NORTHEAST UTILITIES

THE CONNECTICUT LIGHT AND POWER COMPANY
WESTERN MASSACHUSETTS ELECTRIC COMPANY
HOLYOKE WATER POWER COMPANY
NORTHEAST UTILITIES SERVICE COMPANY
NORTHEAST NUCLEAR ENERGY COMPANY

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November 27, 1989

Docket Nos. 50-213
50-245
50-336
50-423
A08329

Re: Generic Letter 89-21
USIs

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Gentlemen:

Haddam Neck Plant
Millstone Nuclear Power Station, Unit Nos. 1, 2, and 3
Response to Generic Letter 89-21
Status of Unresolved Safety Issues

The purpose of this submittal is to provide the status of Unresolved Safety Issues (USIs) in response to Generic Letter 89-21, "Request for Information Concerning Status of Implementation of Unresolved Safety Issue Requirements," received on October 27, 1989.(1) Connecticut Yankee Atomic Power Company (CYAPCO) and Northeast Nuclear Energy Company (NNECO) are providing this information on behalf of the Haddam Neck Plant and Millstone Unit Nos. 1, 2, and 3, respectively, within the requested 30-day response period.

The list of Integrated Safety Assessment Program (ISAP) topics for the Haddam Neck Plant and Millstone Unit No. 1 have included many of the USIs for which plant-specific resolutions have been identified. In addition, a comprehensive discussion of each applicable USI was previously provided in the ISAP Final Report(2),(3) for both plants. These documents proved to be very valuable in

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(1) Nuclear Regulatory Commission letter to All Holders of Operating Licenses and Construction Permits for Nuclear Power Reactors, "Request for Information Concerning Status of Implementation of USI Requirements - Generic Letter 89-21," dated October 19, 1989.

(2) Millstone Unit No. 1--Integrated Safety Assessment Program, Final Report, dated July 31, 1986.

(Footnote Continued)

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determining the current status of the USIs and has once again shown the overall benefit of having a comprehensive and integrated method such as ISAP for addressing issues at the Haddam Neck Plant and Millstone Unit No. 1.

We believe the information provided in Attachments 1, 2, 3, and 4 for Millstone Unit Nos. 1, 2, and 3 and the Haddam Neck Plant, respectively, meets the intent of Generic Letter 89-21. Further, we believe that the information is an accurate representation of the current status of the USIs at our plants. Notwithstanding the above, given the 30-day response interval, relative to the amount of time required to research and analyze data for each plant, estimated at 80 person-hours per plant by the NRC, the information may not be as comprehensive as it could have been, had more time been available to respond.

If you should have any questions, please feel free to contact my staff.

Very truly yours,

CONNECTICUT YANKEE ATOMIC POWER COMPANY
NORTHEAST NUCLEAR ENERGY COMPANY

E. J. Mroczka
Senior Vice President

cc: W. T. Russell, Region I Administrator
M. L. Boyle, NRC Project Manager, Millstone Unit No. 1
G. S. Vissing, NRC Project Manager, Millstone Unit No. 2
D. H. Jaffe, NRC Project Manager, Millstone Unit No. 3
A. B. Wang, NRC Project Manager, Haddam Neck Plant
W. J. Raymond, Senior Resident Inspector, Millstone Unit Nos. 1, 2, and 3
J. T. Shedlosky, Senior Resident Inspector, Haddam Neck Plant

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(Footnote Continued)
(3) Haddam Neck Plant--Integrated Safety Assessment Program, Final Report, dated December 12, 1986.

Docket No. 50-245
A08329

Attachment 1

Millstone Nuclear Power Station, Unit No. 1

Response to Generic Letter 89-21

Status of Unresolved Safety Issues

November 1989

Millstone Nuclear Power Station, Unit No. 1

Unresolved Safety Issues for Which a Final Technical Resolution Has Been Achieved

USI/MPA Number	Title	Reference Document	Applicability	Status/Date*	Remarks
A-1	Water Hammer	SECY 84-119; NUREG-0927, Rev. 1; NUREG-0993, Rev. 1; NUREG-0737, Item I.A.2.3; SRP Revisions	All	I	The resolution of this USI did not involve any hardware or design changes on existing plants, however addressed under ISAP. ISAP Topic 1.43 implementation schedule to be provided in future ISAP/IIS update, as discussed in latest update, dated 9/29/89.
A-2/ MPA D-10	Asymmetric Blowdown Loads on Reactor Primary Coolant Systems	NUREG-0609; GL 84-04; GDC 4	PWR	NA	
A-3	Westinghouse Steam Generator Tube Integrity	NUREG-0844; SECY 86-97; SECY 88-272; GL 85-02 (No Requirements)	W-PWR	NA	
A-4	CE Steam Generator Tube Integrity	NUREG-0844; SECY 86-97; SECY 88-272; GL 85-02 (No Requirements)	CE-PWR	NA	
A-5	B&W Steam Generator Tube Integrity	NUREG-0844; SECY 86-97; SECY 88-272; GL 85-02 (No Requirements)	B&W-PWR	NA	

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I = Incomplete E = Evaluating Actions Required

Millstone Nuclear Power Station, Unit No. 1

Unresolved Safety Issues for Which a Final Technical Resolution Has Been Achieved

USI/MPA Number	Title	Reference Document	Applicability	Status/Date*	Remarks
A-6	Mark I Containment Short-Term Program	NUREG-0408	Mark I-BWR	C (9/84)	Mark I containment long-term program SER dated 09/12/84; FTOL SER Section 6.2.4.[1]
A-7/ D-01	Mark I Long-Term Program	NUREG-0661; NUREG-0661, Supplement 1; GL 79-57	Mark I-BWR	C (9/84)	Mark I containment long-term program SER dated 09/12/84; FTOL SER Section 6.2.4.[1]
A-8	Mark II Containment Pool Dynamic Loads	NUREG-0808; NUREG-0487, Supplement 1/2; NUREG-0802; SRP 6.2.1.1C; GDC 16	Mark II-BWR	NA	
A-9	Anticipated Transients Without Scram	NUREG-0460, Vol. 4; 10CFR50.62	All	C (10/88)	ISAP Topic 1.18 resolved. License Amendment No. 5, dated 7/30/87; ATWS SER dated 10/6/88.

(1) NUREG-1143, Safety Evaluation Report (SER) related to full-term operating license (FTOL) for Millstone Unit No. 1, dated October 1985.

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Millstone Nuclear Power Station, Unit No. 1

Unresolved Safety Issues for Which a Final Technical Resolution Has Been Achieved

USI/MPA Number	Title	Reference Document	Applicability	Status/Date*	Remarks
A-10/ MPA B-25	BWR Feedwater Nozzle Cracking	NUREG-0619; Letter From D. G. Eisenhut dated 11/13/80;	BWR	C (10/81)	NNECO letters to NRC dated 1/22/81 and 10/5/81; NRC letter dated 7/16/81 and NRC SER dated 6/9/83; FTOL SER[1] Section 5.7.
A-11	Reactor Vessel Material Toughness	NUREG-0744, Rev. 1; 10CFR50.60/82-26	All	NC	FTOL SER Section 5.4.1[1]; Tech Spec 4.6.B.5.
A-12	Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports	NUREG-0577, Rev. 1; SRP Revision 5.3.4	PWR	NA	
A-17	Systems Interactions	Letter, DeYoung to Licensees, 9/72; NUREG-1174; NUREG-1229; NUREG/CR-3922; NUREG/CR-4261; NUREG/CR-4470; GL 89-18 (No Requirements)	All	NC	ISAP Topic 1.45 resolved.

(1) NUREG-1143, Safety Evaluation Report (SER) related to full-term operating license (FTOL) for Millstone Unit No. 1, dated October 1985.

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Millstone Nuclear Power Station, Unit No. 1
Unresolved Safety Issues for Which a Final Technical Resolution Has Been Achieved

USI/MPA Number	Title	Reference Document	Applicability	Status/Date*	Remarks
A-24/ MPA B-60	Qualification of Class 1E Safety-Related Equipment	NUREG-0588, Rev. 1; SRP 3.11; 10CFR50.49; GL 82-09; GL 84-24; GL 85-15	All	C (8/87)	ISAP Topic 1.17, resolved as discussed in status report dated 8/4/87. EEQ SER dated 7/30/85.
A-26/ MPA B-04	Reactor Vessel Pressure Transient Protection	DOR Letters to Licensees, 8/76; NUREG-0224; NUREG-0371; SRP 5.2; GL 88-11	PWR	NA	
A-31	Residual Heat Removal Shutdown Requirements	NUREG-0606; RG 1.113; RG 1.139; SRP 5.4.7	All OLs After 01/79	C (10/85)	FTOL SER Section 5.6.2;[1] IPSAR Section 4.0[2]
A-36/ C-10, C-15	Control of Heavy Loads Near Spent Fuel	NUREG-0612; SRP 9.1.5; GL 81-07; GL 83-42; GL 85-11; Letter From D. G. Eisenhut dated 12/22/80	All	C (1/85)	NNECO letter dated 1/14/85; FTOL SER Section 9.5;[1] Generic SER (GL-85-11).

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(1) NUREG-1143, Safety Evaluation Report (SER) related to full-term operating license (FTOL) for Millstone Unit No. 1, dated October 1985.

(2) NUREG-0824, Supplement No.1, Integrated Plant Safety Assessment Report (IPSAR) for Systematic Evaluation Program (SEP) for Millstone Unit No. 1, dated November 1985.

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Millstone Nuclear Power Station, Unit No. 1

Unresolved Safety Issues for Which a Final Technical Resolution Has Been Achieved

USI/MPA Number	Title	Reference Document	Applicability	Status/Date*	Remarks
A-39	Determination of SRV Pool Dynamic Loads and Pressure Transients	NUREG-0802; NUREG-0763; NUREG-0783; NUREG-0802; NUREG-0661; SRP 6.2.1.1.C	BWR	C (9/84)	ISAP Topic 1.46 resolved; Mark I containment long-term program SER dated 9/12/84.
A-40	Seismic Design Criteria	SRP Revisions; NUREG/CR-4776; NUREG/CR-0054; NUREG/CR-3480; NUREG/CR-1582; NUREG/CR-1161; NUREG-1233; NUREG-4776; NUREG/CR-3805; NUREG/CR-5347; NUREG/CR-3509	All	NRC Evaluation Required	ISAP Topic 1.19; NNECO letter dated 2/8/88.
A-42/ MPA B-05	Pipe Cracks in Boiling Water Reactors	NUREG-0313, Rev. 1; NUREG-0313, Rev. 2; GL 81-03; GL 88-01	BWR	NRC Evaluation Required	Response to GL 88-01, dated 7/27/88 and 5/19/89.
A-43	Containment Emergency Sump Performance	NUREG-0510; NUREG-0869, Rev. 1; NUREG-0897; RG 1.82 (Rev. 0); SRP 6.2.2; GL 85-22 (No Requirements)	All	C (5/89)	The resolution of this USI did not require action by existing plants, however addressed under ISAP. ISAP Topic 1.47, completed during Cycle 12 refueling outage, 5/89.

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Millstone Nuclear Power Station, Unit No. 1
Unresolved Safety Issues for Which a Final Technical Resolution Has Been Achieved

USI/MPA Number	Title	Reference Document	Applicability	Status/Date*	Remarks
A-44	Station Blackout	RG 1.155; NUREG-1032; NUREG-1109; 10CFR50.63	All	NRC Evaluation Required	ISAP Topic 1.106; NNECO response to SBO rule submitted to NRC in letter dated 4/17/89. NRC site inspection July 18-21, 1989.
A-45	Shutdown Decay Heat Removal Requirements	SECY 88-260; NUREG-1289; NUREG/CR-5230; SECY 88-260 (No Requirements)	All	NC	Subsumed into IPE program. ISAP Topic 2.28 resolved.
A-46	Seismic Qualification of Equipment in Operating Plants	NUREG-1030; NUREG-1211/GL 87-02; GL 87-03	All	I	Being pursued via SQUG methodology; see NNECO letter dated 9/30/88.
A-47	Safety Implication of Control Systems	NUREG-1217; NUREG-1218; GL 89-19	All	E (3/90)	
A-48	Hydrogen Control Measures and	10CFR50.44; SECY 89-122	All, Except PWRs With	NRC Response Required(3)	

(3) NNECO provided the Staff with a comprehensive discussion of the basis on which our conclusion that combustible gas control for design basis accidents is resolved for Millstone Unit No. 1 and that we fully comply with 10CFR50.44 in its entirety in a letter dated October 15, 1986. On May 1, 1989, the Staff issued a letter which stated that the information supplied in our submittals did not adequately resolve the issue and also requested that a meeting be arranged to review the status of combustible gas control at Millstone Unit No. 1. On May 2, a telephone discussion took place between our Senior Management and Mr. Stephen Varga (and other members of the Staff) to discuss this letter and NNECO's position on the issue, which was that we comply with 50.44, and the NRC has
(Footnote Continued)

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Millstone Nuclear Power Station, Unit No. 1
Unresolved Safety Issues for Which a Final Technical Resolution Has Been Achieved

USI/MPA Number	Title	Reference Document	Applicability	Status/Date*	Remarks
	Effects of Hydrogen Burns on Safety Equipment		Large Dry Containments		
A-49	Pressurized Thermal Shock	RGs 1.154, 1.99; SECY 82-465; SECY 83-288; SECY 81-687; 10CFR50.61/GL 88-11	PWR	NA	

(Footnote Continued)
approved this position via several SERs. At the conclusion of the conversation, the Staff stated that we should not respond to the May 1 letter, and that the Staff will be issuing another letter articulating the Staff's current position.

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Docket No. 50-336
A08329

Attachment 2

Millstone Nuclear Power Station, Unit No. 2

Response to Generic Letter 89-21

Status of Unresolved Safety Issues

November 1989

Millstone Nuclear Power Station, Unit No. 2
Unresolved Safety Issues for Which a Final Technical Resolution Has Been Achieved

USI/MPA Number	Title	Reference Document	Applicability	Status/Date*	Remarks
A-1	Water Hammer	SECY 84-119; NUREG-0927, Rev. 1; NUREG-0993, Rev. 1; NUREG-0737, Item I.A.2.3; SRP Revisions	All	NA	The resolution of this USI did not involve any hardware or design changes on existing plants.
A-2/ MPA D-10	Asymmetric Blowdown Loads on Reactor Primary Coolant Systems	NUREG-0609; GL 84-04; GDC 4	PWR	C (5/88)	NRC SER dated 5/12/88.
A-3	Westinghouse Steam Generator Tube Integrity	NUREG-0844; SECY 86-97; SECY 88-272; GL 85-02 (No Requirements)	W-PWR	NA	
A-4	CE Steam Generator Tube Integrity	NUREG-0844; SECY 86-97; SECY 88-272; GL 85-02 (No Requirements)	CE-PWR	C (6/85)	NNECO letter dated 6/25/85 and SECY 86-97.
A-5	B&W Steam Generator Tube Integrity	NUREG-0844; SECY 86-97; SECY 88-272; GL 85-02 (No Requirements)	B&W-PWR	NA	
A-6	Mark I Containment Short-Term Program	NUREG-0408	Mark I-BWR	NA	
A-7/ D-01	Mark I Long-Term Program	NUREG-0661; NUREG-0661, Supplement 1; GL 79-57	Mark I-BWR	NA	
A-8	Mark II Containment Pool Dynamic Loads	NUREG-0808; NUREG-0487, Supplement 1/2; NUREG-0802; SRP 6.2.1.1C; GDC 16	Mark II-BWR	NA	
A-9	Anticipated Transients Without Scram	NUREG-0460, Vol. 4; 10CFR50.62	All	C (12/88)	NRC SER dated 12/13/88.

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Millstone Nuclear Power Station, Unit No. 2

Unresolved Safety Issues for Which a Final Technical Resolution Has Been Achieved

USI/MPA Number	Title	Reference Document	Applicability	Status/Date*	Remarks
A-10/ MPA B-25	BWR Feedwater Nozzle Cracking	NUREG-0619; Letter From D. G. Eisenhut dated 11/13/80; GL 81-11	BWR	NA	
A-11	Reactor Vessel Material Toughness	NUREG-0744, Rev. 1; 10CFR50.60/82-26	All	C (12/86)	Tech Spec Section B 3/4.4.9, Rev. 113, dated 12/8/86.
A-12	Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports	NUREG-0577, Rev. 1; SRP Revision 5.3.4	PWR	NA	The resolution of this USI contained no backfit requirements, only applied to plants with construction permits issued after October, 1983.
A-17	Systems Interactions	Letter, DeYoung to Licensees, 9/72; NUREG-1174; NUREG-1229; NUREG/CR-3922; NUREG/CR-4261; NUREG/CR-4470; GL 89-18 (No Requirements)	All	NC	
A-24/ MPA B-60	Qualification of Class 1E Safety-Related Equipment	NUREG-0588, Rev. 1; SRP 3.11; 10CFR50.49; GL 82-09; GL 84-24; GL 85-15	All	C (3/85)	NRC SER dated 3/20/85.
A-26/ MPA B-04	Reactor Vessel Pressure Transient Protection	DOR Letters to Licensees, 8/76; NUREG-0224; NUREG-0371; SRP 5.2; GL 88-11	PWR	C (11/88)	Amendment #50 to OL, dated 3/23/76; NNECO letter dated 11/1/88 responding to GL 88-11.

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Millstone Nuclear Power Station, Unit No. 2

Unresolved Safety Issues for Which a Final Technical Resolution Has Been Achieved

USI/MPA Number	Title	Reference Document	Applicability	Status/Date*	Remarks
A-31	Residual Heat Removal Shutdown Requirements	NUREG-0606; RG 1.113; RG 1.139; SRP 5.4.7	All OLs After 01/79	NA	
A-36/ C-10, C-15	Control of Heavy Loads Near Spent Fuel	NUREG-0612; SRP 9.1.5; GL 81-07; GL 83-42; GL 85-11; Letter From D. G. Eisenhut dated 12/22/80	All	C (7/86)	Generic SER (GL 85-11) dated 6/28/85; NNECO letter dated 7/15/86.
A-39	Determination of SRV Pool Dynamic Loads and Pressure Transients	NUREG-0802; NUREG-0763; NUREG-0783; NUREG-0802; NUREG-0661; SRP 6.2.1.1.C	BWR	NA	
A-40	Seismic Design Criteria	SRP Revisions; NUREG/CR-4776; NUREG/CR-0054; NUREG/CR-3480 NUREG/CR-1582; NUREG/CR-1161; NUREG-1233; NUREG-4776; NUREG/CR-3805; NUREG/CR-5347; NUREG/CR-3509	All	NA	No backfitting is required per resolution of USI.
A-42/ MPA B-05	Pipe Cracks in Boiling Water Reactors	NUREG-0313, Rev. 1; NUREG-0313, Rev. 2; GL 81-03; GL 88-01	BWR	NA	

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Millstone Nuclear Power Station, Unit No. 2

Unresolved Safety Issues for Which a Final Technical Resolution Has Been Achieved

USI/MPA Number	Title	Reference Document	Applicability	Status/Date*	Remarks
A-43	Containment Emergency Sump Performance	NUREG-0510; NUREG-0869, Rev. 1; NUREG-0897; RG 1.82 (Rev. 0); SRP 6.2.2; GL 85-22 (No Requirements)	All	NC	In the resolution of this USI the Staff concluded that regulatory analysis did not support any new generic requirements for licensees to perform debris assessments.
A-44	Station Blackout	RG 1.155; NUREG-1032; NUREG-1109; 10CFR50.63	All	NRC Evaluation Required	NNECO response to SBO rule submitted to NRC in letter dated 4/17/89. NRC site inspection July 18-21, 1989.
A-45	Shutdown Decay Heat Removal Requirements	SECY 88-260; NUREG-1289; NUREG/CR-5230; SECY 88-260 (No Requirements)	All	NC	Subsumed into IPE program.
A-46	Seismic Qualification of Equipment in Operating Plants	NUREG-1030; NUREG-1211/GL 87-02; GL 87-03	All	I	Being pursued via SQUG methodology; NNECO letter dated 9/30/88.
A-47	Safety Implication of Control Systems	NUREG-1217; NUREG-1218; GL 89-19	All	E (3/90)	
A-48	Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment	10CFR50.44; SECY 89-122	All, Except PWRs With Large Dry Containments	NA	

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Millstone Nuclear Power Station, Unit No. 2
Unresolved Safety Issues for Which a Final Technical Resolution Has Been Achieved

USI/MPA Number	Title	Reference Document	Applicability	Status/Date*	Remarks
A-49	Pressurized Thermal Shock	RGs 1.154, 1.99; SECY 82-465; SECY 83-288; SECY 81-687; 10CFR50.61/GL 88-11	PWR	C (8/87)	NNECO letters dated 1/23/86, 7/6/87, 7/31/87, and NRC SER dated 8/24/87.

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Docket No. 50-423
A08329

Attachment 3

Millstone Nuclear Power Station, Unit No. 3

Response to Generic Letter 89-21

Status of Unresolved Safety Issues

November 1989

Millstone Nuclear Power Station, Unit No. 3

Unresolved Safety Issues for Which a Final Technical Resolution Has Been Achieved

USI/MPA Number	Title	Reference Document	Applicability	Status/Date*	Remarks
A-1	Water Hammer	SECY 84-119; NUREG-0927, Rev. 1; NUREG-0993, Rev. 1; NUREG-0737, Item I.A.2.3; SRP Revisions	All	C (7/84)	See Section 10.4.9 of the SER.[1]
A-2/ MPA D-10	Asymmetric Blowdown Loads on Reactor Primary Coolant Systems	NUREG-0609; GL 84-04; GDC 4	PWR	C (11/85)	See Section 3.9.3 of the SER Supplement 4.[1]
A-3	Westinghouse Steam Generator Tube Integrity	NUREG-0844; SECY 86-97; SECY 88-272; GL 85-02 (No Requirements)	W-PWR	C (7/84)	See Appendix C of the SER.[1]
A-4	CE Steam Generator Tube Integrity	NUREG-0844; SECY 86-97; SECY 88-272; GL 85-02 (No Requirements)	CE-PWR	NA	
A-5	B&W Steam Generator Tube Integrity	NUREG-0844; SECY 86-97; SECY 88-272; GL 85-02 (No Requirements)	B&W-PWR	NA	
A-6	Mark I Containment Short-Term Program	NUREG-0408	Mark I-BWR	NA	
A-7/ D-01	Mark I Long-Term Program	NUREG-0661; NUREG-0661, Supplement 1; GL 79-57	Mark I-BWR	NA	

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(1) NUREG-1031, Safety Evaluation Report related to the operation of Millstone Unit No. 3, Supplement 1 through 5.

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Millstone Nuclear Power Station, Unit No. 3
Unresolved Safety Issues for Which a Final Technical Resolution Has Been Achieved

USI/MPA Number	Title	Reference Document	Applicability	Status/Date*	Remarks
A-8	Mark II Containment Pool Dynamic Loads	NUREG-0808; NUREG-0487, Supplement 1/2; NUREG-0802; SRP 6.2.1.1C; GDC 16	Mark II-BWR	NA	
A-9	Anticipated Transients Without Scram	NUREG-0460, Vol. 4; 10CFR50.62	All	C (7/89)	NNECO letter dated 7/24/89.
A-10/ MPA B-25	BWR Feedwater Nozzle Cracking	NUREG-0619; Letter From D. G. Eisenhut dated 11/13/80; GL 81-11	BWR	NA	
A-11	Reactor Vessel Material Toughness	NUREG-0744, Rev. 1; 10CFR50.60/82-26	All	C (7/84)	See Section 5.3 of the SER.(1)
A-12	Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports	NUREG-0577, Rev. 1; SRP Revision 5.3.4	PWR	C (7/84)	See Section 5.2 of the SER.(1)
A-17	Systems Interactions	Letter, DeYoung to Licensees, 9/72; NUREG-1174; NUREG-1229; NUREG/CR 3922; NUREG/CR-4261; NUREG/CR-4470; GL 89-18 (No Requirements)	All	NC	

(1) NUREG-1031, Safety Evaluation Report related to the operation of Millstone Unit No. 3, Supplement 1 through 5.

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Millstone Nuclear Power Station, Unit No. 3

Unresolved Safety Issues for Which a Final Technical Resolution Has Been Achieved

USI/MPA Number	Title	Reference Document	Applicability	Status/Date*	Remarks
A-24/ MPA B-60	Qualification of Class 1E Safety-Related Equipment	NUREG-0588, Rev. 1; SRP 3.11; 10CFR50.49; GL 82-09; GL 84-24; GL 85-15	All	C (1/86)	See Section 3.11 of SER Supplement 5.[1]
A-26/ MPA B-04	Reactor Vessel Pressure Transient Protection	DOR Letters to Licensees, 8/76; NUREG-0224; NUREG-0371; SRP 5.2; GL 88-11	PWR	NRC Evaluation Required	NNECO response to GL 88-11 dated 11/1/88.
A-31	Residual Heat Removal Shutdown Requirements	NUREG-0606; RG 1.113; RG 1.139; SRP 5.4.7	All OLs After 01/79	C (7/84)	See Section 5.4.7 of SER.[1]
A-36/ C-10, C-15	Control of Heavy Loads Near Spent Fuel	NUREG-0612; SRP 9.1.5; GL 81-07; GL 83-42; GL 85-11; Letter From D. G. Eisenhut dated 12/22/80	All	C (9/85)	See Section 9.1.5 of SER Supplement 2.[1]
A-39	Determination of SRV Pool Dynamic Loads and Pressure Transients	NUREG-0802; NUREG-0763; NUREG-0783; NUREG-0802; NUREG-0661; SRP 6.2.1.1.C.	BWR	NA	

(1) NUREG-1031, Safety Evaluation Report related to the operation of Millstone Unit No. 3, Supplement 1 through 5.

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Millstone Nuclear Power Station, Unit No. 3
Unresolved Safety Issues for Which a Final Technical Resolution Has Been Achieved

USI/MPA Number	Title	Reference Document	Applicability	Status/Date*	Remarks
A-40	Seismic Design Criteria	SRP Revisions; NUREG/CR-4776; NUREG/CR-0054; NUREG/CR-3480; NUREG/CR-1582; NUREG/CR-1161; NUREG-1233; NUREG-4776; NUREG/CR-3805; NUREG/CR-5347; NUREG/CR-3509	All	C (7/84)	See SER.(1)
A-42/ MPA B-05	Pipe Cracks in Boiling Water Reactors	NUREG-0313, Rev. 1; NUREG-0313, Rev. 2; GL 81-03; GL 88-01	BWR	NA	
A-43	Containment Emergency Sump Performance	NUREG-0510; NUREG-0869, Rev. 1; NUREG-0897; RG 1.82 (Rev. 0); SRP 6.2.2; GL 85-22 (No Requirements)	All	C (11/85)	See Section 6.2.2 of SER Supplement 4.(1)
A-44	Station Blackout	RG 1.155; NUREG-1032; NUREG-1109; 10CFR50.63	All	NRC Evaluation Required	NNECO response to SBO rule submitted to NRC in letter dated 4/17/89. NRC site inspection July 18-21, 1989.
A-45	Shutdown Decay Heat Removal Requirements	SECY 88-260; NUREG-1289; NUREG/CR-5230; SECY 88-260 (No Requirements)	All	C (7/84)	See Section 10.4.9 of SER.(1)

(1) NUREG-1031, Safety Evaluation Report related to the operation of Millstone Unit No. 3, Supplement 1 through 5.

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I = Incomplete E = Evaluating Actions Required

Millstone Nuclear Power Station, Unit No. 3

Unresolved Safety Issues for Which a Final Technical Resolution Has Been Achieved

USI/MPA Number	Title	Reference Document	Applicability	Status/Date*	Remarks
A-46	Seismic Qualification of Equipment in Operating Plants	NUREG-1030; NUREG-1211/ GL 87-02; GL 87-03	All	NA	
A-47	Safety Implication of Control Systems	NUREG-1217; NUREG-1218; GL 89-19	All	E (3/90)	
A-48	Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment	10CFR50.44; SECY 89-122	All, Except PWRs With Large Dry Containments	NA	
A-49	Pressurized Thermal Shock	RGs 1.154, 1.99; SECY 82-465; SECY 83-288; SECY 81-687; 10CFR50.61/GL 88-11	PWR	NRC Evaluation Required	NNECO response to GL 88-11 dated 11/1/88.

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Docket No. 50-213
A08329

Attachment 4

Haddam Neck Plant

Response to Generic Letter 89-21

Status of Unresolved Safety Issues

November 1989

Haddam Neck Plant

Unresolved Safety Issues for Which a Final Technical Resolution Has Been Achieved

USI/MPA Number	Title	Reference Document	Applicability	Status/Date*	Remarks
A-1	Water Hammer	SECY 84-119; NUREG-0927, Rev. 1; NUREG-0993, Rev. 1; NUREG-0737, Item I.A.2.3; SRP Revisions	All	C (2/80) 12/31/86	NUREG 0927, Rev. 1, 3/84 and NRC Staff letter dated 2/26/80 provided final closeout.
A-2/ MPA D-10	Asymmetric Blowdown Loads on Reactor Primary Coolant Systems	NUREG-0609; GL 84-04; GDC 4	PWR	C (7/89) 6/16/89	CYAPCO letter dated 6/16/89; NRC Staff letter dated 7/11/89 provided final closeout on GL 84-04.
A-3	Westinghouse Steam Generator Tube Integrity	NUREG-0844; SECY 86-97; SECY 88-272; GL 85-02 (No Requirements)	W-PWR	C (8/87) 12/31/86	ISAP Topics 1.49 and 2.06 (both topics closed by NRC Staff in NUREG-1185, 8/18/87).
A-4	CE Steam Generator Tube Integrity	NUREG-0844; SECY 86-97; SECY 88-272; GL 85-02 (No Requirements)	CE-PWR	NA	
A-5	B&W Steam Generator Tube Integrity	NUREG-0844; SECY 86-97; SECY 88-272; GL 85-02 (No Requirements)	B&W-PWR	NA	
A-6	Mark I Containment Short-Term Program	NUREG-0408	Mark I-BWR	NA	
A-7/ D-01	Mark I Long-Term Program	NUREG-0661; NUREG-0661, Supplement 1; GL 79-57	Mark I-BWR	NA	

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Haddam Neck Plant

Unresolved Safety Issues for Which a Final Technical Resolution Has Been Achieved

USI/MPA Number	Title	Reference Document	Applicability	Status/Date*	Remarks
A-8	Mark II Containment Pool Dynamic Loads	NUREG-0808; NUREG-0487, Supplement 1/2; NUREG-0802; SRP 6.2.1.1C; GDC 16	Mark II-BWR	NA	
A-9	Anticipated Transients Without Scram	NUREG-0460, Vol. 4; 10CFR50.62	All	NRC Evaluation required	ISAP Topic 1.16 (final information to support closeout provide by CYAPCO in letters dated 5/27/88 and 10/26/88; see ISAP reports dated 11/13/87 and 3/2/89.
A-10/ MPA B-25	BWR Feedwater Nozzle Cracking	NUREG-0619; Letter From D. G. Eisenhut dated 11/13/80; GL 81-11	BWR	NA	
** A-11	Reactor Vessel Material Toughness	NUREG-0744, Rev. 1; 10CFR50.60/82-26	All	N C (10/82)	NUREG-0744, 10/82; CYAPCO letter dated 10/13/82.
A-12	Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports	NUREG-0577, Rev. 1; SRP Revision 5.3.4	PWR	~~C (8/87)~~ NA	NUREG-0577, 10/83; ISAP Topic 1.50 (closed out by NRC Staff in NUREG-1185, 8/18/87).

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Haddam Neck Plant

Unresolved Safety Issues for Which a Final Technical Resolution Has Been Achieved

USI/MPA Number	Title	Reference Document	Applicability	Status/Date*	Remarks
A-17	Systems Interactions	Letter, DeYoung to Licensees, 9/72; NUREG-1174; NUREG-1229; NUREG/CR-3922; NUREG/CR-4261; NUREG/CR-4470; GL 89-18 (No Requirements)	All	NC	ISAP Topic 1.51 (reported as "resolved" by CYAPCO in ISAP report dated 11/13/87 and 3/2/89.)
A-24/ MPA B-60	Qualification of Class 1E Safety-Related Equipment	NUREG-0588, Rev. 1; SRP 3.11; 10CFR50.49; GL 82-09; GL 84-24; GL 85-15	All	~~C (8/87)~~ C December 86	ISAP Topic 1.17 (closed out by NRC Staff in NUREG-1185, 8/18/87). EEQ SER dated 2/13/87.
A-26/ MPA B-04	Reactor Vessel Pressure Transient Protection	DOR Letters to Licensees, 8/76; NUREG-0224; NUREG-0371; SRP 5.2; GL 88-11	PWR	C (5/88)	ISAP Topic 1.52; CYAPCO letter dated 10/13/82; CYAPCO ISAP report dated 11/13/87; NRC Staff letter closing out dated 5/12/88; see also CYAPCO response to GL 88-11 dated 11/1/88.
A-31	Residual Heat Removal Shutdown Requirements	NUREG-0606; RG 1.113; RG 1.139; SRP 5.4.7	All OLs After 01/79	I	NUREG-0826; Generic resolution complete. Follow-up item remains open. ISAP Topic 1.02 remains open; see CYAPCO ISAP report dated 3/2/89.

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Haddam Neck Plant

Unresolved Safety Issues for Which a Final Technical Resolution Has Been Achieved

USI/MPA Number	Title	Reference Document	Applicability	Status/Date*	Remarks
A-36/ C-10, C-15	Control of Heavy Loads Near Spent Fuel	NUREG-0612; SRP 9.1.5; GL 81-07; GL 83-42; GL 85-11; Letter From D. G. Eisenhut dated 12/22/80	All	C (7/86)	CYAPCO letters dated 7/20/81, 4/16/82, 7/30/82, 7/21/83; final resolution in letter dated 7/15/86; Generic SER (GL 85-11).
A-39	Determination of SRV Pool Dynamic Loads and Pressure Transients	NUREG-0802; NUREG-0763; NUREG-0783; NUREG-0802; NUREG-0661; SRP 6.2.1.1.C.	BWR	NA	
A-40	Seismic Design Criteria	SRP Revisions; NUREG/CR-4776; NUREG/CR-0054; NUREG/CR-3480; NUREG/CR-1582; NUREG/CR-1161; NUREG-1233; NUREG-4776; NUREG/CR-3805; NUREG/CR-5347; NUREG/CR-3509	All	~~I~~ NA	ISAP Topics 1.04, 6/92 (for completion); 1.05, 6/92 (for completion); 1.08, resolved, CYAPCO letter 3/2/89 (see 3/2/89 ISAP report for all 3 topics).
A-42/ MPA B-05	Pipe Cracks in Boiling Water Reactors	NUREG-0313, Rev. 1; NUREG-0313, Rev. 2; GL 81-03; GL 88-01	BWR	NA	
A-43	Containment Emergency Sump Performance	NUREG-0510; NUREG-0869, Rev. 1; NUREG-0897; RG 1.82 (Rev. 0); SRP 6.2.2; GL 85-22 (No Requirements)	All	N C ~~(11/87)~~	ISAP Topic 1.53 closed (see CYAPCO ISAP reports dated 11/13/87 and 3/2/89).

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I = Incomplete E = Evaluating Actions Required

Haddam Neck Plant

Unresolved Safety Issues for Which a Final Technical Resolution Has Been Achieved

USI/MPA Number	Title	Reference Document	Applicability	Status/Date*	Remarks
A-44	Station Blackout	RG 1.155; NUREG-1032; NUREG-1109; 10CFR50.63	All	~~NRC Evaluation Required~~ I (6/30/92)	CYAPCO letter 4/17/89 responded to SBO rule; ISAP Topic 1.116 (new), ISAP Topics 2.08 and 2.12, closed (see NUREG-1185, dated 8/18/87).
A-45	Shutdown Decay Heat Removal Requirements	SECY 88-260; NUREG-1289; NUREG/CR-5230; SECY 88-260 (No Requirements)	All	NC	Subsumed into IPE program. CYAPCO letter dated 10/13/82.
A-46	Seismic Qualification of Equipment in Operating Plants	NUREG-1030; NUREG-1211/GL 87-02; GL 87-03	All	I	Being pursued via SQUG methodology, ISAP Topic 1.48 (see CYAPCO letter dated 9/30/88).
A-47	Safety Implication of Control Systems	NUREG-1217; NUREG-1218; GL 89-19	All	E (3/90)	ISAP Topic 1.54 (see ISAP report dated 3/2/89).

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Haddam Neck Plant

Unresolved Safety Issues for Which a Final Technical Resolution Has Been Achieved

USI/MPA Number	Title	Reference Document	Applicability	Status/Date*	Remarks
A-48	Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment	10CFR50.44; SECY 89-122	All, Except PWRs With Large Dry Containments	~~NA~~ NC	
A-49	Pressurized Thermal Shock	RGs 1.154, 1.99; SECY 82-465; SECY 83-288; SECY 81-687; 10CFR50.61/GL 88-11	PWR	C (11/87)	ISAP Topic 1.61 (reported as resolved in CYAPCO ISAP reports, 11/13/87 and 3/2/89); see also CYAPCO response to GL 88-11 dated 11/1/88.

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ENCLOSURE 2

PLANT Haddam Neck DOCKET NO(S). 50-213

PROJECT MANAGER Alan B. Wang TECHNICAL CONTACT A. Serkiz

USI NO. A-1 TITLE Water Hammer

MPA NO. N/A TAC NOS.

ISSUES SUMMARY:

This Unresolved Safety Issue (USI) was resolved in March 1984, with the publication of NUREG-0927, "Evaluation of Water Hammer in Nuclear Power Plants - Technical Findings Relevant to Unresolved Safety Issue A-1." Also on March 15, 1984, the EDO sent the Commissioners SECY 84-119 titled, "Resolution of Unresolved Safety Issue A-1, Water Hammer."

In SECY 84-119, the staff concluded that the frequency and severity of water hammer occurrences had been significantly reduced through (a) incorporation of design features such as keep-full systems, vacuum breakers, J-tubes, void detection systems, and improved venting procedures; (b) proper design of feedwater valves and control systems; and (c) increased operator awareness and training. Therefore, the resolution of USI A-1 did not involve any hardware or design changes on existing plants. It did involve Standard Review Plan (SRP) changes (forward fits) and a comprehensive set of guidelines and criteria to evaluate and upgrade utility training programs (per TMI Task Action Plan Item I.A.2.3). In addition, the assumption was made that for BWRs with isolation condensers (ICs) a reactor-vessel high water-level feedwater pump trip was in place or being installed. This was necessary because calculated values had postulated an IC failure by water hammer that opened a direct pathway to the environment.

IMPLEMENTATION AND STATUS SUMMARY (PLANT SPECIFIC):

Based on the findings of NUREG-0927, CYAPCO considered to USIA-1 be resolved for the Haddam Neck Plant. By letter dated February 26, 1983, the Staff previously found the means for minimizing steam generator waterhammer at the Haddam Neck Plant to be acceptable. The staff concluded that the following actions would limit the consequences of water hammer at the Haddam Neck site:

1) Downward turning elbow on each steam generator nozzle eliminates the horizontal feedwater piping at the entrance to the steam generator.

2) Administrative controls require operators to maintain steam generator water levels at 25% to 50% of narrow range when feedwater is being manually controlled.

REFERENCES:

Haddam Neck
A-1

1. REQUIREMENT DOCUMENTS:

TITLE	NUDOCS NO.	DATE
Letter from Denton to Utilities, "Notice of Issuance and Availability NUREG-0927 Rev. 1, Safety Issue A-1"	8403150310	03/05/84

2. IMPLEMENTATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE
NUREG-0927 "Evaluation of Water Hammer in Nuclear Power Plants-Technical Findings Relevant to Unresolved Safety Issue A-1"	8306060413	05/31/83
NUREG-0993 Rev. 1 "Regulatory Analysis for for USI A-1, Water Hammer"	8306060418	March 1984
SRP Sections: 3.9.3, 3.9.4, 5.4.6, 5.4.7, 6.3, 9.2.1, 9.2.2, 10.3, and 10.4.7		
SECY-84-119, "Resolution of Unresolved Safety A-1, Water Hammer"		03/15/84
Steam Generator Water Hammer, Generic Issue NRC Acceptance of CY Response	8004030365	02/26/80
NU ISAP Final Report	8702040321	12/86
NUREG 1185, ISAR	879090221	07/87

PLANT Haddam Neck DOCKET NO(S). 50-213

PROJECT MANAGER Alan B. Wang TECHNICAL CONTACT Jai Rajan

USI NO. A-2 TITLE Asymmetric Blowdown Loads in RCS

MPA NO. D-10 TAC NOS.

ISSUES SUMMARY:

This USI was resolved in January 1981 with the publication of NUREG-0609, "Asymmetric Blowdown Loads on PWR Primary Systems."

In October 1975, the NRC notified each operating PWR licensee of a potential safety problem concerning the fact that asymmetric LOCA loads had not been considered in the design of any PWR piping system. In June 1976 the NRC informed each PWR licensee that it was required to reassess the reactor vessel support design of its facility. The staff expanded the scope of the problem in January 1978 with a request for additional information to all PWR licensees. NUREG-0609 provided guidance for these analyses. For operating PWRs, Multi-Plant Action (MPA) Item D-10 was established by NRC's Division of Licensing for implementation purposes.

During the course of the work on USI A-2, it was demonstrated that there were only a very limited number of break locations which could give rise to significant loads. Subsequently, after substantial new technical work, it was demonstrated that pipes would leak before break and that new fracture mechanics techniques for the analyzing of piping failures assured adequate protection against failures in primary system piping in PWRs (Generic Letter 84-04). This was reflected in a revision of General Design Criteria (GDC)-4 (Appendix A to 10 CFR Part 50) published in the Federal Register in final form on April 11, 1986, and in a subsequent revision to GDC-4 published in the Federal Register on July 23, 1986. In addition, it has also been satisfactorily demonstrated in the course of the A-2 effort that there is a very low likelihood of simultaneous pipe loading with both LOCA and safety shutdown earthquake (SSE) loads. Therefore, the last revision of GDC-4 represented the final technical action of NRC regarding the issue of asymmetric blowdown loads issue in PWRs primary coolant main loop piping.

IMPLEMENTATION AND STATUS SUMMARY (PLANT SPECIFIC):

GL 84-4 required Haddam Neck to specifically respond to certain criteria. By letter dated 6/16/89, CYAPCO confirmed that they met the two criteria required by GL 84-04 and the "leak-before-break" philosophy is applicable to the Haddam Neck Plant. USI A-2 is considered to be resolved.

REFERENCES:

Haddam Neck
A-2

1. REQUIREMENT DOCUMENTS:

TITLE	NUDOCS NO.	DATE
Generic Letter "Evaluation of Primary Systems for Asymmetric LOCA Loads"		01/20/78
Task Action Plan A-2, "Asymmetric Blowdown Loads on Reactor Primary Coolant System," NUREG-0371 Task Action Plans for Generic Activities		11/78
"Asymmetric Blowdown Loads on PWR Primary Systems," NUREG-0609 US NRC NRR		01/81
GDC-4, "Environmental and Dynamic Effects Design Basis"		
GL 84-04, "Safety Evaluation of Westinghouse Topical Reports Dealing With Elimination of Postulated Pipe Breaks in PWR Primary Main Loops."		

2. IMPLEMENTATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE
Asymmetric LOCA Loads on Reactor Vessel Intervals	8906270017	06/16/89
GL 84-04 Closeout Letter	8907180092	07/11/89

3. VERIFICATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE

PLANT Haddam Neck DOCKET NO(S). 50-213

PROJECT MANAGER Alan B. Wang TECHNICAL CONTACT E. Murphy

USI NO. A-3, A-4, an A-5 TITLE Steam Generator Tube Integrity

MPA NO. TAC NOS.

ISSUES SUMMARY:

USIs A-3, 4, and 5, were resolved in September 1988 with the publication of NUREG-0844 "NRC integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity." USIs A-3, A-4, and A-5 did not result in new generic requirements for industry in view of the small potential for reducing risk.

Steam generator tube integrity was designated an unresolved safety issue in 1978 after it became apparent that steam generator tubes were subject to widespread degradation, frequent leaks, and occasional ruptures (i.e., gross failures). USI Task Action Plans A-3, A-4, and A-5 were established to evaluate the safety significance of these problems for Westinghouse, Combustion Engineering, and Babcock & Wilcox steam generators, respectively. These studies were later combined into a single effort because PWR vendors were all experiencing many of the same problems.

NUREG-0844 provides a generic risk assessment that indicates that risk from steam generator tube rupture (SGTR) events is not a significant contributor to the total risk at a given site, nor to the total risk to which the general public is routinely exposed. This finding is considered indicative of the effectiveness of licensee programs and regulatory requirements for ensuring steam generator tube integrity in accordance with 10 CFR Part 50, Appendices A and B.

NUREG-0844 also identifies a number of staff-recommended actions that can further improve the effectiveness of licensee programs in ensuring the integrity of steam generator tubes and in mitigating the consequences of a SGTR. As part of the integrated program, the staff issued Generic Letter 85-02 encouraging licensees of PWRs to upgrade their programs, as necessary, to meet the intent of the staff-recommended actions; however, such recommended actions do not constitute NRC requirements. The staff's assessment of licensee responses to Generic Letter 85-02 was provided to the Commission in SECY 86-97.

IMPLEMENTATION AND STATUS SUMMARY (PLANT SPECIFIC):

Steam generator tube integrity was reviewed in the ISAR by Topics 1.49, "Steam Generator Tube Integrity," 2.01, "Secondary Side Chemistry Monitoring" 2.02, "Demineralized Water Storage Tank Oxygen Reduction, 2.03, "Additional Atmospheric Steam Pump" and 2.06, "Evaluation of RCS Loop Isolation Valves to Mitigate SGTR." These topics included: 1) installation of new SG chemistry monitoring panels, 2) local and remote readout in the chemistry lab and 3) continuing studies to control oxygen in the main feedwater and provide separate steam dump liner such that one dump line services no more than two steam generators. Based on these ISAP reviews, this USI is considered resolved.

REFERENCES: Haddam Neck
A-3, 4, 5

1. REQUIREMENT DOCUMENTS:

TITLE	NUDOCS NO.	DATE
NUREG-0844, "NRC Integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity"		September 1988
Generic Letter 85-02		04/17/85
SECY-86-97, Steam Generator USI Program - Utility Responses to Staff Recommendations in Generic Letter 85-02		03/04/86

2. IMPLEMENTATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE
NUREG 1185, "ISAR"	8709090221	07/87
NU ISAP Letter	8903080205	03/02/89

3. VERIFICATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE

PLANT Haddam Neck DOCKET NO(S). 50-213

PROJECT MANAGER Alan B. Wang TECHNICAL CONTACT J. Mauck

USI NO. A-9 TITLE ATWS per 10 CFR 50.62

MPA NO. ______ TAC NOS. ______

ISSUES SUMMARY:

This USI was resolved in June 1984 with the publication of a final rule (10 CFR 50.62) to require improvements in plants to reduce the likelihood of failure of the reactor protection system (RPS) to shut down the reactor following anticipated transients and to mitigate the consequences of an anticipated transient without scram (ATWS) event.

The rule includes the following design-related requirements: 50.62(C)(1), diverse and independent auxiliary feedwater initiation and turbine trip for all PWRs; 50.62(C)(2), diverse scram systems for CE and B&W reactors; 50.62(C)(3) alternate rod injection (ARI) for BWRs; 50.62(C)(4); standby liquid control system (SLCS) for BWRs; and 50.62(C)(5), automatic trip of recirculation pumps under conditions indicative of an ATWS for BWRs. Information requirements and an implementation schedule are also specified.

IMPLEMENTATION AND STATUS SUMMARY (PLANT SPECIFIC):

By letter dated August 19, 1986, Haddam Neck requested an exemption from some of the turbine trips of the ATWS rule. This exemption is being reviewed and should be issued shortly.

REFERENCES:

1. REQUIREMENT DOCUMENTS:

TITLE	NUDOCS NO.	DATE
NUREG-0460, and Supplements, "Anticipated Transients Without Scram for Light Water Reactors"		03/80
Federal Register Notice 49 FR 26045 (10 CFR 50.62)		06/26/84

2. IMPLEMENTATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE
ATWS Exemption Request	8608260132	08/19/86
RAI for ATWS Exemption	8903090445	03/02/89
Response to ATWS RAI	880610002501	05/27/88
Response to ATWS RAI		10/26/88
Request for Additional Information on ATWS		10/26/89
Response to ATWS RAI		12/15/89

3. VERIFICATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE

PLANT Haddam Neck DOCKET NO(S). 50-213

PROJECT MANAGER ______ TECHNICAL CONTACT D. Thatcher

USI NO. A-17 TITLE Systems Interactions in Nuclear Power Plants

MPA NO. ______ TAC NOS. ______

ISSUES SUMMARY:

Generic Letter (GL) 89-18, dated September 6, 1989, was sent to all power reactor licensees and constitutes the resolution of USI A-17. The generic letter did not require any licensee actions.

GL 89-18 had two enclosures which (a) outlined the bases for the resolution of USI A-17, and (b) provided five general lessons learned from the review of the overall systems interaction issue. The staff anticipated that licensees would review this information in other programs, such as the Individual Plant Examination (IPE) for Severe Accident Vulnerabilities. Specifically, the staff expected that insights concerning water intrusion and flooding from internal sources, as described in the appendix to NUREG-1174, would be considered in the IPE program. Also considered in the resolution of this USI was the expectation that licensees would continue to review information on events at operating nuclear power plants in accordance with the requirements of TMI Task Action Plan Item I.C.5 (NUREG-0737).

IMPLEMENTATION AND STATUS SUMMARY (PLANT SPECIFIC):

CYAPCO has addressed this issue through the Haddam Neck Probabilistic Safety Study. This was discussed in ISAR Topic 1.51. CYAPCO considers this topic to be resolved pending any future requirements as a results of the staff's technical resolution of this issue.

REFERENCES:

Haddam Neck
A-17

1. REQUIREMENT DOCUMENTS:

TITLE	NUDOCS NO.	DATE
Generic Letter 89-18		09/06/89
NUREG-1174 "Evaluation of Systems Interactions in Nuclear Power Plants"		May 1989
NUREG-1229 "Regulatory Analysis for Resolution of USI A-17"		August 1989
NUREG/CR-3922 "Survey and Evaluation of System Interaction Events and Sources"		January 1985
NUREG/CR-4261 "Assessment of System Interaction Experience in Nuclear Power Plants"		June 1986
NUREG/CR-4470 "Survey and Evaluation of Vital Instrumentation and Control Power Supply Events"		August 1986
NRC Letters to Licensees Informing Licensees of Staff Concerns Regarding Potential Failure of Non-Category I Equipment		9/72

2. IMPLEMENTATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE

3. VERIFICATION DOCUMENTS:

TITLE	NUDOC NO.	DATE
NU ISAP Final Report	8702040321	12/86
NUREG 1185, ISAR	8709090221	07/87
NU ISAP Letter	8903080205	03/02/89

PLANT Haddam Neck DOCKET NO(S). 50-213

PROJECT MANAGER Alan B. Wang TECHNICAL CONTACT P. Shemanski

USI NO. A-24 TITLE Qualification of Class 1E Equipment

MPA NO. ________ TAC NOS. ________

ISSUES SUMMARY:

This USI was resolved in July 1981 with the publication of NUREG-0588, Revision 1, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment." Part I of the report is the original NUREG-0588 that was issued for comment; that report, in conjunction with the Division of Operating Reactor (DOR) Guidelines, was endorsed by a Commission Memorandum and Order as the interim position on this subject until "final" positions were established in rule making. On January 21, 1983 the Commission amended 10 CFR 50.49 (the rule), effective February 22, 1983, to codify existing qualification methods in national standards, regulatory guides, and certain NRC publications, including NUREG-0588.

The rule is based on the DOR Guidelines and NUREG-0588. These provide guidance on (a) how to establish environmental service conditions, (b) how to select methods which are considered appropriate for qualifying the equipment in different areas of the plant, and (c) such other areas as margin, aging, and documentation. NUREG-0588 does not address all areas of qualification; it does supplement, in selected areas, the provisions of the 1971 and 1974 versions of IEEE Standard 323. The rule recognizes previous qualification efforts completed as a result of Commission Memorandum and Order CLI-80-21 and also reflects different versions IEEE 323, dependent on the date of the construction permit Safety Evaluation Report (SER). Therefore, plant-specific requirements may vary in accordance with the rule.

In summary, the resolution of A-24 is embodied in 10 CFR 50.49. A measure of whether each licensee has implemented the resolution of A-24 may therefore be found in the determination of compliance with 10 CFR 50.49. This was addressed by 72 SERs for operating plants issued shortly after publication of the rule and subsequently in operating license reviews pursuant to Standard Review Plan Section 3.11. This was further addressed by the first-round environmental qualification inspections conducted by the NRC.

IMPLEMENTATION AND STATUS SUMMARY (PLANT SPECIFIC):

In a December 28, 1983 letter, CYAPCO requested a schedular exemption for 14 valve motor operators. These were the only components yet to be qualified to achieve compliance with 10 CFR 50.49. On April 5, 1984, the staff granted a schedular extension to March 31, 1985. In a February 28, 1985 letter, CYAPCO requested a schedular extension until November 30, 1985. In a letter dated March 28, 1985, the staff granted this extension and noted any further schedular extensions must be granted by the Commission. During the 1986 refueling outage, all 14 motor operators were replaced to achieve compliance with 10 CFR 50.49. The staff considers this issue resolved.

REFERENCES:

Haddam Neck
A-24

1. REQUIREMENT DOCUMENTS:

TITLE	NUDOCS NO.	DATE
DOR "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors"		
NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety Related Electrical Equipment"		12/79
Commission Memorandum and Order, CLI-80-21, on DOR Guidelines and NUREG-0588		05/23/80
NUREG-0588, Revision 1		07/81
10 CFR 50.49 (48 FR 2730-2733)		01/21/83
Standard and Review Plan 3.11, Environmental Qualification of Mechanical and Electrical Equipment		07/81

2. IMPLEMENTATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE
Schedular Exemption Request From 10 CFR 50.49	831230047	12/28/83
Exemption from 10 CFR 50.49	8404250090	04/05/84
Request for Extension to Schedular Exemption	8503110469	02/28/85
Exemption from 10 CFR 50.49	8504040346	03/28/85
EEQ SER	8702240367	02/13/87

3. VERIFICATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE

PLANT Haddam Neck DOCKET NO(S). 50-213

PROJECT MANAGER Alan B. Wang TECHNICAL CONTACT Chu Liang

USI NO. A-26 TITLE Reactor Vessel Pressure Transient Protection

MPA NO. ______ TAC NOS. ______

ISSUES SUMMARY:

This USI was resolved in September 1978 with the publication of NUREG-0224, "Reactor Vessel Pressure Transient Protection for PWRs," and Standard Review Plan Section 5.2. The licensees of all operating PWRs were requested to provide an overpressure prevention system that could be used whenever the plants were in startup or shutdown conditions. The issue affected all operating and future plants, and the staff established MPA B-04 for implementing the solution at operating PWRs.

Since 1972, there have been numerous reported incidents of pressure transients in PWRs where technical specification pressure and temperature limits have been exceeded. The majority of these events occurred while the reactors were in a solid-water condition during startup or shutdown and at relatively low reactor vessel temperatures. Since the reactor vessels have less toughness at lower temperatures, they are more susceptible to brittle fracture under these conditions than at normal operating temperatures. In light of the frequency of the reported transients and the associated potential for vessel damage, the NRC staff concluded that measures should be taken to minimize the number of future transients and reduce their severity.

Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its Impact on Plant Operations," was published July 12, 1988. This generic letter provides guidance regarding review of pressure-temperature limits and indicates that licensees may have to revise low-temperature-overpressure protection setpoints.

IMPLEMENTATION AND STATUS SUMMARY (PLANT SPECIFIC):

By letter dated March 6, 1978, CYAPCO provided a low temperature overpressure protection system design. By letter dated October 14, 1982, CYAPCO informed the staff that the low temperature overpressure protection system had been installed and the operating procedures had been implemented.

Region I had informed the staff that CYAPCO has had to revise the operating set points for the installed LTOP system because of operational problems at the current set points. Additionally, the Region is concerned with the wording for procedures associated with LTOP. The Region believes the procedures associated with the LTOP need to be clarified to avoid possible confusion in the operation of the LTOP. These concerns are referenced in Inspection Report No. 50-213/87-06 dated March 27, 1987.

IMPLEMENTATION AND STATUS SUMMARY (PLANT SPECIFIC):

These concerns were addressed in IR No. 50-213/88-03 dated April 21, 1988. In particular the IR verified the procedures for LTOP were implemented and reflect the current TS. Based on this IR, a closeout letter was issued on May 12, 1988 for this USI.

REFERENCES:

Haddam Neck
A-26

1. REQUIREMENT DOCUMENTS:

TITLE	NUDOCS NO.	DATE
NUREG-0224 - "Reactor Vessel Pressure Transient Protection for PWRs."		9/78
NRC Letters to Licensees Informing Licensees of Staff Concerns Regarding Overpressure Low-Temperature Conditions in PWRs		August 1976
Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations"		7/12/88
Standard Review Plan Section 5.2		

2. IMPLEMENTATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE
LTOP Design		03/06/78

3. VERIFICATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE
LTOP Implemented		10/14/82
IE Report 50-213/87-06	8704020513	03/27/87
IE Report 50-213/88-03	8805030143	04/21/88
SEP Topics II-3B, II-3.C and ISAR Topic 1.52	8805230168	05/12/88

PLANT Haddam Neck DOCKET NO(S). 50-213

PROJECT MANAGER Alan B. Wang TECHNICAL CONTACT R. Jones

USI NO. A-31 TITLE RHR Shutdown Requirements

MPA NO. ______ TAC NOS. ______

ISSUES SUMMARY:

This USI was resolved in May 1978 with the publication of Standard Review Plan (SRP) Section 5.4.7. Only those plants expected to receive an operating license after January 1, 1979 were affected by this resolution. The USI involved establishment of criteria for the design and operation of systems necessary to take a power reactor from normal operating conditions to cold shutdown.

SRP Section 5.4.7 stated that, for purposes of implementation, plants would be divided into three classes: Class 1 would require full compliance for Construction Permit (CP) or Preliminary Design Approval (PDA) applications which were docketed on or after January 1, 1978. Class 2 required a partial implementation for all plants for which CP or PDA applications were docketed before January 1, 1978, and for which an Operating License (OL) issuance was expected on or after January 1, 1979. Class 3 affected all operating reactors and all other plants for which issuance of the OL was expected before January 1, 1979. The extent to which Class 3 plants would require implementation was based on the combined staff review of related plant features. In general, the outcome of these evaluations were that only plants receiving an OL after January 1, 1979 were affected by this USI resolution, and there were no backfits to operating plants that had received an operating license before January 1, 1979.

IMPLEMENTATION AND STATUS SUMMARY (PLANT SPECIFIC):

The issue was included in the Haddam Neck SEP review as Topic V-10.B. Under this topic, the Haddam Neck Plant cooldown capability equipment was specifically reviewed against the criteria of SRP 5.4.7 and BTP RSB 5-1.

In NUREG-0826 (the Haddam Neck SEP Final Report), Section 4.19, the NRC Staff generally found that equipment was adequate to meet the topic safety objective. However, the SEP topic evaluation did identify five issues for the Haddam Neck Plant requiring resolution. Four of these were resolved, in a manner found acceptable by the Staff in NUREG-0825, by technical specification or operating procedure revisions. A final Staff concern involved installation of interlocks on the RHR-to-core deluge motor-operated valves to prevent opening until the RCS pressure is below design pressure. Installation of high/low pressure valve interlocks are included within the scope of ISAP as Topic No. 1.02 However, as discussed under that topic, the RHR system has already been modified to demonstrate compliance with current licensing criteria. CYAPCO is still studying the need for interlocks or some other alternatives.

REFERENCES:

1. REQUIREMENT DOCUMENTS:

TITLE	NUDOCS NO.	DATE
NUREG-0800 "Standard Review Plan," SRP Section 5.4.7		5/78
NUREG-0606 "Unresolved Safety Issues Summary"		
Regulatory Guide 1.139, "Guidance for Residual Heat Removal"		
Regulatory Guide 1.113		

2. IMPLEMENTATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE
NUREG-0826, "IPSAR, SEP, Haddam Neck"	8601230285	6/83
NUREG 1185, "ISAR"	8709090221	07/87
NU ISAP Letter	8903080205	03/02/89

3. VERIFICATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE

PLANT Haddam Neck DOCKET NO(S). 50-213

PROJECT MANAGER Alan B. Wang TECHNICAL CONTACT J. Wermiel

USI NO. A-36 TITLE Control of Heavy Loads, Phases I & II

MPA NO. C-10, C-15 TAC NOS.

ISSUES SUMMARY:

This USI was resolved in July 1980 with the publication of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," and Standard Review Plan (SRP) Section 9.1.5. The staff established MPAs C-10 and C-15 for the implementation of Phases I and II, respectively, of the resolution of this issue at operating plants.

In nuclear power plants, heavy loads may be handled in several plant areas. If these loads were to drop in certain locations in the plant, they may impact spent fuel, fuel in the core, or equipment that may be required to achieve safe shutdown and continue decay heat removal. USI A-36 was established to systematically examine staff licensing criteria and the adequacy of measures in effect at operating plants, and to recommend necessary changes to ensure the safe handling of heavy loads. The guidelines proposed in NUREG-0612 include definition of safe load paths, use of load handling procedures, training of crane operators, guidelines on slings and special lifting devices, periodic inspection and maintenance for the crane, as well as various alternatives.

By Generic Letters dated December 22, 1980, and February 3, 1981 (Generic Letter 81-07), all utilities were requested to evaluate their plants against the guidance of NUREG-0612 and to provide their submittals in two parts: Phase I (six month response) and Phase II (nine month response). Phase I responses were to address Section 5.1.1 of NUREG-0612 which covered the following areas:

1. Definition of safe load paths
2. Development of load handling procedures
3. Periodic inspection and testing of cranes
4. Qualifications, training and specified conduct of operators
5. Special lifting devices should satisfy the guidelines of ANSI N14.6.6.
6. Lifting devices that are not specially designed should be installed and used in accordance with the guidelines of ANSI B30.9
7. Design of cranes to ANSI B30.2 or CMAA-70

Phase II responses were to address Sections 5.1.2 thru 5.1.6 of NUREG-0612 which covered the need for electrical interlocks/mechanical stops, or alternatively, single-failure-proof cranes or load drop analyses in the spent fuel pool area (PWR), containment building (PWR), reactor building (BWR), other areas and the specific guidelines for single-failure-proof handling systems.

As stated in Generic Letter 85-11, "Completion of Phase II of 'Control of Heavy Loads at Nuclear Power Plants' - NUREG-0612," all licensees have completed the requirement to perform a review and submit a Phase I and a Phase II report. Based on the improvements in heavy loads handling obtained from implementation of NUREG-0612 (Phase I), further action was not required to reduce the risks associated with the handling of heavy loads. Therefore, a detailed Phase II review of heavy loads was not necessary and Phase II was considered completed.

While not a requirement, NRC encouraged the implementation of any actions identified in Phase II regarding the handling of heavy loads that were considered appropriate.

IMPLEMENTATION AND STATUS SUMMARY (PLANT SPECIFIC):

NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" represents the NRC Staff's resolution of USI A-36. Implementation of the generic resolution was planned in two phases. By letter dated December 22, 1980, CYAPCO was specifically requested to implement the interim actions describe by the Staff. The NRC Staff also requested CYAPCO and all licensees of operating plants to review their control for the handling of heavy loads to determine the extent to which the guidelines of NUREG-0612 are satisfied at their facilities. Any long-term modifications identified would constitute Phase II of the resolution.

CYAPCO initially responded to this issue by letters dated July 29, 1981, July 16, 1982 and July 30, 1982. In addition, by letter dated 7/21/83, CYAPCO provided a number of commitments and other information relative to the Phase II evaluation of the NUREG-0612 heavy load issues for the Haddam Neck Plant. As indicated in GL 85-11, dated June 28, 1985, the NRC has concluded that the risk associated with potential heavy load drops is acceptably small and that the objective of NUREG-0612 for providing "maximum practical defense in depth" is satisfied. Therefore, the NRC concluded that a detailed Phase II review of this issue was not necessary and, accordingly, that Phase II is considered complete. By letter dated July 15, 1986, CYAPCO considers this issue resolved.

REFERENCES:

Haddam Neck
A-36

1. REQUIREMENT DOCUMENTS:

TITLE	NUDOCS NO.	DATE
Letter, Darrell G. Eisenhut, NRC, to all licensees, applicants for OLs and holders of CPs transmitting NUREG-0612 and staff positions		12/22/80
Generic Letter 85-11, Hugh L. Thompson, NRC, to all licensees for Operating Reactors, "Completion of Phase II of 'Control of Heavy Loads at Nuclear Power Plants' NUREG-0612"		06/28/85

2. IMPLEMENTATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE
Task Action Plan for Unresolved Safety Issue Task 36, Control of Heavy loads Near Spent Fuel		12/22/80
Response to NRC Request on	8107280271	07/20/81
Control of Heavy Loads		
Control of Heavy Loads	8204220279	04/16/82
Control of Heavy Loads	8208060279	07/30/82
Control of Heavy Loads Draft TER	8304230276	05/10/83
Control of Heavy Loads	8308110023	07/21/83
Control of Heavy Loads		07/15/86

PLANT Haddam Neck DOCKET NO(S). 50-213

PROJECT MANAGER Alan B. Wang TECHNICAL CONTACT A. Serkiz

USI NO. A-43 TITLE Containment Emergency Sump Performance

MPA NO. ______ TAC NOS. ______

ISSUES SUMMARY:

19. USI NO. A-43 TITLE: Containment Emergency Sump Performance

The resolution of this USI was presented to the Commission in October 1985 in SECY-85-349. NUREG-0897, Revision 1, "Containment Emergency Sump Performance," presents the results of the staff's technical findings. These findings established a need to revise current licensing guidance on these matters. RG 1.82 Revision 0 and Standard Review Plan Section 6.2.2, "Containment Heat Removal Systems" were revised to reflect this new guidance. No licensee actions were required.

Initially, an issue existed concerning the availability of adequate recirculation cooling water following a loss-of-coolant accident (LOCA) when long-term recirculation of cooling water from the PWR containment sump, or the BWR residual heat removal system (RHR) suction intake, must be initiated and maintained to prevent core melt.

The technical concerns evaluated under USI A-43 were: (a) post-LOCA adverse conditions resulting from potential vortex formation and air ingestion and subsequent pump failure, (b) blockage of sump screens with LOCA generated insulation debris causing inadequate net positive suction head (NPSH) on pumps, and (c) RHR and containment spray pumps inoperability due to possible air, debris, or particulate ingestion on pump seal and bearing systems.

This revised guidance applies only to future construction permits, preliminary design approvals, final design approvals, standardized designs, and applications for licenses to manufacture. The staff performed a regulatory analysis to determine if this new guidance should be applied to operating plants. The results of this analysis were reported in NUREG-0869 Revision 1, "USI A-43 Regulatory Analysis," issued in October 1985. The staff concluded that the regulatory analysis does not support any new generic requirements for present licensees to perform debris assessments.

IMPLEMENTATION AND STATUS SUMMARY (PLANT SPECIFIC):

By letter dated October 13, 1982, CYAPCO provided its position on each USI applicable to the Haddam Neck Plant. CYAPCO stated that the Haddam Neck Plant emergency sump is of standard design including anti-vortexing features such as a belled mouth suction pipe and a grating located one foot above the recirculation suction. As such CYAPCO has concluded the emergency pump can be operated without endangering public health and safety. CYAPCO considers this issue resolved pending and future requirements that may result from the Staff's technical resolution of this issue.

REFERENCES:

Haddam Neck
A-43

1. REQUIREMENT DOCUMENTS

TITLE	NUDOCS NO.	DATE
NUREG-0869, Rev. 1, "USI A-43 Regulatory Analysis"		10/85
NUREG-0897, Rev. 1, "Containment Emergency Sump Performance"		10/85
GL 85-22, "Potential for Loss of Post-LOCA Recirculation Capability Due to Insulation Debris Blockage"		12/03/85

2. IMPLEMENTATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE
Status of Unresolved Safety Issues	8210270200	10/13/82
NU ISAP Final Report	8702040321	12/86
NUREG 1185, ISAR	8709090221	07/87

3. VERIFICATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE

PLANT Haddam Neck DOCKET NO(S). 50-213

PROJECT MANAGER Al; . B. Wang TECHNICAL CONTACT P. Gill

USI NO. A-44 TITLE Station Blackout

MPA NO. TAC NOS.

ISSUES SUMMARY:

This USI wa, resolved in June 1988 with the publication of a new rule (10 CFR 50.63) and Regulatory Guide 1.155.

Station blackout means the loss of offsite ac power to the essential and nonessential electrical buses concurrent with turbine trip and the unavailability of the redundant onsite emergency ac power systems. WASH-1400 showed that station blackout could be an important risk contributor, and operating experience has indicated that the reliability of ac power systems might be less than originally anticipated. For these reasons station blackout was designated as a USI in 1980. A proposed rule was published for comment on March 21, 1986. A final rule, 10 CFR 50.63, was published on June 21, 1988 and became effective on July 21, 1988. Regulatory Guide 1.155 was issued at the same time as the rule and references an industry guidance document, NUMARC-8700. In order to comply with the A-44 resolution, licensees will be required to:

° maintain onsite emergency ac power supply reliability above a minimum level

° develop procedures and training for recovery from a station blackout

° determine the duration of a station blackout that the plant should be able to withstand

° use an alternate qualified ac power source, if available, to cope with a station blackout

° evaluate the plant's actual capability to withstand and recover from a station blackout

° backfit hardware modifications if necessary to improve coping ability

Section 50.63(c)(1) of the rule required each licensee to submit a response including the results of a coping analysis within 270 days from issuance of an operating license or the effective date of the rule, whichever is later.

IMPLEMENTATION AND STATUS SUMMARY (PLANT SPECIFIC):

CYAPCO responded on 4/17/89. Response is under review. CYAPCO has opened a new ISAP Topic to incorporate specific plant modifications that result from NRC rulemaking and industry initiatives.

REFERENCES:

Haddam Neck
A-44

1. REQUIREMENT DOCUMENTS:

TITLE	NUDOCS NO.	DATE
10 CFR 50.63, "Loss of All Alternating Current Power"		06/21/88
Regulatory Guide 1.155, "Station Blackout"		08/88

2. IMPLEMENTATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE
ISAP Letter	8903080205	03/02/89
Response to 10 CFR 50.63 Loss of all alternating current power		04/17/89

3. VERIFICATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE

PLANT Haddam Neck DOCKET NO(S). 50-213

PROJECT MANAGER Alan B. Wang TECHNICAL CONTACT P. Y. Chen

USI NO. A-46 TITLE Seismic Qualification of Equipment in Operating Plants

MPA NO. B-105 TAC NOS.

REQUIREMENTS SUMMARY:

As an outgrowth of the Systematic Evaluation Program (SEP), the need was identified for reassessment of design criteria and methods for the seismic qualification of mechanical equipment and electrical equipment. The seismic qualification of the equipment in operating plants must, therefore, be reassessed to ensure the ability to bring the plant to a safe shutdown condition when subject to a seismic event. The objective of this issue was to establish an explicit set of guidelines that could be used to judge the adequacy of the seismic qualification of mechanical and electrical equipment at operating plants in lieu of attempting to backfit current design criteria for new plants.

The resolution of USI A-46 was mainly based on work completed by the Seismic Qualification Utility Group (SQUG) and EPRI using the seismic and test experience data approach and reviewed and endorsed by the Senior Seismic was narrowed down to equipment required to bring each affected plant to hot shutdown and maintain it there for a minimum of 72 hours. A walk-through of each plant is required to inspect equipment in the scope. Evaluation of equipment will include: (a) adequacy of equipment anchorage; (b) functional capability of essential relays; (c) outliers and deficiencies (i.e., equipment with non-standard configurations); and (d) seismic systems interaction.

The staff issued Generic Letter 87-02 on February 19, 1987, with associated guidance, requiring all affected utilities to perform an evaluation of the seismic adequacy of their plants. The specific requirements and approach for basis prior to individual member utilities proceeding with plant-specific implementation (see discussion below).

IMPLEMENTATION AND STATUS SUMMARY (PLANT SPECIFICA):

For All Plants:

The Generic Implementation Procedure (GIP), Revision 0, was submitted by SQUG on June 3, 1988. The staff issued a Generic Safety Evaluation (SE) on July 29, 1988 endorsing much of the GIP but with about 70 open items to be resolved. After a series of meetings, SQUG submitted by SQUG on March 17, 1989. The staff has prepared a supplemental SE for GIP, Revision I and has submitted it to the CRGR for review. The target date for issuance of the supplemental SE is November 1989. An additional supplement is schedule for June 1990 and overall closeout of implementation projected for 1993.

REFERENCES:

Haddam Neck
A-46

1. REQUIREMENT DOCUMENTS:

TITLE	NUDOCS NO.	DATE
Generic Letter 87-02, "Verification of Seismic Adequacy of Mechanical and Electric Equipment in Operating Reactors"		02/19/87
NUREG-1211, "Regulatory Analysis for Resolution of Unresolved Safety Issues A-46..."		02/87
NUREG-1030, "Seismic Qualification of Equipment in Operating Plants, Unresolved Safety Issue A-46"		02/87
Letter attached with "Generic Safety Evaluation Report on SQUG GIP, Revision 0," from L. Shao (NRC) to Neil Smith (SQUG)		07/29/88

2. IMPLEMENTATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE
"Generic Implementation Procedure (GIP for Seismic Verification of Nuclear Plant Equipment," Revision 0		06/88
"Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment," Revision 1		12/88
NU ISAP Letter	8903080205	03/02/89

3. VERIFICATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE

PLANT Haddam Neck DOCKET NO(S). 50-213

PROJECT MANAGER Alan B. Wang TECHNICAL CONTACT J. Mauck

USI NO. A-47 TITLE Safety Implication of Control Systems in LWR Nuclear Power Plants

MPA NO. TAC NOS.

ISSUES SUMMARY:

USI A-47 was resolved September 20, 1?89, with the publication of Generic Letter (GL) 88-19.

The generic letter states:

> "The staff has concluded that all PWR plants should provide automatic steam generator overfill protection, all BWR plants should provide automatic reactor vessel overfill protection, and that plant procedures and technical specifications for all plants should include provisions to verify periodically the operability of the overfill protection and to assure that automatic overfill protection is available to mitigate main feedwater overfeed events during reactor power operation. Also, the system design and setpoints should be selected with the objective of minimizing inadvertent trips of the main feedwater system during plant startup, norm ` operation, and protection system surveillance. The Technical Specifications recommendations are consistent with the criteria and the risk considerations of the Commission Interim Policy Statement on Technical Specification Improvement. In addition, the staff recommends that all BWR recipients reassess and modify, if needed, their operating procedures and operator training to assure that the operators can mitigate reactor vessel overfill events that may occur via the condensate booster pumps during reduced system pressure operation."

Also, page 2 of the generic letter provides for additional act.ons for CE and B&W plants. The generic letter provides amplifying guidance for licensees.

The generic letter requires that licensees provide NRC with their schedule and commitments within 180 days of the letter's date. The implementation schedule for actions on which commitments are made should be prior to startup after the first refueling outage, but no later than the second refueling outage, beginning 9 months after receipt of the letter.

IMPLEMENTATION AND STATUS SUMMARY (PLANT SPECIFIC):

In a letter dated March 31, 1986, CYAPCO states that in the development of the CYPSS, CYAPCO addressed potential control system failures or malfunctions by detailed fault tree development, consideration of event initiators and consideration of control system power sources both in terms of a support state system model and in terms of special initiators. The CYPSS provides a system-

IMPLEMENTATION AND STATUS SUMMARY (PLANT SPECIFIC):

matic process to review plant control system failures and their impact on event mitigation. Further evaluation of the safety implications of control systems will be made as a result of further review of the PSS. In a letter dated October 18, 1988, the Staff issued a safety evaluation regarding the Haddam Neck non-LOCA Transient Analysis. The Staff concluded that all of the analysis were acceptable except for the "excess feedwater event." CYAPCO will address the mechanical failure of the main feedwater regulating valves in ISAP. CYAPCO is preparing response to GL 89-19.

REFERENCES:

Haddam Neck
A-47

1. REQUIREMENT DOCUMENTS

TITLE	NUDOCS NO.	DATE
Generic Letter 89-19 "Request for Action Related to Resolution of USI A-47"		09/20/89
NUREG-1217 "Evaluation of Safety Implications of Control Systems in LWR Nuclear Power Plants"		June 1989
NUREG-1218 "Regulatory Analysis for Resolution of USI A-47"		July 1989

2. IMPLEMENTATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE
Non-LOCA Transients SER	8810240243	10/18/88
NU ISAP Letter	8903080205	03/02/89

3. VERIFICATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE

PLANT Haddam Neck DOCKET NO(S). 50-213

PROJECT MANAGER ________ TECHNICAL CONTACT B. Elliott

USI NO. A-49 TITLE Pressurized Thermal Shock

MPA NO. ________ TAC NOS. ________

ISSUES SUMMARY:

The final rule (10 CFR 50.61) on pressurized thermal shock (PTS) was approved by the Commission in July 1985. Regulatory Guide 1.154 , "Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for PWRs," was later published in February 1987. Thus, this issue was resolved and new requirements were established, applicable to PWRs only. The rule required that each operating reactor meet the screening criteria provided in the rule or provide supplemental analysis to demonstrate that PTS is not a concern for the facility.

Neutron irradiation of reactor pressure vessel weld and plate materials decreases the fracture toughness of the materials. The fracture toughness sensitivity to radiation-induced change is increased by the presence of certain materials such as copper. Decreased fracture toughness makes it more likely that, if a severe overcooling event occurs followed by or concurrent with high vessel pressure, and if a small crack is present on the vessel's inner surface, that crack could grow to a size that might threaten vessel integrity.

Severe pressurized overcooling events are improbable since they require multiple failures and improper operator performance. However, certain precursor events have happened that could have potentially threatened vessel integrity if additional failures had occurred and/or if the vessel had been more highly irradiated. Therefore, the possibility of vessel failure due to a severe pressurized overcooling event cannot be ruled out.

IMPLEMENTATION AND STATUS SUMMARY (PLANT SPECIFIC):

By letter, dated January 23, 1986, CYAPCO provided calculated RT_{PTS} valves for the two cases of from the time of this submittal until (1) the current operating license (OL) expiration date and (2) the planned extension to the OL expiration date. Information was provided regarding core loading patterns, the source of the best estimate weight percent of copper and nickel in the vessel material, and the relationship of the material on which any measurements were made to the actual vessel material. CYAPCO concluded that the values were made to the actual vessel material. CYAPCO concluded that the values calculated for the RP_{PTS} for the applicable locations at the Haddam Neck Plant will not exceed the screening criteria of 270°F for plant forgings and axial weld materials and 300°F for circumferential weld materials. During phone conversations in December 1986 and February 1987, the staff requested additional information on the methodology used to determine fluence values. By letters April 1, 1987 and, August 21, 1987 CYAPCO provided this additional information.

IMPLEMENTATION AND STATUS SUMMARY (PLANT SPECIFIC):

The staff issued an SER on October 2, 1987 concluding that the Haddam Neck Plant meets the toughness requirements of 10 CFR 50.61 for 32 effective full power years. The staff has requested that CYAPCO submit periodic reevaluation of the RT_{PTS}.

REFERENCES:

Haddam Neck
A-49

1. REQUIREMENT DOCUMENTS:

TITLE	NUDOCS NO.	DATE
10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Requirements"		7/85
Reg. Guide 1.154, "Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for PWRs"		1/89
SECY 82-465, "Pressurized Thermal Shock"		11/23/82
SECY 83-288, "Proposed Pressurized Thermal Shock Rule"		07/15/83
Regulatory Guide 1.154 "Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors"		02/87
Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations"		7/12/88

2. IMPLEMENTATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE
10 CFR 50.61 Compliance	8602130242	01/23/86
Response to RAI	8704070362	04/01/87
Response to RAI	8708280203	08/21/87
PTS SER	8710080254	10/02/87

3. VERIFICATION DOCUMENTS:

TITLE	NUDOCS NO.	DATE

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A-3, 4, 5	7.0/8.0
A-6	9.0/10.0
A-7	11.0/12.0
A-8	13.0/14.0
A-9	15.0/16.0
A-10	17.0/18.0
A-11	19.0/20.0
A-12	21.0/22.0
A-17	23.0/24.0
A-24	25.0/26.0
A-26	27.0/28.0
A-31	29.0/30.0
A-36	31.0/32.0
A-39	33.0/34.0
A-40	35.0/36.0
A-42	37.0/38.0
A-43	39.0/40.0
A-44	41.0/42.0
A-45	43.0/44.0
A-46	45.0/46.0
A-47	47.0/48.0
A-48	49.0/50.0
A-49	51.0/52.0

ENCLOSURE 3

LISTING OF INCOMPLETE USI DATA
FOR INPUT FROM PROJECT MANAGERS

ISSUE NUMBER	ISSUE DESCRIPTIVE NAME	IMPLEMENT DATE	IMPLEMENT STATUS	LICENSEE COMMENT	STAFF COMMENT
** PLANT NAME: HADDAM NECK					
A-01	WATER HAMMER	12/31/80	C	ISAP MODS	
A-02	ASYMMETRIC BLOWDOWN LOADS ON REACTOR PRIMARY COOLANT SYSTEMS	06/16/89	C	GL-80-04 RESP	GL REQ SPECIFIC HN RESP
A-03	WESTINGHOUSE STEAM GENERATOR TUBE INTEGRITY	12/31/86	C	SG RELATED MODS	ISAP
A-04	CE STEAM GENERATOR TUBE INTEGRITY	/ /	N/A		CE PLANTS ONLY
A-05	B&W STEAM GENERATOR TUBE INTEGRITY	/ /	N/A		B&W PLANTS ONLY
A-06	MARK I SHORT-TERM PROGRAM	/ /	N/A		MK I BWR ONLY
A-07	MARK I LONG-TERM PROGRAM	/ /	N/A		MK I BWR ONLY
A-08	MARK II CONTAINMENT POOL DYNAMIC LOADS - LONG-TERM PROGRAM	/ /	N/A		MK II BWR ONLY
A-09	ATWS	08/19/86	C	EXEMPT REQ 8/19/86	EXEMPTION REQUESTED
A-10	BWR FEEDWATER NOZZLE CRACKING	/ /	N/A		BWR ONLY
A-11	REACTOR VESSEL MATERIALS TOUGHNESS	/ /	NC	> 50 FT-LB	
A-12	FRACTURE TOUGHNESS OF STEAM GENERATOR AND REACTOR COOLANT PUMP SUPPORTS	/ /	N/A		CP AFTER 83 ONLY
A-17	SYSTEMS INTERACTION	/ /	NC		NO REQUIREMENTS
A-24	QUALIFICATION OF CLASS 1E SAFETY-RELATED EQUIPMENT	12/31/86	C		EXTENSION GRANTED
A-26	REACTOR VESSEL PRESSURE TRANSIENT PROTECTION	04/21/88	C	PROCEDURES	LTOPS
A-31	RHR SHUTDOWN REQUIREMENTS	/ /	I	INTERLOCKS	LOW PRIORITY IN SEP/ISAP
A-36	CONTROL OF HEAVY LOADS NEAR SPENT FUEL	07/15/86	C		GL-85-11 ENDED
A-39	DETERMINATION OF SAFETY RELIEF VALVE POOL DYNAMIC LOADS AND TEMPERATURE LIMITS	/ /	N/A		BWR ONLY
A-40	SEISMIC DESIGN CRITERIA - SHORT-TERM PROGRAM	/ /	NC		SUBSUMMED BY A-46
A-42	PIPE CRACKS IN BOILING WATER REACTORS	/ /	N/A		BWR ONLY
A-43	CONTAINMENT EMERGENCY SUMP PERFORMANCE	/ /	NC	SEP/ISAP	INFO ONLY
A-44	STATION BLACKOUT	03/31/93	I		SER 3/31/91
A-45	SHUTDOWN DECAY HEAT REMOVAL REQUIREMENTS	/ /	NC	IPE	SUBSUMED BY SEVERE ACC
A-46	SEISMIC QUALIFICATION OF EQUIPMENT IN OPERATING PLANTS	/ /	I		REQ UNDER DEVEL
A-47	SAFETY IMPLICATIONS OF CONTROL SYSTEMS	03/19/90	E		NEW REQUIREMENTS
A-48	HYDROGEN CONTROL MEASURES AND EFFECTS OF HYDROGEN BURNS ON SAFETY EQUIPMENT	/ /	NC		SEP/ISAP
A-49	PRESSURIZED THERMAL SHOCK	10/02/87	C		