32.50	.77
35.50	.93
38.50	.97
41.50	1.21
44.50	1.28
47.50	1.48
50.50	1.63
53.50	1.54
56.50	1.82
59.50	1.76
62.50	2.15
65.50	2.37
68.50	2.41
71.50	2.60
74.50	2.73
77.50	2.80
80.50	3.06
83.50	3.26
86.50	3.22
89.50	3.49