U.S. NUCLEAR REGULATORY COMMISSION

NRC FOIA REQUEST NUMBER(S): FOIA — 90-561

RESPONSE TYPE: X FINAL | PARTIAL

DATE: JAN 1 6 1991

DOCKET NUMBER(S) (If applicable)

RESPONSE TO FREEDOM OF INFORMATION ACT (FOIA) REQUEST

REQUESTER: Ophelia G. Williams

PART I. – AGENCY RECORDS RELEASED OR NOT LOCATED (See checked boxes)

	No agency records subject to the request have been located.
	No additional agency records subject to the request have been located.
	Requested records are available through another public distribution program. See Comments Section.
	Agency records subject to the request that are identified on Appendix(es) ______ are already available for public inspection and copying in the NRC Public Document Room, 2120 L Street, N.W., Washington, DC 20555.
XX	Agency records subject to the request that are identified on Appendix(es) A are being made available for public inspection and copying in the NRC Public Document Room, 2120 L Street, N.W., Washington, DC, in a folder under this FOIA number and requester name.
	The nonproprietary version of the proposal(s) that you agreed to accept in a telephone conversation with a member of my staff is now being made available for public inspection and copying at the NRC Public Document Room, 2120 L Street, N.W., Washington, DC, in a folder under this FOIA number and requester name.
	Agency records subject to the request that are identified on Appendix(es) ______ may be inspected and copied at the NRC Local Public Document Room identified in the Comments Section.
	Enclosed is information on how you may obtain access to and the charges for copying records placed in the NRC Public Document Room, 2120 L Street, N.W., Washington, DC.
	Agency records subject to the request are enclosed.
	Records subject to the request have been referred to another Federal agency(ies) for review and direct response to you.
XX	You will be billed by the NRC for fees totaling $24.88
	In view of NRC's response to this request, no further action is being taken on appeal letter dated ______, No. ______

PART II.A – INFORMATION WITHHELD FROM PUBLIC DISCLOSURE

	Certain information in the requested records is being withheld from public disclosure pursuant to the exemptions described in and for the reasons stated in Part II, sections B, C, and D. Any released portions of the documents for which only part of the record is being withheld are being made available for public inspection and copying in the NRC Public Document Room, 2120 L Street, N.W., Washington, DC in a folder under this FOIA number and requester name.

COMMENTS

The fees associated with the processing of your FOIA request are as follows:

Professional Search 1 hour = $24.88
Total = $24.88

SIGNATURE, DIRECTOR, DIVISION OF FREEDOM OF INFORMATION AND PUBLICATIONS SERVICES

Donnie H. Grimsley

9102190121 901207
PDR FOIA
WILLIAM90-561 PDR

Re: FOIA-90-561

APPENDIX A
DOCUMENTS BEING PLACED IN THE PDR

NUMBER	DATE	DESCRIPTION
1.	04/10/89	Letter from Stolz to Mroczka (1 page)
2.	07/19/89	Memorandum from Jaffe to Stolz (1 page)
3.	10/05/89	Memorandum from Jaffe to Stolz (1 page)
4.	11/22/89	Memorandum from Boyle to Stolz (1 page)
5.	11/22/89	Memorandum from Wang to Stolz (1 page)
6.	01/10/90	Memorandum from Wang to Stolz (1 page)
7.	02/05/90	Memorandum from Wang to Stolz (1 page)
8.	02/16/90	Memorandum from Stolz to Mroczka (3 pages)
9.	04/18/90	Memorandum from Wang to Stolz (1 page)
10.	04/26/90	Letter from Stolz to Mroczka with enclosed Amendment No. 125 to DPR-61, Safety Evaluation, Notice of Issuance, and Notice of Partial Denial (101 pages)
11.	05/03/90	Letter from Stolz to Mroczka (4 pages)
12.	05/07/90	Memorandum from Kokajko to All NRR Project Managers (3 pages)

April 10, 1989

Docket Nos. 50-213
50-245
50-336
50-423

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PGill
JStone

Mr. E. J. Mroczka, Senior Vice President
Nuclear Engineering and Operations
Northeast Nuclear Energy Company
Connecticut Yankee Atomic Power Company
P. O. Box 270
Hartford, Connecticut 06141-0270

Dear Mr. Mroczka:

SUBJECT: RESPONSE TO BULLETIN 88-10, NONCONFORMING MOLDED-CASE CIRCUIT BREAKERS (TAC NOS. 71317, 71318, 71319, 71320)

On November 22, 1988 the NRC staff issued Bulletin 88-10, "Nonconforming Molded-Case Circuit Breakers", requesting certain actions to be taken in regard to molded-case circuit breakers (MCCB) used in safety-related applications. The bulletin requested that, by April 1, 1989, licensees provide a written response confirming that actions requested by the bulletin had been taken, summarizing the results of those actions and if not completed a schedule for completion.

By letter dated March 16, 1989 you responded to Bulletin 88-10. The NRC staff has completed a review of your response and find the actions taken to be in conformance with the bulletin requests and the schedule of July 1, 1989 for providing a summary of your traceability determinations and for replacement of non-traceable MCCB to be acceptable.

Sincerely,

original signed by Mike Boyle for

John F. Stolz, Director
Project Directorate I-4
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

cc: See next page

[DC 50,213,245,336,423]

*SEE PREVIOUS CONCURRENCE

LA:PDI-4*	PM:PDI-4*	PM:PDI-4*	PM:PDI-4	PM:PDI-4	PD:PDI-4
SNorris	GVissing:cb	DJaffe	MBoyle MB	AWang	JStolz MB for
04/04/89	04/04/89	04/05/89	04/10/89	04/10/89	04/10/89

A/1

~~8904200091~~

July 19, 1989

Docket Nos: 50-213
and 50-245
50-336
50-423

MEMORANDUM FOR: John F. Stolz, Director
Project Directorate I-4
Division of Reactor Projects I/II

FROM: David H. Jaffe, Project Manager
Project Directorate I-4
Division of Reactor Projects I/II

SUBJECT: LICENSEE CONFIRMATION OF ITS RESPONSE TO NRC BULLETIN/GENERIC LETTER 89-08 (TAC NOs. 73491, 73506, 73507, 73508)

By letter dated July 13, 1989 Northeast Utilities (the licensee) responded to NRC Bulletin 88-08, Erosion/Corrosion Induced Pipe Wall Thinning, for Haddam Neck and Millstone Units 1, 2 and 3. In its response, the licensee confirmed that activities required to address the issues discussed in the Bulletin have been performed. Therefore, we consider this action complete.

By copy of this memorandum, Region I is advised of the licensee's position on this matter.

/s/

David H. Jaffe, Project Manager
Project Directorate I-4
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

cc: E. McCabe
E. Wenzinger
W. Raymond
T. Shedlosky

[TAC NOS. 73419, 73506,507,508]
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Docket File
DJaffe
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A/r

October 5, 1989

Docket Nos: 50-213
and 50-423

MEMORANDUM FOR: John F. Stolz, Director
Project Directorate I-4
Division of Reactor Projects - I/II

FROM: David H. Jaffe, Project Manager
Project Directorate I-4
Division of Reactor Projects - I/II

SUBJECT: MILLSTONE UNIT 3 AND HADDAM NECK - RESPONSE TO BULLETIN 88-09, THIMBLE TUBE THINNING (TAC 72669, 72662)

By letter dated September 9, 1988, Northeast Utilities responded to Bulletin 88-09 for Millstone Unit 3 and Haddam Neck. The licensee's response provides details regarding inspection methods, acceptance criteria and inspection frequencies. In addition, by internal memorandum dated August 24, 1989 (Marsh to Jaffe), the results of a plant specific audit of the Bulletin 88-09 program were reported for Millstone Unit 3. The August 24, 1989 memorandum indicated that the Millstone Unit 3 program was adequate.

By copy of this memorandum, Region I is advised of the licensee's position on this matter. The date of this memorandum is considered the closeout date for the TAC numbers.

Sincerely,

/s/

David H. Jaffe, Project Manager
Project Directorate I-4
Division of Reactor Projects - I/II

cc: W. Raymond, SRI
B. Buckley
D. Haverkamp, Region I

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DJaffe
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PDI-4 (Memo File)

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Docket Nos. 50-213
and 50-245
50-336
50-423

NOV 2 2 1989

MEMORANDUM FOR: John F. Stolz, Director
Project Directorate I-4
Division of Reactor Projects I/II

FROM: Michael L. Boyle, Senior Project Manager
Project Directorate I-4
Division of Reactor Pr.iects I/"I

SUBJECT: LICENSEE CONFIRMATION OF ITS RESPC SE TO NRC GENERIC LETTER 89-07 (TAC NOS. 74678, 746 3, 74694 AND 74695)

By letter dated October 6, 1989 Northeast Utilities (the licensee) responded to NRC Generic Letter 88-07, Power Reactor Safeguards Contingency Planning for Surface Vehicle Bombs, for Haddam Neck and Millstone Units 1, 2 and 3. In its response, the licensee confirmed that activities required to address the issues discussed in the Generic Letter have been performed. Therefore, we consider this action complete.

By copy of this memorandum, Region I is advised of the licensee's position on this matter.

/s/
Michael L. Boyle, Senior Project Manager
Project Directorate I-4
Division of Reactor Projects I/II

cc: E. McCabe
E. Wenzinger
W. Raymond
T. Shedlosky

(TAC NOS. 74678, 74693, 74694, 74695)

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[GEN LTR]					

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Docket Nos. 50-213
and 50-423

MEMORANDUM FOR: John F. Stolz, Director
Project Directorate I-4
Division of Reactor Projects - I/II

FROM: Alan B. Wang, Project Manager
Project Directorate I-4
Division of Reactor Projects - I/II

SUBJECT: LICENSEE CONFIRMATION OF ITS RESPONSE TO NRC BULLETIN 89-01: FAILURE OF WESTINGHOUSE STEAM GENERATOR TUBE MECHANICAL PLUGS (TAC NOS. 73173 AND 75241)

By letter dated June 16, 1989, Northeast Utilities on behalf of Millstone 3 and Connecticut Yankee Atomic Power Company on behalf of Haddam Neck (the licensees) responded to NRC Bulletin 89-01. In their response, the licensees confirmed that activities required to address the issues discussed in the Bulletin have been performed. Therefore, we consider this action complete.

By copy of this memorandum, Region I is advised of the licensees' position on this matter.

/S/
Alan B. Wang, Project Manager
Project Directorate I-4
Division of Reactor Projects - I/II

cc: D. Haverkamp, RI
E. Wenzinger, RI
W. Raymond, RI
J. Shedlosky, RI
E. Murphy

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[MEMO TO STOLZ FM WANG]

LA:PDI-4	DJaffe PM:PDI-4	PM:PDI-4	PD PDI-4
SNorris		AWang:bld	JStolz
11 08/ /89	11 08/21/89	11 08/20/89	11/22/89

~~8911290291~~

A/S

January 10, 1990

Docket No. 50-213

MEMORANDUM FOR: John F. Stolz, Director
Project Directorate I-4
Division of Reactor Projects - I/II

FROM: Alan B. Wang, Project Manager
Project Directorate I-4
Division of Reactor Projects - I/II

SUBJECT: HADDAM NECK - RESPONSE TO BULLETIN 88-03, "INADEQUATE LATCH ENGAGEMENT IN HFA-TYPE LATCHING RELAYS MANUFACTURED BY GENERAL ELECTRIC (GE) COMPANY" (TAC NO. 73885)

By letter dated November 22, 1989, Northeast Utilities responded to Bulletin 88-03 for Haddam Neck. The licensee's response included type, quantity and inspections results. Northeast Utilities identified six HFA relays that were considered Class 1E. All six relays were found in satisfactory condition with no required corrective action.

By copy of this memorandum, Region I is advised of the licensee's position on this matter. The date of this memorandum is considered the closeout date for the TAC number.

/s/

Alan B. Wang, Project Manager
Project Directorate I-4
Division of Reactor Projects - I/II

cc: T. Shedlosky, SRI
C. Shiraki
D. Haverkamp, RII

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AWang
SNorris

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NAME	:SNorris	:AWang:bld	:JStolz	:	:	:
DATE	:01/10/90	:01/10/90	:01/10/90	:		

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Document Name: HADDAM NECK MEMO TO STOLZ

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February 5, 1990

Docket No. 50-213

MEMORANDUM FOR: John F. Stolz, Director
Project Directorate I-4
Division of Reactor Projects - I/II

FROM: Alan Wang, Project Manager
Project Directorate I-4
Division of Reactor Projects - I/II

SUBJECT: HADDAM NECK - RESPONSE TO BULLETIN 89-02, "STRESS CORROSION CRACKING OF HIGH HARDNESS TYPE 410 STAINLESS STEEL INTERNAL PRELOADED BOLTING IN ANCHOR DARLING MODEL S350W SWING CHECK VALVES OR VALVES OF SIMILAR DESIGN" (TAC NO. 74261)

By letter dated January 5, 1990, Northeast Utilities responded to Bulletin 89-02 for Haddam Neck. The licensee's response stated that no Anchor Darling Model S350W Swing Check valves or valves of similar design using preloaded Type 410 stainless steel bolts are used at Haddam Neck. Therefore, we consider this action complete. By copy of this memorandum, Region 1 is advised of the licensee's position on this matter. The date of this memorandum is considered the close-out date for the TAC number.

Sincerely,

/s/

Alan Wang, Project Manager
Project Directorate I-4
Division of Reactor Projects - I/II

cc: T. Shedlosky, SRI
D. Haverkamp, RI

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AWang
PDI-4 (Memo File)

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NAME	:SNorris	:AWang:adl	:JStolz R. Tam for	:	:	:
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February 16, 1990

Docket No. 50-213
50-336
50-423

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S. Varga (14E4) S. Norris
B. Boger (14A2) G. Vissing
D. LaBarge (14B2) D. Jaffe
E. Jordan (MNBB 3302) OGC
NRC & Local PDRs Plant File
A. Wang

Mr. Edward J. Mroczka
Senior Vice President
Nuclear Engineering and Operations
Connecticut Yankee Atomic Power Company
Northeast Nuclear Energy Company
P. O. Box 270
Hartford, Connecticut 06141-0270

Dear Mr. Mroczka:

SUBJECT: HADDAM NECK PLANT, MILLSTONE UNITS 2 AND 3 - RESPONSE TO BULLETIN 89-03 (TAC NOS. 75427, 75435 AND 75436)

On November 21, 1989, the NRC staff issued Bulletin 89-03 "Potential Loss of Required Shutdown Margin During Refueling Operations." The bulletin requested that all PWR licensees and PWR construction permit holders take the actions described in the bulletin to ensure that an adequate shutdown margin is maintained during all refueling operations. To accomplish this, three actions were described:

1. Assure that any intermediate fuel assembly configuration (including control rods) intended to be used during refueling is identified and evaluated to maintain sufficient refueling boron concentration to result in a minimum shutdown margin of approximately 5%.

2. Assure that fuel loading procedures only allow those intermediate fuel assembly configurations that do not violate the allowable shutdown margin and that these procedures are strictly adhered to.

3. Assure that the staff responsible for refueling operations is trained in the procedures recommended in Item 2 above and understand the potential consequences of violating these procedures. This training should include the fundamental aspects of criticality control with enriched fuel assemblies.

By letter date January 25, 1990 you responded to Bulletin 89-03 which indicated that programs are in place and will be implemented to address all three actions described above. Therefore, we consider your response to Bulletin 89-03 to be satisfactory and TAC Nos. 75427, 75435 and 75436 to be closed.

Sincerely,

/s/

John F. Stolz, Director
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

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cc: See next page

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NAME	:SNorris:lm	:DJaffe	:GVissing	:AWang	:JStolz	:	:
DATE	:2/7/90	:2/15/90	:2/16/90	:2/15/90	:2/16/90	:	:

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Document Name: TAC 75427

A/8

Mr. Edward J. Mroczka
Northeast Nuclear Energy Company

Haddam Neck & Millstone Nuclear Power Station Unit Nos. 2 & 3

cc:

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Day, Berry and Howard
Counselors at Law
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W. D. Romberg, Vice President
Nuclear Operations
Northeast Utilities Service Company
Post Office Box 270
Hartford, Connecticut 06141-0270

Kevin McCarthy, Director
Radiation Control Unit
Department of Environmental Protection
State Office Building
Hartford, Connecticut 06106

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Office of Policy and Management
80 Washington Street
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C. F. Clement, Unit Superintendent
Millstone Unit No. 3
Northeast Nuclear Energy Company
Post Office Box 128
Waterford, Connecticut 06385

Ms. Jane Spector
Federal Energy Regulatory Commission
825 N. Capitol Street, N.E.
Room 8608C
Washington, D.C. 20426

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c/o Robert E. Fletcher, Esq.
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M. R. Scully, Executive Director
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Mr. Edward J. Mroczka
Northeast Nuclear Energy Company

Haddam Neck & Millstone Nuclear Power
Station Unit Nos. 2 & 3

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D. B. Miller, Station Superintendent
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Connecticut Yankee Atomic Power Company
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East Hampton, Connecticut 06424

G. H. Bouchard, Unit Superintendent
Haddam Neck Plant
RFD #1
Post Office Box 127E
East Hampton, Connecticut 06424

Board of Selectmen
Town Hall
Haddam, Connecticut 06103

J. T. Shedlosky, Resident Inspector
Haddam Neck Plant
c/o U. S. Nuclear Regulatory Commission
Post Office Box 116
East Haddam Post Office
East Haddam, Connecticut 06423

April 18, 1990

Docket No. 50-213

MEMORANDUM FOR: John Stolz, Director
Project Directorate I-4
D' ision of Reactor Projects - I/II

FROM: Alan Wang, Project Manager
Project Directorate I-4
Division of Reactor Projects - I/II

SUBJECT: LICENSEE CONFIRMATION OF ITS RESPONSE TO NRC GENERIC LETTER 89-04: INSERVICE TESTING PROGRAM

By letter dated March 29, 1990 Connecticut Yankee Atomic Power Company (CYAPCO) responded to Generic Letter (GL) 89-04, "Inservice Testing Program (IST)" for the Haddam Neck Plant. In its response, CYAPCO provided their IST program and relief requests as required by the GL 89-04. CYAPCO states that their IST program which includes pump, valve, and augmented IST conforms to the guidance in GL 89-04. In addition, CYAPCO believes they are in compliance with 10 CFR 50.55a with the requested reliefs and no additional relief requests are anticipated. Therefore, we consider this action complete.

By copy of this memorandum, Region I is advised of the licensee's position on this matter.

/s/
Alan Wang, Project Manager
Project Directorate I-4
Division of Reactor Projects I/II

cc: D. Havercamp
E. Wenzinger
T. Shedlosky

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B. Boger
S. Norris
A. Wang
L. Marsh (9H3)

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Document Name: GL 89-04

OFC	LA:PDI-4	PM:PDI-4	EMEB	D:PDI-4
NAME	SNorris:rc	AWang	LMarsh	JStolz
DATE	4/17/90	4/18/90	4/17/90	4/18/90

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GL-84-15
87-09
88-16
89-14

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

April 26, 1990

Docket No. 50-213

Mr. Edward J. Mroczka
Senior Vice President
Nuclear Engineering and Operations
Connecticut Yankee Atomic Power Company
Northeast Nuclear Energy Company
P. O. Box 270
Hartford, Connecticut 06141-0270

Dear Mr. Mroczka:

SUBJECT: ISSUANCE OF AMENDMENT (TAC NOS. 43048, 49109, 66797, 74179, 74180, 74181 AND 74551)

The Commission has issued the enclosed Amendment No. 125 to Facility Operating License No. DPR-61 for the Haddam Neck Plant, in response to your applications titled and dated as follows:

1) Revised Technical Specifications dated October 26, 1988, March 6, June 2, June 23, July 28, and August 4, supplemented by submittals on August 21 and November 22, 1989,

2) Proposed Changes to Technical Specifications Section 3/4.5 of Revised Technical Specifications dated August 2, 1989,

3) Proposed Revision to Technical Specifications Fire Protection dated July 31, 1989,

4) Proposed Changes to Technical Specifications Cycle 16 Reload dated July 28, 1989, supplemented by submittal on September 29, 1989,

5) Proposed Changes to Technical Specifications Electrical Power Systems dated November 16, 1987 and revised August 29, 1988, supplemented by submittals on June 9, July 19, and August 1, 1989, and

6) Proposed Changes to Technical Specifications Reactor Protection System Phase II and Nuclear Instrumentation System Upgrades dated July 28, 1989.

This amendment will revise the entire current set of custom Technical Specifications (TS). These TS revisions include: 1) a format change from custom TS to the Westinghouse Standard-format Technical Specifications (WSTS), 2) changes to reflect modifications to the plant such as the new switchgear room (Appendix R), High Pressure Safety Injection Recirculation Path, and Reactor Protection and Nuclear Instrumentation Replacements, 3) changes as recommended by various Generic Letters and changes associated with NUREG-0737

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and the Systematic Evaluation Program. Details of the TS changes and our conclusion that the proposed TS changes are acceptable are provided in the enclosed Safety Evaluation which is divided into six parts each of which addresses the changes requested in items 1) through 6) above.

The following TS changes were denied or deferred:

1) By submittal dated October 26, 1988, the licensee requested that TS Section 5.3.1, "Fuel Assemblies" be revised to allow insertion of stainless steel filler rods or vacancies as justified by the cycle-specific reload analysis. The staff has deferred the review of this request to the resolution of GL 90-02, "Alternative Requirements for Fuel Assemblies in the Design Features Section of Technical Specifications."

2) By submittal dated June 2, 1989, the licensee requested to add the words "to be repaired" to TS Section 4.4.5.4.a.6, "Plugging Limit."

3) By submittal dated June 2, 1989, the licensee requested that the charging flow indication calibration requirement be removed from the TSs.

4) By submittal dated June 23, 1989, the licensee proposed an additional ACTION (a) to TS Section 3.3.3.2, "The Movable Incore Detector System." The proposed action statement stated that with less than the minimum number of detector thimbles required, the movable incore detector system could be used if penalty factors are applied to the linear heat generation rate or quadrant power tilt; or during recalibration of the system.

The specific evaluations are provided in the enclosed Safety Evaluation.

Our conclusions regarding the Phase II of the Reactor Protection System (RPS) Upgrade are consistent with our Phase I conclusions. The one unresolved area is the potential susceptibility of the new equipment to electromagnetic interference. The licensee shall submit by June 15, 1990, or prior to restart a plan outlining the analysis or testing necessary to demonstrate that the electrical environment of the new equipment is enveloped by the vendor's qualification testing and the schedule by which this work will be completed.

As stated in our letter dated September 5, 1989, this TS upgrade effort does not fulfill CYAPCO's SEP commitment to convert to the WSTS. This review is not a STS conversion because:

1) A conversion would require a more comprehensive and detailed review,

2) A conversion would evaluate all deviations of the current TS from the WSTS. This was not performed as part of this review. The staff confirmed that the current requirements were maintained and therefore, in general, the proposed TS changes could be considered an administrative change.

3) A conversion would result in a "completeness" review to assure all applicable sections of the WSTS were included. This was not performed as part of this review. For this review, the staff did evaluate all new TS proposed by CYAPCO but only to determine if the new TSs maintained the current requirements, were appropriate and in the WSTS format.

Therefore, the staff expects the STS conversion to be scheduled for implementation because it is an SEP commitment.

A copy of the related Safety Evaluation is enclosed. Also enclosed is a copy of the Notice of Issuance and a copy of the Notice of Partial Denial and Opportunity for Hearing which have been forwarded to the Office of the Federal Register for publication.

Sincerely,

John F. Stolz

John F. Stolz, Director
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 125 to DPR-61
2. Safety Evaluation
3. Notice of Issuance
4. Notice of Partial Denial

cc w/enclosures:
See next page

Mr. Edward J. Mroczka
Connecticut Yankee Atomic Power Company — Haddam Neck Plant

cc:

Gerald Garfield, Esquire
Day, Berry and Howard
Counselors at Law
City Place
Hartford, Connecticut 06103-3499

W. D. Romberg, Vice President
Nuclear Operations
Northeast Utilities Service Company
Post Office Box 270
Hartford, Connecticut 06141-0270

Kevin McCarthy, Director
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

CONNECTICUT YANKEE ATOMIC POWER COMPANY

DOCKET NO. 50-213

HADDAM NECK PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 125
License No. DPR-61

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

   A. The applications for amendm- . ..ticut Yankee Atomic Power (the licensee), dated:

      (1) October 26, 1988, as supplemented March 6, June 2, June 23, July 28, August 4, August 21 and November 22, 1989,

      (2) August 2, 1989,

      (3) July 31, 1989,

      (4) July 28, 1989, as supplemented September 29, 1989,

      (5) November 17, 1987, revised August 29, 1988, as supplemented June 9, July 19 and August 1, 1989,

      (6) July 28, 1989,

      comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-61 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 125, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance and shall be implemented within 60-days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Director
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 26, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 125

FACILITY OPERATING LICENSE NO. DPR-61

DOCKET NO. 50-213

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages.

Remove	Insert
All	All

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 125

TO FACILITY OPERATING LICENSE NO. DPR-61

CONNECTICUT YANKEE ATOMIC POWER COMPANY

HADDAM NECK PLANT

DOCKET NO. 50-213

PART 1 - Reviews the reformatting of all current Technical Specification sections except for Sections 3.6, 3.7, 3.12, 4.3 and 4.5.

PART 2 - Review of changes to the Technical Specifications to reflect modifications implemented by the end of Cycle 15.

PART 3 - Review of changes to the Technical Specifications to reflect installation of additional fire protection features.

PART 4 - Review of changes to the Technical Specifications as proposed by Generic Letter 88-16 "Removal of Cycle-Specific Parameter Limits from Technical Specifications.

PART 5 - Review of changes to the Technical Specifications to 1) incorporate degraded grid voltage protection requirements; 2) incorporate emergency diesel generator requirements of Generic Letter 84-15 "Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability;" 3) incorporate industry improvements; 4) change custom Technical Specification format to one that is similar to the Westinghouse Standard Technical Specification format; and 5) incorporate requirements for battery discharge testing as required by the Systematic Evaluation Program Topic VIII-3.A.

PART 5A- Review of changes to the Technical Specifications related to the electrical power systems and the degraded grid undervoltage setpoints.

PART 6- Review of changes to the Technical Specifications to reflect installation of a new reactor protection system and nuclear instrumentation system.

DATE: April 26, 1990

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PART 1 OF SAFETY EVALUATION

RELATED TO AMENDMENT NO. 125

1.0 INTRODUCTION

By submittals dated October 26, 1988, March 6, June 2, June 23, July 28, and August 4, 1989, and supplemented by submittals on August 21, 1989, and November 22, 1989, Connecticut Yankee Atomic Power Company (CYAPCO) proposed to upgrade their current custom format Technical Specifications (TS) to the Westinghouse Standard-format Technical Specifications (WSTS). All sections of the current custom TS will be reformatted in this proposed TS except for Sections 3.6, "Core Cooling Systems," 3.7, "Minimum Water Volume and Boron Concentration in the Refueling Water Storage Tank," 3.12, "Station Service Power," 4.3, "Core Cooling Systems-Periodic Testing" and 4.5, "Emergency Power System Periodic Testing". Sections 3.6, 3.7 and 4.3 were reformatted by Amendment No. 121. Sections 3.12 and 4.5 will be reformatted by amendment request dated August 29, 1988.

2.0 DISCUSSION

As part of the Systematic Evaluation Program (SEP), CYAPCO committed to convert their custom TS to the WSTS. In a meeting on September 20, 1988, CYAPCO proposed to submit the TS conversion packages over a three month period beginning October 1988. With the impending issuance of the revised WSTS (MERITS), the staff proposed that it would be advantageous for CYAPCO to await the issuance of the revised WSTS before addressing the full WSTS conversion. In the interim, the staff agreed that the custom TS format could be upgraded to the current WSTS format. The staff concluded that this interim step would: 1) provide a substantially improved TS while facilitating the future conversion effort to the revised WSTS, 2) provide definitive LCO and Action statements for several safety related systems, 3) eliminate the use of administrative TS, 4) provide a mechanism to close prior TS commitments associated with NUREG-0737, SEP and various other Generic Letter (GL) recommendations and 5) eliminate ambiguities inherent with the wording and format of the current TS. Based on the above, the staff concluded that the revised TS would enhance public safety and therefore justified this interim step to improve the Haddam Neck TS. The staff has informed CYAPCO several times that this TS upgrade does not fulfill CYAPCO's SEP commitment to convert to the WSTS.

This amendment is one of several that is part of the TS upgrade. By letter dated September 22, 1987, the NRC provided Northeast Utilities with an acceptable revision of the WSTS. The TS upgrade will be using the provided WSTS revision as a guide for the format change while maintaining the current TS requirements. Since this upgrade is primarily a format change, the staff did not pursue all deviations and omissions from the provided WSTS with the same intensity as would have been done for a WSTS conversion. Therefore, if the proposed TS omitted portions of the requirements that appear in the provided WSTS revision and these same requirements did not already exist in the current TS, the review of these omissions will be deferred to the full WSTS conversion. However, where new TS statements have been proposed (statements not previously found in the current TS) that deviate from the provided WSTS revision, a review of the deviation will be given. The deviations will be reviewed in part, based on three previously agreed upon criteria: 1) plant specific design, 2) previously approved hardware, structural or organizational changes, and 3) past operating experiences that can be shown to provide an equivalent degree of protection to that provided by the WSTS. Any deviations from the current custom TS will also be reviewed. The format change and the additional restrictions resulting from this amendment make substantial improvements in the clarity and readability of the TS. As a result, the staff considered this TS upgrade beneficial from both a public safety and an operational perspective.

3.0 EVALUATION

The evaluation has been divided into two sections. Section I will address proposed TS that are consistent with the provided WSTS and/or the current TS. In addition, many of these TS sections add restrictions to the current TS. Section II will address proposed TS that relax restrictions from either the current TS or the provided WSTS revision. As noted earlier, the staff did not perform a "completeness" review to ensure that all sections of the WSTS were included in this format change. Therefore, this review will exclude the review of complete omissions of WSTS sections that did not already exist in the current TS. Each of the deviations will be addressed individually. If a GL or a SEP issue has been addressed by the proposed TS change then it will also be noted.

3.1 Section I

Previously, the NRC staff provided a version of the WSTS to CYAPCO and excluding plant specific alterations, stated the provided WSTS would be an acceptable guidance for a STS conversion. Although this amendment is not a STS conversion, the amendment does follow the guidance of this WSTS revision. The logic for this TS upgrade has been stated in the Discussion section of this Safety Evaluation. The staff review has determined that all sections of the proposed TS except for those discussed in Section 3.2 of this Safety Evaluation are consistent with the current TS and/or the WSTS, impose added restrictions to the current TS, and/or add restrictions that do not currently exist. Therefore, the proposed TS sections except for those delineated in Section 3.2, are administrative in nature (format change) or provide additional limitations, restrictions, or controls not previously included in the Haddam Neck TS.

In addition, the NRC staff has provided Table 1 which provides a list of all sections of the current TSs and where those TS sections (TS sections from the custom TS) now exist in the proposed TS. This was done to verify that all sections and requirements of the current TS are incorporated in the proposed TS or that justification for deletion or modification of a current TS is provided. The staff has concluded that the safety significant requirements of the current TS have been maintained in the proposed TS.

Based on the above, the staff concluded that the proposed TS are acceptable and provide an equivalent and in some areas an enhanced set of TS to the current custom TS.

3.2 Section II

The TSs reviewed in this section will be addressed by number and subsection as it appears in the proposed TS. As noted earlier the WSTS refers to the WSTS revision provided to CYAPCO by letter dated September 22, 1987.

A) October 26, 1988 Submittal

1) Section 1, Definition, Table 1.1, Frequency Notation

The definition of "S" in Table 1.1 has been changed from "at least once per 8 hours" to "at least once per 12 hours." The 12 hour limit while consistent with WSTS is a relaxation from the current TS. CYAPCO states that the 8 hour frequency does not provide any latitude within an 8 hour shift in which to perform surveillances that are required once per shift. That is, the once per 8 hour shift checks would have to be performed at exactly the same time interval or less within each shift. CYAPCO maintains that the surveillances notated with an "S" will be performed each shift, with a shift being 8 hours. The 12 hour time limit will provide latitude within the shift to allow for scheduling and operational perturbations which could affect the timing of certain activities. The staff believes the intent of the TS is to require a check once per shift and this requirement will be maintained. Based on the above, the staff concludes that the proposed TS is acceptable.

2) Sections 3.2.3.1.1, 3.2.3.2 and 3.2.4

The existing TSs contain only trip setpoints. The proposed TSs contain both trip and allowable setpoin[illegible] The trip setpoints of the existing TS are equivalent to the allowable [illegible]nts of the proposed TS. The proposed trip setpoint is now 72% instead of [illegible]% as the setpoint was written in the custom TS. The proposed trip setpoint has been set 2% lower to account for instrument drift, expected to be a maximum of 2%. This ensures that the allowable value (74%) is not violated at any time between calibrations. Based on the above, the staff concludes that the proposed TS is acceptable.

3) TS Table 5.7.1

Table 5.7.1 provides a list of reactor vessel design transients and the maximum permissible number of design cycles. The list of transients is different than that provided by the WSTS. CYAPCO states that their list provides a list of all transients which have been analyzed for cyclic design restrictions. As modifications and analysis are revised and updated, Table 5.7.1 will be revised to reflect the latest analysis. This table does not currently exist in the current TS. Based on the above, the staff has concluded that the proposed TS meets the intent of the WSTS and represents all currently analyzed component cyclic or transient limits. The staff concludes that the proposed TS is acceptable.

4) TS 3.9.11

TS 3.9.11 required a minimum of 20 feet of water be maintained over the top of the irradiated fuel, seated in the storage racks. WSTS recommends 23 feet. CYAPCO states 21 feet is the maximum possible due to the design of the spent fuel pool. While it is possible to fill the pool to provide 21 feet of water, this would expose certain equipment and components to water/boric acid and could cause equipment/component failures. The proposed level of 20 feet would limit water/boric acid exposure to various equipment, especially the carbon steel sleeve gate operator. CYAPCO has calculated the decontamination factor (DF) for 20 feet of water as approximately 250. This is conservative compared to the DF of 100 for iodine assumed in the fuel handling accident and the DF of 133 recommended in Regulatory Guide 1.25, Revision 2. While this is a deviation, the 20 feet of water provides an adequate degree of protection for any fuel handling accident. Based on the above the staff concludes that this TS is acceptable.

5) Section 4.9.6.2

This surveillance requirement specifies a load test weight to be 125% of the weight of the load to be lifted be performed. The WSTS requires a load test of a fixed weight. CYAPCO states this TS provides some flexibility in the loads to be lifted. CYAPCO states that the load test weight is consistent with the guidelines of ANSI B30.2 and will not exceed the rated load capacity of the hoist. Based on the above, the staff concludes that the surveillance provides an equivalent degree of protection to the WSTS and therefore the TS is acceptable.

TS 5.3.1

The proposed TS allows the fuel assemblies to consist of 1) fuel rods clad with Type 304 stainless steel, 2) filler rods fabricated from Type 309 stainless steel or 3) vacancies as justified by the cycle-specific reload analysis. The current TS requires that the fuel assemblies consist only of fuel rods clad with Type 304 stainless steel. The proposed change provides flexibility to deviate from a fixed number of fuel rods per assembly. This is desirable because it permits timely removal of fuel rods that are found to be leaking during a refueling outage or are determined to be probable sources of future leakage. Approval of the proposed change will allow improvement in the licensee's fuel performance, which will provide for reductions in future occupational radiation exposure and plant radiological releases. Under the proposed change, limitations on fuel rod substitution or omissions and limitations regarding core locations are those implicit in the justifying analyses required to be performed by the licensee for each fuel cycle using NRC-approved methodology to demonstrate that existing design limits and safety analyses continue to be met.

The term "NRC-approved methodology" includes those methodologies acknowledged in the Final Safety Analysis Report and applied in support of issuance of the original operating license for the Haddam Neck Plant. Additionally, it includes those subsequent methodologies that have been submitted to and accepted by the staff as amendments to the operating license.

The requirement for special reporting is consistent with existing TS 6.9.2 and is necessary to keep NRC informed in the event a significant deviation from past fuel performances should be observed during a refueling outage.

The licensee has proposed changes to Specification 5.3.A that are consistent with the guidance provided in Generic Letter 90-02, "Alternative Requirements for Fuel Assemblies in the Design Features Section of Technical Specifications." Therefore, the staff has deferred approval of this request to the resolution of GL 90-02.

B) March 6, 1989 Submittal

1) TS 3.1.2.2 and 4.1.2.2.d

The proposed TS requirement differs from the WSTS in that the required three flow paths are from the boric acid tanks (BAT) rather than one path from the BAT tank and two paths from the RWST and that the flow test surveillance does not specify a flow rate for the BAT flow paths. The boration system ensures that negative reactivity control is available during each Mode of normal operation and for abnormal operational occurrences. At the Haddam Neck Plant, the boric acid concentration in the RWST is significantly lower than that in the BAT. As a result, the limiting case for operation is when the metering pump is used to inject borated water. The metering pump cannot inject sufficient boric acid into the RCS from the RWST to provide the required shutdown margin. Because of post-LOCA chemistry requirements the boric acid concentration in the RWST is bounded in the TS. Therefore, CYAPCO cannot use the RWST as a required water source for reactivity control; and the boration capability to ensure the shutdown margin in all Modes provided by the proposed TS 3/4.1.2.2 can only be provided by the BAT. Accordingly TS 3.1.2.2, Flow Paths-Operating, only references the three flow paths from the BAT to the charging/metering pumps. Although the RWST flow path to the charging/metering pumps is not credited for reactivity control, the RWST flow path to the charging pumps is required to be available by TS 3/4.5.1, ECCS Subsystem-Tavg Greater Than Or Equal To 350° F and TS 3/4.5.2, ECCS Subsystems-Tavg Less Than Or Equal To 350° F. The licensee also states that no flow instrumentation exist in the BAT lines to determine flow. The licensee states that they will demonstrate that the BAT lines to the charging pump suction are unobstructed. As allowed by the ground rules of the TS upgrade one of the basis for deviation is plant specific design. Based on the above the staff concludes that the proposed TS deviations are a result of plant specific design and to obtain the WSTS format would require modification to the plant. In addition, the proposed applicability and surveillance requirements are more restrictive than the current TS and the Action statement did not previously exist. Based on the above, the staff concludes the the TS meets the intent of the WSTS and provides at least an equivalent degree of protection as the current TS and therefore is acceptable.

2) TS 3.1.2.6

The proposed TS differs from the WSTS because the RWST is not included. As noted in the discussion of TS 3.1.2.2, the RWST is not a required water source for reactivity control consideration at the Haddam Neck Plant. In addition, the equivalent requirements (LCO, applicability, action and surveillance requirements) for the RWST exist in the Emergency Core Cooling Systems section of the proposed TS. Based on the above, the staff concludes that the TS is acceptable.

C) June 2, 1989 Submittal

1) TS 3.7.1.2

This TS is for the auxiliary feedwater system. The proposed TS is equivalent to or more conservative than the current TS and therefore by the groundrules of the conversion is acceptable. However, this TS is also part of the GL 83-37 "NUREG-0737 Technical Specifications." The NRC staff has concluded that the proposed TS does not meet the intent of the GL 83-37. CYAPCO and the staff have agreed that this issue will be resolved in a future license amendment.

2) TS 3.6.1.5 and 4.6.1.5

The proposed TS does not include the specific locations of where the temperature readings are to be made as specified in the WSTS. The locations and methodology for calculating containment average temperature was reviewed in Inspection Report 88-23. The report concluded that the dispersion of the resistance temperature detectors (RTDs) adequately represents containment temperature. However, during containment integrated leak rate test an additional RTD is necessary in the dome above the polar crane. While IR 88-23 has concluded that the calculated temperature adequately represents the containment, the inspectors are still reviewing the RTD placements which will assure that the RTDs will provide a representative temperature of containment. Based on the above, the staff concludes that the exact location of the RTDs need not be specified in the TS as the RTD placement will be confirmed by future inspections.

3) TS Table 3.3-3, Footnote for Items 4a, 4b, and 4c

Table 3.3-3, Footnote for 4a, 4b and 4c states that the device must change state within .95-1.05 seconds when the input voltage to the device goes from normal to zero volts instantaneously. The proposed change requires that the relays actuate when the input voltage decreases instantaneously from normal to 50 percent of the tap setting voltage. By requiring the device to change state within one second, ±5 percent, when the input voltage to the device reduces from normal to 50 percent of tap setting voltage instantaneously, the relay is being challenged to operate in a real degraded voltage situation. If the input voltage were allowed to drop to zero, the time-voltage characteristics of the induction coil in the degraded voltage range would not fully be tested. A loss of all voltage would simply cause the relay to return to its de-energized state. Since the proposed testing requirements will challenge the device in a degraded condition, the proposed change represents a more conservative test. Furthermore, the test is consistent with the plant's standard method of testing undervoltage relays of this type. Based on the above, the staff concludes that the proposed TS change is acceptable.

4) TS 4.4.5.4

CYAPCO added "to be repaired" to the TS. Currently, the staff requires that repairing of tubes requires a TS amendment. The amendment would include the approval of a sleeve specifically for use at the Haddam Neck Plant. The TS upgrade did not provide this information and therefore the staff does not find this change acceptable.

D) June 23, 1989 Submittal

1) Table 3.3-2 (3.b)

The proposed action statement for the auxiliary feedwater system requires that with one less than the minimum channels operable restore the channel to operable status within 24 hours or reduce the thermal power to below 10% of rated thermal power within the following hour. The current TS would imply a shutdown on a loss of one channel with no specified time frame. The WSTS would allow up to 48 hours with one less than the minimum channels operable but require the plant to shutdown if the channel cannot be restored within 48 hours. The WSTS is applicable for MODES 1 and 2 while for Haddam Neck the applicable mode is MODE 1 greater than 10% power. CYAPCO states that below 10% power the plant operators would have more than adequate time to manually initiate the auxiliary feedwater pumps since the decay heat loads below 10% power are small. In accordance with the FSAR, the auxiliary feedwater initiation system is defeated below 10% power.

With one channel inoperable the plant would have 24 hours to repair the channel or reduce power to less than 10% where the auxiliary feedwater initiation system is defeated and the action statement would no longer be applicable. This action is similar to the WSTS which provides a fixed time frame to restore the channel or place the plant in a condition for which the action statement is not applicable.

Due to hardware design, the inoperable channel cannot be placed in a tripped position. Therefore, for a maximum of 24 hours, the plant would be without automatic auxiliary feedwater initiation from the trip of all main feedwater pumps. This is partially compensated for by the fact that automatic auxiliary feedwater initiation is still provided by low steam generator water level. CYAPCO's proposed TS provides a reasonable compromise between the plant configuration and the WSTS. Based on the above, the staff concludes the proposed TS is acceptable.

2) TS 3.3.3.2 Action a

The proposed action statement for the movable incore detector system would allow continued use of the system with less than the minimum number of detector thimbles required if penalty factors are applied to the linear heat generation rate or quadrant power tilt; or during recalibration of the system. The staff currently requires that penalty factors be approved before they can be applied in such cases. Therefore, the staff denies this proposed action statement.

3) Proposed Deletion of Various Current TS Requirements

a) Current TS 3.9.C

Current TS requires that neutron monitors in each range (source, intermediate and power) shall be in continuous operation until at least one decade of reliable indication is verified on the next range of instrumentation. CYAPCO has recently replaced their nuclear instrumentation system (NIS). The new power range instrumentation covers the entire range of the original equipment (from 200% power to 1 X 10^{-8}% power). The new source range and power range instruments are provided data from the same detectors. Therefore, there is no need to verify the decade overlap as the entire range is provided by the power instrumentation. The staff agrees that this requirement can be deleted.

b) Current TS 3.11.E

Current TS 3.11.E requires the containment spray system to be operable whenever the reactor is critical. The containment spray system is an auxiliary system that is not credited for in any safety analysis. Containment heat removal is provided by two 100% Containment Air Recirculation fan systems. The staff agrees that this requirement can be deleted from the TS.

c) Current TS 3.13.A

Current TS 3.13.A requires radiation levels in the containment and fuel storage building to be monitored continuously during refueling. Radiation Monitoring of the containment and spent fuel building are part of the Refueling Procedures. In addition, radiation monitoring is required to be maintained in each area in which such licensed special nuclear material is handled, used, or stored by 10 CFR 70.24. The staff agrees that this requirement can be deleted from the TS.

d) Current TS 3.13.F

Current TS 3.13.F requires that whenever new fuel is added to the reactor core, a 1/m plot be maintained to verify the subcriticality of the core. This requirement is not in the WSTS, and it does not have any corresponding limiting condition for operation. The 1/M surveillance is part of CYAPCO's Refueling Procedure and will be maintained there. The staff agrees that this requirement can be deleted from the TS.

e) Current TS 3.13.H

Current TS 3.13.H forbids the movement of spent fuel cask above the fuel pool or its edge until the NRC has received and approved the spent fuel cask drop evaluation. In a letter dated June 28, 1985 GL 85-11, the NRC staff indicated that all licensees have completed the requirement to perform a review and submit a Phase I and Phase II report regarding NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." The GL further stated that based on the improvements in heavy loads handling obtained from implementation of Phase I of NUREG-0612, further action is not required to reduce the risk associated with the handling of heavy loads (Phase II of NUREG-0612). Therefore, the staff concluded that a detailed Phase II review of heavy loads is not necessary and Phase II of NUREG-0612 is complete. In that GL, the staff recommended each licensee to submit a license amendment to delete any requirements related to heavy loads from the TS citing this GL as the basis. CYAPCO has stated that the only TS related to heavy loads is TS Section 3.13.H. Based on the above, the staff agrees that TS 3.13.H can be deleted. However, the staff recommended that any actions identified by the licensee in regard to Phase II of NUREG-0612 should be implemented. Therefore, all open items identified in CYAPCO's letter dated July 21, 1983 relating to Phase II, should be completed prior to the handling of spent fuel casks in the fuel-handling building.

f) Current TS 3.22, A.2, A.3, B.3, C.3, E.2.b and G.3

The above TS sections require Special Reports be made to the NRC whenever the associated system of the TS is declared inoperable. CYAPCO will review all reportable events in accordance with the requirements of 10 CFR 50.72 as proposed in the upgraded TS Section 6.6.1. The staff agrees that this section can be deleted and the reportability be provided under 10 CFR 50.72.

g) Current TS Table 4.2-1, Item 13

This item requires Charging Flow Indication be calibrated each refueling. While this requirement is not in the WSTS, the staff does not believe sufficient bases has been provided to remove this TS requirement. This surveillance will be maintained in TS Section 4.5.1f(4).

h) TS Table 4.2-1, Item 20

This TS item requires calibration of the boric acid control system each refueling. This system is used during normal operation of the plant for boric acid control and is not credited for in any design basis analysis. When and if it becomes necessary to make a rapid addition of boric acid to the RCS, this flow element is bypassed as boric acid from the boric acid mix tank flows through a pump directly to the charging pump suction. This system is calibrated routinely by procedure. The staff concludes that this TS item can be deleted.

i) Current TS Table 4.2-2, Item 10

This TS item requires Refueling System Interlocks to have a function check each refueling. The testing of these interlocks is performed as part of the Refueling Procedures and there is no credit taken for these interlocks in any design basis analysis. There are 13 interlocks to control motion of such things as the crane, bridge, fuel upender and the gripper tube. The staff concludes that this item can be deleted from the TS.

E) July 28, 1989 Submittal

1) TS 3.0.4, 4.0.3, 4.0.4 and associated Bases

These statements deviate from WSTS and do not exist in this form in the current TS. The proposed TSs reflect NRC guidance as recommended in GL 87-09 for improved wording and clarity. The proposed wording recommended by the GL were incorporated in verbatim by the proposed TS. These changes represent part of the improved TS effort as encouraged by the staff and therefore are found to be acceptable.

2) TS 4.0.2 and associated Bases.

This statement deviates from the WSTS and does not exist in this form in the current TS. The proposed TS reflect NRC guidance as recommended in GL 89-14 for improved wording and clarity. The proposed wording recommended by the GL were incorporated in verbatim by the proposed TS.

Experience has shown that the 18-month surveillance interval, with the provision to extend it by 25 percent, is usually sufficient to accommodate normal variations in the length of a fuel cycle. However, the NRC staff has routinely granted requests for one-time exceptions to the 3.25 limit on extending refueling surveillances because the risk to safety is low in contrast to the alternative of a forced shutdown to perform these surveillances. Therefore, the 3.25 limitation on extending surveillances has not been a practical limit on the use of the 25-percent allowance for extending surveillances that are performed on a refueling outage basis.

The use of the allowance to extend surveillance intervals by 25 percent can also result in a significant safety benefit for surveillances that are performed on a routine basis during plant operation. This safety benefit is incurred when a surveillance interval is extended at a time that conditions are not suitable for performing the surveillance. Examples of this include transient plant operating conditions or conditions in which safety systems are out of service because of ongoing surveillance or maintenance activities. In such cases, the safety benefit of allowing the use of the 25-percent allowance to extend a surveillance interval would outweigh any benefit derived by limiting three consecutive surveillance intervals to the 3.25 limit. Also, there is the administrative burden associated with tracing the use of the 25-percent allowance to ensure compliance with the 3.25 limit. On the basis of these considerations, the staff concluded that removal of the 3.25 limit will have an overall positive impact on safety.

This alternative to the requirements of Specification 4.0.2 will remove an unnecessary restriction on extending surveillance requirements and will result in a benefit to safety when plant conditions are not conducive to the safe conduct of surveillance requirements. The removal of the 3.25 limit will provide greater flexibility in the use of the provision for extending surveillance intervals, reduce the administrative burden associated with its use, and have a positive effective on safety. Therefore, the staff concludes the proposed TS is acceptable.

4.0 SUMMARY

The staff has reviewed the proposed TS and as stated in Section 3.1 has determined that all of the safety significant current TS requirements will be maintained by the proposed TS. Furthermore, the proposed amendment is an improved format over the current TS and incorporates numerous new TS limitations, restrictions or controls to plant operation. Based on the considerations discussed in the above evaluation, the staff concluded that the proposed amendment will make overall improvements in the operational safety of the plant while maintaining the current safety analysis. Therefore, the staff finds the proposed amendment to be acceptable.

5.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact have been prepared and published in the Federal Register on February 23, 1990 (55 FR 6563). Accordingly, based upon the environmental assessment, we have determined that the issuance of the amendment will not have a significant effect on the quality of the human environment.

5.0 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: A. Wang
G. Garten

Table 1 - Current T.S. # With
Corresponding Proposed T.S. #

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Existing T.S. #	Description	Proposed RTS
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Existing T.S. #	Description	Proposed RTS
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Existing T.S. #	Description	Proposed RTS
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Existing T.S. #	Description	Proposed RTS
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Existing T.S. #	Description	Proposed RTS
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*See Section 3.2 of SER

Existing T.S. #	Description	Proposed RTS
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Existing T.S. #	Description	Proposed RTS
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*Done procedurally in accordance with Regulatory Guide 1.52 Rev. 2

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PART 2 OF SAFETY EVALUATION

RELATED TO AMENDMENT NO. 125

INTRODUCTION

By letter dated August 2, 1989, Connecticut Yankee Atomic Power Company (CYAPCO/licensee) requested changes to Technical Specification (TS) 3/4.5.1, "Emergency Core Cooling System - ECCS Subsystem - Tavg Greater Than or Equal to 350°F" to reflect modifications to be implemented by the end of the 1989 refueling outage. These modifications will resolve single failure concerns identified on three separate occasions. In addition, TS Section 3/4.5.2, "Emergency Core Cooling System - ECCS Subsystem - Tavg Less Than 350°F," Section 5.3 "Emergency Core Cooling System - Refueling Water Storage Tank," and Section 5.4, "Emergency Core Cooling System - pH Control System" have been renumbered to be consistent with the Westinghouse Standard Technical Specifications (WSTS) format.

DISCUSSION

This TS will address three single failure vulnerabilities that were discovered by CYAPCO. These single failure vulnerabilities are 1) Small Break LOCA, 2) Medium Break LOCA and 3) Charging Pump Flow Paths.

Small Break LOCA

On March 31, 1986, CYAPCO submitted a probabilistic safety study (PSS) in conjunction with the Integrated Safety Assessment Program (ISAP) for the Haddam Neck Plant, which identified a small range of break sizes in one loop of the Reactor Coolant System (RCS) for which safety injection flow in the high pressure recirculation mode may be insufficient to provide adequate core cooling. To respond to these Small Break (SB) LOCAs, CYAPCO took temporary measures which were approved by the NRC. The emergency operating procedures were revised to provide an alternate flow path utilizing the High Pressure Safety Injection (HPSI) pumps for core cooling during the high pressure recirculation mode. The use of this flow path required realignment of two valves which did not satisfy the single failure criterion. Therefore, CYAPCO requested an exemption from the single failure criterion for these valves, pending implementation of the permanent modifications. On April 28, 1986, the NRC granted the requested exemption and requested that CYAPCO provide by September 1986, a description of the long term resolution and a schedule for completion of any modifications. By letters dated September 30, 1986 and April 1, 1987, CYAPCO submitted a description of the proposed modifications and requested a one-cycle extension of the exemption because some of the modifications could not be completed until the end of the Cycle 15 outage. On September 2, 1987, the NRC granted an extension until the end of the Cycle 15 outage. This TS change will incorporate the new valves necessary for HPSI recirculation.

Medium Break LOCA

While analyzing the design for the small break LOCA modifications, a medium size break in the core deluge system was identified which would not be sufficiently mitigated during sump recirculation. Procedures were developed, a flow control valve was repositioned and the TS were changed to provide a temporary resolution to this problem. This TS change will incorporate a new valve necessary for the resolution of this issue.

Charging Pump Flow Paths

During routine plant inservice inspection, CH-MOV-257, volume control tank (VCT) outlet valve failed to operate. As part of the root cause analysis and subsequent evaluation, CYAPCO identified two single failure vulnerabilities (failure of CH-MOV-257 or BA-MOV-373, Suction line from Reactor Water Storage Tank (RWST)) which could impact charging system performance. A temporary resolution consisting of automatically tripping both charging pumps on a safety injection signal (SIAS) was implemented. CYAPCO will resolve these single failure vulnerabilities by adding redundant valves for CH-MOV-257 and BA-MOV-373. These valves will be included in the TS.

EVALUATION

A) Small Break LOCA

By letter dated September 7, 1987, (Attachment 1) the staff approved the permanent ECCS modifications necessary to resolve the small break LOCA problem. Details of the proposed modification can be found in the attachment. This TS change will implement the modifications previously approved, to provide HPSI recirculation capability at the Haddam Neck site. During the 1987 outage an eight inch cross-tie connection between the RHR pump discharge and the HPSI pump suction was added. In addition motor operated valves (MOVs) SI-MOV-854A, 854B, 901, 902 and 873 were installed. During the 1989 refueling outage the remaining modifications necessary for implementation of the HPSI recirculation will be completed. These modifications will include removing valves SI-V-857A and B and SI-FCV-875 and the installation of valves: SI-MOV-903, SI-MOV-904, SI-V-919, SI-V-920, SI-CV-921, SI-CV-922, SI-CV-923, SI-CV-924, SI-V-925, SI-V-926, SI-V-927, SI-V-928, SI-V-929, SI-V-930 and SI-V-931.

The implementation of the HPSI recirculation will require the following specific TS changes:

1) TS Section 4.5.1.a

a) Valve SI-FCV-875, HPSI miniflow line, has been deleted. This valve has been physically removed from the HPSI miniflow line.

b) The asterisk and footnote for valve RH-MOV-874, RHR recirculation line, has been deleted. The note required this valve to be cycled every 31 days. This requirement was part of the compensatory measures taken because the temporary HPSI recirculation path was not single failure proof. To insure reliability of the path CYAPCO had agreed to increased surveillance on this valve and SI-MOV-24, RWST line isolation valve. With the completion of the HPSI recirculation modifications this testing is no longer needed.

c) The positions for valves SI-MOV-854A and B in the TS have been changed to "Open-Manual Operator is Locked" from "Locked Open." These valves have been installed since the 1987 outage, however, power is not available for them until the completion of the new switchgear room. Since the HPSI recirculation modifications were incomplete, these valves were locked open as required for the current plant safety configuration. With the completion of the HPSI recirculation modifications and the new switchgear room, these valves will be powered with the manual operator locked. During HPSI recirculation these valves will need to be closed to provide redundant isolation of the RWST.

d) The positions for valves SI-MOV-90. and 902, RHR/HPSI crosstie, in the TS have been changed to "Closed. Manual Operator is Locked" from "Locked Closed." The situation with these valves is exactly the same as with the SI-MOV-854A and B valves except the required position for these valves was locked closed. For HPSI recirculation these valves will be opened to provide suction for the HPSI pumps from the discharge of the RHR pumps.

e) Valves SI-MOV-903 and 904, HPSI miniflow, have been added to the TS and are required to be open with the manual operator locked. These valves were added to provide remote redundant isolation valves in the HPSI pump minimum flow line. These valves replaced SI-FCV-875, HPSI miniflow line valve, SI-V-857A and B, manual HPSI pump minimum flow line valves. During the recirculation phase these valves would be closed to isolate the RWST and prevent backfilling of the RWST with containment sump water.

2) TS Section 4.5.1.c

a) Surveillance c.2, which currently requires valves SI-MOV-24 and RH-MOV-874 to be cycled every 31 days, will be deleted. As noted in 1.b this surveillance was part of the compensatory measures taken because of the single failure vulnerability of the temporary HPSI recirculation path. With the completion of the permanent HPSI recirculation path this increased surveillance is no longer necessary. In addition, c.3 and c.4 were renumbered because c.2 has been deleted.

b) Surveillance c.4 is being added to require monthly verification that containment sump valve RH-MOV-22 can be cycled manually from the control room and valve RH-V-808A can be manually cycled locally. To assure the reliability of the recirculation path, CYAPCO has increased the surveillance interval of these valves to monthly from 18 months.

3) TS Section 4.5.1.f

Surveillance f.2 is being revised to require all remote manual valves, which are required to change position during a LOCA, to be cycled once per 18 months. These valves are also in the inservice testing (IST) program. The original TS included only valves RH-MOV-22 and RH-V-808A. As noted in 2.b these valves are now cycled monthly and no longer included in this surveillance. CYAPCO proposed this additional surveillance in the TS to highlight the importance of these valves.

4) TS Table 4.5.2

This table lists all the valves to be tested by TS Section 4.5.1.f

B) Medium Break LOCA (Core Deluge Line Break)

1) TS Section 4.5.1.a

The position of valve RH-FCV-796 in the TS has been changed to "Blocked open position" from "Blocked in throttled position." As part of the temporary resolution to the core deluge line break, CYAPCO determined that RH-FCV-796 had to be throttled to prevent RHR pump runout. SI-MOV-873, a remote, redundant valve to isolate the core deluge line from the ECCS in the event of a core deluge line break has been added. Therefore, RH-FCV-796 no longer needs to be throttled and can be returned to the full open position.

2) TS Section 4.5.1.b

The position of valve SI-MOV-873 in the TS has been changed to "Locked Open. Operator circuit breaker locked open" from "Valve is locked open and electrically disconnected." As part of the permanent solution to the core deluge line break valve SI-V-873 was replaced with SI-MOV-873. Since all the modifications necessary to resolve the LOCA problems were incomplete, SI-MOV-873 was locked open and electrically disconnected, which was consistent with the plant's current safety configuration. With the completion of the ECCS modifications during the 1989 refueling outage SI-MOV-873 will be provided with electrical power and an open breaker which will allow electrical energization to permit remote closure while still preventing inadvertent valve closure. As noted before, this valve provides remote, redundant isolation capability for the core deluge line break.

3) TS Section 4.5.1.i

Valve RH-FCV-796 is being deleted from the list of throttled valves. As noted earlier RH-FCV-796 is no longer throttled during normal operation and is blocked in the full open position.

4) TS Section 4.5.1.j

Surveillance j.2, RHR pumps discharge flow balance test is being deleted. This test was necessary to verify that valve RH-FCV-796 was throttled in the correct position as part of the temporary solution to the core deluge line break. This test is no longer necessary since valve RH-FCV-796 is no longer throttled. In addition, this surveillance is being editorially changed to incorporate surveillance j.1 into surveillance j.

C) Charging Pumps Flow Paths

1) TS Table 4.5-1

This table has been revised to include valves BA-MOV-32, CH-MOV-257B and CH-SOV-242B and their safety injection positions. Valves CH-MOV-257B and CH-SOV-242B were added to provide redundant isolation of the VCT from the charging pumps. In addition valve BA-MOV-32, charging pump suction from the RWST, will be modified to receive an automatic open signal on a safety injection actuation signal (SIAS) and have a faster stroke time. This will assure an adequate suction supply to the charging pumps on a SIAS. These modifications will allow the current charging pumps trip on SIAS to be removed, as this trip was the temporary solution to the charging pump single failure vulnerabilities.

2) TS Section 4.5.1 Bases

The Bases section is being revised to reflect plant modifications and the associated proposed TS changes.

The staff has determined that all of the above TS changes are consistent with our Safety Evaluation dated September 2, 1987 relating to the ECCS modifications. Therefore, the staff concludes that the proposed TS changes are acceptable.

ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. We have determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The staff has previously published a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: Alan B. Wang

Attachment:
NRC letter dated 9/2/87

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

Docket No.: 50-213

September 2,1987

Mr. Edward J. Mroczka, Senior Vice President
Nuclear Engineering and Operations
Connecticut Yankee Atomic Power Company
Post Office Box 270
Hartford, Connecticut 06141-0270

Dear Mr. Mroczka:

SUBJECT: EXTENSION OF THE TEMPORARY EXEMPTION FROM THE SINGLE FAILURE CRITERION (GENERAL DESIGN CRITERION NO. 35)

Re: Haddam Neck Plant

On March 25, 1986, Connecticut Yankee Atomic Power Company (CYAPCO) reported that the results of analyses of a small limited range of break sizes in one loop of the reactor coolant system (RCS) showed that safety injection flow during only the high pressure recirculation mode may be insufficient to provide adequate long-term core cooling. By letter dated April 10, 1986, CYAPCO proposed a temporary high pressure safety injection (HPSI) recirculation mode to provide adequate cooling flow to the core for the above range of pipe breaks until CYAPCO could identify and establish a permanent solution. Because the proposed temporary solution did not satisfy the single failure criterion for two valves located outside containment, CYAPCO requested a temporary exemption from General Design Criterion No. 35 (GDC 35) for the subject valves.

By letter dated April 28, 1986, the Commission granted the temporary exemption from the requirements of GDC 35 for the subject valves for the period of Cycle 14 operation. Further, the Commission specified that CYAPCO was to provide a description of the long-term resolution of this issue and a schedule for completion of plant modifications by the end of September 1986. The Commission also stated that the exemption could be extended provided that good cause exists for an extension.

By letters dated September 30, 1986 and April 1, 1987, CYAPCO provided a description of the proposed resolution of the problem and a schedule for completion of all plant modifications. Because some of the required plant modifications could not be implemented until the end of the Cycle 15 outage, CYAPCO also requested a one cycle extension to the exemption granted on April 28, 1986.

CYAPCO's proposed resolution provides for high pressure coolant injection and recirculation by aligning the Residual Heat Removal (RHR) pump discharge to the HPSI pump suction piping. This permits high pressure injection into the RCS from the containment sump. The cross-tie connection between the RHR pump discharge and the HPSI pump suction will be accomplished by the addition of two eight inch lines, each with a separate motor-operated valve. Modifications to

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the HPSI suction piping will be located in the Primary Auxiliary Building (PAB) pipe trench and high pressure safety injection/low pressure safety injection HPSI/LPSI pump pit. Modifications to the HPSI mini-flow line will be in the HPSI/LPSI pump pit. Modifications to the four injection lines for flow balancing will be in containment. The tie-in to the RHR system will require that the RHR system be drained and disabled and will require a full core offload This offload is already planned for the Cycle 14 outage to support the 10-year core barrel inspection. Consequently, CYAPCO should be able to complete all piping and valve modifications during the Cycle 14 outage.

Following completion of all piping and valve modifications, manual operation of the RHR/HPSI injection alignment can be accomplished should the need arise. CYAPCO has stated that new procedures and appropriate training will be implemented to assure that the new valves outside containment can be manually operated when required. CYAPCO has determined that there will be adequate manpower, access and time to implement these operations. Further, CYAPCO intends to conduct periodic surveillance of the valves pending completion of the electrical modifications during Cycle 15 outage. The need for the extension arises because there are no safety-related motor control center (MCC) compartments available at the Haddam Neck Plant to power the new motor-operated valves. However, a limited number of MCC compartments will become available upon the completion of the new switchgear room by the end of the Cycle 15 outage.

The staff has reviewed your proposed modifications and your extension request and concludes that you have shown good faith in that CYAPCO will implement all necessary mechanical modifications during the current Cycle 14 outage. These modifications include installation and testing of the new valves, piping and supports previously described. The staff has reviewed the proposed modifications and concluded that the installed piping modifications will not affect the ability of the existing core cooling systems to provide adequate core cooling during the additional cycle of operation. Because the piping modifications are passive (not powered), the Haddam Neck Plant will operate over the next cycle using the ECCS configuration evaluated at the time the initial exemption was granted. Further, revisions to the emergency operating procedures including operator training in the manual use of the HPSI/RHR configuration (under extreme circumstances) will be completed prior to Cycle 15 operation. A Safety Evaluation on the final ECCS design modifications is enclosed.

The staff has also concluded that the only item preventing completion of this activity during the current outage is the ability to provide a safety-related power supply to the new motor-operated valves from existing power sources. Based upon information presented to the staff, the staff concludes that completion of the new switchgear room will provide additional motor control center compartments and, therefore, a safety-related power source to the new valves.

Based upon the information presented above and in your letters dated September 30, 1986 and April 1, 1987, the staff finds that your extension request was filed on a timely basis and demonstrates good cause for an extension of time to install the permanent plant modifications to resolve the small break LOCA issue. Therefore an extension to the temporary exemption from the requirements of General Design Criterion No. 35 of Appendix A to 10 CFR Part 50 and the Interim Acceptance Criteria for valves RH-MOV-874 and SI-MOV-24 is hereby granted until startup from the Cycle 15 outage. This is consistent with the planned completion date of the new switchgear room at the Haddam Neck Plant and the CYAPCO request for a one cycle extension.

Sincerely,

Original signed by
Dennis M. Crutchfield

Dennis M. Crutchfield, Director
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Enclosure:
Safety Evaluation

cc: See next page

Mr. Edward J. Mroczka
Connecticut Yankee Atomic Power Company

Haddam Neck Plant

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

SAFETY EVALUATION

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO EXTENSION OF THE TEMPORARY EXEMPTION FROM THE

SINGLE FAILURE CRITERION (GENERAL DESIGN CRITERION NO.35)

CONNECTICUT YANKEE ATOMIC POWER COMPANY

HADDAM NECK PLANT

DOCKET NO. 50-213

1.0 INTRODUCTION

On March 25, 1986, Connecticut Yankee Atomic Power Company (CYAPCO) reported that results for small break loss of coolant accident (LOCA) analyses at the Haddam Neck Plant indicated that the safety injection flow during the high pressure recirculation mode could be inadequate for long term core cooling. By letter dated April 10, 1986, CYAPCO proposed a temporary high pressure safety injection (HPSI) recirculation mode to provide adequate core cooling.

To implement the HPSI recirculation mode, CYAPCO requested, by letter dated April 22, 1986, a temporary exemption from the single failure criteria for two emergency core cooling system (ECCS) valves. These valves are located outside of containment and would be used under procedurally defined conditions to respond to small break LOCAs. By letter dated April 28, 1986, the Commission granted a temporary exemption from the single failure criterion for the operation of the Haddam Neck Plant during Cycle 14.

On September 30, 1986, CYAPCO proposed system modifications to meet the single failure requirements for the ECCS. The submittal included a proposed schedule for the completion of all the modifications. Because of facility limitations, some ECCS safety related electrical equipment would have to be installed in a new switchgear room. Since the switchgear room is to be completed by the end of

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the Cycle 15 outage, CYAPCO requested a one cycle extension to the exemption granted on April 28, 1986.

By letter dated December 17, 1986, CYAPCO advised that the original design for low pressure sump recirculation through the Core Deluge System was reexamined for other breaks. That review identified an intermediate break LOCA in the core deluge lines for which adequate core cooling during sump recirculation would not be provided. CYAPCO proposed an interim solution for the intermediate break LOCA which would allow blocking flow control valve FCV-796 in a throttled position to assure adequate core injection flow during sump recirculation. By letter dated December 24, 1986, the staff concurred with the interim modification.

On April 1, 1987, CYAPCO proposed system modifications to assure adequate core cooling for small and intermediate break LOCAs. This included a description of the modifications to the ECCS with supporting single failure analysis for the spectrum of small to large breaks. The licensee reiterated the difficulty in completing the modifications in the Cycle 14 outage and the need for extension of the exemption granted on April 28, 1986.

By letter dated June 1, 1987, CYAPCO proposed Technical Specification (TS) changes for the ECCS. These changes were requested because hardware modifications to the ECCS related to the long term system modifications are being made during the Cycle 14 outage. The TS changes would require the newly installed motor-operated valves (MOVs) to remain in fixed positions so as to not change the present ECCS flow configuration during injection and sump recirculation. In addition, the four newly installed High Pressure Injection System (HPSI) manual valves would be locked in their proper throttled position.

2.0 EVALUATION

2.1 Permanent Modifications To Meet GDC 35

CYAPCO proposed changes to the ECCS to provide a single failure proof means to mitigate potential small and intermediate break LOCAs.

The existing HPSI pump suction would be cross-connected with the Residual Heat Removal (RHR) pump discharge. This will allow a redundant means for high pressure injection into the Reactor Coolant System (RCS) from the containment sump in series with the RHR pump(s) acting as a booster pump for the HPSI pumps.

The core deluge line would also be provided with a redundant motor-operated valve which, in the event of a core deluge line break, would allow its isolation from the ECCS. This is necessary during long term recirculation to prevent RHR pump runout.

The proposed changes to the ECCS require permanent modifications and are described in detail below.

1. The existing HPSI pump suction manual valves will be replaced with motor-operated valves (MOVs) 845A and 854B. These valves are in parallel lines and are required to close during HPSI operation in the recirculation mode to assure a series flow path from the containment sump to the RCS via the RHR and HPSI pumps. An existing valve, SI-MOV-24, will provide redundancy for either of these valves. The required closure of these valves also prevents backflow of potentially contaminated water to the RWST. We find this acceptable.

2. An eight inch cross-tie connection between the RHR pump discharge and the HPSI pump suction will be added. This piping addition will allow HPSI operation during recirculation. We find this acceptable.

3. A three inch manual throttle valve will be installed in each of the four HPSI injection lines. The injection flow path after leaving the throttle valve(s) is via a check valve and MOV to the reactor. In the event of LOCA, a safety actuation signal will cause all four MOVs to proceed to the full open position. During sump recirculation with HPSI operation, two of the injection lines will be blocked closed by remote operator action. The throttle valves in the remaining two available lines are installed to provide adequate flow resistance to prevent the RHR design flow from being exceeded. We find this acceptable.

4. The existing core deluge manual valve SI-V-873 will be replaced with a MOV. This provides a redundant, remote means to isolate the core deluge from the ECCS in the event of a core deluge line break. This is necessary to prevent RHR pump runout during long term recirculation. Presently FCV-796 is throttled to prevent excessive RHR flow and resultant RHR pump damage. At the time the new MOV is placed in service, following the Cycle 15 outage, CYAPCO stated that FCV-796 will be fully opened to assure adequate net positive suction head for proper HPSI pump performance. The fully open position of FCV-796 during long term recirculation has raised staff concern with regard to excessive RHR pump flow. CYAPCO advised that in this mode of operation one of the redundant core deluge paths would be blocked closed and the other path throttled by remote operation of a gate valve. Throttling with a gate valve is not a recommended industrial practice and the staff questions the acceptability of long term operation in this mode. The staff considers this an open issue related to acceptability of the permanent modification.

5. Remote operation from the control room of redundant isolation valves in the HPSI pump minimum flow line will be provided. The concept is acceptable to the staff. However, finalized details have not been provided. The HPSI pump minimum flow final design remains an open item, subject to staff review and approval.

CYAPCO performed a simplified failure modes and effects analysis to assure compliance with the single failure criteria for the ECCS. The staff reviewed the results and is in general agreement with them. Some specific concerns are addressed below.

1. Addition of the cross-tie interconnect for RHR/HPSI raises the potential for the Low Pressure Safety Injection (LPSI) system to discharge to the suction of the HPSI pumps during HPSI operation and recirculation from the containment sump. The licensee's letter of April 1, 1987 states that the MOVs (MOV 901 and MOV 902) on the HPSI/RHR cross-tie will be interlocked to prevent their inadvertent opening while the HPSI pump breakers are

closed to prevent overpressurization of the HPSI suction line. The staff also expressed concern with regard to the potential for HPSI overpressurization resulting from a LPSI start during HPSI operation in the recirculation mode. CYAPCO, by letter dated July 20, 1987, stated that prior to HPSI operation during recirculation, the LPSI pumps are to be breakered out. Thus, the potential for overpressurization in that mode is reduced. We find this acceptable.

2. The licensee's single failure analysis did not consider the potential for three injection flow paths with HPSI during recirculation (two injection paths plus single failure). The staff expressed concern that this condition could cause RHR design flow to be exceeded. By letter dated July 20, 1987, and subsequent discussions, CYAPCO addressed this concern. They stated that: during recirculation if a single failure opens an additional MOV, then three HPSI injection paths are open and can result in excessive RHR flow. However, procedures require that no more than two of the four HPSI injection paths be open for coolant flow to the reactor. Valve position lights, which are powered from emergency buses would give indication to the control room operator that more than two paths (MOVs) are open and the operator would then take appropriate action to close one MOV. With two remaining MOVs then open, the resulting flow resistance seen by the RHR pump prevents its runout. This mitigates the concern of exceeding the design capability and subsequent RHR pump or motor damage. Further, adequate core cooling is assured with only one injection line should the break be in one of the two flow paths. We find this acceptable.

CYAPCO also provided the results of a LOCA analyses associated with the permanent modifications to the ECCS to demonstrate adequate core cooling for the injection and recirculation phases. The licensee stated that the bases for the acceptability of ECCS flows during the injection phase are the previously approved Westinghouse analyses which demonstrate acceptable ECCS performance in accordance with the Interim Acceptance Criteria for the full spectrum of breaks.

For the recirculation phase of the LOCA, the licensee presented data which illustrated that the ECCS delivery is larger than the RCS boil-off rate. Thus, no additional analyses is required since ECCS long term cooling is assured following a LOCA. We find this acceptable.

2.2 ECCS Modifications - Stress Analysis

By letter dated September 30, 1986, CYAPCO submitted a request for extension of single failure exemption for Haddam Neck Plant's modification of the emergency core cooling system (ECCS) until the end of the cycle 15 outage. Subsequent information with regard to the core deluge systems' piping stress analysis, was provided by letters April 1, 1987, June 9, 1987, July 20, 1987, and August 12, 1987, per requirement of NRR's Safety Evaluation Report dated December 24, 1986, "Supporting Amendment No. 88 to Facility Operating Licensee No. DPR-61."

The stress analysis is based on the provisions of ANSI B.31.1 Power Piping Code, 1973 Edition, Summer 1973 Addenda and approved design criteria by the NRC per D. M. Crutchfield letter to W. G. Counsil, "SEP Topic III-6, Seismic Design Considerations Haddam Neck Plant," dated February 25, 1983.

The stress analysis review of the core deluge system consisted of: (1) comparison of a sample of calculated maximum stress valves to specified allowable code limits, (2) implementation of approved codes, (3) verification that the 8 inch cross-tie piping and new valve masses were considered in the analysis.

In each case, the staff found the calculated stress values were within the specified allowable limits and ANSI B.31.1 Power Piping Code, 1973 Edition, Summer 1973 Addenda was used for the analysis. In addition, the licensee confirmed by letters dated July 20, 1987 and August 12, 1987, that the 8 inch cross-tie piping and new valve masses were included in the analysis.

2.3 New Modification Effects On Present ECCS Configuration and TS Changes

The newly installed MOVs, with the exception of SI-MOV 873, will not be electrically connected following the Cycle 14 outage. CYAPCO has proposed TS changes to fix positions of the new MOVs and thus assure that the present ECCS configuration for all modes of operation will be unaffected. The proposed changes also require the four three inch throttle valves installed in the HPSI injection lines to be locked in their preset position. The changes to assure proper ECCS configuration are discussed below.

1. SI-MOV-854A AND B. These HPSI pump suction isolation valves are replacing manual valves which are normally open during operation. The proposed change TS requires verification that these valves are in the locked open position once per twelve hours. We find this acceptable.

2. SI-MOV-901 AND 902. These valves isolate the new cross-tie line and are required to be closed to assure the present ECCS flow configuration. The proposed TS change requires verification that these valves are to in the locked closed position once every twelve hours. We find this acceptable.

3. SI-MOV-873. This valve is located in the core deluge line and will serve as a redundant means of isolating the core deluge line when the permanent modifications are completed and approved. It replaces a manual valve which is open during operation. The proposed TS change requires verification that this valve is locked in the open position prior to startup from cold shutdown and also electrically deenergized. We find this acceptable.

4. SI-V-905,906,907, and 908. These are manual throttle valves which are set to balance flow in the four HPSI injection lines. The valves are also set to prevent HPSI flow from exceeding RHR design flow during HPSI recirculation. The correct positions will be established by test. The

proposed TS change requires verification that these valves are locked in the throttled position prior to start up from cold shutdown. Correct position verification is also to be required within four hours following maintenance on these valves. We find this acceptable.

5. RH-FCV-796. The proposed TS change requires verification of the correct position of this valve within four hours after stroking or maintenance. We find this acceptable.

CYAPCO has also proposed a TS change for a flow balance test, which is to be performed during Mode 5 or 6, following completion of modifications to the ECCS subsystems that alter flow characteristics. This TS change would become effective during the Cycle 14 outage. This would verify that HPSI pump injection lines with a single pump running and two lines isolated would have a flow rate through each line of 1000 + or - 100 gpm. We find this acceptable.

A RHR pump flow test was also proposed. This test is to assure that with a single pump running, the RHR pump flow is equal to 1500 + or - 280 gpm. We find this acceptable, however, the means of throttling to assure proper pump flow remains an open item.

2.4 Extension of Exemption to GDC 35

CYAPCO has attempted to complete modifications required to assure compliance of the ECCS with GDC 35. Construction of a building that would house safety-related switchgear for the ECCS is not to be completed before Cycle 15 operation. Because CYAPCO has made a good faith effort to resolve the single failure problem, the staff considers their request for an extension to the previously granted exemption as reasonable. Our review of the modifications indicates that operation of the ECCS without the electrical power for the newly added MOVs will be no different that that of Cycle 14 which was found acceptable by letter dated April 28, 1986. We find extension of the exemption to GDC 35 acceptable for Cycle 15.

3.0 SUMMARY

The licensee has proposed interim fixes to mitigate the possibility of inadequate core cooling resulting from small and intermediate LOCAs. The staff has previously reviewed the interim fixes and found them to be acceptable. The licensee has proposed long term solutions to provide adequate core cooling resulting from the above LOCAs. These require permanent modifications to the ECCS, some of which cannot be completed during the Cycle 14 outage. Thus CYAPCO requires an extension to the previously granted exemption. We find the proposed modifications will not adversely affect the ECCS operation during Cycle 15 and it is therefore acceptable to extend the temporary exemption of April 28, 1986 until the completion of Cycle 15.

The staff finds that the proposed TS changes assure that the present ECCS configuration remains unchanged and are therefore acceptable.

Based upon our review, we find the proposed changes for long term resolution to be acceptable provided the open items of the throttling requirements for the RHR pump and the finalized design for the HPSI minimum flow line discussed above is resolved prior to start up of Cycle 16.

4.0 CONCLUSION

We have reviewed the information provided by CYAPCO relative to their request for extension of the single failure exemption to GDC 35 for Cycle 15. The staff has concluded, based on the considerations discussed in previous sections, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulation, and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 ACKNOWLEDGEMENTS

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PART 3 OF SAFETY EVALUATION

RELATED TO AMENDMENT NO. 125

1.0 INTRODUCTION

By letter dated July 31, 1989, Connecticut Yankee Atomic Power company (the licensee) requested approval of an amendment to the Haddam Neck Plant Technical Specifications. The proposed changes reflect the installation of additional fire protection features associated with the licensee's efforts to conform with the requirements of Appendix R to 10 CFR 50.

2.0 DISCUSSION

The Technical Specification Amendment includes the following changes:

1. The inclusion of additional fire detection instruments and an increase in the minimum number of required operable fire detectors in several fire zones;

2. The addition of new fire suppression systems in the new Switchgear Building;

3. The installation of new fire hose stations in the new Switchgear Building; and

4. Editorial changes to reflect reconfigured fire detector zones.

3.0 EVALUATION

The staff initially, had several concerns with the licensee's proposed amendment. The first was assurance that all of the fire protection features which were installed in conjunction with the licensee's efforts to conform with Appendix R to 10 CFR Part 50 would be reflected in the proposed Technical Specification changes. Based on its review of the relevant design criteria documents, the staff finds that the proposed amendment is comprehensive in this regard.

The second staff concern was that the numbers of additional fire detectors and fire hose stations identified in the amendment request reflected an adequate design. Based on its review of the system design details, the staff finds that these fire protection features conform with the relevant criteria contained in Appendix A to Branch Technical position (BTP) APCSB 9.5-1.

Based on its review, the staff concludes that the licensee's proposed fire protection Technical Specification changes satisfy the guidelines of Appendix A to BTP APCSB 9.5-1 and are, therefore, acceptable.

4.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact have been prepared and published in the Federal Register on February 23, 1990 (55 FR 6563). Accordingly, based upon the environmental assessment, we have determined that the issuance of the amendment will not have a significant effect on the quality of the human environment.

5.0 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: D. Kubicki

PART 4 OF SAFETY EVALUATION

RELATED TO AMENDMENT NO. 125

1.0 INTRODUCTION

By letter dated July 28, 1989, supplemented September 29, 1989, Connecticut Yankee Atomic Power Company (CYAPCO/licensee) proposed changes to the Technical Specifications (TS) for the Haddam Neck Plant. The September 29, 1989 letter provided several pages of the TSs that were inadvertently not included in the July 28, 1989 submittal. These additional TS pages were within the scope of the original notice and do not affect the staff's determination in the original notice. The proposed changes would modify specifications having cycle-specific parameter limits by replacing the values of those limits with references to the Technical Report Supporting Cycle Operation (TRSCO) for the values of those limits. The proposed changes also include the addition of TRSCO to the Definitions section and to the reporting requirements of the Aministrative Controls section of TS. Guidance on the proposed changes was developed by NRC on the basis of the review of a lead-plant proposal submitted on the Oconee plant docket that was endorsed by the Babcock and Wilcox Owners Group. This guidance was provided to all power reactor licensees and applicants by Generic Letter 88-16, dated October 4, 1988. In addition, CYAPCO has made changes to TS Section 3/4.2.1.1, "Axial Offset-4 Loops," 3/4.2.1.2, "Axial Offset-3 Loops," 3/4.2.5, "DNB Related Parameters," 3.1.3.5, "Shutdown Rod Insertion Limit," 3.4.1.4.1, "Cold Shutdown Loops Filled" 3.4.9.1, "Pressure/Temp. Limits-RCS," 3.4.9.3, "LTOP," Figures 3.4-3, 3.4-4 and 3.4-5 and Bases 3/4.23 and 3/4.4.1. These changes were basically clarification or administrative changes.

EVALUATION

The licensee's proposed changes to the TS are in accordance with the guidance provided by Generic Letter 88-16 and are addressed below.

(1) The Definition section of the TS will be modified to include a definition of the TRSCO that requires cycle/reload-specific parameter limits to be established on a unit-specific basis in accordance with an NRC-approved methodology that maintains the limits of the safety analysis. The definition notes that plant operation within these limits is addressed by individual specifications.

(2) The following specifications will be revised to replace the values of cycle-specific parameter limits with a reference to the TRSCO that provides these limits.

TS Section	Title
4.1.1.1.1	Shutdown Margin - 4 Loops
4.1.1.4.1	Shutdown Margin - 3 Loops
3.1.1.5	Mod. Temp. Coeff.
3.1.3.1	Moveable Cont. Assemblies
3.1.3.6.1	Cont. Group Ins. Limit 4 Loops
3.1.3.6.2	Cont. Group Ins. Limit 3 Loops
3/4.2.1.1	Axial Offset
3/4.2.1.2	Axial Offset
3/4.2.2.1	LHGR - 4 Loops
3/4.2.2.2	LHGR - 3 Loops
3/4.2.3.1	$F^N_{\Delta H}$ - 4 Loops
3/4.2.3.2	$F^N_{\Delta H}$ - 3 Loops

(3) Specification 6.9.1.9 "Technical Report Supporting Cycle Operation," will be added to the reporting requirements of the Administrative Controls section of the TS. This specification requires that the TRSCO be submitted, upon issuance, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector. The report provides the values of cycle-specific parameter limits that are applicable for the current fuel cycle. Furthermore, this specification requires that the values of these limits be established using the NRC-approved methodology in the references provided below and are consistent with all applicable limits of the safety analysis. Finally, the specification requires that all changes in cycle-specific parameter limits be documented in the TRSCO before each reload cycle or remaining part of a reload cycle and submitted upon issuance to NRC, prior to operation with the new parameter limits.

a. F. M. Akstulewicz to E. J. Mroczka, "Review of NUSCO Topical Report on Physics Methodology for PWR Reload Design (NUSCO-152)," August 3, 1987.

b. A. B. Wang to E. J. Mroczka, "Safety Evaluation for Northeast Utilities Topical Report 140-1, NUSCO Thermal Hydraulic Qualification, Volume I (RETRAN)," July 26, 1988.

c. F. M. Akstulewicz to J. F. Opeka, "NUSCO Thermal Hydraulic Model Qualification, Volume II (VIPRE), Topical Report NUSCO 140-2," October 16, 1986.

d. A. B. Wang to E. J. Mroczka, "Safety Evaluation of Northeast Utilities Topical Report 151, Haddam Neck Non-LOCA Transient Analysis," October 18, 1988.

e. Supplement to the Safety Evaluation by the Directorate of Licensing, U.S. Atomic Energy Commission Docket No. 50-213, Connecticut Yankee Atomic Power Company, Haddam Neck Plant, December 27, 1974.

(4) The following Figures were deleted and the information provided in the TRSCO:

Figure	Title
3.1-1	Rod Ins. Limit vs. Power Level
3.1-2	Rod Ins. Limit vs. Power Level
3.2-1a	Power Level vs. Axial Offset
3.2-1b	Power Level vs. Axial Offset
3.2-2a	Power Level vs. Axial Offset
3.2-2b	Power Level vs. Axial Offset

(5) The following Bases Sections were changed to reflect that certain operational limits will be provided by TRSCO:

Bases Section	Title
3/4.1.3	Moveable Cont. Assemblies
3/4.2.3	$F^N_{\Delta H}$

On the basis of the review of the above items, the NRC staff concludes that the licensee provided an acceptable response to those items as addressed in the NRC guidance in Generic Letter 88-16 on modifying cycle-specific parameter limits in TS. Because plant operation continues to be limited in accordance with the values of cycle-specific parameter limits that are established using an NRC-approved methodology, the NRC staff concludes that this change is administrative in nature and there is no impact on plant safety as a consequence. Accordingly, the staff finds the proposed changes acceptable.

The following two changes to the TSs were proposed to clarify certain surveillance requirements during plant start-up following a refueling outage.

Axial Offset

The applicability statement of Technical Specifications 3.2.1.1, "Axial Offset-Four Loops" and 3.2.1.2, "Axial Offset-Three Loops" requires monitoring the axial offset when operating above 40% of rated power. However, the excore/incore axial offset correlation cannot be accurately performed until a minimum of three days operation at 80% power (50% power for three loop operation) after start-up. While the proposed TS surveillance requirement specifies continuous monitoring using the excore/incore axial offset correlation above 40% power, proposed Surveillance Requirement 4.2.1.3 does not require the correlation to be determined after a refueling or major change in excore Power Range instrumentation until exceeding 80% power. The revised proposed TS will not require continuous monitoring of the Axial Offset after a refueling or major change in excore Power Range instrumentation using the excore/incore Axial Offset correlation until the excore/incore correlation can be determined and implemented prior to exceeding 80% of Rated Thermal Power. The requirement of not exceeding 80% power (50% power for three loop operation), combined with the successful completion of the zero power testing will provide assurance that the LHGR will not exceed the initial conditions assumed for the loss of coolant accident (LOCA) analyses prior to determining the correlation. All other required surveillances have been maintained. Therefore, the staff concludes that the proposed changes are acceptable.

DNB Parameter

The present Surveillance Requirement 4.2.5.1.c requires verification of the reactor coolant system (RCS) total flowrate once per 12 hours when operating in MODE 1. However, Surveillance Requirement 4.2.5.2 allows the RCS total flow rate to be determined by heat balance within seven EFPD of Achieving Rated Thermal Power after a refueling. In addition CYAPCO states that Surveillance 4.2.5.2 cannot be accurately performed until achieving 100% power. The revised proposed TS transfers the RCS flow rate Surveillance 4.2.5.1c to Surveillance 4.2.5.2. This will clarify the TS by stating that the RCS total flow rate need not be verified at least once per 12 hours until after the RCS total flow rate has been established. The maintenance of the two other DNB-related parameters will prevent departure from DNB prior to establishing the RCS flow rate. Therefore, the staff concludes that the proposed changes are acceptable.

Control Rod Insertion Limits

The proposed change to TS 3.1.3.5 redefines the fully withdrawn position to be 317 steps instead of 320 steps. All the physical models used in the cycle design and determination of safety analysis input parameters assume that the "all rods out" position to be 317 steps. The 317 step position is based on the interface between the fuel assemblies and the control rods. This change will allow greater operational flexibility in the positioning of control rods to minimize future control rod wear concerns and provide additional margin to accomodate drift in the individual rod position indicators. Based on the above, the staff concludes that the proposed change is acceptable.

RCS Heatup

TS 3.4.1.4.1 requires that at least one RHR loop be in operation in MODE 5. One of the recommendations which resulted from analysis of the thermal shield repair was that no more than two reactor coolant pumps be operated at temperatures less than 350°F. Recent experience has demonstrated that the RCS heatup is very slow with two reactor coolant pumps and one RHR pump operating. The proposed change allows the RHR pump to be deenergized during heatup provided the following constraints are met:

1) The deenergized RHR pump and LOOP are OPERABLE,

2) The reactor coolant pumps in at least two unisolated loops are operating, with steam generator secondary side narrow range water level greater than 25%,

3) No operations are permitted that would cause dilution of the reactor coolant system boron concentration, and

4) Core outlet temperature is maintained at least 10°F below saturation temperature.

These constraints provide an adequate heat sink for operation in MODE 5 because of the low decay heat. Deenergizing the operating RHR pump in MODE 5 will allow a controlled RCS heatup without affecting the protective boundaries. Based on the above, the staff concludes that the proposed changes are acceptable.

RCS Hydrostatic and Leak Testing

The proposed changes to TS 3.4.9.1 allow the low temperature overpressure protection system (LTOPS) to be isolated during performance of RCS hydrostatic and leak testing. In addition, the applicability of the LCO has been changed to apply during heatup, cooldown inservice leak and hydrostatic testing but not during criticality. TS 3.4.9.3, "LTOPS" has also been changed to reflect that the LTOPS can be isolated during performance of RCS hydrostatic and leak testing. CYAPCO has stated it is not possible to perform the RCS hydrostatic and leak testing with the LTOPS inservice. The failure mode of a low temperature, overpressurization event occuring below 315°F while the LTOPS is isolated has been evaluated. It was determined that the 10 CFR Part 50 Appendix G margin of safety is maintained during the tests if the hydrostatic and/or leak test are performed above the required minimum temperatures of 245°F and 235°F respectively and a heatup rate of less than or equal to 10°F/hour for one hour prior to and during the tests is maintained. The minimum operating temperature requirement while critical is maintained by TS 3.1.1.6, "Minimum Temperature for Criticality" and therefore the reference to criticality in this TS can be removed. TS 3.4.9.3," LTOPS restates that the LTOPS can be isolated during hydrostatic and leak testing. Based on the above, the staff concludes that the proposed changes are acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. This amendment also involves changes in recordkeeping, reporting, or administrative procedures or requirements. We have determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The staff has previously published a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be preapred in connection with the issuance of the amendment.

4.0 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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PART 5 OF SAFETY EVALUATION

RELATED TO AMENDMENT NO. 125

1.0 INTRODUCTION

On November 17, 1987, Connecticut Yankee Atomic Power Company (CYAPCO/licensee) submitted a proposed amendment to Facility Operating License No. DPR-61, to add operability requirements for onsite and offsite power sources with limiting conditions for operation (LCO) and time requirements for corrective actions to Technical Specification (TS) 3.12, "Station Service Power." In addition, TS 4.2 "Operational Safety Items," was modified to include requirements for testing and channel calibration of the undervoltage instruments. As a result of a meeting with the NRC on February 25, 1988, CYAPCO revised and combined TS 3.12, "Station Service Power" and TS 4.5, "Emergency Power System Periodic Testing" into a newly titled TS 3/4.8, "Electrical Power System." The new TS submitted August 29, 1988 will: 1) incorporate the degraded grid voltage protection requirements, 2) incorporate emergency diesel generator requirements of Generic Letter (GL) 84-15, "Proposed Staff Actions To Improve and Maintain Diesel Generator Reliability," 3) incorporate industry improvements, 4) change the custom TS format to one that is similar to the Westinghouse Standard Technical Specification (WSTS) format, and 5) incorporate requirements for battery discharge testing as required by the Systematic Evaluation Program (SEP) Topic VIII-3.A, "Station Battery Test Requirements." In addition the degraded grid undervoltage setpoints in the TS were changed. These changes reflect the new station service transformers that were installed during the 1987 refueling outage. The proposed TS were supplemented by additional information provided in letters dated June 9, 1989 July 19 and August 1, 1989. The supplemental letters provided additional bases for several of the TS request. The supplemental information were within the scope of the original notice and did not affect the staff's determination in that original notice. This evaluation relates only to items (1) through (5). A separate Safety Evaluation has been prepared for the degraded grid undervoltage setpoints TS changes.

2.0 DISCUSSION

As part of the SEP, CYAPCO committed to convert their custom formatted TS to the WSTS. Since the conversion effort did not start until October 1988 and with the impending issuance of a newly revised WSTS (merits), the staff proposed that it would be advantageous to await the issuance of the revised WSTS before addressing the full WSTS conversion. In the interim, the staff and CYAPCO agreed that the custom TS format could be upgraded to the WSTS format. The staff concluded that this interim step would: 1) provide a substantially improved TS while facilitating a future conversion effort to the revised WSTS, 2) provide definitive LCO and action statements for several safety related systems, 3) eliminate the use of administrative TS at the Haddam Neck Plant, 4) provide a mechanism to close prior TS commitments associated with NUREG 0737, SEP and various other GL recommendations, and 5) eliminate the ambiguities inherent with the wording and format of the current TS. Based on the above, the staff concluded that the improved TS would enhance public safety and therefore justified this interim step to improve the Haddam Neck TS. The staff has informed CYAPCO several times that this TS upgrade will not fulfill CYAPCO's SEP commitment to convert to the WSTS.

This amendment is one of several that is part of the TS upgrade. By letter dated September 22, 1987, the NRC provided Northeast Utilities with an acceptable revision of the WSTS. The TS upgrade will be using the provided WSTS revision as a guidance while maintaining its current TS requirements. Since the overall upgrade is primarily a format change, the staff did not pursue all deviations and omissions from the provided WSTS with the same intensity as would have been done for a normal WSTS conversion. Therefore, if the proposed TS omits portions of the requirements that appear in the provided WSTS revision and these same requirements did not already exist in the current TS, the review of these omissions will be deferred to the full WSTS conversion. However, where new TS statements have been proposed (statements not previously found in the current TS) that deviate from the provided WSTS revision, a review of the deviation will be given. The deviations will be reviewed in part, based on three previously agreed upon criteria: 1) plant specific design, 2) previously approved hardware, structural or organizational changes, and 3) past operating experiences that can be shown to provide an equivalent degree of protection to that provided by the WSTS. Any deviations from the current custom TS will also be reviewed. The format change and the additional restrictions resulting from this amendment make substantial improvements in the clarity and readability of the TS. As a result, the staff considered this TS upgrade beneficial from both a public safety and an operational perspective.

3.0 EVALUATION

The evaluation has been divided into two sections. Section I will address proposed TS that are consistent with the provided WSTS and/or the current TS. In addition, many of these TS sections will add restrictions to the current TS. Section II will address proposed TS that relax restrictions from either the current TS or the provided WSTS revision. As noted earlier, the staff did not perform a "completeness" review to ensure that all sections of the WSTS were included in the proposed amendment. Therefore, this review will exclude complete omissions of WSTS sections that did not already exist in the current TS. Each of the deviations will be addressed individually. If a GL or a SEP issue has been addressed by the proposed TS change then it will also be noted.

3.1 Section I.

Previously, the NRC staff provided a version of the WSTS to CYAPCO and excluding plant specific alterations, has stated that the provided WSTS would be an acceptable guidance for a STS conversion. Although this amendment is not intended as a STS conversion, CYAPCO has submitted the amendment following the guidance of this WSTS revision. The logic for this TS upgrade has been stated in the Discussion section of this Safety Evaluation. Figure 1 provides a list of proposed TS that are consistent with the provided WSTS and/or the current TS. In many cases the proposed TS impose added restrictions to the current TS or add restrictions that do not currently exist. In all cases, the proposed TS listed in Figure 1 do not relax any of the restrictions found in the current custom TS. Based on the above, the staff has concluded that the TS changes associated with Figure 1 are purely administrative (format change) or provide additional limitations, restrictions, or controls not previously included in the Haddam Neck TS. Therefore, the staff concludes that the proposed TS listed in Figure 1 are acceptable.

3.2 Section II.

The TSs reviewed in this section will be addressed by number and subsection as it appears in the proposed TS. The following clarifications have been provided for this section of the review:

1) The "current (or existing) TS" refers to the TS that is currently part of CYAPCO's operating license.

2) The "admin TS" refers to an administratively controlled TS that CYAPCO has been using in conjunction with the current TS. The admin TS is used by CYAPCO to clarify the current TS and to provide additional requirements that CYAPCO has found advantageous, through past operating experience.

3) The "proposed TS" refers to the TS that CYAPCO has submitted for NRC review as part of the TS upgrade.

4) The "WSTS" refers to the copy of the Westinghouse Standard Technical Specifications that was provided by the NRC to Northeast Utilities. This revision of the WSTS was provided with a letter dated September 22, 1987 and has been used by CYAPCO as a guidance in the proposed TS upgrade. Hereafter, "WSTS" will refer to this revision.

3.8.1.1 LCO b.1

The purpose of the LCO is to require that the diesel generator (DG) be equipped with a separate engine mounted fuel oil tank and to require that a minimum of 400 gallons of fuel oil be maintained in this tank.

The proposed LCO is consistent with the WSTS except that it allows the fuel volume in the tank to drop below the stated minimum volume during DG operation. The fuel oil transfer pumps take suction from the underground fuel oil storage tanks and transfer the fuel oil to the engine mounted tanks. The transfer pumps are controlled by level switches that are set to maintain a level of 400 gallons in the engine mounted tanks. However, the differential setting of the level switches will allow the tank level to drop below the 400 gallons before activating the transfer pumps. Once activated, the pumps will refill the tanks to the required 400 gallon level. Therefore, the TS exception statement is necessary to prevent inadvertent TS violations that would result from the transfer pump controller design. The staff determined that the proposed TS has met the intent of the WSTS and finds the proposed TS to be acceptable.

3.8.1.1 Action a

The principal intent of this Action statement is to limit the time allowed for continued power operation with less than two offsite AC power sources operable. If the failed circuit is not restored within 72 hours then the unit must be in Cold Shutdown within the following 36 hours. During this time, the Action statement requires breaker alignment checks and DG operability tests. The purpose of these checks is to insure that alternate AC power sources are available to maintain the safety function of critical systems.

The proposed TS meets the 72 and 36 hour requirements that are specifically stated in the WSTS and recommended by Regulatory Guide (RG) 1.93. However, the proposed TS deviates from the WSTS in the surveillance intervals between breaker checks and DG tests. In the first deviation, the proposed TS requires the breaker alignment to be checked within 1 hour and every 12 hours thereafter. The WSTS requires the breaker alignment to be checked within 1 hour and every 8 hours thereafter. Both the WSTS and the proposed Action statement assume that an operable offsite circuit and both DGs are available. Following that assumption, there would be an alternate and diverse means to provide AC power to the safety related loads. The intention of the breaker alignment surveillance is to insure that the preferred, operable offsite AC source is available. The proposed TS checks this alignment six times during the 72 hour interval and thereby, does provide assurance that the operable offsite source would be available if needed. In addition, the WSTS 8 hour interval implies that the surveillance should be performed once per shift. CYAPCO has stated that the intent of the proposed 12 hour interval, is that the surveillance will be performed once per shift while allowing some latitude in timing during that shift in which to perform the surveillance. Therefore, the staff finds the proposed deviation to be acceptable. The second deviation from the WSTS is that the DG needs to be demonstrated operable only if either DG has not been successfully tested within the past 24 hours. The WSTS would require that the DGs be tested every 8 hours during the 72 hour interval. As a result, the WSTS requirement could lead to nine DG tests. Following the guidance of GL 84-15 on DG reliability and testing frequency, the staff concluded that nine DG tests would be excessive in this time frame. Futhermore, GL 84-15 states that frequent fast start testing from ambient conditions could result in an increased probability of DG failure. Therefore, after reviewing the basis of a similiar proposed TS change that was previously approved for the North Anna Power Station, Unit 2 (Amendment No. 48 issued April 25, 1985) and using the guidance of GL 84-15, the staff concluded that this deviation is acceptable.

The current TS contains no such Action statements and only requires one offsite power source and one DG to to be operable for power operation. However, CYAPCO currently uses a supplemental admin TS that has similar requirements to the proposed TS and has operated successfully in the past using this supplemental TS. Based on the above, the current TS requirements and the availability of alternate AC sources, the staff has determined that the proposed TS meets the intent of the WSTS Action statement. Therefore, the staff finds the proposed Action statement to be acceptable.

3.8.1.1 Action b

The principal intent of this Action statement is to limit the time allowed for continued power operation with less than two DGs operable. If the inoperable DG is not restored to operable status within 72 hours then the unit must be in Cold Shutdown within the following 36 hours. During this time frame, the Action statement requires breaker alignment checks and the testing of the remaining operable DG. The purpose of these surveillances is to insure that alternate AC sources are available to maintain the safety function of critical systems.

The proposed TS meets the 72 and 36 hour requirements that are specifically stated in the WSTS and are recommended in RG 1.93. However, the proposed TS deviates from the WSTS in the surveillance intervals between breaker alignment checks and DG testing. Following the guidance of RG 1.93, one inoperable onsite source (DC) offers the same severity as the loss of one offsite source.

Since the surveillance intervals for breaker alignment checks and the DG testing are the same as those stated in 3.8.1.1 Action a, the evaluation of these two deviations is consistent with the evaluation of 3.8.1.1 Action a. In addition, a statement has been added to this Action statement that does not require the operable DG to be challenged if the inoperable DG was rendered inoperable due to preplanned maintenance or surveillance testing. If the DG is inoperable due to preplanned maintenance it is assumed that the staggered testing frequency as recommended by GL 84-15, is sufficient to insure that the redundant DG is operable. However, if one of the DGs has become inoperable due to some anomaly, it is necessary to test the remaining operable DG to insure that it has not also been similarly affected. Determining that the redundant DG is operable insures that the critical safety system loads can be powered should they be required. The proposed Action statement is specific and does require that the redundant DG be tested in this situation. This same exception statement was previously approved in Amendment No. 48 issued April 25, 1985 for the North Anna Power Station, Unit No. 2 and the basis for that approval is applicable to Haddam Neck. Currently, CYAPCO's TS do n directly specify an action, or place a time constraint on operation while the plant is in this degraded condition. Based on the above, the current TS requirements, and the evaluation of 3.8.1.1 Action a, the staff has concluded that the proposed TS has met the intent of the WSTS Action statement. Therefore, the staff finds the proposed Action statement to be acceptable.

3.8.1.1 Action c

The principal intent of this Action statement is to limit the time allowed for continued power operation with one offsite AC source and one onsite AC source (DG) inoperable. In addition, the Action statement provides a time constraint during which all AC sources must be made operable. If at least one of the inoperable sources is not restored to operable status within the 12 hours then the unit must be in Cold Shutdown within the following 36 hours. In addition, if all AC sources are not restored within 72 hours then the unit must be in Cold Shutdown within the following 36 hours. During this time frame, the Action statement requires breaker alignment checks and the testing of the remaining DG. The purpose of these surveillances is to insure that the remaining AC sources are operable and available to maintain the safety function of critical systems.

CYAPCO's current TS only requires that one offsite and one onsite AC source be available during power operation. One of the design basis events (DBE) of the plant is a LOCA with a loss of offsite power and a loss of a DG. With one AC offsite and one AC onsite source operable, redundancy is still provided by two diverse sources of power and this factor is considered in the DBE. However, the allowed time for continued operation in this configuration should be kept minimal. The intent of the WSTS is to recognize the severity of the loss of both an onsite and offsite AC power source and to address it accordingly. The proposed TS meets the 12 and 72 hour requirements that are specifically stated in the WSTS and recommended by RG 1.93. As a result, the proposed TS is imposing an additional requirement over the current TS. Therefore, the staff concluded that CYAPCO has recognized the severity of this condition by imposing the added restrictions and by meeting the 12 and 72 hour WSTS requirements. The proposed TS deviates from the WSTS in the surveillance intervals between breaker alignment checks and DG testing and by adding a statement that does not require the operable DG to be challenged if the inoperable DG was rendered inoperable due to preplanned maintenance or surveillance testing. These deviations are consistent with the proposed Action statements a and b. Since the deviations are consistent with the previously proposed .. and the proposed Action statement meets the intent of the WSTS by recognizing the severity of this operating condition and imposing added restrictions to the current TS, the staff finds the proposed Action statement to be acceptable.

3.8.1.1 Action d

The principal intent of this Action statement is to provide assurance that a loss of offsite power event will not result in a complete loss of the safety function of critical systems while one DG is inoperable. The Action statement requires that with one DG inoperable, in addition to Action b or c, the operability of the charging pump, HPSI pump, LPSI pump and RHR which depend on the remaining operable DG as a source of emergency power must be verified. In addition, if these conditions are not satisfied within 2 hours the unit must be in Cold Shutdown within the following 36 hours.

The proposed TS meets the 2 and 36 hour time requirements for continued operation as specifically stated in the WSTS. The deviation from the WSTS arises from the wording of which equipment should be verified operable. The wording of the WSTS provides a general description of equipment that must be operable. The proposed TS provides a specific list of equipment to be verified operable. The listed equipment in the proposed TS is the equipment that the operable DG must carry to maintain the safety function of critical systems. Furthermore, since the intent of the Action statement is to insure that the safety function of critical systems is not lost, the wording of the proposed TS does reflect that intent. In addition, the proposed TS deletes the WSTS references that require verification that the steam-driven auxiliary feedwater pump is operable. This deletion can be justified due to the Haddam Neck Plant design. Unlike the standard Westinghouse plant that has two electric driven and one steam-driven auxiliary feedwater pumps, Haddam has two steam-driven auxiliary feedwater pumps. Therefore, having one inoperable DG would would not significantly affect auxiliary feedwater availability. As a result, the deletion will have no adverse impact on plant safety.

The current TS provides a similar restriction to the proposed TS and lists the same equipment to be verified operable. However, the current TS does not have the shutdown time requirements that the proposed TS has added. Based on the above, the current TS and the additional proposed time constraints, the staff finds the proposed TS to be acceptable.

3.8.1.1 Action e

The principal intent of this Action statement is to minimize the risk associated with two DGs (onsite sources) inoperable while avoiding the risk associated with an immediate shutdown. The Action statement allows 2 hours in which to restore one of the DGs to operable status or be in Hot Standby within the next 6 hours and in Cold Shutdown within the following 30 hours. In addition, if both DGs are not restored to operable status within 72 hours the unit must be in Cold Shutdown within the following 36 hours. During the allowed time for continued operation, the Action statement requires that the offsite AC sources be demonstrated operable by performing breaker alignment surveillances.

The proposed TS meets the 2 and 72 hour requirements that are specifically stated in the WSTS and are recommended by RG 1.93. The deviation from the WSTS is in the surveillance intervals between breaker alignment checks. The surveillance interval in this Action statement is consistent with the intervals found in Action statements a, b and c. Although the severity of this Action statement differs from that of the other Action statements, the staff concluded that the additional restrictions imposed by this Action statement do meet the intent of the WSTS. Since the proposed TS does meet the intent of the WSTS by providing shutdown time requirements where none currently exist and is consistent with previously proposed surveillance intervals, the staff finds the proposed TS to be acceptable.

4.8.1.1.2 Surveillance a.1

The purpose of this surveillance requirement is to verify that the fuel volume in the engine mounted fuel tank is at least 400 gallons.

The proposed TS is consistent with the WSTS except that it allows the fuel volume in the tank to drop below the stated minimum volume during DG operation. This same exception statement appears in proposed TS 3.8.1.1 LCO b.1) and the design circumstances that apply to that TS are also applicable to this surveillance. As a result, the evaluation for 3.8.1.1 LCO b.1) is applicable to this surveillance. Therefore, the staff finds the proposed TS to be acceptable.

4.8.1.1.2 Surveillance a.4

The purpose of this surveillance requirement is to verify that the DG starts from an ambient condition and within 10 seconds is at a designated speed, voltage and frequency. This surveillance is footnoted to provide limitations on the frequency of fast start surveillance testing and to specify that the mechanical stress and wear created by these tests be minimized.

The proposed TS deviates from the WSTS in the wording of the footnote and by not listing the start signals that are listed in the WSTS. The WSTS footnote states that the surveillance testing should be preceded by an engine prelube period and/or manufacturer recommended procedures. The proposed TS states that the testing shall allow for gradual acceleration to reduce stress and wear on the DG. The intent of this footnote is to reflect the recommendations of GL 84-15 and current industry standards for the reduction of wear on DGs. GL 84-15 concluded that an overall improvement in diesel engine reliability can be gained by performing DG starts for surveillance testing using manufacturer recommended procedures. Rather than make the general statement of following the manufacturer's recommendations, CYAPCO has stated that the proposed TS reflects their manufacturer's recommendation of gradual acceleration. Therefore, the proposed TS does meet the intent of the WSTS and GL 84-15 through the proposed wording. In addition, the proposed TS deviates from the WSTS by not providing a specific list of start signals for the DG test. The WSTS provides a diverse list of possible start signals and allows the operator to use any one of the listed signals for test initiation. CYAPCO's current operating procedures already designate how the DG surveillance should be initiated. CYAPCO has operated in the past with the existing procedures and found them to be effective in demonstrating DG reliability. As a result, CYAPCO did not include a list of possible start signals as part of the TS. In view of the diversity of the WSTS list, the staff determined that CYAPCO's operating procedures for DG starting do provide an equivalent level of protection to that of the WSTS. Therefore, the staff finds the proposed deviation to be acceptable.

The proposed TS also deviates from the current TS. The current TS requires that a DG surveillance must be performed monthly. The proposed TS (in the footnote) requires the DG surveillance to be performed once in 184 days. The increased surveillance intervals result in a reduction of DG fast starts which is consistent with the guidance of GL 84-15 and the WSTS. GL 84-15 determined that frequent fast cold starts resulted in undue wear and stress on engine parts. However, GL 84-15 also stated that the demonstration of fast start capability for DGs cannot totally be eliminated. Combining these two conclusions, GL 84-15 provided an acceptable TS to reflect the findings. The sample TS provided by GL 84-15 did specifically state the 184 day interval.

The proposed TS has met the intent of the WSTS by providing criteria to determine whether or not a DG start is successful. Futhermore, it provides additional restrictions over the current TS and incorporates the guidance of GL 84-15. Therefore, the staff finds the proposed TS to be acceptable.

<u>4.8.1.1.2 Surveillance a.5</u>

This surveillance requires verification that once the DG is synchronized and connected to the bus, it is manually loaded to between 2750 KW and 2850 KW in less than or equal to 60 seconds and that it operates in that range for at least 60 minutes. The surveillance statement is footnoted to limit the testing frequency and to require gradual loading for limiting mechanical stress and wear.

The proposed TS deviates from the current TS in the length of time in which the DG is required to remain loaded. The current TS requires that the DG be loaded for 2 hours while the proposed TS requires that the DG be loaded for at least 60 minutes. The intent of the surveillance requirement is to provide sufficient assurance that the DG is available and can successfully operate in a steady state condition. Although the proposed change reduces the length of DG operation required by the current TS, it is consistent with the WSTS, GL 84-15 and the manufacturer's recommendations. In addition, CYAPCO's operating procedures require a 2 hour running time consistent with the current TS and past operating experiences. CYAPCO has stated that they intend to continue running the DG for the 2 hour period but have followed the WSTS in the wording ("at least 60 minutes") of the proposed TS. Therefore, the staff concluded that the proposed TS does meet the intent of the current surveillance, the intent of the WSTS and does reflect the guidance of GL 84-15. The evaluation of the footnote for 4.8.1.1.2 surveillance a.4) also applies to this footnote. Therefore, the staff finds the proposed TS to be acceptable.

4.8.1.1.2 Surveillance b

This surveillance requires the verification that the automatic load sequence timers are within 10% of their design intervals.

The proposed TS deviates from the WSTS in the wording of the surveillance. The WSTS provides a general statement that the interval between each load block is within 10% of its design interval. The proposed TS provides a list of equipment to be sequenced on by the automatic timer and their perspective allowable elapsed times. The allowable elapsed times that are listed in the proposed TS are the exact times as required by the plant design basis. The current TS has no such requirements but CYAPCO has been operating with Admin TS that contain similar requirements to the proposed TS. The staff concluded that the proposed TS meets the intent of the WSTS and finds the proposed TS to be acceptable.

4.8.1.1.2 Surveillance d and e

These two surveillances specify API, water, sediment, viscosity and testing requirements for fuel oil upon delivery and during underground storage.

The proposed TS deviates from the WSTS in wording and API gravity. In particular, the proposed TS does not specify that new fuel oil will be sampled in accordance with ASTM-D4057. Since 1976, CYAPCO has procedurally followed the recommendations of RG 1.137 which references all ASTM procedures necessary to meet the standards. In following RG 1.137, CYAPCO does sample in accordance with ASTM-D4057 and has stated that they will continue to sample in accordance with RG 1.137. However, CYAPCO chose not to include the specific procedural number as a part of TS which may be subject to frequent revisions or replacement. Since the current TS requires no such testing and CYAPCO has successfully used the recommendations of RG 1.137 through its operating procedures in the past, the staff determined that the proposed TS does offer an equivalent level of protection to that of the WSTS. The second deviation is in differing numerical values for API gravity. The values for API gravity in the proposed TS differed from the WSTS by only a small amount. These values were obtained through plant specific data and based on considerable past operating experience. Since no such requirements exist in the current TS and CYAPCO has used these numerical requirements successfully in the past, the staff determined that the numerical variations were acceptable based on the ground rules of this TS upgrade. The staff determined that the added restrictions of the proposed TS do meet the intent of the WSTS and are acceptable.

4.8.1.1.2 Surveillance f

The purpose of this surveillance is to verify that with a loss of offsite power coincident with a Safety Injection Actuation Signal (ESF) the:

1) emergency busses will deenergize and shed load;

2) DG will auto-start and energize the emergency busses with permanently connected loads within 10 seconds and energize the auto-connected shutdown loads and will operate for greater than or equal to 5 minutes (loaded); and maintains voltage and frequency requirements;

3) correct DG trips are bypassed; and

4) DG capability to reject a load of greater than the largest single load.

CYAPCO has submitted the proposed TS f.1 as an equivalent TS to WSTS 4.8.1.1.2.f.4 and WSTS 4.8.1.1.2.f.6. WSTS 4.8.1.1.2.f.4 requires the above mentioned surveillances while simulating a loss of offsite power by itself. WSTS 4.8.1.1.2.f.6 requires the above mentioned surveillance while simulating a loss of offsite power in conjunction with an ESF actuation test signal.

The Haddam Neck Plant uses discrete time delay relays for loading safety injection motors onto the electrical system. Whether a DG start is initiated by an ESF signal or loss of offsite power signal, the same diesel start and loading logic are used. The difference between the signals results from the fact that an ESF signal will also initiate the loading of the safety injection loads. Since a loss of offsite power signal alone or a loss of offsite power in conjunction with an ESF actuation signal will initiate the same diesel start and loading logic, one of the tests will verify the operability of the diesel start and loading logic. By performing the surveillance requiring both the loss of offsite power and the ESF actuation signal, verification of the loading of the safety injection loads and the verification of the diesel start and loading logic are both accomplished. Through surveillance procedures, CYAPCO initiates the proposed TS surveillance first by an undervoltage condition which initiates the DG, and then by an ESF signal which initiates a second DG and the safety injection loads. By initiating the surveillance in this fashion, both initiation signals are tested. Considering the proposed TS in conjunction with the surveillance procedures and the plant hardware design, the staff determined that the proposed TS does meet the intent of the WSTS.

The proposed TS in part 1.b and part 2 deviate from the WSTS by omitting the statement requiring that the voltage and frequency shall be maintained within set limits after the bus is energized. The design of the Haddam Neck Plant on-site power system utilizes two GM/EMD 20 cylinder, turbo charged, low impedance generators. This design uses power current transformers to supply the needed energy to the exciter during motor starts while the voltage is depressed to as low as 50% of the DG rated value. In addition, this design allows for frequency swings during motor starts (loading). During the 1980 refueling outage, a special test was conducted that simulated runout safety injection flow and worst case DG loading. CYAPCO has stated that the test successfully demonstrated the on-site power systems capability to start and run the design basis loads without maintaining the voltage and frequency guidelines as set forth in the WSTS. Based on the above, the staff determined that the plant design would not permit the precise wording of the WSTS without incurring unwarranted TS violations. Therefore, the staff concluded that the proposed TS f.1.b and f.2 are acceptable.

The current TS only requires the demonstration of the readiness of the emergency power system to automatically start and restore power to the vital equipment by initiating a loss of normal AC power to each emergency bus. The detailed requirements of the proposed TS and the WSTS are not in the current TS. However, CYAPCO has been performing the proposed surveillance through their Admin TS in the past. Based on the above reviews of the individual parts of this surveillance, the absence of similar surveillance criteria in the current TS and the added restrictions imposed by the proposed TS, the staff has determined that the proposed TS is acceptable.

4.8.1.1.3 Surveillance-Reports

This TS requires the licensee to report all DG failures to the Commission and include the information recommended in RG 1.108. Additional information is required based on the number of failures within a valid test sample.

The proposed TS deviates from the WSTS by requiring that if the number of failures in the last 20 valid tests is greater than three, additional information will be reported in accordance with RG 1.108. The WSTS requires that if the number of failures in the last 100 valid tests is greater than seven, additional information will be reported in accordance with RG 1.108. Through the guidance of GL 84-15, the testing requirements for DGs were changed. Subsequently, the reporting requirements were also changed. The reporting requirements of GL 84-15 are different from both the WSTS and the proposed TS. However, the proposed TS does incorporate a portion of the reporting requirements found in GL 84-15. Although the proposed TS does not completely follow GL 84-15, the staff determined that it does meet the intent of the GL reporting requirements. Therefore, the staff finds the proposed TS to be acceptable.

Table 4.8-1 Diesel Generator Test Schedule

This table determines the DG testing frequency based on the number of failures in the last 20 valid tests.

As a result of GL 84-15, the testing requirements for DGs were changed to improve reliability and reduce unnecessary DG wear. The GL reduced the testing frequency of the DGs and based the testing criteria on the number of failures per valid tests. GL 84-15 provided a sample modified WSTS that reflected a number of the recommendations found throughout the GL. Included with the sample TS, was a DG test schedule table. The sample TS table included the reduced testing frequency based on the number of failures in the last 20 valid tests. The proposed TS follows the guidance of GL 84-15 and the sample TS table. The WSTS revision used in this TS upgrade presents the testing frequency in a different form and includes tests not required in the GL 84-15. Since the proposed TS table does follow the guidance of GL 84-15, the staff finds it to be acceptable.

3.8.1.2 Action a

When in MODES 5 and 6, the purpose of this Action statement is to immediately suspend all operations involving Core Alterations, positive reactivity changes, movement of irradiated fuel or crane operation with less than one DG and one offsite circuit operable. The Action statement also requires immediate action in MODE 5 if less than two steam generators are operable and in MODE 6 if water level is less than 23 feet above the reactor vessel flange.

The proposed TS deviates from the WSTS by omitting the statement requiring RCS venting. The Haddam Neck Plant has a separate, dedicated system called the Low Temperature Overpressure Protection (LTOP). The LTOP is a system capable of protecting the RCS against pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 315 degrees F. The operation of the LTOP is currently covered by TS 3.3.4.2. The LCO, Applicability and Action statements of TS 3.3.4.2 do coincide with the plant conditions in proposed TS 3.8.1.2. Since the LTOP is capable of venting the RCS and since by TS, the LTOP is required to be operational in the plant conditions of proposed TS 3.8.1.2, the staff determined that the proposed TS does meet the intent of the WSTS. Therefore, the staff finds the proposed TS to be acceptable.

3.8.2.1 Action

The purpose of this Action statement is to limit the time allowed for continued operation with the available onsite DC supplies one less than the LCO. The Action statement allows a short time interval in which the affected DC supply must be restored. If the affected DC supply is not restored within that time, the unit must be in Cold Shutdown within the following 36 hours.

The proposed TS meets the 36 hour requirement that is specifically stated in the WSTS and recommended by RG 1.93. The proposed TS deviates from the WSTS in the time allowed for continued power operation. The proposed TS allows 24 hours of continued operation in comparison to the WSTS which allows 2 hours of continued operation. The primary intent of the 2 hour requirement stated by the WSTS is to minimize the risks associated with only one operable DC source and provide constraints on continued operation. By comparison, the current Haddam Neck [illegible] only requires that one battery charger must be in service and provides no direct Action statements for a degraded condition. However, the proposed Action statement follows from a proposed LCO requiring two battery banks and associated chargers to be operable. It was the opinion of the staff, that when considering the current TS, the added restrictions and clarity of the proposed TS are a substantial improvement over the current TS and do reduce the current risk associated with this degraded condition. Furthermore, the licensee contends that the proposed 24 hour period would allow time to attempt successful repairs on the inoperable DC supply while minimizing the risks associated with continued operation and a forced shutdown with no redundant onsite DC supply. Based on the above and the increased level of safety resulting from the proposed TS, the staff determined that the proposed TS does meet the intent of the WSTS. Therefore, the staff finds the proposed TS to be acceptable.

4.8.2.1 Surveillance b

This surveillance requires that once per 92 days and within 10 days after a battery discharge or overcharge, that specified battery parameters be verified and that the resistance of terminals or connectors be verified to be less than a specified value.

The proposed TS meets the 92 day interval that is specified by the WSTS but deviates in the surveillance interval after a battery discharge or overcharge. The WSTS requires that within 7 days after a discharge or overcharge, that this surveillance be performed. The proposed TS allows a 10 day interval. Currently, CYAPCO has procedures that follow both the manufacturer and IEEE 450 recommendations. These procedures require that an equalizing charge may take as long as 6.5 days to complete at which time the batteries are placed on a float charge for 3 days. CYAPCO has operated with these procedures in the past and has found them to be an effective means in which to verify battery surveillance parameters. Since CYAPCO is following both the manufacturer and industry recommendations and has an effective procedure already in place for this surveillance, the staff finds the proposed TS to be acceptable.

Table 4.8-2 Battery Surveillance Requirements

This table lists the parameters for battery surveillance requirements for weekly and quarterly surveillances.

The proposed TS table is consistent with the WSTS except for slight numerical deviations. The numerical values in the proposed TS reflect both CYAPCO's past operating experiences and the manufacturer's recommendations. Since the intent of this table is to insure that the batteries are maintained in a reliable operating condition, the staff concluded that the plant specific and manufacturer's data warranted the numerical deviations. Based on the above, the staff determined that the proposed TS does meet the intent of the WSTS. Therefore, the staff finds the proposed TS to be acceptable.

3.8.2.2 Action a

The purpose of this Action statement is to immediately suspend all operations involving core alterations, positive reactivity changes, movement of irradiated fuel or crane operation with less than one battery bank and associated charger operable. In addition, the Action statement requires the RCS to be vented.

The proposed TS deviates from the WSTS by omitting the statement requiring RCS venting. As with the proposed Action statement of 3.8.1.2 the LTOP system is also used in this case as an equivalent vent path. Since the deviation and operating conditions of that TS are consistent with this LCO, the evaluation of 3.8.1.2 applies to this LCO. Based on the evaluation of proposed TS 3.8.1.2 the staff finds the Action statement of 3.8.2.2 to be acceptable.

3.8.3.1 Action b

The purpose of this Action statement is to require operator action with the loss of a vital bus and/or its associated inverter. The Action statement provides time constraints in which to restore the vital bus to its normal configuration or be in COLD SHUTDOWN within the following 36 hours.

The proposed TS deviates from the WSTS by allowing an optional means in which to energize a failed vital bus from another source. The proposed optional means is a Haddam Neck Plant specific design feature. The WSTS assumes that there is available an alternate, independent source of power for the vital busses (other than the associated inverter). Accordingly, the WSTS provides a limited time in which to reenergize the vital bus and restore it to its normal operating conditions. The Haddam Neck Plant design does not have an alternate, independent source that can be used to reenergize the vital busses. However, the Haddam design does allow the crosstying of vital busses between inverters. CYAPCO has proposed this option as part of the proposed Action statement.

The current configuration would allow the vital busses to be crosstied across safety divisions. After review, the staff found the crosstying between safety divisions to be unacceptable. However, as part of SEP, CYAPCO committed to the addition of a new separated switchgear room and bus arrangement. The design would permit the DC system to meet current plant design and separation criteria. Along with the new design, the bus arrangement would be altered such that vital busses would have the ability to be crosstied with another inverter within the same safety division. Haddam has built the new switchgear room and intends to put the new configuration in service during the current outage. The electrical portion of the new switchgear room and bus design has been reviewed and approved. Based on the new design the staff has analyzed the proposed course of action. The staff finds the proposed Action statement to be acceptable for the following reasons:

1) The new switchgear room and bus design will maintain the separation (both electrical and physical) between the two safety divisions. Therefore, the crosstying of two vital busses will only be within one safety division. Based on the staff's analysis of the provided information, the staff concluded that it is not acceptable to crosstie between two safety divisions at power.

2) CYAPCO has performed an analysis and determined that a single inverter can adequately carry the loads of two vital busses for the duration of the Action statement.

3) For the duration of the Action statement, a compensatory measure will be taken. This measure will consist of placing the reactor protection system channel of the failed bus in the tripped condition.

4) CYAPCO has performed analysis and determined that the isolation devices at the output of each inverter will protect the crosstied inverter from a faulted condition that may exist on the failed vital bus.

5) The length of continued operation in this configuration will be limited to 72 hours. After 72 hours the plant will shutdown if the vital busses have not been restored to their normal configuration.

6) CYAPCO has stated that the time to reenergize the failed vital bus (8 hours), results from the method by which the loads of the failed vital bus will need to be loaded onto the nonfailed (crosstied) vital bus.

7) A prior Safety Evaluation of SEP topic VI-7.C.1 has stated that with an acceptable new bus separation design (the switchgear room and bus configuration changes), such a crosstie would be permitted.

Furthermore, the staff compared the severity of a failed vital bus without any means to be reenergized with a limited continued operation time for a crosstied configuration. Based on the above review and this comparison, the staff determined that this Action statement is acceptable.

The Haddam Neck Plant design also permits the crosstying of other redundant busses between safety divisions (trains). This evaluation should not be construed as to find such a procedure acceptable. In fact, the staff has found it to be unacceptable to crosstie redundant busses between safety divisions.

3.8.3.1 Action c

This Action statement addresses the operator response and time constraints with one DC bus not energized from its associated battery bank.

The proposed TS deviates from the WSTS in the time allowed for continued operation in this condition. The proposed TS allows 24 hours of continued operation whereas the WSTS allows 2 hours. The current TS simply requires one battery charger to be operable with no definitive Action statements. The added restriction does show CYAPCO's recognition of the severity of this operating condition and does define a course of action for this condition. Furthermore, the proposed time is consistent with the LCO and Action statements of 3.8.2.1. Based on the above, the staff concluded that the proposed TS is acceptable.

3.8.3.2 Action a

During MODES 5 and 6, this Action statement requires operator action with the loss of the electric service busses as listed in the LCO.

The deviation from the WSTS is in both wording and the RCS venting requirement. The operating conditions are consistent with the LCOs 3.8.1.2 and 3.8.2.2. Based on the evaluations of those Action statements, the staff concluded that the proposed Action statement a, of 3.8.3.2, is acceptable.

4.0 SUMMARY

After checking the current TS sections 3.12 and 4.5, the staff determined that the current TS requirements have been maintained by the proposed TS. Furthermore, the proposed amendment offers not only an improved format over the current TS but also adds numerous TS restrictions to plant operation. Based on the considerations discussed in the above evaluation, the staff concluded that the proposed amendment will make overall improvements in the operational safety while maintaining the current safety analysis. Therefore the staff finds the proposed amendment to be acceptable.

5.0 ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. We have determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The staff has previously published a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

We have concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Figure 1

LCO - Limiting condition for operation
APP - Applicability
SURV - Surveillance
ACT - ACTION

TS NUMBER	SUBSECTION	TYPE
3.8.1.1	a	LCO
3.8.1.1	b-2, 3	LCO
3.8.1.1	-	APP
4.8.1.1.1	a	SURV
4.8.1.1.2	a-2,3,6	SURV
4.8.1.1.2	c	SURV
4.8.1.1.2	f-3	SURV
3.8.1.2	a,b	LCO
3.8.1.2	-	APP
3.8.1.2	b	ACT
4.8.1.2	-	SURV
3.8.2.1	a,b	LCC
3.8.2.1	-	APP
4.8.2.1	a	SURV
4.8.2.1	c	SURV
4.8.2.1	d,e,f	SURV
3.8.2.2	-	LCO
3.8.2.2	-	APP
3.8.2.2	b	ACT
4.8.2.2	-	SURV
3.8.3.1	a,b,c,d,e	LCO
3.8.3.1	f,g,n	LCO
3.8.3.1	-	APP
3.8.3.1	a	ACT
4.8.3.1	-	SURV
3.8.3.2	a,b,c,d	LCO
3.8.3.2	-	APP
3.8.3.2	b	ACT
4.8.3.2		SURV
bases		

PART 5A OF SAFETY EVALUATION

RELATED TO AMENDMENT NO. 125

1.0 INTRODUCTION

Connecticut Yankee Atomic Power Company (CYAPCO) replaced station service transformers during the 1987 refueling outage to eliminate a potential PCB hazard. Because of differences in the replacement transformer impedances, the degraded grid voltages available to Haddam Neck safety equipment are different than those previously approved by NRC in the degraded grid operating procedure safety evaluation letter dated July 2, 1985. Evaluations have been made by the licensee with the new transformers in the system under various plant and grid conditions including conditions of degraded grid voltage. As a consequence of these evaluations, CYAPCO proposed by letter dated November 17, 1987, as revised August 29, 1988, to amend the Technical Specification (TS) degraded grid undervoltage setpoints. This safety evaluation covers these changes.

2.0 BACKGROUND

By letter dated October 21, 1981, CYAPCO proposed technical specification changes to include the additional requirements and limiting conditions for operation associated with a degraded grid voltage protection system proposed in response to NRC staff positions letter dated June 3, 1977. The NRC safety evaluation dated July 9, 1982 concluded that the proposed technical specification modifications for degraded voltage were acceptable. However, since manual operator actions were required in response to degraded grid conditions, the staff requested submission of appropriate operating procedures. Accordingly, CYAPCO submitted Abnormal Operating Procedure AOP-3-2-25 on February 3, 1983. By letter dated July 2, 1985, the staff provided a safety evaluation of the AOP procedure, finding that it was acceptable. However, the degraded grid voltage action level numerical values in the procedure were not consistent with those in the TS. Therefore, the staff requested that CYAPCO revise and resubmit the TS to reflect the proper numerical values as contained in the approved procedure. CYAPCO submitted the proposed Technical Specification degraded grid voltage changes by letter dated November 17, 1987. However, due to voltage differences caused by replacement of the feeder transformers, the numerical values are different from those previously approved. This Safety Evaluation (SE) is only for the numerical voltage setpoint change values. A separate SE will evaluate the remaining portions of the licensee's November 17, 1987 (as revised) submittal (Part 5).

3.0 PROPOSED CHANGES

The proposed changes consist of revising Technical Specifications Sections 3.12 and 4.2 as follows:

3.1 Section 3.12, Station Service Power

- Section 3.12 B)(1), revise the 4160 volt emergency bus specification level three undervoltage setpoint range from "below the level three undervoltage setpoint (3980V), but above 3642 volts" to "below the level three undervoltage setpoint (4019V) but above 3684 volts."

- Section 3.12 B)(2), revise the 4160 volt emergency bus specification level two undervoltage setpoint from "3642 volts" to "3684 volts."

- Section 3.12 B) Basis, revise the 4160 volt emergency bus basis undervoltage values from 3980 and 3642 volts, respectively, to 4019 and 3684 volts, respectively.

3.2 Section 4.2, Operational Safety Items

Revise the Table 4.2-1 undervoltage protection calibration setpoints as follows:

- Channel 31, 4.16kV Emergency Bus Undervoltage - Level 2; change both the 4.16kV emergency bus undervoltage level two trip setpoint and allowable value from "3642 volts" to "3684 volts."

- Channel 32, 4.16kV Emergency Bus Undervoltage - Level 3; change both the 4.16kV emergency bus undervoltage level three trip setpoint and allowable value from "3980 volts" to "4019 volts."

4.0 REVIEW CRITERIA/REQUIREMENTS

- NUREG-0452, Standard Technical Specifications for Westinghouse Pressurized Water Reactors, Revision 4.

- Code of Federal Regulations, 10 CFR Part 50 Appendix A, General Design Criteria 17 - Electric Power Systems.

- NRC letter to CYAPCO, Safety Evaluation and Statement of Staff Positions Relative to the Emergency Power Systems for Operating Reactors, June 3, 1977.

5.0 TECHNICAL EVALUATION/DISCUSSION

Review of the November 17, 1987 CYAPCO proposed TS degraded grid voltage revisions consisted of an evaluation of the licensee's basis for the numerical values for the level three and level two undervoltage allowable values and the degraded grid voltage instrumentation setpoint values.

The licensee's basis for revising these values are electrical system impedance changes due to replacing feeder transformers which provide the power to the Class 1E safety-related systems. As a consequence of these changes, the voltages available to the loads are diffe[illegible] and they vary depending upon the conditions of the grid and the magnitude a[illegible] characteristics of the load. The licensee has conducted evaluation case study analyses involving a total of 35 different electric grid supply and load configurations including both steady state and transients in order to envelope the range of voltages which could occur on the 4160 volt safety related buses. Based upon the analyses and upon previously established minimum starting and operating voltages required for the safety-related equipment, the licensee has established the revised 4019 volt level three and 3684 volt level two setpoints and allowable values. The staff has reviewed the licensee analysis and voltage values resulting from the impedance changes due to the replacement of the feeder transformers and find the new values to be acceptable.

6.0 ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. We have determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The staff has previously issued a proposed finding that his amendment involves no significant hazards consideration and there has been no public comment on finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

7.0 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

8.0 REFERENCES DOCUMENTS

- CYAPCO Degraded Grid Voltage Protection Response Letters to NRC June 3, 1977, letters within the period August 1, 1977 to April 21, 1982.
- NRC letter to CYAPCO, Safety Evaluation of Proposed Degraded Grid Voltage Protection System, July 9, 1982.
- CYAPCO Degraded Grid Voltage Protection Systems Proposed Operating Procedures, February 3 and 14, 1985.
- NRC letter to CYAPCO, Safety Evaluation of Degraded Grid Voltage Protection System Operating Procedures, July 2, 1985.
- CYAPCO Revised Degraded Grid Voltage Protection System letter to NRC, November 17, 1987.

Principal Contributor: C. H. Woodard, Region I

Part 6 of Safety Evauation
Related to Amendment No. 125

1.0 INTRODUCTION

Connecticut Yankee Atomic Power Company (licensee) has upgraded portions of the Reactor Protection System (RPS) in two phases. Phase I was evaluated in a previous Safety Evaluation (SE). This SE will provide the evaluation of the second phase of the upgrade. Certain aspects of this evaluation are the same as previously performed for Phase I and will refer to the previous SE where appropriate to reduce repetition. The Phase I RPS upgrade SE was issued to the licensee March 21, 1990. This SER will also evaluate the Nuclear Instrumentation System (NIS) upgrade and the associated Technical Specification (TS) changes for both parts of the Phase II upgrade.

By letter dated September 1, 1988 the licensee submitted preliminary information concerning the RPS Phase II and NIS upgrades. The licensee stated that the information was provided for information purposes only and was not requesting NRC review or approval. The licensee has stated that these changes will be implemented in compliance with 10 CFR 50.59. In addition to the technical evaluation of the physical changes this SE will also address the appropriateness of making the changes via the 10 CFR 50.59 rule.

By letter dated July 28, 1989 the licensee submitted the proposed changes to Technical Specifications associated with the RPS Phase II and NIS upgrades. These changes were described using the new Standard Technical Specification (STS) format. This SE will address only those changes specifically associated with the described upgrades and is not intended to review the remainder of the STS format changes which will be evaluated by a separate SE.

2.0 DESCRIPTION AND EVALUATION

This section will describe the physical changes being implemented, discuss the NRC review criteria and provide our evaluation of the changes.

2.1 Reactor Protection System Phase II Upgrade Description

The RPS Phase II changes are a continuation of the modernization effort of Phase I which includes the replacement of sensors, transmitters and Main Control Board equipment. Phase II is being installed via Plant Design Change Record (PDCR) No. 952. The following Systems are affected:

a) Reactor Coolant System Flow
b) Reactor Coolant System Pressure
c) Primary Containment Pressure

d) Steam Generator Narrow Range Level (transmitter replacement only)
e) Steam Generator Steam Flow (transmitter replacement only)

The Reactor trip relay logic system is being replaced with a solid state Foxboro Spec 200 Micro logic system. This change will involve changing the logic implementation, the field interfaces, bypass and defeat abilities and on-line testing capabilities. In addition, Power Dependent Insertion Limit (PDIL) circuitry is being added to the Rod Control System. The details of each change are listed below

The Reactor Coolant System Low Flow trip circuit has been substantially changed from the existing system. The four (one channel per loop) flow transmitters are being replaced with twelve (three channels per loop) new qualified Foxboro transmitters. The three transmitters per loop will use the same tap so there are no additional pressure boundary penetrations. Each of the three transmitters per channel will be powered from one of the A, C or D vital power buses. The output of each transmitter is input to individual Foxboro Spec 200 microprocessors which compare the flow to the setpoint and provide an electrically isolated (via Foxboro L2CR isolator) output to each of four separate Spec 200 micros. Each of the four microprocessors receives the output from each of the three transmitters and does a 2 out of 3 coincidence which if satisfied provides an isolated output trip signal. Each of the 2/3 comparators is powered from a different vital bus. Each channel (total of 4 channels, one per fluid loop) has four isolated separate trip outputs for a total of sixteen trip output signals.

One output from each loop then is input to another set of four microprocessors (also powered from each of the four vital buses) where the P7 and P8 permissive are compared with the transmitter low flow trip signals. This section is the same as the existing design except that it is accomplished with software within the microprocessor and there are four complete sets of coincidence logic. Two of the four isolated outputs are hardwired together (two out of two) for the Train A breaker trip and the other two are combined (also hardwired two out of two) for the Train B breaker trip. This total logic train from transmitter to breaker is designed such that there is no single failure of sensor, transmitter, microprocessor, cable or power supply that would cause a trip or prevent a valid trip signal. This configuration also allows increased bypass and testing abilities without a single failure during testing causing a reactor trip. This logic configuration is acceptable to the staff.

PDCR 952 will remove the two existing Reactor Coolant System Pressure wide range (0-3000 psig) transmitters from loop 4 and will install qualified wide range (0-3000 psig) and narrow-range (0-600 psig) transmitters on loop 4 and add a redundant pair of wide and narrow range transmitters to loop 1. In addition to the added redundancy the narrow-range transmitter will provide a more accurate pressure signal to the Residual Heat Removal (RHR) system and the Low Temperature Overpressurization System (LTOPS) interlocks. This modification is acceptable to the staff.

The existing six Primary Containment Pressure Switches (mercoid) will be replaced with four Primary Containment Pressure Transmitters. The logic will change from a 2 out of 3 taken twice configuration to 2 out of 4. The new pressure transmitters are expected by the licensee to provide higher accuracy and better repeatability. This change is acceptable to the staff.

The Steam Generator Narrow Range Level transmitters will be replaced with the new qualified transmitters. This change is primarily to replace obsolete equipment with new qualified reliable equipment and is acceptable to the staff.

The Steam Generator Feedwater Flow transmitter upgrade described in the September 1, 1988 submittal has been postponed by the licensee until other feedwater modifications are scheduled are not included in PDCR 952 and therefore are not considered as part of this SE.

2.1.1 System and Hardware Evaluation

The changes described above will use Foxboro transmitters and input/output modules. The Foxboro Spec 200 Micro equipment used are digital microprocessors which use software to implement the various functions. The changes described above will also require new wiring, cable, instrument racks and tubing which are seismically qualified. The requirements that this equipment must meet are the same as the Phase I RPS upgrade and were discussed in the previous SE. The equipment used is the same as Phase I and the applications are similar. The equipment is acceptable to the staff for use in the Phase II upgrade.

The licensee has made many changes which will reduce the possibility of inadvertent reactor trip from a single component or power supply failure. The remaining open issue is potential common mode problems which would defeat the redundancies added by the licensee. The staff reviewed potential common mode failure mechanisms. Use of qualified Class-1E components and verification and validation of software reduces the potential for common mode mechanistic or programming errors to an acceptable level. The one open area which the staff believes was not adequately addressed was the potential for electromagnetic interference or voltage perturbations on the power buses to cause unacceptable operation of several microprocessors at one time.

As described in the Phase I SER, Foxboro had performed specific testing which established a level of electrical environment qualification. The staff required that the licensee determine that the electrical environment at the installed equipment was enveloped by the vendor testing. The staff required that a plan for determining this be submitted to the NRC for review prior to startup from the Phase II installation. The conclusions from Phase I apply for Phase II with additional emphasis due to the new use of microprocessors to implement two out of two reactor trip logic. EMI induced problems in microprocessors may be more complex than a loss of power or electromechanical problem in an analog system.

2.1.2. Software Assessment

The Phase I SE described the verification and validation of the software used in the Foxboro Spec 200 Micro and the Haddam Neck Plant configuration control and found them acceptable for that application. As described in that SE a few notes of caution were listed that some applications may be extremely complex or require extreme speed and were not addressed in that SE. The staff has reviewed the Phase II RPS upgrade and has concluded that the situations which were cautioned against are not being used in Phase II and therefore are not a concern. Each specific segment of the logic which has been implemented with this software is relatively simple which provides a high degree of confidence that the verification and validation reviewed for the Phase I upgrade is adequate for Phase II.

2.2 Nuclear Instrumentation System Upgrade Description

This change will involve replacement of the existing ex-core Nuclear Instrumentation System (NIS) and is implemented with PDCR 954. The ex-core detectors, cabling, scaler/timer, rod disconnect panel, preamplifiers and main control board equipment will be replaced. The Boger Communication Module and the Refueling Cavity Level Indicator will be relocated. The primary equipment supplier will be Gammametrics.

This upgrade will retain the four power range channels but there will now be ten reactor trip steps instead of the previous three. The new ranges are selected by a ten position switch on the Power Range Drawer. Rod Stops will also be increased to ten. The Power Range channels will use fission chambers instead of the previous uncompensated ion chambers.

Four wide range channels will replace the existing two intermediate range channels. These channels will use fission chambers to replace the current compensated ion chambers. Reactor trip on Hi SUR will now be a 2/4 logic.

Four Source Range Channels will replace the existing three source channels and will share the same detectors as the wide range channels.

2.2.1 NIS System and Hardware Evaluation

The new NIS is expected by the licensee to be much more reliable than the old system, easier to maintain and less noise sensitive. This equipment has been previously accepted for use at other facilities.

Northeast Utilities performed a reliability analysis (dated June 23, 1989) for the NIS which concluded that the overall NIS reliability would be improved due to increased redundancies and modern equipment. The previous equipment had become obsolete and was having an increasing negative effect on system and plant availability.

The existing relay matrix logic system which develops the permissives and reactor trips is being replaced with two completely redundant coincidentors. These coincidentors will develop the trips and permissives utilizing solid state electronics. The resultant signal from each coincidentor will then go to the RPS logic cabinets. The NIS upgrade is acceptable to the staff.

2.3 Technical Specifications

The changes to the TS were provided by letter dated July 28, 1989. This letter described the changes using the new Standard Technical Specification (STS) format. This review addresses only those changes specifically related to the RPS Phase II and NIS upgrades. The proposed changes are described below.

1) A definition of Reactor Trip System Response Time was added. A surveillance of response time was also added. A new table providing the RTS instrumentation response times was added.

2) The RTS/ESF and Accident Monitoring tables were revised to reflect the new NIS LCO's and surveillance requirements were added.

3) The NIS Analog Channel Operational test was changed from every 14 to every 41 days. A 41 day requirement of trip actuation and device operational testing was added. The 14 day requirement is no longer required since the equipment operational difficulties have been resolved via replacement with the new NIS. The TS changes are acceptable to the staff.

2.4 RPS Response Time

As a result of the RPS upgrade the time responses for the power range nuclear flux, start-up rate and low-flow reactor trips have been lengthened relative to previous analysis. The change in response times are as follows:

1) Power Range Nuclear Flux(Overpower trip)	0.25 sec. to 0.5 sec.
2) Start-up Rate Reactor Trip	0.4 sec. to 1.0 sec.
3) Low-flow Rate Reactor Trip	1.15 sec. to 1.85 sec.

The licensee determined that of the thirteen transients evaluated for the Chapter 15 non-LOCA transient analysis only seven are affected. CYAPCO has evaluated the affect of the response time delays on the following transients: 1) uncontrolled rod withdrawal from power, 2) uncontrolled rod withdrawal from subcritical, 3) steam line break, 4) RCCA ejection, 5) loss of reactor coolant flow, 6) locked rotor/sheared shaft and 7) idled and isolated loop start-up.

1) Steam Line Break

The steam line break(SLB) is a cooldown event which is concerned with post trip return to power. In the SLB analysis, reactor trip is on high power, caused by positive moderator feedback resulting from the break. The stored energy in the fuel and the initial negative Doppler and moderator reactivity insertion due to fuel and moderator heat up prior to trip are minimized by assuming no delay on the reactor trip signal. If the RPS delay were included, the pre-trip heat-up would result in a slight negative Doppler and more negative moderator reactivity insertion. In addition, the stored energy in the fuel would also be increased. The slight negative reactivity insertion would reduce the positive reactivity contribution from the cooldown. For steam line break transients the consequences are more severe by having an earlier trip. Therefore, delaying the trip will still result in minimum DNBRs and peak fuel centerline temperatures that are bounded by previous analyses.

2) Loss of Reactor Coolant Flow, Locked Rotor/Sheared Shaft and Idled and Isolated Loop Start-Up

For the loss of reactor coolant flow and locked rotor/sheared shaft transients the increase in the low-flow trip delay time has been compensated by a reduction in the conservatism assumed in the radial peaking factors. For the idled and isolated loop start-up transient the increase in the overpower trip response time is also compensated by the reduction in the conservatisms assumed in the radial peaking factors. In all three cases the radial peaking factors, assured by the TSs, continue to bound the analysis assumptions and the predicted minimum DNBRs and peak fuel centerline temperatures are bounded by previous analysis.

3) Uncontrolled Rod Withdrawal from Subcritical

For the uncontrolled rod withdrawal from subcritical transient the increase in the high start-up rate trip response time is compensated for by the new wide range channels. The new wide range channels will be able to detect and trip the event at initiation, while the original equipment could not detect and trip the event until the power levels reached the intermediate range channels. This will result in a trip at a much lower power level. The longer delay times with the expanded sensitivity of the new wide range channels resulted in minimum DNBRs and peak fuel centerline temperatures that are bounded by previous analyses.

4) RCCA Ejection

For the RCCA ejection transient the increase in the overpower trip delay is compensated for by an improved pin census for Cycle 16 and the use of a less conservative, but still bounding, gap conductance. The pin census is a distribution of the number of fuel pins as a function of post ejection radial peaking factor. This census is used in combination with the critical heat flux (CHF) analysis to determine how many fuel pins have a radial peaking factor greater than or equal to that radial peaking factor which results in a calculated DNBR below that which is assumed to result in DNB. The pin census became less severe in Cycle 16. Even though the longer RPS response time resulted in a lower radial peaking factor leading to DNB, the pin census improved so much that the total number of fuel pins calculated to enter DNB is lower than previously calculated. On a monthly basis the axial and radial power distribution are measured. The measured values are compared to calculated values of axial and radial power distribution to confirm they are bounded by the actual values. This provides assurance that the codes are still accurately predicting core behavior. Axial offset, nuclear enthalpy rise hot channel factor, quadrant power tilt ratio, and a core reactivity balance are factors assured by TSs, which confirm that the analysis assumptions and predicted minimum DNBRs and peak fuel centerline temperatures are bounded by the reload analysis during operation. The peak fuel centerline temperatures for the hot full power and hot zero power cases, assuming the increase in response times, are bounded by the previous analysis.

5) Uncontrolled Rod Withdrawal from Power

For the uncontrolled rod withdrawal from power transient the increase in the overpower trip response time does not impact minimum DNBR since it occurs immediately prior to reactor trip. The RCCA withdrawal event is a relatively slow transient. In the minimum DNBR analysis, reactor trip occurs at approximately 83 seconds following event initiation. The rate of temperature rise over this time is slow. Power is also rising slowly. The core power increases an additional 0.1% of full power due to the increased delay time. As reactor trip on high power occurs, system pressure is increasing (2240 psia to 2260 psia in 2 seconds). Reactor trip on high pressure is not credited in order to assess the high power and variable low pressure trips. The rise in pressure compensates for the further increase in core power resulting from the 0.25 second increase in trip delay time. Although the predicted peak fuel centerline temperature increases by 6 F out of 4400 F, the increase is insignificant and does not impact consequences of the transient.

3.0 SUMMARY

The equipment upgrades for both the RPS Phase II and the NIS are acceptable. In addition the staff has concluded the increase in response times for the power range nuclear flux, start-up rate and low-flow reactor trips do not impact the consequences of any design basis event.

3.1 RPS Phase II Upgrade Conclusions

The staff has concluded that the RPS upgrade is acceptable with the exception that qualification to the electrical environment has not been determined. Care must be taken to assure that there is no common mode EMI/SWC problems which could prevent a reactor trip when required. It is also important to assure that no inadvertent trip can occur due to EMI/SWC. The staff requires that the installed configuration of the Foxboro Spec 200 Micro equipment be shown to be enveloped by the vendor testing. The staff requires that the licensee determine the method to be used to verify the electrical environment qualification and document the plan to the staff prior to restart with the RPS Phase II operational.

3.2 NIS Upgrade Conclusions

The NIS upgrade using the Gammametrics system has been previously approved for use at several other plants and is acceptable to the staff for use at Haddam Neck.

3.3 Technical Specification

The TS are consistent with the equipment changes, conform to the STS and are acceptable to the staff as shown in the July 28, 1989 letter.

3.4 10 CFR 50.59 Evaluation

The staff has been aware that the RPS Phase II upgrade would involve the use of Microprocessors since the original audit for Phase I. The Phase I SER described the staff conclusion that the RPS upgrade should not have been done under 10 CFR 50.59 because the change from an analog system to a digital microprocessor system has inherent (software) failure modes which present a malfunction of a different type than previously evaluated and is, therefore, an unreviewed safety question. Since the equipment has been found acceptable and additional guidance to the industry via generic communication is being considered, the staff does not consider this a significant violation. The effects of EMI/SWC may also have a greater impact on the digital systems than on the original analog system. These conclusions also apply to the RPS Phase II.

4.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact have been prepared and published in the Federal Register on February 23, 1990 (55 FR 6563). Accordingly, based upon the environmental assessment, we have determined that the issuance of the amendment will not have a significant effect on the quality of the human environment.

5.0 CONCULSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: J. Stewart
A. Wang

UNITED STATES NUCLEAR REGULATORY COMMISSION

CONNECTICUT YANKEE ATOMIC POWER COMPANY

DOCKET NO. 50-213

NOTICE OF ISSUANCE OF AMENDMENT TO

FACILITY OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (Commission) has issued Amendment No. 125 to Facility Operating License No. DPR-61 issued to Connecticut Yankee Atomic Power Company (the licensee), which revised the Technical Specifications for operation of the Haddam Neck Plant located in Middlesex County, Connecticut. The amendment is effective as of the date of issuance.

The amendment revises the entire current set of Technical Specifications (TS). These TS revisions include: 1) a format change from custom TS to the Westinghouse Standard-format Technical Specifications (WSTS), 2) changes to reflect modifications to the plant such as the new switchgear room (Appendix R), High Pressure Safety Injection Recirculation Path, and Reactor Protection and Nuclear Instrumentation Replacement, 3) changes as recommended by various Generic Letters and changes associated with NUREG 0737 and the Systematic Evaluation Program.

The applications for the amendment comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

9005040159

Notices of Consideration of Issuance of Amendment and Opportunity for Hearing in connection with this action were published in the FEDERAL REGISTER as follows:

(1) Application dated October 26, 1988, as supplemented March 6, June 2, June 23, July 28, August 4, August 21 and November 22, 1989 published on September 11, 1989 (54 FR 37521).

(2) Application dated July 31, 1989 published on September 11, 1989 (54 FR 37519)

(3) Application dated July 28, 1989 published on September 11, 1989 (54 FR 37520).

No request for hearing or petition for leave to intervene was filed following the notices.

The Commission has prepared an Environmental Assessment related to the above items and has determined not to prepare an environmental impact statement. Based upon the environmental assessment, the Commission has concluded that the issuance of this amendment will not have a significant effect on the quality of the human environment.

Notices of Consideration of Issuance of Amendment to Facility Operating License and Proposed no Significant Hazards Consideration Determination and Opportunity for Hearing in connection with this action were published in the FEDERAL REGISTER as follows:

(1) Application dated August 2, 1989 published on September 20, 1989 (54 FR 38763).

(2) Application dated July 28, as supplemented September 29, 1989 published September 20, 1989 (54 FR 38761)

(3) Application dated November 17, 1987, revised August 29, 1988, as supplemented June 9, July 19 and August 1, 1989, published on December 14, 1988 (53 FR 50323).

The supplemental submittals noted above did not affect the staff's initial proposed no significant hazards consideration determination. No request for a hearing or petition for leave to intervene was filed following the notices.

For further details with respect to the action see (1) the applications for amendment as revised and supplemented noted in items (1) through (6) above, (2) Amendment No. 125 to License No. DPR-61, (3) the Commission's concurrently issued Safety Evaluation and (4) the Commission's Environmental Assessment dated February 14, 1990. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street N.W., Washington, D.C. and at the Local Public Document Room located at the Russell Library, 123 Broad Street, Middletown, Connecticut. A copy of items (2), (3) and (4) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Reactor Projects - I/II.

Dated at Rockville, Maryland this 26th day of April 1990.

FOR THE NUCLEAR REGULATORY COMMISSION

Alan Wang

Alan B. Wang, Project Manager
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

UNITED STATES NUCLEAR REGULATORY COMMISSION

CONNECTICUT YANKEE ATOMIC POWER COMPANY

DOCKET NO. 50-213

NOTICE OF PARTIAL DENIAL OF AMENDMENT TO FACILITY OPERATING LICENSE

AND OPPORTUNITY FOR HEARING

The U.S. Nuclear Regulatory Commission (the Comnision) has denied requests by Connecticut Yankee Atomic Power Company (the licensee) for amendment to Facility Operating License No. DPR-61, issued to the licensee for operation of the Haddam Neck Plant located in Middlesex County, Connecticut. The Notice of Consideration of Issuance of this amendment was published in the FEDERAL REGISTER September 11, 1988 (54FR37521).

The NRC staff has concluded that the requests listed below cannot be granted:

1) By submittal dated October 26, 1988, the licensee requested that TS Section 5.3 ., "Fuel Assemblies" be revised to allow insertion of stainless steel filler rods or vacancies as justified by the cycle-specific reload analysis. The staff has deferred the review of this request to the resolution of GL 90-02, "Alternative Requirements For Fuel Assemblies In The Design Features Section Of Technical Specifications".

9005040162

2) By submittal dated June 2, 1989, the licensee requested to add the words "to be repaired" to TS Section 4.4.5.4.a.6, "Plugging Limit".

3) By submittal dated June 2, 1989, the licensee requested that the charging flow indication calibration requirement be removed from the TSs.

4) By submittal dated June 23, 1989, the licensee proposed an additional ACTION (a) to TS Section 3.3.3.2, "The Movable Incore Detector System". The proposed action statement stated that with less than the minimum number of detector thimbles required, the movable incore detector system could be used if penalty factors are applied to the linear heat generation rate or quadrant power tilt; or during recalibration of the system.

By June 4, 1990, the licensee may demand a hearing with respect to the denial described above. Any person whose interest may be affected by this proceeding may file a written petition for leave to intervene.

A request for hearing or petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, N.W., Washington, D.C., by the above date. A copy of any petitions should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, and to Gerald Garfield, Esquire, Day, Berry and Howard, Counselors at Law, City Place, Hartford, Connecticut 06457, attorney for the licensee.

For further details with respect to this action, see (1) the applications for amendment dated October 26, 1988, June 2, and June 23, 1989, and (2) the Commission's letter to the licensee dated April 26, 1990 .

These documents are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, N.W., Washington, D.C. and at the Russell Library, 123 Broad Street, Middletown, Connecticut. A copy of Item (2) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Document Control Desk.

Dated at Rockville, Maryland, this 26th day of April 1990.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Director
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

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Docket Nos. 50-213
50-245
50-336

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Mr. Edward J. Mroczka
Senior Vice President
Nuclear Engineering and Operations
Connecticut Yankee Atomic Power Company
Northeast Nuclear Energy Company
P. O. Box 270
Hartford, Connecticut 06141-0270

Dear Mr. Mroczka:

SUBJECT: RESPONSE TO NRC GENERIC LETTER 89-06 ON THE SAFETY PARAMETER DISPLAY SYSTEM FOR THE HADDAM NECK PLANT AND MILLSTONE NUCLEAR POWER STATION, UNITS 1 & 2 (MPA F-072, TAC NOS. 73664, 73675, 73676)

NRC Generic Letter (GL) 89-06, dated April 12, 1989, requested you to provide certification regarding the implementation of a Safety Parameter Display System (SPDS) at your facility. The GL and its attachment, NUREG-1342, provided clarification of the requirements for an acceptable SPDS as originally defined in NUREG-0737, Supplement 1, issued January 1983. The GL further requested you to complete a checklist and take photographs of your SPDS and to retain these records for three years from the date of certification.

On July 21, 1989, Connecticut Yankee Atomic Power Company and Northeast Nuclear Energy Company certified that the SPDS at Haddam Neck Plant and Millstone Nuclear Power Station, Units 1 & 2 fully meets the requirements of NUREG-0737, Supplement 1, and is consistent with the majority of the information provided in NUREG-1342. Based upon this certification, the NRC staff concludes that your facility has satisfactorily met all the requirements for an SPDS specified in NUREG-0737, Supplement 1. Therefore, staff review and licensee implementation of the SPDS are considered complete for your facility. Please contact me if you have any questions.

Sincerely,

John F. Stolz, Director
Project Directorate I-4
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

cc: See next page

LA:PDI-4	PM:PDI-4	PM:PDI-4	PM:PDI-4	D:PDI-4
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5/3/90	5/8/90	5/3/90	5/3/90	5/3/90

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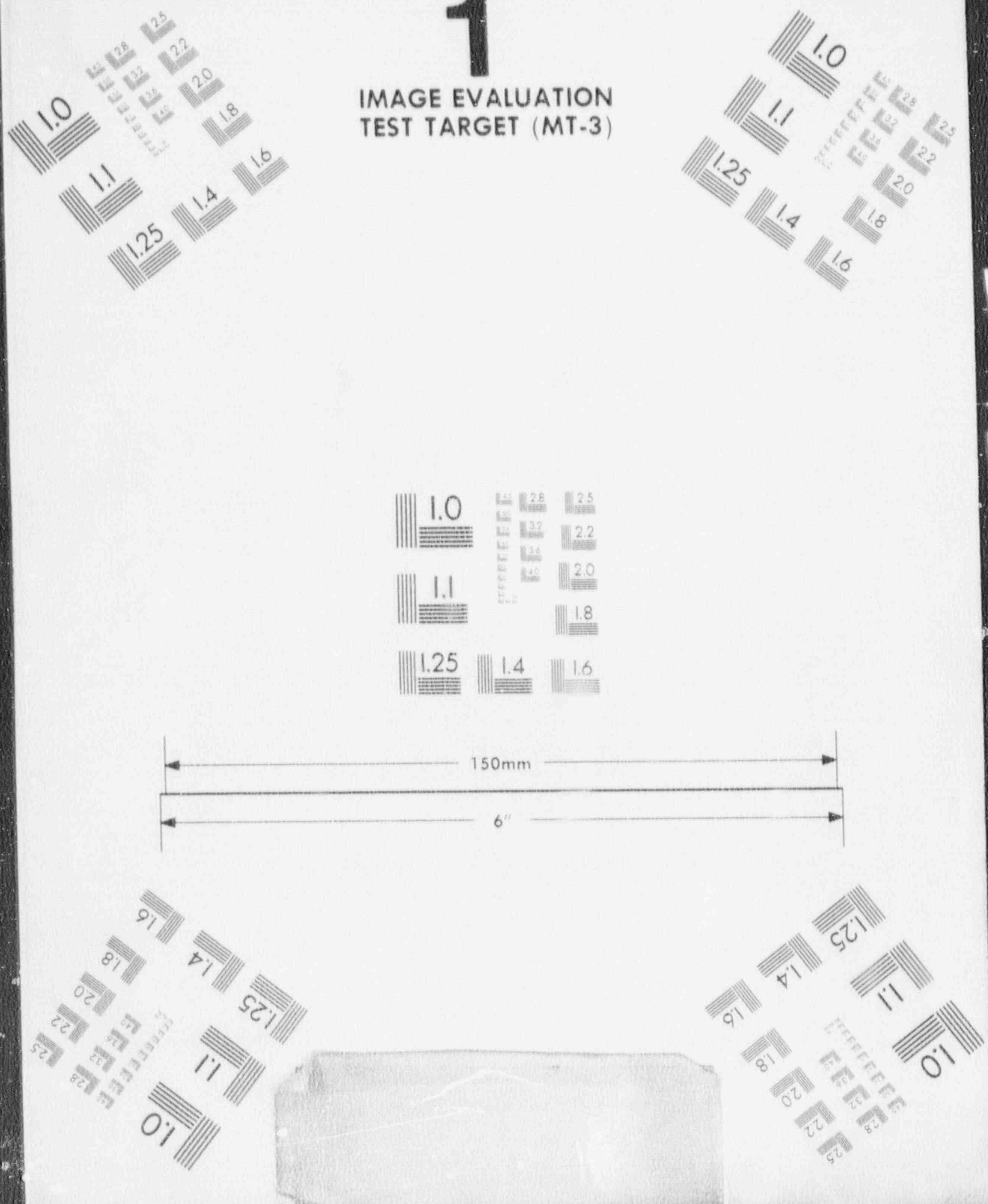
IMAGE EVALUATION TEST TARGET (MT-3)



1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6″

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Proj. Manager

May 7, 1990

MEMORANDUM FOR: All NRR Project Managers

THRU: John T. Larkins, Acting Director
Project Directorate V
Division of Reactor Projects - III,
IV, V, and Special Projects

J.T.L.

FROM: Lawrence E. Kokajko, Project Manager
Project Directorate V
Division of Reactor Projects - III,
IV, V, and Special Projects

SUBJECT: CLOSEOUT OF MPA-X805 (NRC BULLETIN 88-05: Nonconforming Materials Supplied by Piping Supplies, Inc., at Folsom, New Jersey, and West Jersey Manufacturing Company at Williamstown, New Jersey)

Background

NRC Bulletin 88-05, with Supplements 1 and 2, addressed nonconforming materials supplied by Piping Supplies, Incorporated, West Jersey Manufacturing Company, and Chews Landing Metal Manufacturers, Incorporated, during the period of January 1, 1976 to the present. While the staff determined that there was no immediate safety concern at the affected plants due to the inherent safety margins in the specifications of the referenced material, an industry representative (NUMARC), with staff concurrence, agreed to submit a report to resolve this issue.

On July 22, 1988 and October 27, 1989, NUMARC presented reports that addressed the issue of nonconforming materials at nuclear power plants. By letter dated August 30, 1989, NUMARC submitted an additional report which responded to the staff's concerns outlined in our letters dated December 9, 1988 and February 15, 1989. The staff reviewed the report and determined that the report was responsive to the staff's concerns with the exception of a few items. By letter dated November 28, 1989, NUMARC responded to these open items identified in the staff's letter dated November 2, 1989. Upon review of all reports and related documentation, the staff judged the NUMARC responses and methodology to be satisfactory. As a result, MPA-X805 (NRC Bulletin 88-05) is ready to be closed out.

NUREG-1402, entitled "Closeout of NRC Bulletin 88-05: Nonconforming Materials Supplied by Piping Supplies, Inc., at Folsom, New Jersey, and West Jersey Manufacturing Company at Williamstown, New Jersey," was prepared by the staff and closes this issue for operating plants. It is attached for your review as Enclosure 1. NUREG-1402 will be sent to the licensees and other interested parties by the NRC's Regulatory Publications Branch.

In summary, NUREG-1402 states that activities in response to NRC Bulletin 88-05 can be closed for fittings and flanges for all operating plants. Activities can be closed for product forms other than fittings and flanges for operating plants that did not receive such material. For operating plants that did

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receive such material, licensees should determine its location and perform an engineering evaluation where it was used in safety-related applications. These evaluations are to be performed in accordance with 10 CFR 50.59 and are not required to be reviewed by the NRC staff for approval. These evaluations are subject to audits and inspections at the NRC's discretion.

However, Palo Verde Nuclear Generating Station and Perry Nuclear Power Plant must provide a Bulletin 88-05 followup engineering evaluation for nonconforming material other than fittings and flanges. These items were not specifically addressed in the NUMARC program or in NUREG-1402, and were beyond the scope of the original bulletin. Enclosure 2 to this memorandum is a draft letter that should be sent to each licensee by the respective Project Manager. These licensees will be required to submit their evaluations for NRC staff review. These engineering evaluations will be reviewed by EMEB.

Moreover, activities in response to Bulletin 88-05 will remain open for unreviewed NTOL plants. A plant-specific evaluation shall be performed for each NTOL plant before it is licensed.

Project Manager Responsibilities

In order to closeout this item on a plant-specific basis, I will have a global update made for this issue on the WISP database by placing "5/07/90CA" in the Licensing Action Complete field, and "N/A" in the Licensee Implementation field for all appropriate plants. This will include those plants that may have closed out this item due to their respective plants not having any nonconforming material. The new WISP database information should appear on the WISP PMRs within the next 2 weeks. The Implementation Accession Number will be entered when it becomes available, which I will arrange as well.

For those plants that may have closed this item during the initial licensing process, the information that now exists will not change on the WISP database. The plants that are in this category include South Texas 2, Vogtle 2, Shoreham, Seabrook 1, Limerick 2, and Comanche Peak 1.

The Project Managers should review their WISP PMR to verify that this information has been entered correctly. If the information is incorrect, please correct this data, either manually or electronically, to indicate the above-stated information. Also, the Project Managers should verify that their respective plants did receive a copy of NUREG-1402. No further action on the part of the Project Managers is necessary.

This issue will be closed for Palo Verde and Perry on the WISP database also. However, the respective Project Managers for Palo Verde and Perry should send the letter found in Enclosure 2. Upon receipt of the engineering evaluations in response to the letter, the Palo Verde and Perry Project Managers should open a new TAC number (entitled "NRCB 88-05 Followup Engineering Evaluations") for the respective plants and route the information to EMEB for review. Further guidance will come from EMEB based upon the responses provided by the licensees.

If you have any questions regarding NRC Bulletin 88-05, NUREG-1402, or the closeout methodology, please contact me on extension 21380.

Lawrence E. Kokajko, Project Manager
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Division of Reactor Projects - III,
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Enclosures:
1. NUREG-1402
2. Draft Letter to
Palo Verde and Perry

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