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November 15, 1990
BYR 90-150

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

Attention: Mr. Patrick Sears
Senior Project Manager
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

References:
(a) License No. DPR-3 (Docket No. 50-29)
(b) Yankee Letter to NRC, dated December 20, 1989
(c) NRC Letter to Yankee, dated July 22, 1990
(d) Yankee Letter to NRC, dated October 16, 1990

Subject: Response to NRC Request for Additional Information Regarding the YNPS Severe Accident Closure Submittal

Dear Mr. Sears:

In Reference (d), Yankee identified its schedule for responding to the staff's request for additional information regarding the YNPS Severe Accident Closure Submittal (Reference (b)). In keeping with that schedule, please find responses to those questions in List (1) of Reference (c) regarding Yankee's Individual Plant Examination (IPE), with the exception that the response to Question No. 22 will be submitted by December 31, 1990.

As noted in Reference (d), we will respond to the remaining questions (Nos. 1, 2, 4, 5, 6, 7, 9, 10, 11, 25, 26, 29, 31, 32, 34) in List (1), which relate to Containment Performance Improvement (CPI), in the February 1991 time frame. We will be prepared to discuss responses to questions in List (2) and List (3) after December 15, 1990. We request that phone calls and/or meetings, as appropriate, be scheduled by the NRC on, or as soon as possible after, December 15, 1990.

With regard to the IPE External Events (IPEEE), we request the NRC provide the date for issuance of staff questions on the IPEEE portion of our December 20, 1989 submittal to aid us in allocating resources for review and resolution.

Finally, we request a meeting with the NRC in the first quarter of 1991 to discuss the Accident Management content of our submittal and the basis for NRC closure of severe accidents for Yankee.

Sincerely,

Jane M. Grant

Jane M. Grant
Senior Engineer
License Renewal Activities

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JMG/gjt/WPP77/215
Attachment

Question 3

Page 10-4 of PSS: You acknowledge that "there is a high likelihood of the operating staff of YNPS taking action to align and actuate manual systems." Are there approved procedures in place for taking these actions?

Response 3

Page 10-4 of the PSS identifies certain systems which were not credited and other systems for which recovery was not credited, in the baseline core damage sequences for the PSS (Section 8.0 "Event Trees"). This information was provided to support the statement that there are conservatisms in the core damage sequences. Approved procedures are in place for several of these systems. There are also other systems and actions not identified which demonstrate conservatism and have approved procedures. The following table provides identification of which systems, although not credited, are proceduralized.

Table 3.1

ACTIONS NOT CREDITED BUT PROCEDURALIZED

ACTION	EMERGENCY OPERATING PROCEDURE
Charging System to Primary (for LOCAs)	ECA-1.1 Loss of Emergency Coolant Recirculation FR-C.1 Response to Inadequate Core Cooling FR-H.1 Response to Loss of Secondary Heat Sink
Emergency Feedwater To Primary	FR-C.1 Response to Inadequate Core Cooling
Emergency Feedwater with Safe Shutdown System Secondary Makeup Pump	ECA-0.0 Loss of all AC Power ECA-0.1 Loss of all AC Power, Recovery without SI required ECA-0.2 Loss of all AC Power, Recovery with SI required FR-H.1 Response to Loss of Secondary Heat Sink
Addition of Makeup to Safety Injection Tank	ECA-1.1 Loss of Emergency Coolant Recirculation

Question 8

Page 10-88 of PSS: You appear to consider containment isolation failure about half-way through your event trees. Normally it is considered much earlier in the event tree. The most important time to consider this so-called "beta" failure mode is for cases that would otherwise not fail, that is those that result in Release Cat. #5 from Event Tree Figure 10-6.

Why did you exclude containment isolation failure from the most benign and most probable release category? Also, noting the Event Tree in Figure 10-9, why do you assign a release category of #5 to a situation where you have a failed containment, only passive containment cooling, and a 3 MW heat source in containment? How does 98% of the noble gases remain in the containment?

Response 8

The answer to this question involves, in essence, an explanation of the "containment isolation failure" event for YNPS. As indicated in Chapter 9 "Fault Trees" section 9.4.6 "Containment Isolation System (CIS), page 9-39:

> "A single CIS fault tree was drawn with the top event the failure to isolate all paths from the VC atmosphere to the outside atmosphere. The probability of this occurrence was required by the containment event trees discussed in Section 10. A review of the valves closed by the CIAS indicated that there are no CIS valves which control lines that connect the VC atmosphere to the outside atmosphere. However, there are four valves that control lines which are periodically open to the atmosphere for testing or sampling purposes. In each case, there is a manual valve (or valves) downstream of the CIS valve which is normally closed but is opened for the periodic testing. The CIS valves analyzed were: VD-TV-203, SA-TV-213, HV-SOV-1, and HV-SOV-2."

Thus, only these four valves were modeled to contribute to "containment isolation failure". Any other valve failure in CIS would also require an additional breach of the closed loop system in which the valve functions, to result in containment isolation failure. (This information is also discussed in Appendix B "Detailed System Fault Tree Analysis" section B.4.5.3 "CIS Fault Tree Development", pages 158, 159.)

The four valves indicated are:

(1) VD-TV-203 Pressurizer Capillary Bleed
(2) SA-TV-213 Bleed Line Sample
(3) AV-SOV-1 Containment Hydrogen Vent
(4) HV-SOV-2 Containment Hydrogen Vent

Valve 1 controls a capillary with insignificant flow. Valve 2 is in a 3/8" line. Valves 3 and 4 are in 2" lines. Therefore, failure of any of these valves would result in negligible to minor flow. Further, YNPS is operated at positive pressure by technical specification (minimum 0.75 psig to maximum 3.0 psig), which would indicate any prior failure of especially the larger two valves. Note also that valves 1 and 2 do not specifically connect VC atmosphere to outside atmosphere but require primary boundary breach to establish this path.

Therefore, without occurrence of the event "containment isolation failure", leakage is based on technical specification leakage of 0.2% by weight of containment air per 24 hours at 31.6 psig, and even with "containment isolation failure" the leakage rate is still minor as described above. Thus, both events were assigned to release category 5, "Radionuclide release . . . without containment failure" described in section 10.9.1 "Methodology for Quantifying Release Frequencies", page 10-96 as well as described in Chapter 11.0 "Fission Product Release and Behavior", page 11-11.

It should also be noted that the dominant contributor to "containment isolation failure" was valve SA-TV-213 failure to close on demand coupled with manual valve SA-HCV-210 downstream being in the open position (94.9% of total). This is in the 3/8" line, reinforcing the minimal consequence of this event.

Regarding 98% of the noble gases remaining in containment for release category 5, Chapter 11.0 "Fission Product Release and Behavior", section 11.4 "Discussion of Results", page 11-11 states

> ". . . release category (RC5) represents accident sequences in which core melt occurs but containment integrity is preserved. Release occurs due to normal allowable leakage from the containment. The release fractions for RC5 are much smaller than those for any of the other release categories."

thus,

> "The resulting atmospheric releases of the various radionuclide groups for the six release categories are shown in Table 11-5. For all of the release categories, except for RC5, essentially all of the noble gas core inventory is released into the atmosphere."

Note that Table 11-5 indicates RC5 has a release fraction for noble gases of 0.02.

QUESTION 12

Provide a thorough discussion to justify why the Yankee IPE was performed using a PRA that is out-of-date.

Discuss how Yankee incorporated the most current design and operations information into the IPE and whether the results and conclusions presented reflect the current plant.

Certify that the IPE represents the as-built, as-operated plant.

Provide the basis for conclusion that the IPE as submitted accounts for the effect of all modifications made to the plant subsequent to the freezing of the design (1981) to perform the Yankee Rowe PSS. Your IPE submittal (Section 3.3.4) describes several plant modifications performed since publication of the PSS in 1983. Provide a concise list of all safety related plant modifications (both those motivated by the PSS and those made for other reasons) since 1981 (when the PSS design was frozen) and describe the potential downside, if any, of these modifications.

RESPONSE 12

The Individual Plant Examination (IPE) conducted as part of the Yankee Nuclear Power Station Severe Accident Closure Submittal is based on current methods using the most current design and operations information for the plant. The conclusions presented reflect the current plant.

The Yankee Severe Accident Closure Submittal ("the Submittal") includes as a part of its basis the Yankee Nuclear Power Station (YNPS) Probabilistic Safety Study (PSS) docketed on January 3, 1983 and associated ongoing programs. Section 3.3.4 ("Confirmation of PSS Results") of the IPE portion of the Submittal summarizes the impact of examples of plant modifications implemented since publication of the PSS.

A. Plant Modification Evaluations

Each change to the plant since publication of the PSS has been evaluated with no resulting change in the conclusions of the PSS. These evaluations consisted of assessment of each plant change for impact on any segment of the analysis, specifically:

- Initiating Events
- Data
- Accident Event Trees
- System Fault Trees
- Core, Vessel, and Containment Response
- Fission Product Release and Consequence Analysis

Since the evaluations of plant changes did not impact the validity of the PSS, the IPE, which is based on the PSS and these ongoing evaluations, also reflects the current plant design.

B. PSS Review

Note that the major plant modifications since publication of the PSS have been made for purposes of external events. In particular, the addition of the dedicated Safe Shutdown System and the seismic modifications constitute the majority of changes at the plant, both in terms of positive impact on safety and analysis extent/modification cost. Each of these modifications has been specifically probabilistically evaluated. These and other external modifications and analyses are listed and described in Section 4.0 of the Submittal.

Section 3.3.4 also describes an additional formal internal review of the Level I and II portions of the PSS itself which has been in process since 1987 and which is documented in internal calculation files and notebooks. As noted in Section 3.3.4, this additional review is being performed:

- To update the models to reflect the current plant configuration and operating history (e.g., addition of the Safe Shutdown System and other plant modifications listed in Section 3.3.4).

- To update methods to allow future modifications in the plant to be more readily incorporated.

- To enhance and extend the scope of the study.

- To train additional personnel in support of YNPS.

Results to date have not altered the conclusions of the PSS.

In particular, by updating the PSS Models to reflect the current plant configuration during this formal internal review, we have confirmed that the cumulative effects of the modifications have not altered the conclusions of the PSS docketed in 1983. (Note: Each modification has been evaluated as a part of our ongoing program, individually and cumulatively, using the YNPS PSS prior to installation.) The review and update of the PSS has accounted for the integrated cumulative change because it consists of the following:

1. Initiating Events

   Review and update of plant and industry generic data which has confir- completeness of initiating events categorization and has not appreciably changed initiating event frequencies. For example, update of the initiating event data with almost ten additional years of history continues to support the fact that no unrecovered loss of feedwater has occurred at YNPS. This, coupled with inspection of industry LOCA data, confirms the continued dominant contribution of LOCAs as initiating events.

2. Data

   Review a d update of plant-specific and industry-generic data including systems, common cause and human performance data, which has not appreciably altered plant specific failure rates and indicates that the YNPS PSS human performance modeling is conservative. For example, similar to initiating events, the inclusion of approximately an additional ten years of history in the data used for quantification of all systems continues to confirm the dominance of ECCS Systems (HPSI, LPSI, Recirculation) contribution to core damage frequency.

3. Accident Event Trees

Review and conversion of event trees from modular to functional as well as verification that conversion of EOPs from procedure/event-based to standardized symptom/critical safety function-based has not resulted in change to PSS conclusions.

4. System Fault Trees

Review and conversion of dependency treatment, from Fussell-Vesely factors to an auxiliary event tree approach, as well as expansion and revision of systems modeled to account for plant modifications since the issuance of the YNPS PSS, resulting in same dominant system contributions to core damage frequency (HPSI, LPSI, Recirculation).

5. Core Vessel and Containment Responses

Analysis of results of Level II portion of PSS on a separate effects basis as described in the containment performance improvements portion of the Submittal (Section 5) resulting in direct potential and actual modifications as described in the section.

Section 3.7 "Ongoing IPE/PRA Programs" of the YNPS Severe Accident Closure Submittal included commitment via a formal procedure, PRA 13 "Review of Plant Changes and Documents," to continue the process of regularly reviewing plant documents and changes for impact on the PSS in the future.

Also, in response to an NRC request for comments on the "Risk-Based Inspection Guide" produced for NRC by EG&G Idaho, Inc., Yankee provided an additional evaluation of plant changes since publication of the YNPS PSS and the reason for the changes. The review of changes included a summary of additional insights.

Thus, the IPE is based on current methods and information and has been performed on the most current design of the plant. This is because the IPE is based on use of the PSS which has been evaluated for and used to evaluate plant changes and has recently been subjected to a formal and ongoing review.

The following table provides a concise list of all major safety-related plant modifications (both those motivated by the PSS and those made for other reasons) since 1981 and an evaluation of the safety impact of each. The table is an expanded format and content version of the evaluation of significant changes presented in Section 3.3.4 (Pages 15 and 16) of the submittal and includes those items.

Plant Modification	Year	Safety Impact: Improvement	Safety Impact: Downside
Additions		Improvement by Additional:	
1. Containment Isolation Valves	83	Containment Isolation Capability	None
2. Four Emergency Atmospheric Steam Dump Valves	84	Secondary Heat Removal - Steam Removal Capability	Increased Probability of Minor Cooldown
3. Shutdown Cooling Valve Interlock	84	Prevention of Interfacing LOCA, Primary Reactivity and Inventory Control Capability	None
4. Safe Shutdown System	86	Secondary Heat Removal and Primary Inventory Control Capability	None
Replacements		Improvement by Increasing:	
5. Battery Chargers No. 1 and No. 2	84	DC System Reliability	None
6. Feedwater Control System	85	Feedwater Control System Reliability	None

Plant Modification	Year	Safety Impact Improvement	Safety Impact Downside
7. Pressurizer Safety Relief Valves	86	Primary Pressure Control	None
8. Safety Injection System Relief Valves	87	Primary Inventory Control	None
9. Battery No. 3	89	DC System Reliability	None
10. Emergency Diesel Generators	90	Emergency Power Reliability	None
Upgrades		Improved Reliability of:	
11. Seismic Upgrade for Battery Racks, EFW System, Hot Shutdown System, Etc.	80/86	Seismic Capability	None
12. Station Vital Bus	85	Instrumentation/Human Actions	None
13. Emergency Diesel Generator Cooling System and Ventilation	85	Emergency Diesel Generator Support Systems/Emergency Power	None
14. Safety Injection Building Ventilation	85	Long-Term Core Heat Removal	None
15. Reactor Protection System (RPS)	86	Reactor Protection System	None
16. Turbine Trip Logic	87	Reactivity Control and Reactor Pressure Vessel Integrity	None
17. Nuclear Instrumentation	90	Reactor Protection System	None

QUESTION 13

Discuss how an independent in-house review was conducted to ensure the accuracy of the IPE documentation package and to validate both the IPE process and its results. Provide, as a minimum, a description of the internal review performed, the results of the review team's evaluation, and a list of the review team members.

Describe the walkthrough/walkdown activities (e.g., initial walkthrough for plant familiarization; special ones to verify logic trees, dependencies, or aspects of systems interactions; to examine spatial interactions such as internal flooding) including scope and team makeup. Describe Yankee Nuclear's involvement in the plant walkthrough/walkdowns.

RESPONSE 13

A. An independent in-house review of the four part Severe Accident Closure Submittal, including the IPE was conducted.

In particular, the primary reference for the IPE, the Yankee Nuclear Power Station (YNPS) Probabilistic Safety Study (PSS), was extensively reviewed; and the PSS contains a description of the review.

PSS

Early in the study, the PSS plant system descriptions, logic models (system fault trees and accident event trees) and success criteria were reviewed by Design, Operations, Maintenance, and Analysis personnel both in formal sessions and via informal communications. During development of the PSS, sessions to review event trees and initiating events were conducted. Finally, at the end of the study, a three-week, independent formal review session of the YNPS PSS was conducted by plant and corporate office personnel. Design, Operations, Maintenance, and Analysis personnel were the independent reviewers. This review is documented in files and in an audio recording of the discussions. Recommendations from this review were incorporated in the final PSS.

As noted in Section 3.3.2, "YNPS Probabilistic Safety Study (PSS)," a further independent review of the PSS was conducted by a Technical Review Board of acknowledged experts who impartially critiqued the PSS. The reviewers were (Section 3.3.2.1):

- Professor Norman C. Rasmussen, Massachusetts Institute of Technology.
- Dr. Salomon Levy, S. Levy, Incorporated.
- Mr. Garry Thomas, Electric Power Research Institute (EPRI).
- Dr. Robert L. Ritzmann, Science Applications Incorporated (now EPRI).

Example comments involved recommendations to investigate impacts of spatial interactions of initiating events on systems performance, as well as to investigate loss of DC. These investigations were performed and are included in Appendix G, "Environmental and DC Power Top Event Reviews." The entire record of their comments and resolutions is documented in our files.

SEVERE ACCIDENT CLOSURE SUBMITTAL

In addition to the continuous review process described in the response to Question 1, an internal review of the four-part Severe Accident Closure Submittal was also conducted and consisted of comments and resolution by authors and non-participants at the outline, development and final document stages. Table 1 provides the reviewers titles, disciplines, functional positions and their independence status correlated with the sections they critiqued. The results of the reviewer's evaluations are retained in files and are reflected in the submittal made to NRC on December 12, 1989.

TABLE 1

Severe Accident Closure Submittal Review Matrix

Submittal Section	Title/Discipline/Functional Position	Participant	Independent
Full Submittal	3 Executives		x
"	2 Project Managers		x
"	2 Generic Licensing Engineers	x	x
"	Nuclear Engineering Dept. Director		x
"	Manager PRA	x	
"	Plant Superintendent YNPS		x
"	Manager of Operations YNPS		x
"	Lead Engineer PRA YNPS	x	
"	Lead Engineer Lic. Renewal/Severe Accidents YNPS	x	
"	PRA Engineer YNPS	x	
"	Lead PRA Engineer Maine Yankee		x
"	Consultant Containment	x	
IPEEE	Manager Environmental Engineering	x	
IPEEE	Lead Mechanical Engineer (Seismic) YNPS	x	
IPEEE	Lead System Engineer YNPS	x	
AM	Lead Engineer Seabrook Project		x
AM	Manager Emergency Planning		x
AM	Emergency Planning Engineer		x
AM	Lead Transient Analysis Engineer YNPS	x	
AM	PRA Manager Seabrook		x
FIRES/AM	Fire Specialists	x	

TABLE 1
(Continued)

Severe Accident Closure Submittal Review Matrix

Submittal Section	Title/Discipline/Functional Position	Participant	Independent
CPI/AM	Manager Transient Analysis		x
CPI/AM	Manager LOCA		x
CPI/AM	LOCA Engineer YNPS	x	
CPI/AM	Lead PRA Engineer Vermont Yankee		x

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B. All systems at Yankee included in the PSS and IPE have been walked down by Yankee Analysts, Systems Engineers, and/or Plant Personnel. The walkdowns performed in support of the PSS were:

- Systems walkdowns performed in support of the fault tree models (PSS, Chapter 9).

- A spatial interactions walkdown performed to determine the environmental effects (e.g., steam or other line breaks) on equipment which was modeled or which could cause an initiating event (PSS, Appendix G).

Subsequent walkdowns in support of the review and update of the PSS described in the response to Item A above have also been performed. Specifically, as part of the Systems Analysis Review and Update all systems were walked down by Yankee PRA Analysts. In addition, a comprehensive spatial interactions walkdown including additional external hazards (such as fire, internal flood) has also been performed by Yankee PRA Analysts, Systems Engineers, and Plant Personnel. These walkdowns were performed per a formal procedure PRA 01, "PRA System Walkdown Procedure," and are documented in system and spatial effects notebooks.

This procedure provides instructions for planning and performing system walkdowns. The basic steps are to:

- Gather and review information pertinent to systems being walked down.

- Consolidate information into a pre-walkdown checklist organized in the order of the planned walkdown path.

- Record information required on walkdown form during the walkdown.

Both sets of walkdowns (PSS and PSS review/update) in conjunction with the ongoing verification investigations per another formal procedure, PRA 13 "Review of Plant Changes and Documents," described in the response to Item A above constitute the process to confirm that the IPE/PSS represent the as-built, as-operated plant (to account for the impact of plant modifications).

QUESTION 14

Provide a thorough discussion of the evaluation of the decay heat removal function to address resolution of the USI A-45 "Decay Heat Removal Requirements." The discussion should identify and quantify the contributions of USI A-45 to core damage frequency or unusually poor containment performance.

RESPONSE 14

Generic Letter No. 88-20 requests that potential decay heat removal vulnerabilities be identified, as part of the IPE, to resolve Unresolved Safety Issue (USI) A-45, "Shutdown Decay Heat Removal Requirements." It also provides decay heat removal insights in an appendix. Generic Letter No. 88-20, Supplement No. 1, requests evaluation of the decay heat removal function.

The decay heat removal function has been evaluated as part of the PSS/IPE and no vulnerabilities have been identified. The basis is as follows:

An inspection of the dominant contributors to core melt frequency in the PSS results in the following conclusions:

- The overall KNPS decay heat removal capability is high because of the low total core damage frequency.

- Major contributors (although on an absolute basis low) are LOCAs and ECCS.

Specifically, PSS Table 8-2, "Initiating Event Importance Ranking with Respect to Core Melt Frequency," Table 8-3, "Dominant Accident Sequences and Their Contributions to Core Melt Frequency" and Table 8-4, "System Contributions to Core Melt Frequency," all indicate the dominance of LOCAs and ECCS. However, the absolute values of the contributions of LOCAs and ECCS are seen to be very low. Other initiators and systems which can affect the decay heat removal function essentially do not appear at any significant level in the list of contributors to core damage frequency.

The specific reasons for this are:

- A low frequency of transient initiating events. For example, no unrecovered loss of feedwater in 29 years of operation.

- Diverse and mitigative secondary heat removal system (more than 10 trains of feedwater delivery capability and multiple delivery paths).

- Three (3) emergency diesel generators.

- A separate dedicated Safe Shutdown System with its own diesel generator.

- A full feed and bleed capability. (ECCS through the PORV as well as charging pumps which can lift the PORV or the pressurizer code safety valves.)

- Simple design, including passive containment heat removal (no spray).

- Minimal reliance on support systems (e.g., air cooling of diesel generators, ECCS and electric emergency feedwater pumps).

- Carned main coolant pumps.

- Substantial thermal hydraulic margins (e.g., large primary volume, large secondary heat capacity, large containment free volume to core thermal power ratio).

Note that several enhancements to the ECCS System have already been incorporated as a result of these PSS findings and are listed in Item 1 of Table 3-9 of the Submittal, specifically:

- Relief valves changed.
- Check valves investigated.
- Ventilation modified.

- Alternative recirculation path from containment implemented.
- High pressure header division from low pressure header.

Resolution of Unresolved Safety Issue (USI) A-45, "Shutdown Decay Heat Removal Requirements," is summarized in Section 3.6 of the Submittal which indicates that the diversity, high reliability, and extensive capability for shutdown decay heat removal at YNPS is based on the following design features:

- Multiple cooling sources.
- Multiple pumps.
- Multiple flow paths.

Section 3.6 of the Submittal also lists the items which were addressed in the resolution based on the guidance provided both in the Generic Letter 88-20 Appendix 5 insights and in NUREG/CR-5230, "Shutdown Decay Heat Removal Analysis: Plant Case Studies and Special Issues."

Specifically, the following items were addressed:

Insight/Item	Treatment/Disposition
• Transient and LOCAs	• Included in Analysis, LOCAs Dominated
• Plant Support System Design and Reliance	• Minimal Support System Reliance at YNPS
• Human Performance	• Errors of Omission Modeled
• Recovery Actions	• Selected Actions Credited in PSS, Section 13.3.3, "Core Melt Frequency Sensitivity to Recovery and Operator Errors in Responding to Events"
• Loss of Off-Site Power	• Explicitly Modeled as a Separate Event Tree
• "Feed and Bleed" Capability	• Full Capability exists at YNPS
• Cost Effective Improvements	• ECCS Enhancements Incorporated

Thus, the investigation of potential decay heat removal vulnerabilities (none identified), consideration of prior decay heat removal insights and evaluation of the decay heat removal function as requested in Generic Letter 88-20 and

Supplement No. 1 have been performed during the plant-specific YNPS PSS and IPE. Therefore, USI A-45, "Shutdown Decay Heat Removal Requirements," is resolved for YNPS.

Question 15

The Safe Shutdown System (SSS) appears to enhance Yankee's safety capability significantly, especially with regard to external events. Provide the probabilistic assessment of the availability of the SSS and treatment for human recovery actions at the SSS for the leading sequences requiring use of the facility.

Response 15

Total unavailability of the Safe Shutdown System (SSS) was quantified using a fault tree. Results are presented below:

Total unavailability: 8.3 E-2

Major contributors:

(1)	SSS Diesel Generator failure to run (24 hrs.)	65%
(2)	SSS Diesel Generator failure to start	14%
(3)	Human error to start and align system	9%
(4)	SSS secondary make-up pump failure to start	5%

Quantification is based on both plant-specific and generic data.

Human action to start and align system is part of SSS Fault Tree Model.

Note: The Safe Shutdown System is not credited in "internal events" analysis because system installation was finished in 1986, after YNPS PSS was completed. In the seismic and tornado/high wind analysis, unavailability of the system was estimated since the SSS was not yet fully accepted.

Tornado/High Wind:	SSS Unavailability = 1.0 E-2
Seismic:	SSS Unavailability = 1.0 E-1

In the seismic study, human error to align system correctly was modeled separately and quantified to be 1.73 E-2.

Question 16

(A) Define core damage as used in the Yankee IPE. (B) Provide a description of how vulnerabilicies were defined and identified. (C) Discuss the fundamental causes of any vulnerabilities identified. (D) List the core damage and containment failure sequences that were selected by the screening criteria (Appendix 2) of Generic Letter 88-20. (E) Provide a concise discussion of the level at which the criteria were applied (e.g., system or train).

Response 16

(A) Core damage is the condition of the reactor fuel which results from failure to maintain the following critical safety functions for a period of 24 hours (minimum):

1. Reactivity Control
2. Core Cooling
3. Primary Inventory Control

(B) The NRC policy statement on severe reactor accidents defined "vulnerability/outlier" as " . . . possible significant risk contributors (sometimes called outliers) that might be plant specific and might be missed absent a systematic search". Generic Letter 88-20 states that for the IPE, ". . . reporting guidelines include: a concise discussion of the criteria used by the utility to define vulnerability . . . the utility should decide if it has identified a specific vulnerability . . .".

The YNPS IPE consists of the plant specific internal events PRAs and NRC reviews cited in Section 3.3 of the submittal. Vulnerabilities would be, as stated above, plant specific significant risk contributors.

(C) No vulnerabilities were found. Key or dominant contributors to core damage frequency were determined via the analyses, even though on an absolute basis they are very low. Improvement opportunities for some key contributors were thus identified, and selected modifications have already been made. (Table 3-9 of Submittal)

The decision process on whether an identified potential change is warranted is depicted in Figure 1 and the criteria are detailed in Table 1.

(D) The tables on pages 2-15 and 2-17 of the PSS provide core melt frequencies and release frequencies ranked by initiating event. The core damage frequencies represent the sum of all functional sequences stemming from each initiating event. On a best estimate basis or "expected" level as specified in GL 88-20, Appendix 2, none of the sequences exceeds reporting selection criteria numbers 1, 3, or 4 since the sum of all sequence by initiator do not exceed the criteria. Similarly, on a best estimate basis, Table 3-8 of the submittal, which is also the sum of sequences by initiator, indicates only LOCAs and ATWS (marginally) and Reactor Vessel Rupture exceed reporting selection criterion number 2, and were thus included in the submittal. Note that Reactor Vessel Rupture does not appear in the conservative assessment and is on an absolute basis very low. Sequence sums by initiator were used and individual sequences were not listed (although, they are available in Table 8-3) since all were low in absolute frequency. Nevertheless, Table 3-9 of the submittal lists modifications/ lessons/uses which have resulted, demonstrating actual performance which satisfies the intent of criterion 5.

(E) Note that had any initiating event category exceeded the criteria, its dominant sequences would have been investigated at the system and/or train level, both of which are variously modeled, to determine if any significant contributors existed.

DEPICTION OF DECISION PROCESS

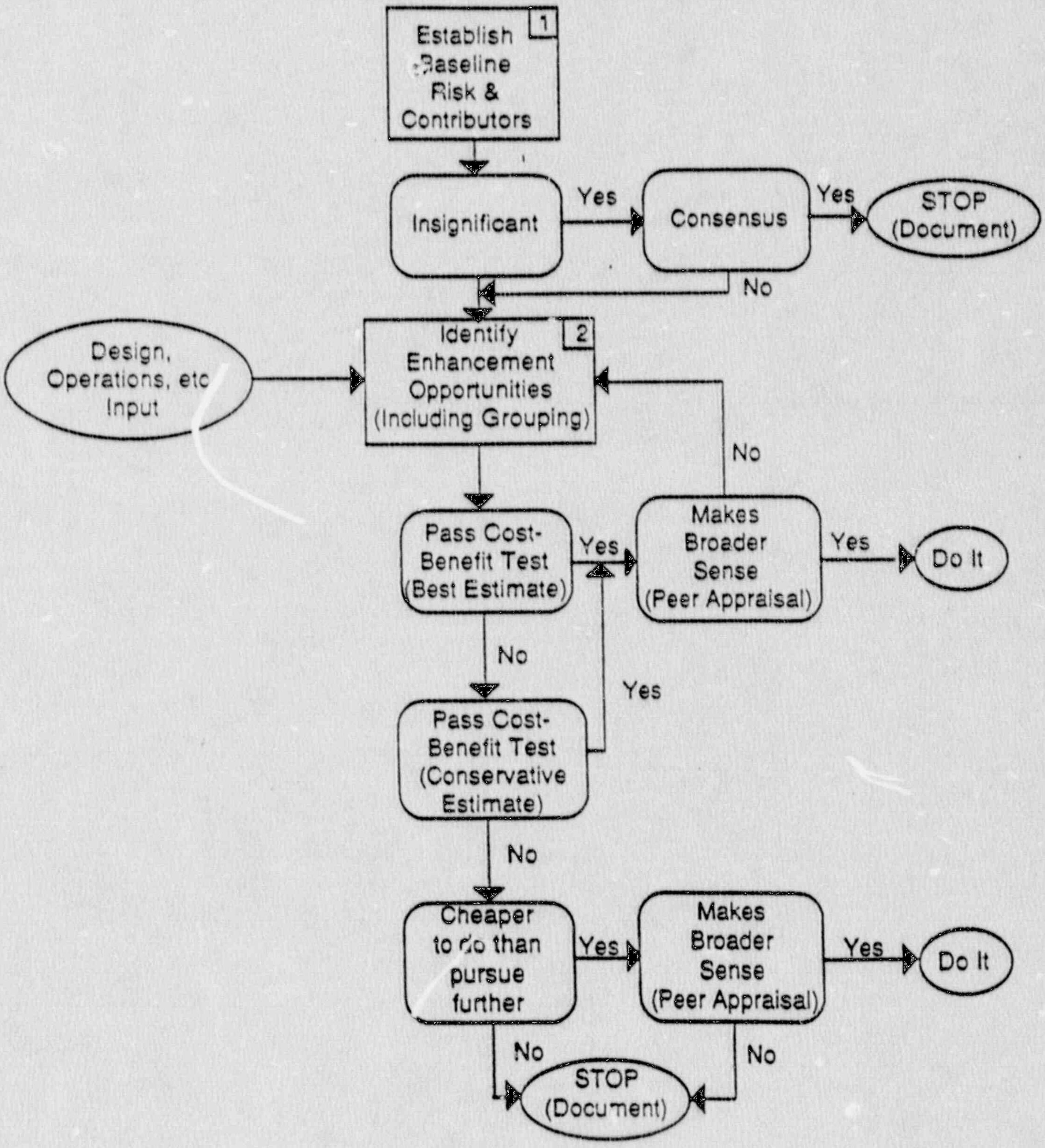

Figure 1

Table 1

DECISION CRITERIA

If < $ 1000/person	Do It

Unless:

Extenuating Circumstances:

Personnel Health $ Safety (ALARA)

Economics

If > $ 1000/person	Do Not Do It

Unless:

Absolute $ Low, with positive safety value

Cheaper to do than to analyze

Question 17

The IPE Generic Letter requested licensees to discuss unique plant features that contribute significantly to improved or reduced core damage frequency or good containment performance. Although your IPE submittal did not address this area explicitly, we believe that there are several plant features (e.g., the Safe Shutdown System and the passive Vapor Containment) that should be highlighted. Provide a concise discussion of those unique plant features that Yankee Rowe believes provide a substantial safety benefit. Describe those features that require special attention by operating personnel.

Response 17

The following is a concise discussion of YNPS unique plant features that contribute significantly to improved or reduced core damage frequency or good containment performance. Features that require special attention by operating personnel are addressed.

(A) The Safe Shutdown System (SSS) provides a remote, independent, additional means of primary inventory, pressure and reactivity control capability as well as an additional secondary heat removal capability. The SSS has independent instrumentation, motive/control power and water source (Fire Water Storage Tank - FWST) and a separate building.

The SSS is designed for operation during fire or tornado/high wind or a seismic event or a flood. The SSS consists of a positive displacement primary makeup pump with a Boron mix tank and a secondary makeup centrifugal pump, both taking suction from the FWST and both powered by an independent diesel.

The SSS is placed into operation and alignment of flow paths is accomplished by manual operator actions.

(B) YNPS has Main Coolant System (MCS) loop isolation valves (one on each hot and cold leg) which are used for MCS inventory control during a steam generator tube rupture event. Each valve is remote manually operated from the control room. Two separate switches must be operated to achieve movement of each valve, to minimize possibility of inadvertent closure.

(C) YNPS has the capability to achieve Main Coolant System (MCS) inventory control by makeup to the primary system from several secondary systems. Specifically, Main Boiler Feedwater, Electric Driven Emergency Feedwater or Steam Driven Emergency Feedwater are capable of being manually aligned to supply MCS makeup.

(D) Main Coolant System heat removal at YNPS is able to be accomplished by use of the charging pumps in conjunction with either the PORV or the Pressurizer Code Safety Valves. These positive displacement pumps are capable of lifting either of these type of valves.

(E) Secondary Heat Removal and Inventory Control may be accomplished at YNPS by use of Alternate Emergency Steam Generator Feedwater, in addition to normal Emergency Feedwater. These systems include the Safety Injection System or Charging System aligned to feed a steam generator, the Safe Shutdown System, previously mentioned, as well as use of the condensate pumps if the steam generators are depressurized. The systems are manually aligned by the operators via either the normal feedwater paths or via the Steam Generator blowdown paths.

(F) The containment function at YNPS is accomplished by a Vapor Container (VC) which requires no Spray System to dissipate containment atmosphere heat, since it is capable of passive heat transfer through the uninsulated VC metal which dissipates heat to the ultimate heat sink (the outside atmosphere). Since it is passive, no operator actions are required.

(G) Recirculation at YNPS is possible via normal ECCS recirculation or by an alternate recirculation using the charging pumps which are remote manually aligned by the operators.

(H) YNPS has canned Main Coolant Pumps because of its low thermal power rating. Thus, it is not susceptible to a Main Coolant Pump Seal LOCA, which results in a significant reduction in overall risk.

(I) There is minimal reliance on support systems by frontline systems at YNPS. Examples are Emergency Diesels, Emergency Core Cooling System Pumps and Electric Driven Emergency Feedwater Pumps which do not rely on Component Cooling Water.

(J) YNPS has substantial margin to design limits. Examples are larger than typical Primary System volume, Secondary System Heat Capacity and Inventory as well as containment free volume to core size ratio.

Question 18

Provide train level dependency tables/matrices for dependencies between front-line and support systems as well as for dependencies among support systems. (Note: this is not a single-failure analysis.) Particular attention should be paid to dc power, component cooling, service water, room cooling, control air and pump lubrication.

Identify where spec..al dependencies were accounted for in the IPE internal events evaluation. What sequences leading to core damage were affected by spatial dependencies?

Response 18

Dependency matrices for systems and components are given in YNPS PSS (Table 8-1 or B-3; "YNPS System Interdependency Matrix" and Table B-2; "YNPS Component Interdependency Matrix"). The dependencies were not treated as the frontline-frontline or frontline-support system dependencies because that methodology was not available in time when the original YNPS PSS was completed. In the YNPS PSS, system interdependencies were addressed at three points:

(1) Initiating event dependencies were included in event trees.

(2) Environmental effects of the initiating events were also addressed and quantified.

(3) Shared component dependencies were addressed during the quantification of the event trees.

The system dependencies were incorporated in event tree quantifications by using Fussel-Vesely importance measures and a pairwise approximation (probability of the entire sequence is set equal to the probability of the minimum pair, where a pair is defined as a common functional failure contribution between two systems). Conservatism of this approach is proved through ongoing YNPS PSS update where frontline/support system methodology is applied.

Dependencies between electrical buses (involving dc buses) and "frontline" systems are defined in Table 18.1, from Table B-2, YNPS PSS. Effect of losses of component cooling, service water, control air and heating, ventilation and air conditioning (HVAC) on YNPS systems are defined in Tables 18.2, 18.3, 18.4 and 18.5, respectively. As it can be seen from the tables, those systems have a high redundancy and diversity of back-up systems. Therefore, dependency between them and "frontline" systems is only minor.

Spatial dependencies were accounted for in an Environmental Effects study (Chapter 8.5 and Appendix G, YNPS PSS). A vital areas analysis was performed for each of the initiating events that could create an adverse environment. For each unique location and initiating event, the top events of the event tree were examined to:

- define equipment and instrumentation in the area,
- determine if equipment/instrumentation is qualified for environment,
- determine if equipment/instrumentation is affected,
- determine effect severity,
- determine if the operator is restricted from operating necessary equipment locally.

If possible adverse effect was identified, the failure probability of the system was set to 1. If this resulted in a substantial impact on the core melt frequency, the important systems were carefully reviewed to assess the realistic impact and the core melt frequency was recalculated.

The results of a quantitative analysis of the possible environmental effects associated with each initiating event are summarized in Table 18.6.

Table 18.1

YNPS Interdependency Matrix
Between Electrical Components and Systems

(from Table B-2, YNPS PSS)

Component Type	Component Descr.	CIS	LPSI	HPSI	ACC	SB Recirc.	SITOSG	2MU	1MU	CSD	EBF	MFW	NRV	RPS	LB Recirc.	SIAS
Elec. Buses & Motor Control Centers	480V Bus 6-3					x		o	o	o		x			x	
	480V Bus 4-1					x		o	o	o					x	
	480V Bus 5-2					x		o	o	o	x	x			x	
	480V Emerg. Bus E-1		x	x		x			x	x					x	
	480V Emerg. Bus E-2		x	x		x	o								x	
	480V Emerg. Bus E-3		x	x		x			x	x					x	
	EMCC1								x							
	EMCC2									x	x					
	125V Bus B				x			x	x		x					x
	125V Vital Bus							x		x						
	125V DC Bus 1	x	x	x	x	x							x	x	x	x
	125V DC Bus 2	x	x	x		x	o						x		x	x
	125V DC Bus 3	x	x	x		x							x		x	
	125V DC Bus 3A				x											
	MCC1 Bus								x							
	MCC1 Bus 2								x							
	MCC2 Bus 1							o	o	o						
	MCC4 Bus 2							o	o	o						
	MCC4 Bus 2							o	o	o						
	2400V Bus 2										x	x				
	2400V Bus 3										x	x				

o Component Failure will cause system failure
x Component Failure alone will not cause system failure

Table 18.2

Effects of Loss of Component Cooling

System	Effect	Back-up
Main Coolant Pumps	Pump will be inoperable within 3 minutes of loss of Cooling Water	An alternate source is available (Fire Protection System), connections are provided and they are redundant, OR If loss is unrecoverable, plant will be brought to a normal shutdown condition by normal means on natural circulation
Shutdown Cooling Cooler/ Low Pressure Surge Tank Cooler	Loss of Cooling Water	Alternate sources are available through hose connection to Fire System, OR If loss is unrecoverable, YNPS procedures provide for alternate methods of cooling the MCS: (1) Primary Feed and Bleed (2) Restoring SG Cooling

Table 18.3

Effects of Loss of Service Water

System	Effect	Back-up
Boiler Feedwater Pumps	Loss of Cooling Water to the Lube Oil Coolers	Electric and Steam Driven Emergency Feedwater pumps, ECCS and CVCS pumps
Charging Pumps	Loss of Cooling Water to the Speed Controllers of the No. 1 and No. 3 Charging Pumps	The Speed Controllers can be manually locked at full speed
Component Cooling Coolers	Results in Loss of Cooling Water to Primary Plant Components. Require shutdown of Main Coolant Pumps	Fire System connection from fire house to Service Water side of cooler could restore cooling with fire water
Shutdown Cooling Pumps	Loss of Cooling Water to Mechanical Seal and the Lube Oil Coolers	Fire System Connection can provide diverted water for limited service
Low Pressure Surge Tank Cooling Pumps	Loss of Cooling Water to the Mechanical Seal and the Lube Oil Coolers	Fire System Connection can provide diverted water for limited service
Control Air	Loss of Cooling to Control Air Compressors	No. 1 and No. 2 Compressors can be supplied with Emergency Cooling from the Fire System
Turbine Bypass	Loss of SW implies Loss of Circulating Water and Loss of Control Air resulting in Loss of Auxiliary Steam, this results in Loss of Condenser Vacuum	EASD and SG SVs

Table 18.4

Effects of Loss of Control Air

System	Effect	Back-up
Charging Pumps	Disables Speed Controllers of the No. 1 and No. 3 Charging Pumps (they fail as is)	The Speed Controllers can be manually locked at full speed
Turbine Bypass	Loss of Control Air results in Loss of Control for Turbine Bypass, also can result in Loss of Condenser Vacuum due to Loss of Auxiliary Steam	Could be operated locally with handwheel, Turbine Bypass is backed up by motor operated Emergency Atmospheric Steam Dump Valves and Steam Generator Code Safety Valves
Main Feedwater Controllers	Loss of Control Air results in Loss of Main Feedwater Controllers	Main Feedwater Controllers are backed up by manual operator control of the Flow Control Valves
VC Isolation Trip Valves	Loss of Control Air effects many Vital VC Isolation Trip Valves (fail closed)	For some valves Control Air is backed up by an alternate standby nitrogen system, Valves fail closed on Loss of Air
Steam Driven Emergency Feedwater - Steam Supply	Disables the Steam Reducer from the Main Steam Boiler (AS-TV-405)	Operator repositions the 3 way valve to supply nitrogen from the Emergency Station (Control Air is backed up by nitrogen system, AS-TV-405 also has a manual bypass valve)2

Table 18.5

Effects of Loss of Heating, Ventilation and Air Conditioning (HVAC)

System	Effect	Back-up
Main Control Board (MCB)	Loss of MCB AC can result in slow heatup and instruments drift and possible failure	Opening of both MCB Doors (proceduralized) provides sufficient flow for cooling
Main Control Room (MCR)	Loss of MCR AC can result in slow heatup, affecting instruments and operator actions	Opening of the door to to Southwest (SW) staircase and the door from SW staircase to outside, (or East turbine floor doors) will provide natural circulation cooling
Switchgear Room (SWGR)	Loss of SWGR ventilation will result in slow heatup, which in long [illegible]rm may affect equipment	Opening of the door to SW staircase (and door from SW staircase to outside) or the door to turbine building will provide natural circulation cooling
Turbine Building (TB) Pump Room	Can result in overheating of BFW pumps	Opening of TB doors or running of fans of BFW pumps
SI Building	Loss of HVAC results in heatup, slow if injection mode, rapid if in recirculation mode	Opening of building louvers, door to PAB, #3 DG cubicle door to outside, Battery Room 3 door, running of DGs with door open (note: HVAC is two train system)
NRV Enclosure	Loss of ventilation - NRVs failure	Opening of the door and hatch
	Loss of heating - NRVs failure	Portable Heater

Table 18.6

Frequency of Environmental Induced Sequences

Initiating Event	Calculated frequency of environmental induced sequences leading to core melt.
Excessive Cooldown	
• Inside Containment	1.9×10^{-7}
• Outside Containment	1.8×10^{-7}
Steam Line Break	5.3×10^{-9}
Very Small LOCA	Negligible
Small LOCA	No Effects
Intermediate LOCA	7.5×10^{-9}
Steam/Generator Tube Rupture	No Effects
Plant Trip	No Effects
Loss of AC Power	No Effects
Decrease in Feedwater Flow	
• Pipe Breaks inside VC	No Effects
• Pipe Breaks outside VC and Turbine Building	1.0×10^{-9}
• Pipe Breaks inside the Turbine Building	5.3×10^{-9}
Decrease in Steam Flow - Loss of Vacuum	
• Non-Pipe Breaks	No Effects
• Pipe Breaks	1.0×10^{-8}
Decrease in Steam Flow - NRV Closure	No Effects
Decrease in Steam Flow - Turbine Trip	No Effects

Question 19

Provide the appropriate minimum success criteria for event trees/frontline systems used in the IPE. Indicate the basis for these criteria and the degree of conservatism (or whether best-estimate or optimistic) used.

Provide the success criteria for initiating events developed in the master logic diagram. A statement for example that "high pressure safety injection (is required)" does not indicate if one, two, or three charging pumps are required or if only two safety injection pumps are required or perhaps a combination of both is satisfactory.

Response 19

The minimum success criteria for event trees frontline systems used in the IPE as well as their bases are provided in Chapter 9 "Fault Trees". Specifically, Table 9-1 provides the YNPS System Failure Probabilities and provides the failure criteria for each system. In addition, the corresponding text in sections 9.4.1 through 9.4.8, pages 9-14 through 9-42, provides the bases for these criteria.

Chapter 13 "Sensitivity Evaluations" provides the degree of conservatism (e.g., in an additional set of evaluations, certain of the conservative system success criteria were replaced with best estimate criteria). In particular, in section 13.3.2, pages 13-22 through 13-36, core melt frequency sensitivity to success criteria is thoroughly discussed. The evaluation primarily addressed safety injection and charging system failures for various LOCA sizes.

As stated in project approach, section 3.1.4 "Preparation of

Master Logic Diagram", the MLD "was used as the fundamental means of searching for accident initiating events. In essence, it is a fault tree of the plant in broad overview with 'excessive off-site release' as the top event." The MLD is provided in figures 3-2 and 3-3 on pages 3-57,58. The 27 basic events at level 10 of the MLD fault tree constitute the initiating event categories and are listed on page 3-19, as well as in Chapter 5 "Initiating Events", pages 5-2,3. The categories yielded 19 specific initiating events which were developed into frontline event trees as identified on page 5-15. Thus success criteria are given for the top events in the frontline tree rather than for the initiating events resulting from the MLD fault tree. The success criteria for major systems are also given in response to Question 28.

Question 20

Provide a concise description of how and why the component cooling water, service water and control air system failures were lumped into the Plant Trip initiating event tree.

Response 20

The reason for treating component cooling water, service water and control air system failures as initiating events in the plant trip event tree is stated in section 8.3.1.8 Plant Trip - Event Tree 13, pages 8-44,45 of the PSS. Specifically these events were quantified separately because of their potential "common mode" failure impact on the mitigative systems represented by the top events of Event Tree 13.

The method for treating these system's failures in the plant trip event tree is stated in detail in sections 8.3.2.2, 8.3.2.3 and 8.3.2.4 Complete Loss of Service Water, Component Cooling and Control Air, pages 8-65 through 8-76. In summary, the initiating event frequency was quantified for these systems and the effect of the loss of the system on each of the subsequent top events of the plant trip tree was accounted for by assuming failure probability of 1.

Question 21

Provide a concise discussion of how the initiating events frequency for non-isolable LOCAs was determined for the IPE.

Provide a concise discussion of how intersystem LOCAs were evaluated under the IPE for the shutdown cooling system.

Response 21

The identification and quantification of non-isolable LOCAs is thoroughly discussed in Section 5.4.2.6 of the PSS, pages 5-33 through 5-48. The identification process resulted in the list given on page 5-34. Each was evaluated in the subsequent sections. Section 5.4.2.6.9 "Non-isolable LOCA Event Frequency Summary" indicates that the total frequency is less than 2×10^{-7}/year. This total frequency is basically the combination of the only two significant contributors from the evaluations detailed in sections 5.4.2.6.1 through 5.4.2.6.8 (most of which were not significant). Specifically the two contributors are (1) Shutdown Cooling System Isolation Valve Failures, equal to 1.8×10^{-7}/year, and (2) Safety Injection System Check Valve Failures, less than 10^{-8}/year.

A thorough discussion of how intersystem LOCAs were evaluated under the IPE for the Shutdown Cooling System (SCS) is provided in PSS section 5.4.2.6.2 "Shutdown Cooling System Isolation Valve Failures" as part of section 5.4.2.6 "Non-isolable LOCAs Outside the Vapor Container". (Please refer to pages 5-36 through 5-41.) A concise summary of the discussion follows. The isolation valves between the Main Coolant System and the SCS were investigated. Failure modes were determined. (Expressed as cut sets.) The impact of operator errors was assessed by reviewing procedures to determine probability of failure to close valves. Failure rates were determined and the probability of a non-isolable LOCA occurring for the SCS was calculated to be in the range of 1.8 x 10^{-7} to 10^{-10}/year. The former value is consistent with WASH-1400 failure rate distributions and the latter value is consistent with valve disc rupture probabilities.

Question 23

Discuss the impact of loss of service water on plant systems, and estimate its contribution to core damage frequency.

Response 23

YNPS Service Water System (SWS) provides cooling for many systems required for normal operation and shutdown. The effect of loss of this system was reviewed against each top event in the YNPS "plant trip" event tree. The system and equipment of interest from the YNPS PSS are listed in Table 23.1 with an explanation of the effect loss of Service Water has upon them and with possible back-ups. Table 23.1 is based on information presented in Tables 18.3 and 18.4 in Response 18.

Given systems/equipment backups and the redundancy and diversity of the Service Water System, loss of Service Water doesn't appear among the top 40 core melt sequences and the total contribution to the core melt frequency is less than 0.1 percent.

Service Water System functions and impacts on plant systems are analyzed in Sections 4 4.4, 5.4.2.2 and 8.3.2.2 of YNPS PSS.

Table 23.1

Effects of Loss of Service Water

System	Effect	Back-up
Boiler Feedwater Pumps	Loss of Cooling Water to the Lube Oil Coolers	Electric and Steam Driven Emergency Feedwater pumps, ECCS and CVCS pumps
Charging Pumps	Loss of Cooling Water to the Speed Controllers of the No. 1 and No. 3 Charging Pumps	The Speed Controllers can be manually locked at full speed
Component Cooling Coolers	Results in Loss of Cooling Water to Primary Plant Components. Require shutdown of Main Coolant Pumps	Fire System connection from fire house to Service Water side of cooler could restore cooling with fire water
Shutdown Cooling Pumps	Loss of Cooling Water to Mechanical Seal and the Lube Oil Coolers	Fire System Connection can provide diverted water for limited service
Low Pressure Surge Tank Cooling Pumps	Loss of Cooling Water to the Mechanical Seal and the Lube Oil Coolers	Fire System Connection can provide diverted water for limited service

Table 23.1 (cont'd.)

Effects of Loss of Service Water

System	Effect	Back-up
Control Air	Loss of Cooling to Control Air Compressors	No. 1 and No. 2 Compressors can be supplied with an emergency cooling from the Fire System.
	Loss of Control Air Results in Loss of Control for Turbine Bypass	Turbine Bypass is backed up by motor operated Emergency Atmospheric Steam Dump valves and Steam Generator Code Safety valves.
	Loss of Control Air Results in Loss of Main Feedwater Controllers	Main Feedwater controller are backed up by manual operator control of the flow control valves.
	Loss of Control Air Effects Many Vital VC Isolation Trip Valves (to hold open)	For some valves Control Air is backed up by an alternate standby nitrogen system. Valves fail closed on loss of air.
	Loss of Control Air disables the Steam Reducer from the Main Steam Boiler (AS-TV-405) affecting Steam Driven EFW Pump	Control Air is backed up by an alternate standby nitrogen system, AS-TV-405 also has a manual bypass valve.

Question 24

Discuss the need for feed and bleed, the probability of success and the impact on core damage frequency from loss of this function.

Response 24

Feed and bleed is needed if all other means of maintaining the Core Cooling Critical Safety Function by Secondary Heat Removal (SHR) have failed. Note that the probability of failure of SHR and hence the probability of demand for feed and bleed are very low at YNPS because of the numerous diverse and redundant mitigative systems available for SHR. The probability of success of feed and bleed is high. This is because feed and bleed is a bona fide capability at YNPS since is consists of not only ECCS operation with the PORV but also Charging System operation with either PORV or Pressurizer Code Safety Valves resulting again in diverse, redundant capabilities. These features are described on page 17 of the submittal.

Conversely, the impact on core damage frequency of the loss of the feed and bleed function is very low at YNPS. This is because of the numerous other means of core cooling at YNPS and their corresponding high availability, as stated previously.

Note, also, that Appendix G provides an assessment of the sensitivity of the mitigative functions/actions to:

(1) the environment resulting from each initiating event, and
(2) each possible combination of the DC bus losses.

Feed and bleed, in particular the PORV, was assessed for effects of potential harsh environments caused by the initiating events for each event tree. For those instances where there was an effect, credit for success was appropriately adjusted. However, the effect on overall results was minimal because of the charging pump pressurizer code safety valve capability for feed and bleed, as previously mentioned.

With regard to the various combinations of loss of DC buses, no credit was taken for feed and bleed since many of the systems that were included in re-establishing secondary feedwater flow were included in Decay Heat Removal.

Question 27

Provide a concise description of the additional systems being considered to retain the reactor core within the vessel, as mentioned in the CPI portion of the submittal presented at the May 3, 1990 meeting.

Response 27

Note that brief descriptions of the additional systems being considered are provided in sections 5.3.2, 5.3.3 and 5.5 of the submittal. The following is additional information.

Depressurization System

As a result of a conceptual study, two options for depressurizing the reactor warrant further evaluation.

Option 1

The depressurization system option consists of four electrically actuated (triggered) valves. The series/parallel arrangement provides redundant isolation as well as redundant depressurization capacity. Two normally open MOVs would be provided in the common discharge pipe to allow isolation upon spurious operation of the trigger valves. The valve inlet would be from the pressurizer safety valve inlet piping. The valve outlet would be a pipe stub which would include a rupture disk and small relief valve. Optionally, the discharge could go to the existing safety valve discharge header piping which is equipped with a rupture disk with a small bypass to the low pressure surge tank to collect any minor valve leakage.

A portable electric source would be used to trigger the valves when required.

Option 2

This option consists of five air-operated valves which fail closed on loss of air. The valves would be arranged in two parallel trains of two series valves with an additional valve cross connecting the trains between the series valves. This arrangement provides redundant isolation as well as redundant depressurization capacity. The valve inlet would be off the pressurizer safety valve inlet piping. The valve outlet would be a pipe stub which would include a rupture disk and relief valve. Optionally, the discharge could go to the safety valve discharge header piping which is equipped with a rupture disk with a small bypass to the low pressure surge tank to collect any minor valve leakage.

Motive power would be provided by three nitrogen bottles which would provide redundancy located in an accessible location. The bottles would not be connected to the operator tubing until operation is required. A normally closed isolation valve and pressure regulator would be located in the bottle discharge tubing. Operation would require connecting the tubing and opening the manual isolation valve.

Injection System

A number of options with variations are under consideration for an injection system. The system would consist of a water source, 60 GPM pump, piping and valves, motive power for the pump valves and any required accessories.

The required pump head would be determined by the point chosen for injection. Either the reactor head, charging system piping, ECCS piping or shield tank cavity could be the injection point depending on the final design criteria required to be met.

The water source would be from either Sherman Pond, an existing site tank, new outside tank or wells.

Motive power for the pumps and MOV would be supplied from the security diesel generator which would be modified to allow the generator to power these loads or an additional diesel generator which would be installed.

The pump location, pipe routing and valves required would be dependent on the required injection point, water source and design criteria.

Question 28

Provide the remaining summary sheets for major systems similar to the ones provided for main feedwater and recirculation. The major plant systems identified are: HPSI, LPSI, accumulator, chemical shutdown, emergency feedwater, reactor protection system, and the containment isolation system. As depicted in the May 3, 1990, meeting handouts, the system summary sheets include: mission times, success criteria, failure probability, and major cut set contributors.

Response 28

Attached are summary sheets for:

HPSI

LPSI

Accumulator

Chemical Shutdown

Emergency Feedwater

Reactor Protection System

Containment Isolation System

System:	**HPSI (2)**
	(2 HPSI/1 LPSI Pump)
Mission Time:	48 hours
Success Criteria:	Delivery of Water
From:	The SI Tank with level greater than 11 feet.
By:	At least 2 (of 3) HPSI pumps boosted by at least 1 (of 3) LPSI pumps.
Through:	All 4 cold leg injection lines (3 lines required for success and one is assumed lost by LOCA).
To:	The MCS
Failure Probability:	3.54×10^{-3}

Major Cut Set Contributors

(1)	1 of 12 check valves fails (to open)	> 41%
(2)	1 of 14 manual valves closed (during maintenance)	> 23%
(3)	SI tank out for maintenance	17%
(4)	1 safety valve fails (to close)	14%

System:	**LPSI (2)**
	(2 LPSI Pumps)
Mission Time:	3 hours
Success Criteria:	Delivery of Water between 120° F and 130° F
From:	The SI Tank with level greater than 11 feet.
By:	At least 2 (of 3) LPSI pumps.
Through:	All 4 cold leg injection lines (3 lines required for success and one is assumed lost by LOCA).
To:	The MCS
Failure Probability:	3.48×10^{-3}

<u>Major Cut Set Contributors</u>

(1)	1 of 12 check valves fails (to open)	44%
(2)	1 of 13 manual valves closed (during maintenance)	25%
(3)	SI tank out for maintenance	15%
(4)	1 safety valve fails (to close)	13%

System:	**Accumulator**
Mission Time:	12 hours
Success Criteria:	Delivery of Water after approximately 11 second delay
From:	The Accumulator
By:	Adequate N^2 Pressurization
Through:	All 4 (of 4) cold leg injection lines (3 lines required for success and one is assumed lost by LOCA).
To:	The MCS
Failure Probability:	4.13×10^{-3}

Major Cut Set Contributors

(1)	1 of 3 relief valves fails (to reseat)	33%
(2)	1 of 12 check valves fails (to open)	30%
(3)	1 of 12 manual valves closed (during maintenance)	26%

System:	**Chemical Shutdown**
Mission Time:	1 hour
Success Criteria:	Delivery of borated water
From:	The BAMT (includes isolation of alternative sources)
By:	3 of 3 charging pumps
Through:	The normal charging lines
To:	The MCS
Failure Probability:	2.44×10^{-2}

<u>Major Cut Set Contributors</u>

(1)	Loss of power (busses or breakers fail)	56%
(2)	Pump Failures	17%
(3)	1 of 3 safety valves fails (to close)	5.5%

System:	**Emergency Feedwater**
Mission Time:	10 hours
Success Criteria:	Delivery of water
From:	The DWST or the PWST
By:	At least 1 (of 1 steam driven and 2 electrical driven) emergency feed pump
Through:	The normal steam generator feed lines or the steam generator blowdown lines
To:	At least 1 (of 4) steam generators
Failure Probability:	4.8×10^{-6}

Major Cut Set Contributors

(1)	Failure of the SDEFW combined with an operations error in which both motor driven pumps are improperly returned from maintenance	62.4%
(2)	Failure of the SDEFW combined with failure of 1 motor driven pump and the other pump out for maintenance	21.7%
(3)	Failure of the SDEFW combined with failure of both motor driven pumps	1.77%

System:	**Reactor Protection System 1**
	(for Cooldown Transients)
Mission Time:	On demand
Success Criteria:	No failure of any two adjacent control rods to insert

Failure Probability:

Independent Failures:	1.41×10^{-4}
Total (including common mode):	1.51×10^{-4}

Major Cut Set Contributors (to independent failures)

(1)	Probability of failing to insert two adjacent control rods	99.8%
(2)	Failure of the two scram breakers to open	0.16%

System:	**Reactor Protection System 2 (for Non-Cooldown Transients)**
Mission Time:	On demand
Success Criteria:	Insertion of 12 of 24 control rods

Failure Probability:

Independent Failures:	3.03×10^{-7}
Total (including common mode):	1.0×10^{-5}

Major Cut Set Contributors (to independent failures)

(1) Failure of the two scram breakers to open 99.9%

System: **Containment Isolation System**

Mission Time: On demand

Success Criteria: Isolation of all paths from the VC atmosphere to the outside atmosphere

Failure Probability: 9.86×10^{-5}

<u>Major Cut Set Contributors</u>

(1)	Failure of bleed line sample isolation valve (SA-TV-213) to close on demand, given down-stream manual valve left open after testing	94.9%
(2)	Failure of main coolant vent header trip valve (VD-TV-203) to close on demand, given down-stream manual valve left open after testing	2.39%
(3)	Common mode failure of the CIAS and SIAS pressure detectors due to miscalibration and/or valving out of the detectors	3%

Question 30

Steam generator tube rupture has emerged as a major contributor to bypass leakage. It is listed as one of the initiating events that were examined in the Yankee PSS. No mention was found, however, concerning the possibility of induced steam generator tube rupture. Discuss the extent to which steam generator dryout induced SGTR was considered.

Response 30

As explained in the IPE submittal (section 5.) "Containment Performance Improvements", page 117 and Figure 5-1, page 137), the investigations and results of, in particular, the PSS led to a set of separate effects analyses (described in section 5.2.3) which explicitly treated the phenomenon of steam generator dryout induced steam generator tube rupture. The discussion can be found on pages 4-2 through 4-4 of the internal report "Containment Performance Investigation for the Yankee Nuclear Power Station" (attached). The results of this investigation have led to consideration of both procedural changes (Main Coolant Pump Operation) and hardware modifications (additional depressurization capability), as further described in sections 5.3 and 5.5 of the submittal and the response to Question 27.

be very useful in extending the plant capabilities for preventing core damage. Specifically, the procedures are very effective in guiding the operator(s) to use the core injection systems as well as the extensive capabilities for water injection to the secondary side of the steam generators for maintaining adequate core cooling. With the comparatively large secondary side water inventory for this smaller reactor system, the time to steam generator dryout is much longer than for the larger plants considered in the NRC (NUREG-1150 and NUREG-0956) and IDCOR analyses. Thus, the secondary side systems accessed by the operator have a longer interval over which they can be implemented to protect the core cooling function. It follows that they are more influential in reducing the likelihood of core damage. The emergency procedures aid the operators in bringing these systems into service when necessary, including the YNPS plant specific Safe Shutdown System (SSS).

The review included conditions associated with inadequate core cooling in which the core exit thermocouples could potentially reach elevated levels, i.e. in access of 1200°F. Under these conditions, the water inventory within the primary system would be localized to the lower half of the core, the lower plenum and a small amount of water in the pump suction piping for each reactor coolant loop. When the core exit thermocouples record temperatures greater than 1200°F, the operator is instructed to start the main coolant pumps (MCP), with the intent being to add the small amount of water to the core inventory and to take advantage of the available secondary side cooling in the steam generators.

At this point in the accident, the temperatures within the core would likely be much higher than 1200°F and initiation of the reactor coolant pumps could transfer hot gases into the steam generators at an accelerated rate. For station blackout like conditions, the steam generator tubes would be protected because counter-current natural circulation would govern the flow of high temperatures gases between the core and the steam generators. This issue of consequential steam generator tube rupture has been investigated (using MAAP) as part of the Seabrook submittal for a reduced emergency planning zone [4-1] by INEL using the RELAP code [4-2] and by the NUREG-1150 expert opinion teams [4-3]. All of these analyses focused on the natural circulation behavior. While the temperature of the tubes increased to about 700 K (800°F), this was not sufficient to cause creep rupture, i.e., about 1100 K (1520°F). However, the temperatures in the upper plenum are well above this level and are separated from the tubes by the single phase (gas counter-current natural circulation flow. The water seal on the suction side of the reactor coolant pumps is an important feature in forcing counter-current flow. The increase in temperature when the flow changes to once through natural circulation is addressed in Ref. [4-1].

This natural circulation flow between the core, the upper plenum and the steam generators is directly dependent on the primary system pressure, i.e. the energy transfer rate would be reduced for all sequences by decreas- ing the pressure. Thus, if such elevated temperatures are observed in the core, it is recommended that the primary system be depressurized such that the energy transfer rate would be reduced. Depressurization could also promote additional water injection from the accumulator or the low pressure injection systems. These are two major reasons for implementing this ac- tion.

If there were essentially no water on the secondary side of the steam generators, starting the main coolant pumps would override the natural circulation flow and rapidly increase the energy transfer rate to the steam generator tubes. Without depressurization and secondary side cooling, the temperature of the tubes would quickly increase to levels where creep rup- ture failure could be anticipated. Since the steam generator tubes are the containment boundary, integrity of the tubes must be protected.

The procedures call depressurization if the available reactor coolant pumps have been started. For the reasons stated above, it is recommended that the procedures be altered to initiate depressurization on elevated core temperatures and instruct the operator to <u>not</u> restart the reactor coolant pumps unless a normal level is measured in the steam generator connected to the main coolant pump being started or until the temperatures have been reduced below this level through water injection to the primary system. Through this approach, the system response will be optimized to protect the primary system pressure boundary, including the steam generator tubes and will focus the operators attention on reducing the primary system pressure to both provide additional water sources to the primary system and to reduce the energy transfer from the high temperature core to other regions in the primary system such as the steam generator tubes.

With respect to accident management, such depressurization instructions are consistent with both accident recovery and accident mitigation. As stated above, this would:

- enhance the potential for system recovery since additional lower pressure water injection sources would be potentially made available,
- the integrity of the steam generator tubes would be protected, and
- the potential dynamic response associated with a postulated reactor vessel failure that could possibly occur later in the accident would be reduced or eliminated.

With respect to the last point, depressurization would substantially mitigate the primary system blowdown and the potential influence on the shield tank surrounding the reactor vessel. This is equivalent to addressing the NRC concern regarding direct containment heating (DCH) by depressurizing the primary system. Therefore, the procedural response already addresses the depressurization issue discussed for DCH and in particular, reduces the influence in uncertainties with respect to in-vessel core melt progression and the mode of RPV failure.

Another section of the procedures to be addressed is the end of ECA-0.0, "Loss of All AC Power". This procedure currently ends with recovery instructions. In keeping with the above discussion, and the current plant changes considered, it would be beneficial to include instructions to depressurize if core temperatures greater than 1200°F are observed.

Question 33

Discuss, in a paragraph or two, the structural analysis based on a failure criterion of 0.9 yield stress, used to estimate the containment overpressure capability. Was it an in-house analysis? Finite element analysis? When was it performed? Have containment modifications been made since the time of the original analysis, and if so, are the results unchanged?

Response 33

Appendix E "Containment Integrity and Leakage Evaluation" of the PSS provides a description of the YNPS vapor container design/construction and the determination of higher pressure capability.

The vapor container is an ASME Code Section VIII Vessel designed by rule. The concept behind the design by rule approach is to design the basic vessel for code stress limits and then provide excess material to compensate for openings and other discontinuities.

To determine the containment over-pressure capacity, the ASME Code equation for a sphere was used, but instead of the code allowable stress, 90 percent of actual material yield stress was used based on material data records.

The justification for use of the failure criterion of 0.9 yield stress is the similar treatment in the YNPS Systematic Evaluation program based on NUREG/CR-0098 "Development of Criteria for Seismic Review of Selected Nuclear Power Plants", June 1978.

This conservative analysis was performed by Yankee in-house in February 1982 and did not use finite element techniques. No containment modifications have been made since the original analyses, which would change the results.

Question 35

What factors influenced the decision to use plant-specific data, generic data, or a combination of both in various applications in PSS?

How would the conclusions of the IPE change if a more up-to-date data base instead of the NPRDS reliability data base were used to provide generic data? What is your basis for this conclusion?

What were the sources of generic error data from which bounding values were derived for estimating human errors? Why were these sources selected?

Response 35

The factors influencing use of plant-specific vs. generic data are documented in the YNPS PSS Section 7. The following is a brief description of major steps in the process of YNPS PSS data base development:

1. Available generic data sources were identified and searched for the most representative. A generic data base was developed from this data.
2. All important components were identified.
3. For important components, plant-specific data was collected, where available, from plant records in the form of number of failures, number of demands or cycles and total operating time.
4. For those components with no plant-specific data, the generic data were taken directly from the generic data base developed for YNPS. For those components with insufficient plant-specific data, as well as those components with sufficient plant-specific data, the historical plant-specific data were used to update generic distribution using Bayes' theorem. Alse see response to question 46 for more detailed information.

The NPRDS reliability data base was used for less than ten percent of the total components listed in the YNPS PSS data base. Using a more up-to-date data base for those components results in minimum change, if any. The basis for this conclusion lies in the fact that 1) the failure rates used in YNPS PSS data base are conservative and would not be significantly affected by using a more up-to-date data base and 2) NPRDS reliability database was not used for any component which shows as significant contributor to the risk.

The source of generic error data was NUREG/CR-1278,"Handbook on Human Reliability Analysis with Emphasis on Nuclear Power Applications," by Swain and Guttman, from April, 1980. This source was selected because at that time, the THERP method was "state-of-the art".

Question 36

In light of our understanding today of human reliability, manufacturing defects and maintenance errors, provide a concise discussion of why common mode/common cause failures were found to be negligible contributors to unavailability/failure in each of the 40 major fault trees. List the component groups subjected to CCF analysis. Provide a concise discussion of the sources of CCF rates used in the IPE.

When was the human reliability analysis first conducted? When was it updated? Was it requantified to account for the change to symptom based procedures? To what extent does it consider common cause? Does it take advantage of recent work on common cause?

Response 36

Common mode/common cause failure were not found to be negligible contributors in some of the major fault trees. On the contrary, they are major contributors in double and triple diesel generators failures and in Reactor Protection System failure in the case of non-cooldown events.

The common cause contribution to the system unavailabilities explanation is given in the next Table 36.1. The component groups subjected to CCF analysis are also listed in the Table 36.1.

At the time the YNPS PSS was completed, insufficient generic data was available for CCF rates. The applied β and γ factors were the result of expert judgement and comparison with other studies. A modified WASH-1400 methodology was applied to RPS dependent failure analysis.

Table 36.1

Common Cause Contribution

System	Common Cause Group	Common Cause Contribution to the System Unavailability	Comments
RPS 1 Cooldown Transients	Detectors Logic Control Rods Breakers	6.6%	Human interaction during the test and calibration of detectors - dominates.
RPS 2 Non-cooldown Transients		100%	
MFW	-	Negligible	"Single" failures are dominant
EFW	Maintenance on piping and valves	7.6%	-
ECCS	-	Negligible	"Single" failures are dominant
CVCS	-	Negligible	"Single" failures are dominant
2 DGs	Diesel Generators	80%	-
3 DGs	Diesel Generators	66%	-
DC buses	Busses/Ventilation	Negligible	Human error is dominant

The human reliability analysis was conducted in the period from 1981 to 1983. Update of the analysis is ongoing as part of the YNPS PSS update. Update of the human actions during the event sequences is based on the symptom based procedures. All action credited in the original YNPS PSS are covered by the new procedures. The YNPS PSS considers common cause in maintenance related human errors (test, alignment, calibration). The original work was completed before the most recent work on common cause. (Also see responses to Questions 37 through 44.)

Question 37

What percent of the core damage frequency was due to human error? What inferences (insights) are drawn regarding the contribution of human error on overall plant risk?

Identify those sequences that, but for low assumed human error rates in recovery actions, would have been above the screening criteria of Appendix 2 of Generic Letter 88-20.

Response 37

The contribution to core damage frequency from the human error is approximately 26% (see Table 37.1). Human error is the second most dominant contributor to the core damage frequency (after the Safety Injection System). The human errors were estimated conservatively since no credit was taken for recovery actions and no credit was taken for multiple operation personnel in the control room. More recent review in conjunction with the update of the PSS by specialists with experience in human reliability analysis has confirmed the conservatism of these results.

Since no credit is taken for recovery actions, there is no conservativism which will influence sequences frequencies reported in the baseline results of the PSS reported in the submittal. (Note: Some recovery actions were credited in the Best-Estimate Analysis, Chapter 13, YNPS PSS.)

Table 37.1

System Contributions to Core Melt Frequency

System (or action)	Contribution to Core Melt Frequency (%)	Human Contribution (%)
HPSI and LPSI Systems	48.8	negligible
Recirculation System	10.1	negligible
Reactor Protection System and Chemical Shutdown System	10.0	3.7
Operator failure to manually depressurize MCS for small LOCAs if HPSI fails*	9.5	9.5
Accumulator	6.2	negligible
Operator Errors in Initiating/ Controlling Feedwater*	5.3	5.3
Failure of MCS Loop Isolation Valves to close during Steam Generator tube rupture plus operator errors in responding to event*	4.6	4.6
Diesel Generators plus Steam-driven Emergency Feed Pump (including operator errors) - Loss of AC -*	2.4	~2.4
Pressurized Thermal Shock induced Reactor Vessel failure due to operator errors during degradation of DC power events*	0.5	0.5
TOTAL	97.4	26

* Human Error Contribution Dominates

Question 38

List the most dominant human recovery actions identified in the Yankee IPE, along with the task analysis performed for each.

What type of human systems analysis was performed to support plant model development and to identify pertinent human task actions for inclusion in the event and fault trees.

What types of task actions (both cognitive and physical) were analyzed as part of each accident sequence? How were they chosen?

Response 38

The most dominant human recovery actions identified in the Yankee IPE, are listed below, together with contribution to core damage frequency. (see Table 37.1, Response 37)

Human Action	Contribution to Core Melt Frequency (%)
Manual Depressurization of MCS following Small LOCA with loss of HPSI	9.5%
Initiation and Controlling of Feedwater (Restart Main Feedwater after trip. Start Emergency Feedwater System.)	5.3%
Isolation of Main Coolant System Loop following Steam Generator Tube Rupture	4.6%
Initiation of Chemical Shutdown	3.7%
Initiation of Steam Driven Emergency Feedwater	2.4%

Task analysis and classification necessary for action quantification is documented in Table 7-6, with reference to Table 7-3, of the YNPS PSS.

The human systems analysis to support plant model development was based on NUREG/CR-1278, "Handbook on Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications," by Swain and Guttman, from April, 1980. Human task actions were identified as part of system/event tree analysis. Operator response for any proceduralized action during response to an initiating event was assessed. Human actions involved:

- Manual actions during the event sequences (manual initiation of the system, actions following malfunction in automatic system). These actions are included in event trees.

- Interaction with equipment during routine plant operation (testing, maintenance, valve alignment, calibration, etc.). These actions are included in fault trees.

No credit was taken for correcting system alignments during an event. All human actions were reviewed by YAEC and plant personnel who were familiar with YNPS operation, procedures and training.

The human actions which were analyzed as part of each accident sequence are listed in Tables 7.4, 7.5 and 7.6 of the YNPS PSS. Their selection is based on YNPS operating procedures. Most of these individual actions contain both a cognitive and an execution element. They are quantified using the "generic" data from NUREG/CR-1278 at two levels: (1) Operator decision to perform action. (2) Actions necessary to perform function given operator decision.

Question 39

Provide a concise discussion to justify the operator actions without procedures for which the IPE takes credit. Quantify their contributions to the likelihood of core damage or containment failure.

Response 39

The selection of the human action which were analyzed as part of accident sequences was based on YNPS operating procedures. Therefore, the IPE didn't take credit for any operator actions without procedure. (Also see Response 44.)

Question 40

What person-centered (e.g., experience, fatigue, stress), task-centered (e.g., training, procedures), and environment-centered (e.g., supervision, team support, organizational support) performance shaping factors (PSFs) were scaled for human task actions included as precursors, initiators, or mediators in the event and fault trees? How were they chosen? What methods were used to scale each PSF? Why were those methods chosen?

What quantification methods (e.g., THERP, HCR, SLIM-MAUD) were used to estimate human errors on task actions selected for analysis? Why were these quantification methods chosen?

Response 40

The human factors analysis performed for the YNPS PSS used techniques which were then "state-of-the-art". The specific Performance Shaping Factors (PSFs) evaluated for the YNPS PSS human analysis were hostile environment, operator stress, parameter display clarity, complexity of manual action and the time available for action. Evaluation forms are available at YAEC. In addition to this, a questionnaire was completed by the control room operators in order to access their opinions about quality of procedures, training, displays, annunciators/alarms, environment, etc. No outliers were uncovered for YNPS. The only PSF applied for quantification of dynamic human actions was stress. Stress levels taken into account were very low, optimum, moderate and high. NUREG/CR-1278 methods were used to scale each stress level (see Tables 7.3 and 7.6, YNPS PSS). (Note: Some other PSFs, e.g., team support, were taken into account when determining stress level by adjusting the stress level downward on level, because of the increased opportunity for supervision and error correction.)

The THERP method was used to estimate human error in YNPS PSS. At that time, the THERP method was "state-of-the-art".

Question 41

Was a sensitivity analysis of human error performed? What characterization or behavioral model of plant personnel was used to identify multiples and dividends of the base or point estimates of human error for the sensitivity analysis?

Response 41

Sensitivity analysis was only performed for a limited number of recovery actions (e.g., recovering of main feedwater because one to two hours are available for the operator to initiate feedwater). The sensitivity analysis is explained in Chapter 13 of the YNPS PSS.

No behavioral model of plant personnel was applied in the YNPS PSS human analysis. No credit was taken for multiple operations personnel in the control room (except in a few cases by adjusting stress level).

Question 42

To what extent were results documented to allow for auditing and/or replicating, or to allow for combining with data from other PRAs?

How was the completeness of the set of human faults verified?

Response 42

Definition and quantification of human actions represented in the event trees is documented in the YNPS PSS (see Table 7.3, 7.4, 7.5 and 7.6). Quantification of human actions which are significant contributors to system unavailabilities is also documented in the Table 7.6. For example, quantification of operator actions necessary to start Chemical Shutdown, Electric or Steam Driven EFW, Aligning ECCS or CVCS to SGs, etc, These actions can be easily audited and/or replicated. Other minor human actions and test and maintenance errors are part of the fault tree system analyses and are documented in the system fault tree analyses files (available at YAEC).

The completeness of the set of human faults was verified during review sessions between the contractor, corporate analysts and key plant staff knowledgeable in each area modeled in the specific fault and event trees. There were individual review sessions for each tree. The plant personnel were trained on the development and background of the trees before the review. Each session was documented by a secretary and recorded on audio tape. The contractor and corporate analysts then completed and corrected the trees and each was reviewed by an independent member of the team, responsible for a different tree, to ensure accuracy.

Question 43

Does the PSS consider maintenance induced events?

Response 43

As stated in response to question 38, test and maintenance induced events are analyzed as human actions during routine plant operation and are modeled as fault tree basic events.

For human recovery actions during routine operation (maintenance, testing, calibration, valve alignment, etc.) a review of plant procedures was performed to determine the practices used to perform the function. Procedure use and type of checkoff, information display and function review process were examined. Using information provided in Table 7.3 YNPS PSS, the HEP was determined and used to quantify associated fault trees.

For many systems an adequate time exists to correct any valve misalignment. However, no credit was taken for corrective actions, which was a very conservative assumption.

Question 44

Provide written assurance that the procedures and operator actions for which the IPE takes credit are in place at Yankee Rowe and that the operators have received training on these procedures.

Response 44

Table 7-4 (page 7-53) presents the "Manual Actions Represented in Event Trees". Each of the operator actions for which the IPE (PSS) takes credit are currently proceduralized at YNPS and the operators have received training on these procedures with the following notations.

The event in Table 7-4 titled "Manually Controlled Main Coolant System Pressure with Letdown, Drains or PORV" was quantified such that no credit was given to use of letdown or drains. Thus, the event actually involves just use of PORV which is proceduralized.

The action to "Reopen Main Steam Line Non-Return Valves to Established Condensor Heat Sink" was modeled as a means of steam removal at the time of the PSS. The current EOPs and model, however, include use of Emergency Atmospheric Steam Dumps, (EASDs), which is proceduralized. Installation of the EASDs was not completed until after docketing of the PSS. There is no significant impact on conclusions of the PSS as a result of the effects of the subsequent installation of the EASDs.

It should also be noted that in Appendix G, "Environmental and DC Power Top Event Reviews", certain operator actions were evaluated if an extended outage or harsh environment was expected. These actions, although not proceduralized, are not significant and are:

Tree	Event	Operator Action
Loss of AC Power	EBF (page G-44)	If an extended outage is expected, the doors to the switch gear room can be opened, and if required, the covers on the electrical enclosure can be opened or removed for electrical equipment for motor driven EBF pumps.
Loss of AC Power	Effect on Control Room (page G-47)	If an extended outage is expected, the operators will minimize any temperature rises by opening outside doors and removing panel covers where possible. Any temperature rise would not cause undo discomfort to the operators. The effect of temperature rise on equipment required in this event tree is minimal.

Question 45

Provide a list of the equipment for which plant-specific data were used in the PSS and provide the plant-specific data failure or initiating event rates particular to the equipment.

Response 45

Tables 7-2 and 5-11 of the YNPS PSS provide component failure rates (both, generic and plant-specific) and initiating event frequencies for all the equipment and initiating event groups modeled in the study. A list of the equipment for which plant-specific failure data was developed has been extracted from the above stated Table 7-2 and is shown in Table 1. The YNPS PSS initiating event frequencies are given in Table 5-11. Section 5.4," Quantification of Initiating Event Frequencies," (page 5-16) indicates that the first 13 event tree initiating event frequencies (i.e., event trees 1-18) used plant-specific data.

Table 1

YNPS PSS Plant Specific Data
(extracted from PSS Table 7-2)

EQUIPMENT	FAILURE MODE	MEAN	VARIANCE
Diesel Generator	Fails to Start Fails to Run	8.61 E-3 1.40 E-3	2.06 E-5 1.68 E-6
Condenser	Loss Vacuum	3.38 E-6	2.40 E-11
Feedwater Heater	Excess Leakage Rupture	4.12 E-5 1.50 E-7	2.38 E-10 4.01 E-14
Piping (per section) Less than 2" 2"- 6" Greater than 6"	 Rupture Rupture Rupture	 2.30 E-7 8.30 E-8 3.00 E-8	 3.21 E-13 4.16 E-14 5.27 E-15
Condensate Pump	Fails to Run	4.58 E-5	2.10 E-10
Circulating Water Pump	Fails to Run	3.06 E-5	1.43 E-10
Feedwater (BF) Pump	Fails to Run	6.49 E-5	4.00 E-10
Charging Pump	Fails to Start Fails to Run	1.78 E-3 1.87 E-4	8.39 E-7 2.97 E-8
Emergency Feedwater (EBF) Pump	Fails to Start	1.07 E-3	6.48 E-7
Service Water Pump	Fails to Run	8.91 E-6	4.84 E-11
LP Safety Injection Pump	Fails to Start	8.38 E-4	3.95 E-7
HP Safety Injection Pump	Fails to Start	8.38 E-4	3.95 E-7
Shutdown Cooling Pump	Fails to Start Fails to Run	1.40 E-3 1.25 E-5	7.28 E-7 7.95 E-11
Battery	All Modes (hr^{-1}) All Modes (d^{-1}) No Output (hr^{-1})	7.20 E-6 3.61 E-4 3.80 E-8	4.42 E-12 1.20 E-8 8.73 E-15
Charger Battery Static (Inverter) Motor-Generator	 All Modes No Output	 1.23 E-5 7.07 E-6	 1.84 E-12 1.78 E-11
Transformers Station Srv. (15-115KV) Main 3Ø (115-242KV) XFER Tie 3Ø (31-72KV) Substation 3Ø (2-30KV) Substation 3Ø (31-72KV) Satation Service (0.48-2.4KV)	 All Modes All Modes All Modes All Modes All Modes All Modes	 6.50 E-7 1.53 E-6 8.22 E-7 6.75 E-7 9.37 E-7 4.07 E-7	 6.00 E-13 6.12 E-13 2.26 E-13 8.44 E-14 2.07 E-13 9.39 E-14

Question 46

Provide a concise description of how plant-specific data were combined with generic data, particularly when there were significant differences between the rates or when there was a statistically significant amount of plant-specific data for a particular component or system.

Response 46

The response to question 35 explains how, in general, plant-specific data were combined with generic data. The following is a brief description of the plant-specific data base development in two particular cases a) when there were significant differences between the rates and b) when there was a statistically significant amount of plant-specific data for a paticular component or system.

a) A careful review of the YNPS PSS data base shows a few components for which generic data rates are significantly different from plant-specific rates. Combining plant-specific data with generic data for these components as well as most other components for which plant-specific rates were developed has resulted in more conservative numbers. Using more conservative rates was one of YNPS PSS data base development objectives because:

   o Using plant-specific data alone to generate failure rate distribution in most cases was resulting in less conservative numbers due to zero or limited number of failures. Having zero or limited number of failure was mainly because of a) size of the plant making it easier to maintain and operate b) size of the components which are generally smaller than industry average c) quality of workmanship and material used at the time of plant construction, and d) excellent plant crew, most of them working at the plant since early operation.

   o The YNPS PSS was developed to be a living document, therefore any data used in this study should be conservative enough to be unaffected when some components are to be replaced.

b) For the same reasons as described in part a), for those components with statistically significant amount of plant-specific data, generic data was used to produce more conservative data.