SEP 1 0 1985

Docket No.: 50-423

Mr. John F. Opeka
Senior Vice President
Nuclear Engineering and Operations
Northeast Nuclear Energy Company
P. O. Box 270
Hartford, Connecticut 06141-0270

Dear Mr. Opeka:

Subject: Request for Additional Information for Millstone Nuclear Power Station, Unit No. 3

Enclosure 1 contains requests for additional information which the staff requires to complete its evaluation of your application for an operating license for Millstone 3.

Enclosure 2 contains meeting summaries also detailing information which the staff requires to complete its evaluation of your application for an operating license for Millstone 3.

Please submit your responses no later than 30 days of the date of this letter.

Enclosure 3 contains the staffs response to your requests for approval of use of 3 Code Cases at Millstone 3.

Enclosure 4 contains information related to the staffs review of the Millstone Nuclear Power Station Emergency Plan, Draft 2 to Revision 0.

For further information or clarification, please contact the Licensing Project Manager, Elizabeth L. Doolittle at (301) 492-4911.

Sincerely,

B. J. Youngblood, Chief
Licensing Branch No. 1
Division of Licensing

cc: See next page

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EDoolittle/mac
9/10/85

LB#1/DL
For BJYoungblood
9/10/85

8509130299 850910
PDR ADOCK 05000423
A PDR

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

Docket No.: 50-423

SEP 10 1985

Mr. John F. Opeka
Senior Vice President
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For further information or clarification, please contact the Licensing Project Manager, Elizabeth L. Doolittle at (301) 492-4911.

Sincerely,

T. M. Novak
For

B. J. Youngblood, Chief
Licensing Branch No. 1
Division of Licensing

cc: See next page

Mr. J. F. Opeka
Northeast Nuclear Energy Company

Millstone Nuclear Power Station
Unit No. 3

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Millstone III
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Project Management Department
Massachusetts Municipal Wholesale
Electric Company
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Regional Administrator
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631 Park Avenue
King of Prussia, Pennsylvania 19406

Mr. Karl Abraham
Public Affairs Office
U. S. Nuclear Regulatory Commission,
Region I
King of Prussia, Pennsylvania 19406

ENCLOSURE 1

REQUEST FOR ADDITIONAL INFORMATION

MILLSTONE NUCLEAR POWER STATION, UNIT 3

NORTHEAST NUCLEAR ENERGY COMPANY

DOCKET NO. 50-423

Licensing Branch No. 1, Generic Letter 83-28

Item 1.1 Post-Trip Review

By letter dated November 8, 1983, you responded to Generic Letter 83-28 with regard to required actions based on generic implications of Salem ATWS events. With regard to Action Item 1.1, "Post-Trip Review Program Description and Procedure," you indicated that an operating procedure containing responses to staff positions would be available for staff review by October 1984. Please provide this procedure so that the staff may complete its review of the Post-Trip Review Program.

REQUEST FOR ADDITIONAL INFORMATION
REGARDING CONTAINMENT ISOLATION PROVISIONS FOR
MILLSTONE UNIT 3

480.0 Containment Systems Branch

480.7 General Design Criteria for the design of containment isolation provisions have been developed with the intent of providing redundant protection against the loss of a containment atmosphere during a loss-of-coolant accident (LOCA). Several incoming lines at Millstone Unit 3 are provided with a single active valve outside containment and a check valve inside containment to satisfy the requirements of the General Design Criteria. For the subatmospheric-type containment, consideration should also be given to the possible loss of the ability to reduce the containment to subatmospheric pressure after a LOCA as a result of a single active failure of the outside valve in those lines which are open to containment atmosphere by design or by the assumed failure of non-seismic Class I equipment. In this regard, provide a discussion regarding the rationale for using simple check valves in certain lines in Millstone Unit 3 in lieu of weight- or spring-loaded check valves as traditionally used in subatmospheric containments. Include a listing of all containment penetrations where in-leakage to the containment is precluded by provisions other than redundant active isolation valves, and a description of the method of maintaining containment vacuum assuming the worst single active failure to occur. For lines that rely on a loop seal to provide the sealing function, or the pressure of a column of water to hold the check valve closed, provide justification that sufficient water is available to provide a sealing function for 30 days after onset of an accident. The effect of the containment temperature transient and piping system pressure surges should be considered.

ENCLOSURE

MILLSTONE UNIT 3
REQUEST FOR ADDITIONAL INFORMATION
CONTAINMENT LINER REVIEW
STRUCTURAL AND GEOTECHNICAL ENGINEERING BRANCH

220.0 Structural and Geotechnical Engineering Branch, Structural Section

Reference

Stone and Webster Engineering Corporation Report NERM-59 "Evaluation of Anchor Stud Spacing, Containment Structure Steel Liner" Millstone Unit 3 copyright 1984.

220.39 In obtaining liner boundary displacements, tension strength of the concrete was assumed to be negligible. Even though such assumption is conservative for most of the reinforced concrete design, it may be unconservative in the liner analysis since concrete together with reinforcing steel acts as a restraint to liner temperature expansion. Please provide a justification for the assumption.

220.40 The worst case loading was selected as combination of temperature differential of 120°F and negative pressure of 6.7psi. It is implied that higher temperature differential with positive internal pressure will produce lower stresses in anchors as well as in liner since pressure will prevent liner from buckling. Provide a justification with a supporting evidence which demonstrates that applied positive pressure is sufficient to increase buckling load of the liner significantly.

220.41 Millstone 3 FSAR criteria (Table 3.8.1J) specify that SSE be included in the liner and anchor analysis. Provide justification for not including SSE.

630.0 Licensee Qualifications Branch

630.12 In Section 13.1.2 of the SER we stated that it appeared "that at least one senior operator on each shift must have at least six months of hot operating experience at a similar type plant or that a Shift Advisor must be provided who has had at least one year of hot operating experience and who is adequately qualified to advise the shift crew." This position is described in Generic Letter 84-16, "Adequacy of On-Shift Operating Experience of Near Term Operating License Applicants." The resumes of the Shift Supervisors in the Millstone Unit 3, FSAR, do not indicate that you will have enough individuals with hot operating experience on a comparable nuclear power plant to staff each shift with individuals that meet the position described in Generic Letter 84-16.

In a letter to Hugh L. Thompson, dated March 19, 1984, Northeast Nuclear Energy Company submitted an operating shift experience survey which contains experience information by job title for the positions of Shift Supervisors, Senior Control Operator and Control Operator. This letter states that Northeast Nuclear Energy Company does not intend to use shift advisors. The survey also indicates that you do not have an adequate number of personnel with hot operating experience to meet Generic Letter 84-16 for each shift.

In a letter to B. J. Youngblood, dated June 24, 1985, Northeast Nuclear Energy Company restated its position that it does not intend to use shift advisors. You also provided additional information on the experience level of proposed operating shifts to justify the position that you meet the intent of Generic Letter 84-16 for the startup of Millstone Unit 3. However, the information presented does not provide adequate detail for us to determine whether each shift meets the intent of Generic Letter 84-16. In order for us to perform our evaluation, we need the following information: (1) resume type information on each SRO for which credit will be taken in meeting Generic Letter 84-16, containing the individuals past experience as an SRO and/or RO by job position (control operator/shift supervisor, or other), (2) the length of time, (3) the reactor at which this experience has been obtained, and (4) the specific shift to which the individual will be assigned. Identification of individuals should be sufficient so that we can correlate past information provided with any new information provided.

260.0 Quality Assurance Branch

260.59 Table 1.4 of the Safety Evaluation Report related to the operation of Millstone Nuclear Power Station Unit No. 3 (NUREG-1031, July 1984) showed "QA program commitments" as confirmatory item 70. By letters to the NRC dated December 10, 1984 and March 1, 1985, the applicant has provided information to resolve confirmatory item 70.

FSAR Amendment 12 added the following alternative to the applicant's commitment to comply with R.G. 1.123:

> Certain standard catalog or non-engineered items may be procured without seller qualification as described in Section 7 of the Millstone 3 Quality Assurance Program Manual referenced in FSAR Section 7.1.2.

We believe it is the applicant's intent to use this alternative during the operations phase of Millstone 3. However, FSAR Section 7.1.2 (and 17.1.2, if there has been a misprint) does not address QA for operations. Further, the quoted alternative references the Millstone 3 Quality Assurance Program Manual which is not reviewed by the Licensing Section of the QA Branch and which can be changed without NRC notification. Therefore, the applicant should describe its controls for the procurement of "certain standard catalog or non-engineered items" in either the Millstone 3 FSAR or in the Northeast Utilities Quality Assurance Program Topical Report (NU-QA-1) which is referenced in FSAR Section 17.2. The controls should address item 7B4 on page 17.1-16 of the Standard Review Plan (NUREG-0800) which states: "For commercial 'off-the-shelf' items where specific quality assurance controls appropriate for nuclear applications cannot be imposed in a practicable manner, special quality verification requirements shall be established and described to provide the necessary assurance of an acceptable item by the purchaser."

260.60

The QA Branch has reviewed the NU August 6 revised response to question 260.58. The review included discussion with involved NRR technical reviewers of SER table 3.2.1. As a result of this review, we are ready to change SER open item 19 to a confirmatory item of limited scope. In its response to question 260.58, NU indicates several times that "FSAR Table 3.2-1 will be revised to include...(specific items)." Since Table 3.2-1 lists Millstone 3 structures, systems, and components which are classified QA Category I, we find such commitments acceptable to close SER open item 19. NU's response to part e of question 260.58, however, states: "FSAR Table 3.2-1 will be expanded to include major components of the SLCRs (supplementary leak collection and release system) and ESF (engineered safety features) filter systems. This is considered a confirmatory item until the table is so expanded and the reviewer verifies that NU's definition of major components brings the table to the comparable level of detail as found in FSARs for other recently licensed nuclear power plants.

Enclosure 1 - ICSB Comments On Millstone Unit 3 Technical Specification

1. We have reviewed the Westinghouse blue prints dated July 31, 1985 which contain preliminary setpoint data for Millstone Unit 3 protection system. Based on this information, we marked up values in Table 2.2-1 reactor trip system instrumentation trip setpoints and Table 3.3-4 engineered safety feature actuation system instrumentation trip setpoints. The marked up copy or Tables 2.2-1 and 3.3-4 is attached. Because these are the preliminary data, the final values may be changed subject to our review of the final setpoint methodology documents.

2. In Table 2.2-1 Notes 1 and 3, the equations for overtemperature ΔT and overpower ΔT are different from FSAR Section 7.2.1.1.2. The applicant should provide the clarification for these discrepancies.

3. Based on a generic letter dated July 24, 1985 from NRR Director to Westinghouse Owner's Group Chairman on Westinghouse Standard Technical Specifications, the following items should be corrected.

   (a) In Table 3.3-1, Item 15b, the minimum channel operable should be "1" instead of "4."

   (b) In Table 3.3-1 Items 11, 12, and 15, the action statement number should be "6" instead of "7."

4. There are some discrepancies on safety analysis limits between the setpoint document and the FSAR assumptions. For example:

	Setpoint Document	FSAR (Table 15.0-4)
(a) Pressurizer low pressure	1845 psig	1840 psig
(b) Steam generator high level	85.7%	90%

It appears that the Technical Specification limit is less conservative than the FSAR limit. The applicant should provide the clarification for these discrepancies and verify that no other discrepancies exist between the setpoint document and the FSAR assumptions.

5. There are some discrepancies on response time limits between the Technical Specification and the FSAR assumptions. For example:

	Tech Spec (Table 3.3-2)	FSAR (Table 15.0-4)
(a) Overtemperature ΔT	4 sec	2.0 sec
(b) Overpower ΔT	4 sec	2.0 sec

It appears that the Technical Specification limit is less conservative than the FSAR limit. The applicant should provide the clarification for these discrepancies and verify that no other discrepancies exist on response time limits between the Technical Specification and the FSAR assumptions.

6. In Table 3.3-9 Remote shutdown system instruments, there are three items which were not listed in FSAR Table 7.4-1, which is supposed to list all

safety-related instruments on Auxiliary Shutdown Panel (ASP). These instruments are source range count rate (2 channels), Intermediate flux (2 channels) and RHR return loop temperature (2 channels). The applicant is required to verify the design and update the FSAR information.

7. In Table 3.3-10 Accident Monitoring Instrumentation, the items should include all the Type A variables. Three items should be added: (a) Containment Hydrogen Monitor, (b) RCS Subcooling Monitor, and (c) Neutron Flux. A marked up copy of Table 3.3-10 is attached.

8. We have not completed our review on N-1 loop operation. Additional comments on Technical Specification for N-1 loop operation will be provided later.

ENCLOSURE 2

MEETING SUMMARIES FOR 4 AUDITS AT MILLSTONE 3

MILLSTONE NUCLEAR POWER STATION, UNIT 3

NORTHEAST NUCLEAR ENERGY COMPANY

DOCKET NO. 50-423

SUMMARY OF THE STAFF'S AUDIT

OF THE MILLSTONE 3 SPDS

JULY 30-31, 1985

On July 30 and 31, 1985 the staff performed an audit of the Millstone 3 SPDS. The purpose of the meeting was to attempt to resolve outstanding questions regarding the Verification and Validation (V&V) program, to confirm that the V&V program is being correctly implemented, to audit the V&V results to date, and to audit the installed SPDS at Unit 3.

The NRC team leader, George Lapinsky of the Human Factors Engineering Branch, was assisted in the audit by consultants from Lawrence Livermore National Laboratory and EG&G, San Ramone. A list of attendees is included here as Enclosure 1.

On July 30, 1985 representatives of the Northeast Utilities presented information regarding the Millstone 3 SPDS design and implementation. Copies of Vu-graphs used during the presentations are included here as Enclosure 2. In addition, the audit team examined a number of reports documenting the V&V program. A full listing of those documents will be provided in the staff's audit report at a later date.

On July 31, 1985 the staff witnessed a demonstration of the SPDS functions and conducted an audit of the display page formats. In the afternoon the staff toured the Unit 3 control room. The audit concluded in an exit briefing, at which time the staff presented the following observations, notes, and conclusions:

1. In order to complete its review, the staff needs the following items for confirmatory review,

    a. Integrated Test Results (in summary form)
    b. Man-in-the-Loop Test Results
    c. Training Plan and Schedule
    d. List of scenarios used in Man-in-the-Loop Testing
    e. Results of the 100 Hour Test
    f. Commitment to a formal procedure for assuring that the SPDS and the plant Emergency Operating Procedures are consistent with each other.

2. Both the design program and the V&V program appear to be well planned and executed. The single exception was the lack of a test procedure for testing worst-case computer loading and possible effects on the operation and response time of the SPDS.

3. The SPDS interface devices and display formats were simple and easy to use and understand. The staff's audit discovered only one area of concern -- the system is vulnerable to disruption from outside the control room because of a simulation capability that can be initiated from the programmer's console. As presently designed, personnel at the programmer's console could conceivably put all five control room

consoles into a simulation mode without the knowledge and consent of the control room operators. In addition, the simulation mode displays were not distinctively identified. The simulation mode is necessary for the man-in-the-loop testing that is yet to be done. However, the staff feels that the method of identifying simulated data should be improved (red, flashing identifier) and access to the simulation mode should be strictly controlled, e.g., by password, administrative control, and keylocking scenario tapes. In addition, the staff suggested that once the need for the simulation capability no longer exists, it should be deleted from the system.

4. The staff and NU personnel discussed parameter selection and the acceptability of using non-SPDS information to supplement the current Millstone 3 parameter set. The only conclusion drawn was that further discussion was necessary and that a representative of the Procedures and Systems Review Branch needs to provide further input before a decision can be made.

ENCLOSURE 1

SPDS AUDIT JULY 29 1985

ENTRANCE MEETING

G. W. Lapinsky	NRC/HFEB
G. Johnson	LLNL
W. O. Wade	EG&G/LLNL
G. Caccavale	GE&C
T. Rebelowski	NRC/SRI MILL 3
R. R. Viviano	NUSCO Asst. Proj. Engr. MP-3
J. J. Festa	NUSCO, PM
K. J. Spitzner	NUSCO, PCE
V. I. Swisher	GEE, Consultant
E. Babij	NUSCO, PCE
S. J. Sorrentino	Process Computer Engineer
D. Wilkinson	NUSCO, Consultant
P. Blanch	NUSCO, I&C
P. Blasioli	NUSCO, Licensing
M. Kai	NUSCO, REB
A. Stave	NUSCO, NSE/Human Factors
P. Callaghan	NUSCO, NSE/Human Factors
P. Slowik	NUSCO, IRD-PCE
M. Blancafcor	NUSCO, Special Studies
A. J. Masto	NUSCO, GEE-PE-CY
M. J. Whitelaw	NUSCO, GEE-PE-MP2

Millstone 3 Enclosure 2 Handout @ SPDS audit 7/29/85

SPDS Verification & Validation

V & V Plan	D. Wilkinson
Design Documents Verification	S. Sorrentino
Specification Verification	P. Blanch

V & V Responsibilities

V & V Team

Develop V & V Plan

Review/Audit Specific V & V Activities

Provide Overall Guidance on V & V

Functional Organization

Prepare V & V Procedures

Conduct V & V In Assigned Areas

V & V Team - Adminstrative

- Independent Makeup

- Functional Interfaces

- Team Records/Files

- Policy Interface - VP Nuclear Operations

SPDS V & V Team

John Becker	Operations Engineer	MP-2
Mike Bigiarelli	Sr. Reactor Engineer	MP-1
Rick Borg	Computer Engineer	CY
Sonny Sorrentino	Supervisor	Computer Engineering
Terry Mulder	Sr. Engineer	Nuclear Operations
Wolf Schubler	Computer Engineer	MP-1
Zen Ufnal	Computer Project Engineer	MP-1
Dan Wilkinson	Consultant	

V & V OBJECTIVES

Meet Regulatory Requirements

Assure complete system documentation of SPDS implementation

Early identification and correction of system or documentation deficiencies

Assure design reflects requirements

Assure installed system meets functional requirements.

V & V SCOPE

- Verification of System Documentation

  Functional Specification

  H/W&S/W Design Requirements

  H/W&S/W Design Documents

- Validation Testing

  Factory Testing

  Installation and Acceptance Testing

  Man-in-the-Loop Testing

V & V ELEMENTS

Review System Functional Specifications

- Appropriate Functions
- Consistent with requirements

Evaluation of Implementation Steps

- Correct Translation from previous step

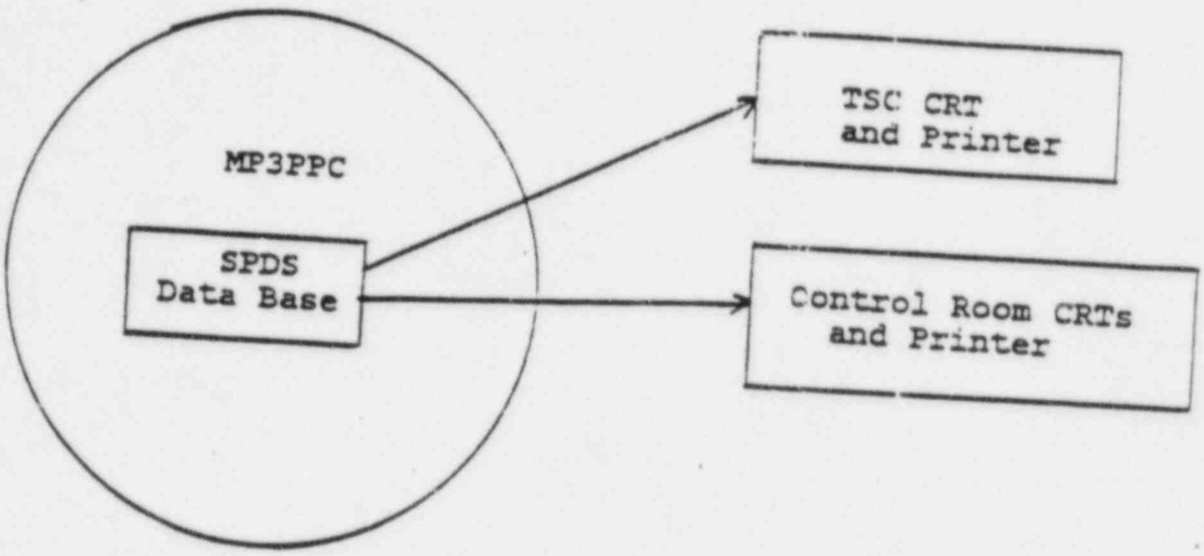
Adequate System Documentation

Adequate Configuration Management

Handout @ SPDS audit
7/29/85

MILLSTONE POINT - 3

PLANT PROCESS COMPUTER/SPDS



SPDS PROJECT OVERVIEW

SAR

H/W DESIGN, PROCUREMENT, INSTALLATION & TESTING

FUNCTIONAL SPECIFICATION

DISPLAY DEVELOPMENT

SOFTWARE REQUIREMENTS DOC.

SOFTWARE DESIGN DOCUMENTS

SOFTWARE DEVELOPMENT

SOFTWARE TEST PROCEDURES

S/W INST. & MODULE TESTS

INTEG. TEST

HF AUDIT

VERIFICATION & VALIDATION AND AUDITS

MAN-IN-THE-LOOP VALIDATION PLAN SCENARIO DEV. PROGRAM VALIDATION

HOT FUNCT. TESTS

AUG. 1, 1985

SPDS PROJECT ORGANIZATION

V & V TEAM

DAN WILKINSON

MP-3 PROJECT ENGINEER

JOHN FESTA

LICENSING

PAUL BLASIOLI

MP-3 OPERATIONS

KEN BURTON
GENE OLSEN

GENERATION ELECTRICAL ENGINEERING

PAUL BLANCH
MARIO BLANCAFLOR
VICKI SWISHER
JOHN MEYERS

PROCESS COMPUTER ENGINEERING

KLAUS SPITZNER
PHIL SLOWIK
GENE BABIJ
SONNY SORRENTINO

MP COMPUTER SERVICES

CHUCK SCOPELITIS
BRUCE BODNER

NUCLEAR SAFETY/ HUMAN FACTORS

PAUL CALLAGHAN
AL STAVE

REACTOR ENGINEERING

MIKE KAI

TRAINING

JOHN SAUGER

SPDS
FUNCTIONAL SPECIFICATION

- SPDS Scope
- General Requirements
  - Flexibility
  - Quality
  - Availability
  - Historical Capabilities
- CSF Algorithms
- Signal Validation Algorithms
- Displays
- Testi
- Documentation

Millstone 3

hardout @ SPDS aud
7/29/85

INDEPENDENT VERIFICATION OF FUNCTIONAL SPECIFICATION

- SPDS FUNCTIONAL SPECIFICATION SP-EE-149A REV. 2

- INDEPENDENT REVIEWER

- QUALIFIED - PRIOR SPDS EXPERIENCE

- VERIFICATION GUIDELINE

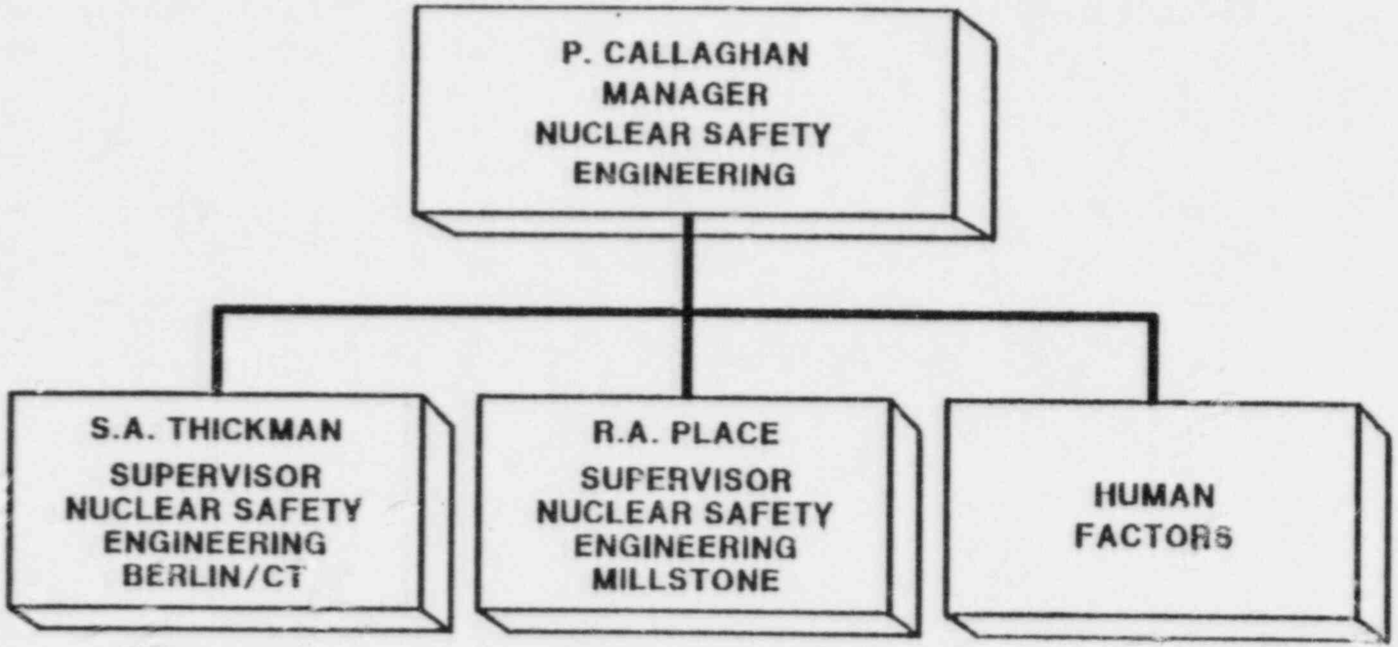
- NUREG 0737 SUPPLEMENT-1 BASIS

- NUREG 0696, 0700, 0800, 0835 USED TO AID INTERPRETATION OF 0737

RESULTS:

FUNCTIONAL SPECIFICATION INCORPORATES ALL SPDS REQUIREMENTS OF SUPPLEMENT-1

NUCLEAR SAFETY ENGINEERING SECTION



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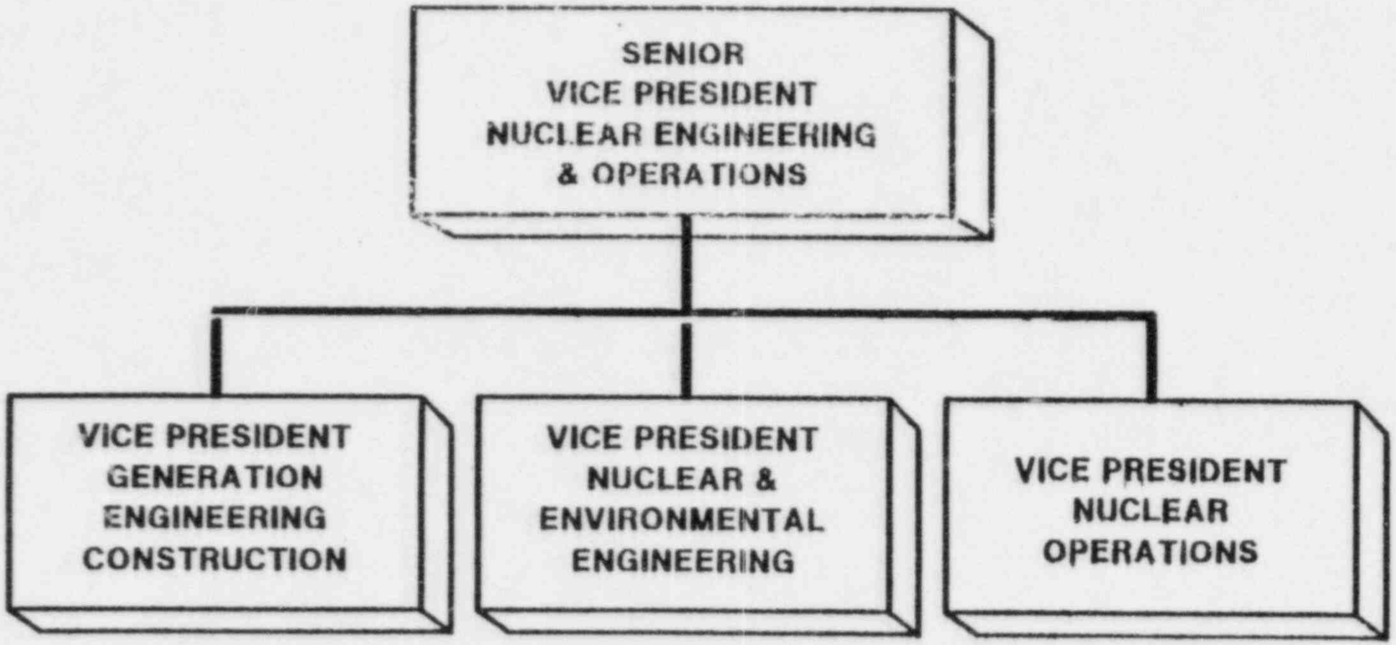

SENIOR VICE PRESIDENT NUCLEAR ENGINEERING & OPERATIONS
VICE PRESIDENT GENERATION ENGINEERING CONSTRUCTION
VICE PRESIDENT NUCLEAR & ENVIRONMENTAL ENGINEERING
VICE PRESIDENT NUCLEAR OPERATIONS

NUCLEAR SAFETY ENGINEERING FUNCTIONS

- LER EVALUATIONS (INTERNAL)
- PIR EVALUATIONS
- SEE-IN PROGRAM EVALUATIONS
- REVIEW PLANT DATA FOR TRENDS
- NUCLEAR REVIEW BOARD (NRB) SUPPORT
- HUMAN FACTOR ASSESSMENTS
- INDEPENDENT SAFETY ENGINEERING GROUP FUNCTIONS

NU

HUMAN FACTOR ACTIVITIES

- O CRDR ACTIVITIES - MP-3, MP-2
- O SPDS ACTIVITIES - MP-3, OPERATING UNITS
- O CR/ERF LAYOUT/MMI
- O HUMAN PERFORMANCE/ERROR EVENTS
- O TRAINING (HUMAN FACTORS)
- O CRT DISPLAY DEVELOPMENT GUIDELINES
- O PDCR REVIEWS -
- O INDEPENDENT ASSESSMENTS
- O INDUSTRY ACTIVITIES
    - O EPRI -
    - O IEEE -
    - O INPO -
    - O WOG -
    - O BWRCG -

PERSONNEL

ALLAN STAVE - 30 YEARS EXPERIENCE
- MASTERS DEGREE PSYCHOLOGY
- PHD CANDIDATE INDUSTRIAL PSYCHOLOGY
- JOINED NU - 1983
- BWROG DISPLAY DEVELOPMENT ACTIVITIES
- IEEE - SC-7

EVERETT PERKINS - 9 YEARS EXPERIENCE
- NUCLEAR ENGINEERING DEGREE
- HUMAN FACTORS COURSES UNIV. MICHIGAN - HARVARD -

RAYMOND SABEH - 30 YEARS EXPERIENCE
- MASTERS DEGREE INDUSTRIAL PSYCHOLOGY
- NU EMPLOYEE 1981 - 1983
- WOG - SUBCOMMITTEE CHAIRMAN GENERIC SYSTEM FUNCTION AND TASK ANALYSIS

MP-3 SPDS PROJECT

- O NSE/HUMAN FACTORS PROVIDED THE LEAD IN DESIGN, REVIEW AND APPROVAL OF PHASE I DISPLAYS.

- O NSE/HUMAN FACTORS PROVIDED SUPPORT FOR:

  - O PROGRAM DEVELOPMENT
  - O SPECIFICATION PREPARATION
  - O VERIFICATION AND VALIDATION
  - O MAN IN THE LOOP VALIDATION
  - O MAN MACHINE INTERFACE
  - O INDEPENDENT ASSESSMENT

OUR DIRECT INVOLVEMENT IN THE SPDS, CRDR AND EMERGENCY RESPONSE FACILITIES PROVIDES FOR A UNIFORM INDEPENDENT ASSESSMENT OF THE HUMAN FACTORS ASPECTS OF THE SUPPLEMENT I TASKS.

SUMMARY

DISPLAY DESIGN METHODOLOGY

SPDS USE PHILOSOPHY

DISPLAY DESCRIPTION

DISPLAY DESIGN METHODOLOGY

TASK DEFINITION

EQUIPMENT CONSIDERATIONS

VIEWING ENVIRONMENT

DISPLAY CONCEPT DEVELOPMENT

DISPLAY DESIGN

DESIGN REVIEW

SPECIFICATION INPUTS

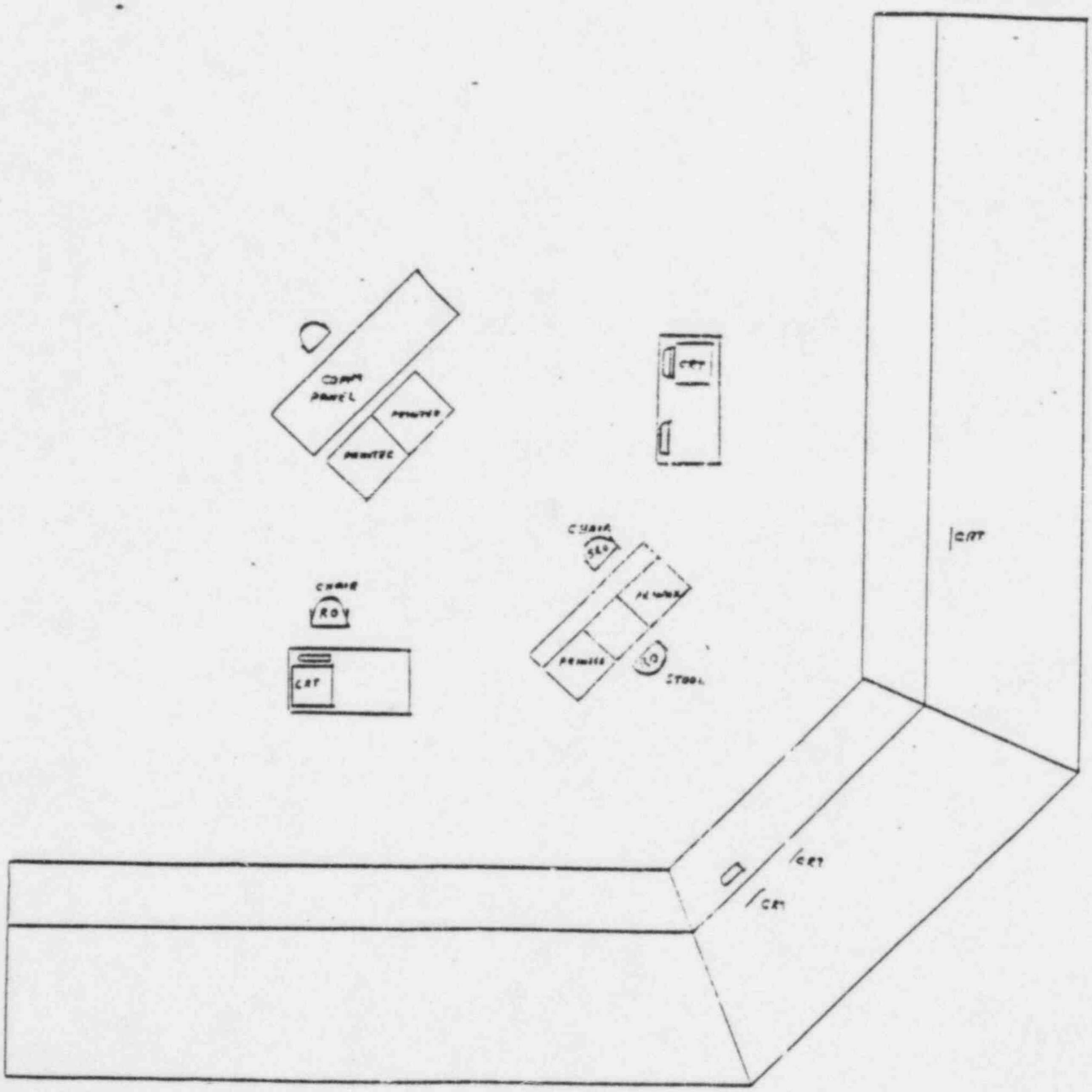
SPDS USE PHILOSOPHY

FOR SCO USE

INDEPENDENT CSF MONITORING

REQUIRED INFORMATION

CSF Challanged?
How did computer decide?
Raw sensor readings?



MP 3 CONTROL ROOM

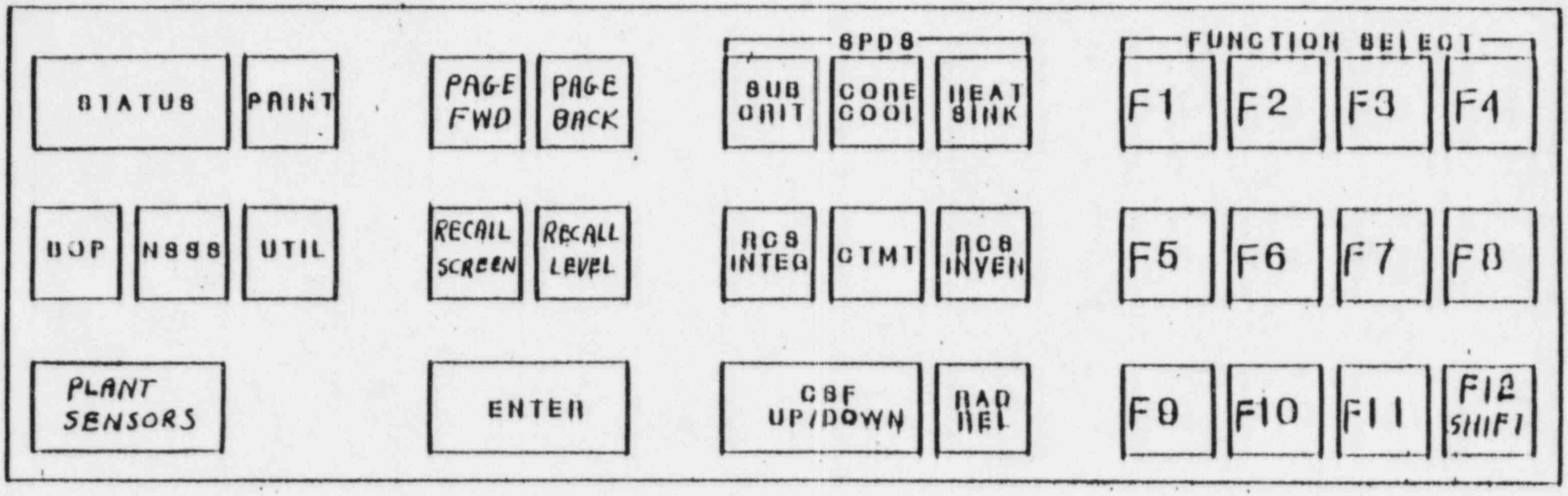


FIGURE 1 PROCESS COMPUTER FUNCTION KEYBOARD

SUBCRITICALITY CSF

F1 STATUS TREE	F2 STATUS VARIABLES	F3 SUPPORTING VARIABLES	F4 TIME HISTORY
F5	F6	F7	F8
F9	F10	F11	F12

FIGURE 3,0

Millstone 3 handout @ SPDS audit 7/29/85

SPDS MAN-IN-THE-LOOP VALIDATION

- o GOALS AND OBJECTIVES
  - DEMONSTRATE SPDS AIDS IN MONITORING THE STATUS OF CSFs
  - DEMONSTRATE SPDS DISPLAYS ARE COMPATIBLE WITH EOPs
  - SPDS IS TECHNICALLY CORRECT, UNDERSTANDABLE, JOB COMPATIBLE, COMPLETE
- o VALIDATION METHOD
- o SCENARIO SELECTION
  - SCENARIO SELECTION CRITERIA
  - SCENARIO DEVELOPMENT
- o RESOURCES
  - MANPOWER
  - EQUIPMENT
  - TRAINING

- Checklists
  - Biographical Data Sheets
  - Observer Checklist
  - Operator Questionnaires

- Performance of Validation
  - Dry Run
  - Man-in-the-Loop Validation

- Data Reduction and Analyses
  - SPDS Response
  - Operator/SPDS Interactions
  - Operator Opinion

- Documentation

SPDS Audit Agenda

July 29, 1985 (Location: Room S-105 in Berlin)

9:30 - 9:45	Introduction (P. A. Blasioli) Entrance Briefing (NRC)
9:45 - 10:15	Overview of SPDS Program (P. M. Blanch)
10:15 - 10:45	Project Documentation Overview (P. M. Blanch, K. J. Spitzner)
10:45 - 11:30	Verification & Validation (V&V) Overview (C. D. Wilkinson) Design Document Verification (S. J. Sorrentino) Functional Specification Verification (P. M . Blanch)
11:30 - 12:00	Human Factors (P. Callaghan/A. M. Stave)
12:00 - 1:00	Lunch
1:00 - 1:30	Man-in-the-loop (V. Swisher)
1:30 - 2:00	Signal Validation Methodology (M. M. Biancafor)
2:00 - 2:30	Availability Calculation (P. P. Slowik)

July 29, 1985 (Location: Building 3333 and N-3 in Berlin)

2:30 - 4:30	Review of V&V Documentation

July 30, 1985 (Location: Building 3333)

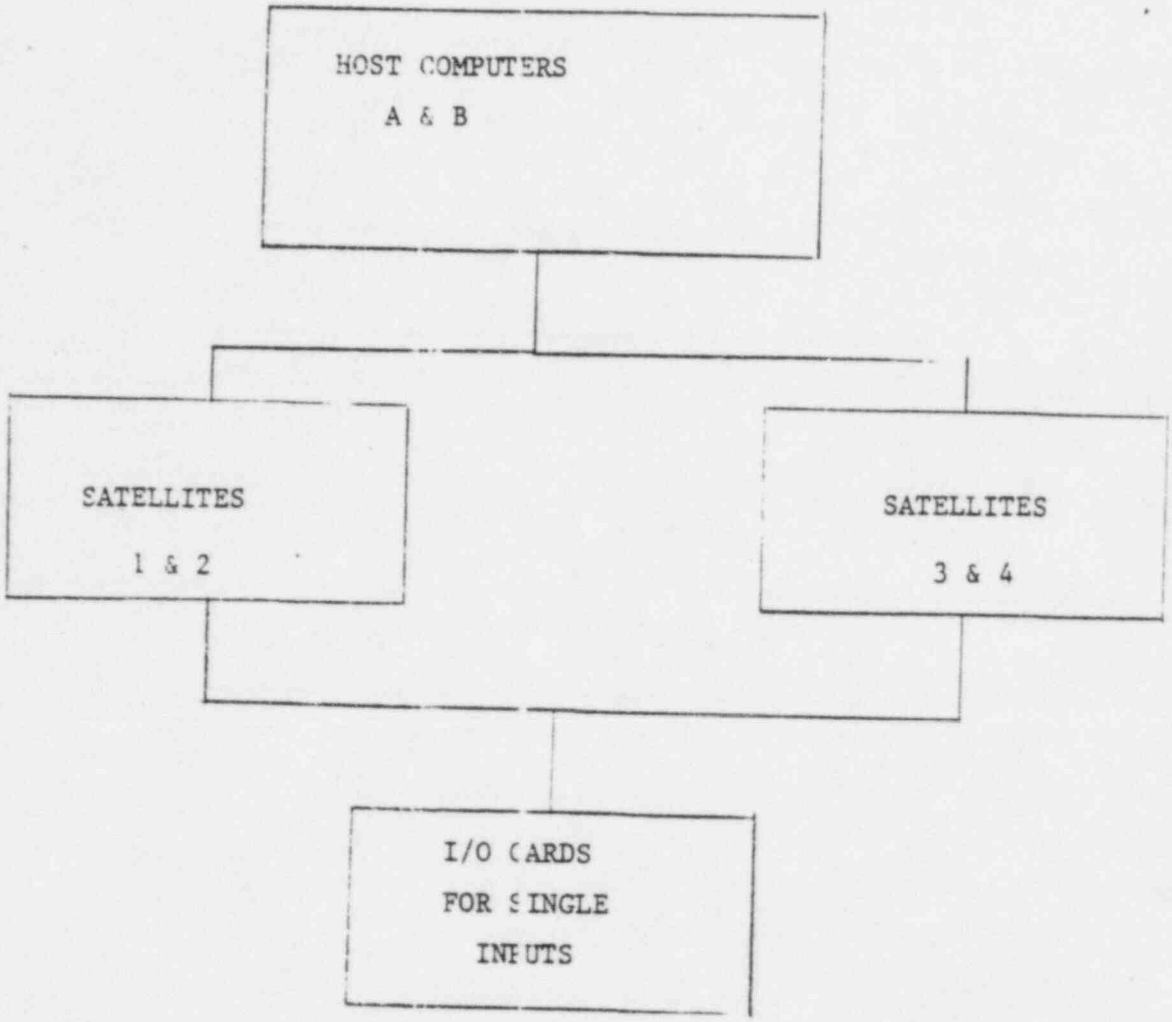
9:00 - 12:00	Demonstration of SPDS NRC Audit of Displays, etc.

July 30, 1985 (Location: Millstone Unit No. 3)

2:00 - 3:00	Tour of Control Room
3:00 - 3:30	NRC Caucus
3:30	Exit Briefing (Room NB-3)

Millstone 3 Handout @ SPDS audit 7/24/85

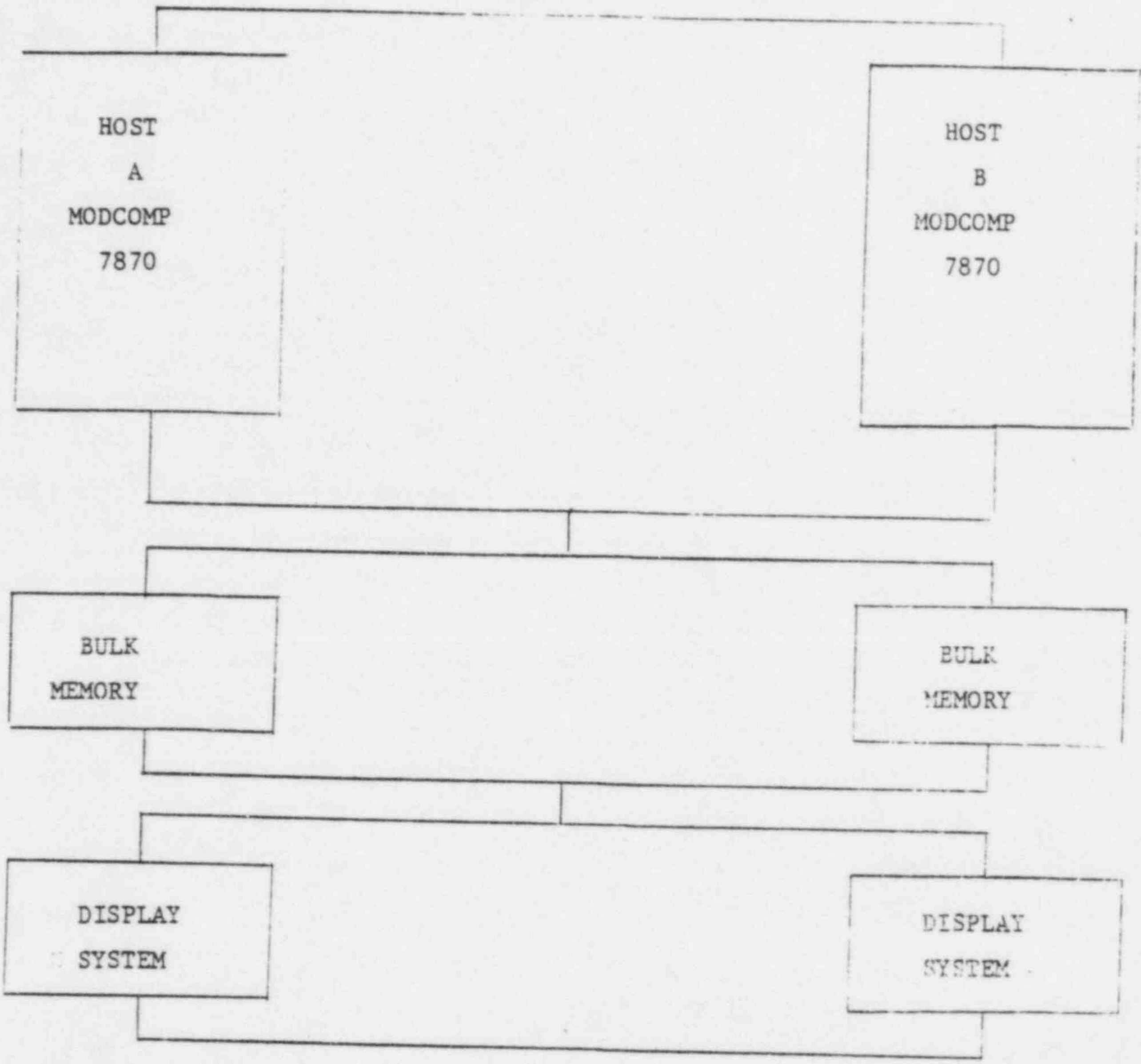
SPDS AVAILABILITY DIAGRAM



AVAILABILITY %

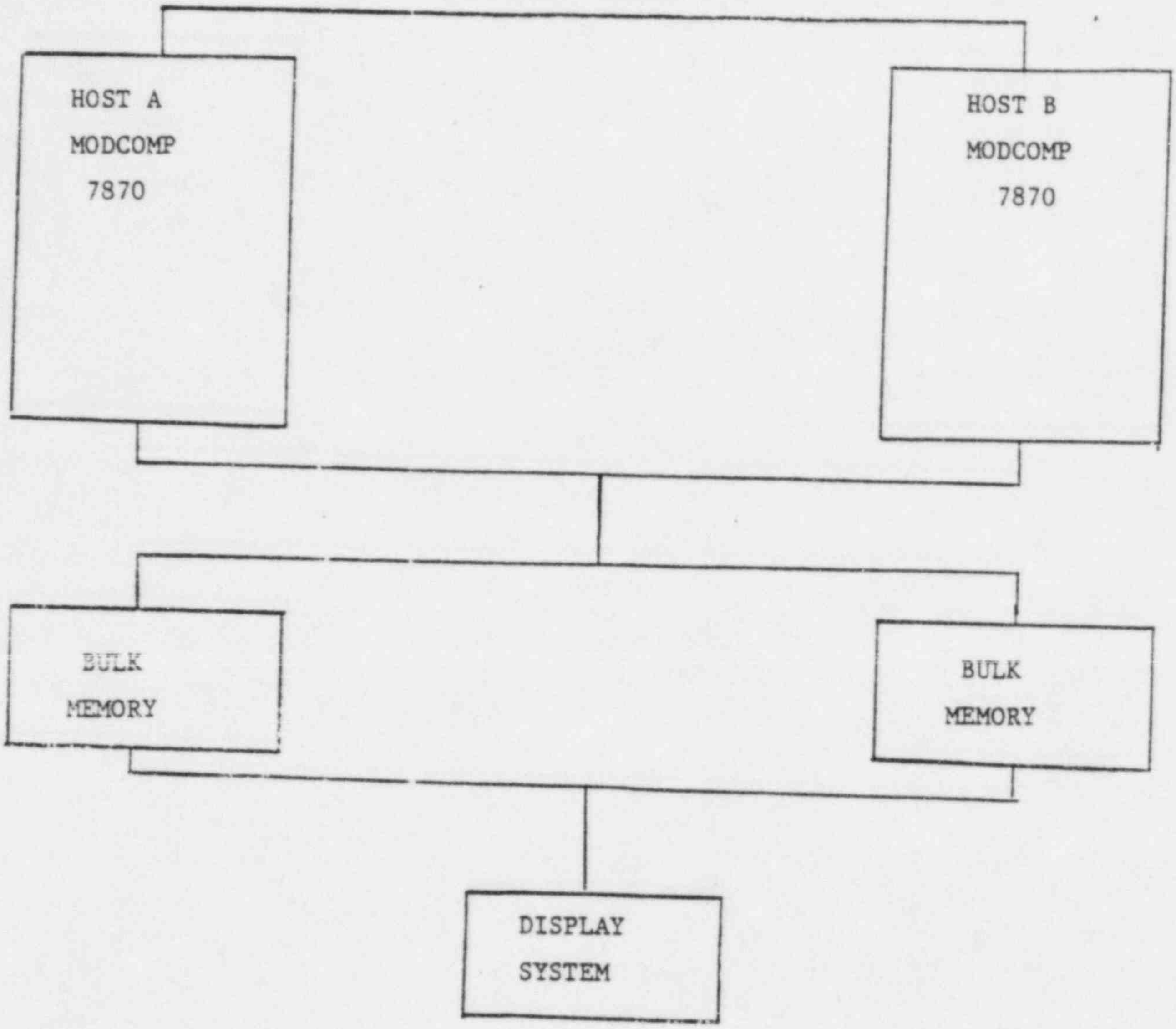
	MANUFACTURES MTTR	MTTR = 8 HR
SPDS	99.975	99.543
TSC	99.939	99.103

HOST COMPUTERS



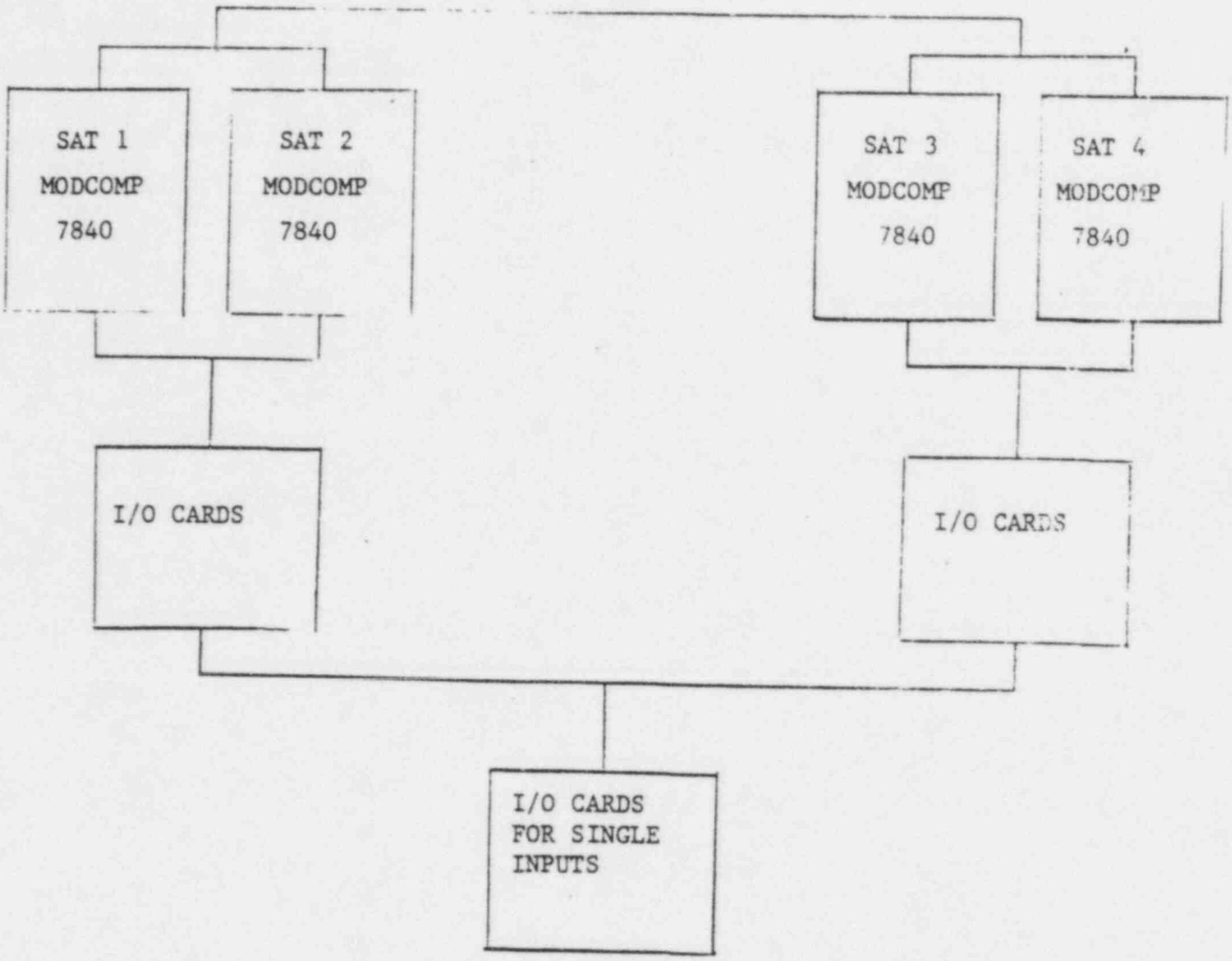
	AVAILABILITY %	
	MANUFACTURE'S MTTR	MTTR = 8 HR
HOST COMPUTERS	99.998	99.943

HOST COMPUTERS (TSC)



	AVAILABILITY %	
	MANUFACTURE MTTR	MTTR = 8 HR
HOST COMPUTER (TSC)	99.62	99.501

SATELLITE COMPUTERS



	AVAILABILITY %	
	MANUFACTURE MTTR	MTTR = 8 HR
SAT 1 & 2	99.861	98.052
SAT 3 & 4	99.898	98.635
SINGLE INPUT I/O CARDS	99.977	99.627

SUMMARY OF THE NRC STAFF
POWER SYSTEMS BRANCH AUDIT
AT MILLSTONE 3

April 10 and 11, 1985

A site audit was held at the Millstone Nuclear Power Station, Unit 3 on April 11 and 12, 1985. Representatives from Northeast Utilities Service Company and Stone and Webster Engineering Corporation attended. The purpose of the audit was to view the installation and arrangement of electrical equipment and cables and to discuss matters related to the electrical power systems at Millstone 3 Site.

A status summary of all items discussed and a summary of the discussion of each item is enclosed.

ENCLOSURE

SUMMARY OF THE NRC STAFF

POWER SYSTEMS BRANCH AUDIT

AT MILLSTONE 3

APRIL 10 AND 11, 1985

A status summary of all items discussed at the Millstone Unit 3 site visit meeting is given below:

SER Items	Status
OI #2	applicant action required
OI #13	applicant and NRC staff action required
CI #43	Closed
CI #44	applicant action required
CI #45	applicant action required
CI #46	applicant action required
CI #47	Closed
CI #48	Closed
CI #49	NRC staff action required
CI #50	applicant action required
CI #51	NRC Region action required
CI #52	applicant action required
CI #53	NRC staff action required
CI #54	NRC staff action required

New Items	Status
NI #1	Closed
NI #2	Closed
NI #3	Closed
NI #4	Closed
NI #5	NRC staff action required

A summary of the discussion for each of the above items is presented below:

OI #2 (SER Section 8.3.1.11)

Additional analysis or test data was not available to demonstrate the diesel generator's design capability (in accordance with the requirements of GDC 17) to accept design load after operation at no load. This item remains open.

OI #13 (SER Section 8.3.3.)

The applicant indicated that the design for the PORV and Block valve power supplies is being changed such that it will be in conformance with the staff position stated in the SER. This item remains open pending staff review of the proposed change.

CI #43 (SER Section 8.2.1.1)

This item has been closed by SSER #1.

CI #44 (SER Section 8.2.2.1)

Design description, documented by letters dated May 3, 1983 and August 29, 1983 (response to NRC question number 430.4), has not been incorporated into Section 8.2 of the FSAR. This item remains confirmatory.

CI #45 (SER Section 8.2.2.2)

The offsite power cables between the normal station service transformer and the Class 1E buses and between the Reserve station service transformer and the Class 1E buses were traced as part of the confirmatory site visit. Based on the cable tracing, the staff concludes that the cables are routed in accordance with Millstone separation criteria defined in the FSAR with the following exception. The A division cables from the normal station service transformer are not routed in embedded conduit when they pass through the B division cable tunnel shown on figure 8.3-7 of the FSAR. This item will become open.

CI #46 (SER Section 8.2.2.5)

Description of surveillance, operability requirements, and analysis presented by letter dated June 12, 1984 has not been documented in response to question 430.7 or in Section 8.2 of the FSAR. The applicant indicated that the analysis presented for the protection scheme is being changed. When the proposed change is received, this item will become open.

CI #47 (SER Section 8.2.2.6)

This item has been closed by SSER #1.

CI #48 (SER Section 8.3.1.2)

This item has been closed by SSER #1.

CI#49 (SER Section 8.3.1.2)

The staff confirmed that information presented in a letter dated June 12, 1984 was documented in the FSAR by amendment 9. This item is closed.

CI #50 (SER Section 8.3.1.3)

The NRC staff reviewed a draft logic diagram for the second level of undervoltage protection. The staff indicated that the logic diagram was consistent with their description presented by letter dated June 12, 1984 and is acceptable. Amendment to the FSAR showing incorporation of the June 12, 1984 description in Section 8.3 of the FSAR was not available. This item remains confirmatory. In addition, electrical schematic drawings of the second level of undervoltage protection were unavailable for NRC staff review. This item remains confirmatory.

CI #51 (SER Section 8.3.1.5)

The NRC staff reviewed the results of the Millstone voltage drop analysis and found the results to be satisfactory. Based on these results, the applicant stated that grid voltage limits are 331 kV to 362 kV for operability of the offsite circuits through the normal station service transformer and 334 kV to 362 kV for operability of the offsite circuits through the reserve station service transformer. The staff stated that these grid limits for operability will be included in the Millstone technical specifications. The staff also stated that we would request the NRC Regional office to verify the test results which substantiate the analysis. This item remains confirmatory pending completion of the Region verification.

CI #52 (SER Section 8.3.3.3)

Amendment to the FSAR was unavailable for NRC staff review. This item remains confirmatory.

CI #53 (SER Section 8.3.3.3.10)

The staff evaluation (of the applicant's test results and design provisions to ensure that non Class 1E circuits are sufficiently isolated and will not cause unacceptable influence on any Class 1E circuits) was inadvertently left out of the staff's SER. SER Section 8.3.3.3.10 will be corrected as follows and will be included as part of our next supplemental input for the Millstone plant.

8.3.3.3.10 Transformer Used as an Isolation Device

By Section 8.3.1.1.2 (Item 3) and Figure 8.3-3 of the FSAR, the applicant indicated that non Class 1E nuclear steam supply system loads are connected to the Class 1E 120V vital ac buses through transformers that are qualified as isolation devices. The staff disagrees that the transformers are qualified isolation devices.

By letters dated August 29, 1983 and June 12, 1984, the applicant provided results of tests and design provisions to ensure that non Class 1E circuits are sufficiently isolated and will not cause unacceptable influence on any Class 1E circuit.

These results of tests and design provisions included the following items:

1. The transformer is Class 1E and is protected by a fuse and a circuit breaker that are physically separated.

2. The circuit from the transformer to the loads are protected by transformer output fuses and feeder circuit fuses.

3. The output circuit of the transformer is run in dedicated conduit to the 120 volt non Class 1E distribution panel.

4. The loads are limited to control and instrument loads.

5. The circuits from the 120 volt non Class 1E distribution panel are routed in raceways designated nonsafety; thus, circuits associated with redundant safety division are intermixed. The staff found this aspect of the design to be unacceptable.

6. The test report, with respect to a bolted short on the output of the transformer demonstrated that associated Class 1E circuits and power supplies were not adversely affected.

7. The test report, with respect to hot short, indicated electrical transients may adversely affect Class 1E circuits. The staff found this aspect of the design to be unacceptable.

Subsequently, by letter dated July 18, 1984 the applicant committed to perform additional testing to demonstrate the hot short capability of the isolation transformer. Given the reverse assumption that the isolation transformer passes the additional testing, the staff concludes that the above design provisions and the isolation capability of the transformer meet the guidelines of RG 1.75 and is, therefore, acceptably. Given the assumption that the isolation transformer fails to pass the additional testing, the applicant has committed to either route the associated cables independently so that redundant associated cables are not intermixed or to remove the subject non Class 1E circuits from Class 1E power sources. The staff concludes that either of these commitments would provide adequate protection for and independence between Class 1E circuits and are, therefore, acceptable. Pending staff review of the results of the additional testing to be performed, this item is acceptably resolved.

During the site visit the applicant provided Appendix B of test No. T3345BP002. The NRC staff found the results to be satisfactory. This item is, therefore, considered complete.

CI #54 (SER Section 8.3.3.3.16)

The staff confirmed that information presented by letter dated June 12, 1984 or by proposed amendment 8 to the FSAR was included in amendment 8 to the FSAR dated May 1984. This item is, therefore, considered complete.

NI #1 (SER Section 8.3.1.7)

This item was identified as confirmatory in Section 8.3.1.7 of the SER but was not assigned a confirmatory item number in Table 1.4 of the SER. The staff reviewed with the applicant S&W drawings numbered: 12179-ESK-8KK (Revision 2), 12179-ESK-8KF (Revision 6), 12179-ESK-5DS (Revision 11), and 12179-ESK-8KG (Revision 6). The staff confirmed that the design for bypassing diesel generator protective relaying under accident conditions meets the staff position. This item is, therefore, considered complete.

NI #2 (SER Section 8.3.1.12)

The staff reviewed with the applicant S&W drawing number 12179-ESK-5DS (Revision 11). The staff confirmed that the Millstone diesel generator (when in the test mode and paralleled with offsite power) are capable of responding to an accident signal. This item is, therefore, considered complete.

NI #3 (SER Section 8.3.3.3.2)

This item was identified as confirmatory in Section 8.3.3.3.2 of the SER but was not assigned a confirmatory item number in Table 1.4 of the SER. The staff

reviewed the cable's color code identification to determine that the 15 ft. marking interval is sufficient to facilitate visual verification that the cables are installed in conformance with separation criteria. Because the majority of cables were continuously marked (solid color cable) such that they contrasted with black cables marked at approximately 15 foot intervals, the staff found that visual verification was not a problem. This item is, therefore, considered complete.

NI #4 (SER Section 8.3.3.3)

The cable routings that were traced as part of the confirmatory site visit included power cables associated with the two motor driven auxiliary feedwater pumps and control cables associated with the steam supply valves for the steam driven auxiliary feedwater pump. Based on the cable tracing, the staff concluded that these redundant cables are routed in separate plant areas separated by walls. The staff, therefore, confirmed that these cables are routed in accordance with Millstone separation criteria and meet the requirements of IEEE Standard 384. In addition, the staff confirmed that this cable routing meets the requirements of Appendix R to 10 CFR 50. This item is, therefore, considered complete.

NI #5 (SER Section 8.3.3.3)

The applicant's March 22, 1985 letter addressing their electrical separation testing program was briefly discussed with the applicant. Based on these discussions, the staff stated that the separation test results appear to provide adequate justification for lesser separation between Class 1E and non Class 1E circuits than allowed by IEEE Standard 384. In addition, the staff stated that the results of the testing program would be reviewed in further detail and that the results of the staff review would be reported in a future supplement to the SER.

MILLSTONE UNIT 3 SITE VISIT SUMMARY

INSTRUMENTATION AND CONTROL SYSTEMS BRANCH

We conducted a site visit at Millstone Unit 3 station on May 6-8, 1985 to assure that the installation of safety related instrumentation and control systems were implemented in accordance with the design criteria specified in the FSAR. We also conducted a drawing review and discussed the confirmatory items identified in our safety evaluation report.

A. Plant Walk Through

The following areas were observed:

1. Control room main control board instrumentation.
2. Reactor protection system and engineered safety feature actuation system cabinets and their testing provisions.
3. Consoles and panels cable routing and separation.
4. Cable spreading area and isolation cabinets.
5. Vital instrumentation and control power supply installation.
6. Remote shutdown panel and transfer panels.
7. Electrical penetration area.
8. Switchgear rooms and safety related motor control centers.
9. Engineered safety feature equipment and pump rooms.
10. Instrument sensors and transmitters inside containment.
11. Cable tunnel area.

B. Circuit traces from sensors to protection cabinets

1. We traced the signals for the redundant channels of the steam generator low level inputs to the protection system. The tracing started from the steam generator level measurement instrument taps to the sensing lines to the transmitters, then the cable from the transmitters to the cable conduits, the cable trays, the electrical penetrations, the cable tunnel, the cable spreading room area, and the entry into the protection cabinets in the instrument room. We verified that channel separation and channel identification are properly maintained.

2. We traced the signals for four redundant channels of the turbine stop valve inputs to the protection system starting from the turbine building. Two channels have dedicated conduits with proper separation distance. The other two channels run through two separated cable trays. These two cable trays enter the reactor building cable spreading area at least thirty feet apart. These signals are terminated at isolation cabinets. Digital isolators are used to isolate these contact inputs from the reactor protection system cabinets which receive inputs from these contacts.

C. Confirmatory items (CI) discussion

1. CI-30, Design modification to automatic reactor trip system using shunt coil trip.

We reviewed the updated schematic diagrams which show the required changes. We also inspected the reactor trip breaker cabinets in the switchgear room to verify that the field modification was completed.

2. CI-31, Reactor coolant pump underspeed trip.

The applicant stated that Westinghouse is working on analysis to address the staff's concern.

3. CI-34, Steam generator level control and protection.

We reviewed the updated logic diagrams. The applicant stated that these logic diagrams will be submitted to the docket by FSAR Amendment 14.

4. CI-35, Confirmatory test related to IE Bulletin 80-06.

We reviewed the pre-operational test program which describes the confirmatory tests related to IE Bulletin 80-06.

5. CI-36, Control building isolation reset.

We reviewed the updated logic diagrams. The applicant stated that the logic diagrams will be submitted to the docket by FSAR Amendment 14.

6. CI-40, Sequencer deficiency report.

A copy of letter dated April 30, 1985 was provided to the staff during the meeting. This letter addresses the concern of CI-40. The applicant performed a demonstration test on sequencer during the site visit. No problem was observed during the test.

7. CI-42, Instrument accuracy related to Position (4), (5), and (6), NUREG-0737, Item II.F.1.

A copy of letter dated March 4, 1985 was provided to the staff during the meeting. This letter addresses the concern of CI-42. The staff stated that the justification for containment hydrogen monitor accuracy should be provided. The applicant stated that additional information will be provided later.

D. <u>Exit meeting summary</u>

During the exit meeting on May 8,1985, the staff expressed the following concerns:

1. We observed that computer console has not been installed in the main control room. The software for the safety parameter display system (SPDS), inadequate core cooling display system, and other operator aid programs are not completed. Many electrical and instrumentation cables are still being pulled.

2. We observed that the plant security system needs fine tuning in order to adequately support the calibration and preoperational test activities required to meet the November 1, 1985 fuel load schedule date.

3. We observed that main control boards are still being modified to satisfy the human factor engineering review effort. Consequently, the cable separation inside the console in some cases does not conform to the design criteria. We were told that cable separation will be reinspected by the construction department about 6 weeks prior to the building turnover to the operation department. The resident inspector will be informed when this reinspection is completed.

4. We observed that safety grade Inadequate Core Cooling instrumentation cabinets are located in the instrument room. No safety grade back-up display is located in the control room. The operator can only observe the inadequate core cooling status from the SPDS console which is not a safety grade system. The applicant stated that the SPDS computer system has 99% availability and is therefore acceptable on this basis. The Core Performance Branch will evaluate this during the review of Inadequate Core Cooling Instrumentation (ICCI) system.

5. We observed that the lighting system at the remote shutdown panel area is inadequate. The applicant stated that additional emergency lights will be installed before fuel load. The resident inspector will be informed when the installation is completed.

6. We observed that the circuit for remote shutdown control function is fed from the same fuse for control room control function. In case of fire, the fuse may be damaged. It will degrade the remote shutdown capability. We understand that this is still an open issue in fire protection review.

7. We observed that in the cable tunnel area some safety related cables trays cross over non-safety related cable trays without proper separation. We were told that cable tray covers will be installed before internal reinspection. The resident inspector will be informed when the installation is completed.

8. We observed that steam generator reference leg thermal insulation has not been installed. We were told that proper insulation will be installed before fuel load. The resident inspector will be informed when the installation is completed.

ENCLOSURE 2

NRC - INSTRUMENTATION AND CONTROL SYSTEMS BRANCH
MILLSTONE-3 SITE AUDIT, MAY 6-8, 1985
EXIT MEETING, MAY 8, 1985

LIST OF ATTENDEES

Faust Rosa	NRC/NRR/DSI/ICSB
Hulbert Li	NRC/NRR/DSI/ICSB
Beth Doolittle	NRC/NRR/OL/LB 1
T. Rebelowski	NRC/SR1/MILL 3
Arvao Roby	Northeast Utilities
D. Robinson	NNECO
Mark Samek	NUSCO/I&C
John Festa	MP 3 Project
T. A. Shaffer	NUSCO/I&E Eng.
R. T. Laudenat	NUSCO
P. M. Beanch	NUSCO
R. G. Joshi	NUSCO-Licensing
G. M. Olsen	NUSCO/I&C Eng.
W. Lepper	NUSCO
G. R. Pitman	NUSCO Mgr. Elect.
W. Brown	NNECO