NORTHEAST UTILITIES

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

General Offices • Selden Street, Berlin, Connecticut

P.O. BOX 270
HARTFORD, CONNECTICUT 06141-0270
(203) 665-5000

November 30, 1990

Docket No. 50-245
B13651
Re: Integrated Safety Assessment Program

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

Gentlemen:

Millstone Nuclear Power Station, Unit No. 1
Integrated Safety Assessment Program

In a letter dated April 30, 1990,(1) Northeast Nuclear Energy Company (NNECO) submitted to the NRC Staff an updated report on the Millstone Unit No. 1 Integrated Safety Assessment Program (ISAP).

Since the April 30, 1990 ISAP report was submitted to the NRC Staff, NNECO has performed in-depth reviews and updates of previous active ISAP topics. In those cases where there was a substantial change in scope, a full ISAP reevaluation was performed. Several new topics were added and evaluated if sufficient project scope was available. As a result, revised Analytical Ranking Model (ARM) scores were defined and a new Integrated Implementation Schedule (IIS) was developed.

NNECO continues to take an aggressive approach to maximize the closure of open IIS commitments. The April 30, 1990 report contained information for the closure of 6 topics. This submittal represents closure documentation for an additional 6 topics. There are now 43 active ISAP topics, for which reviews and/or modifications are pending.

The revised IIS, provided in this report, is fully consistent with the Program Plan submitted to the NRC Staff by letter dated March 24, 1988.(2) NNECO is hereby providing the attached "six-month" ISAP update in accordance with the proposed license condition. This document is intentionally being submitted within seven months of our April 30, 1990 submittal for internal resource management compatibility with the September 28, 1990 Haddam Neck Plant ISAP submittal.

---

(1) E. J. Mroczka letter to U.S. Nuclear Regulatory Commission, "Integrated Safety Assessment Program (ISAP)," dated April 30, 1990.

(2) E. J. Mroczka letter to U.S. Nuclear Regulatory Commission, "Integrated Implementation Schedule--License Amendment," dated March 24, 1988.

A001
1/1

Attachment 1 to this letter provides a list of all the ISAP topics including both open topics and those topics considered closed by NNECO. Attachment 2 provides an update on those open, active ISAP topic reviews discussed previously. Attachment 3 describes new topics being added to the Millstone Unit No. 1 ISAP, or existing ones which have recently been reevaluated in ISAP. Attachment 4 provides the updated IIS, including both old and new topics. Attachment 5 provides a summary table of the ISAP ARM scores and installation man-rem for each project recently reevaluated.

NNECO respectfully requests that the NRC Staff review and respond back to NNECO within 60 days of receipt of this letter as to whether or not you concur with our positions outlined herein. We will remain available to discuss these issues with you at your convenience.

Please contact us if you have any questions.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

FOR: E. J. Mroczka
Senior Vice President

BY: CFSears
C. F. Sears
Vice President

Attachment 1--List of ISAP Topics
Attachment 2--Updates on Existing ISAP Topics
Attachment 3--Evaluation of New ISAP Topics or Reevaluation of Existing Topics
Attachment 4--Integrated Implementation Schedule
Attachment 5--Summary Table of ISAP ARM Scores

cc: T. T. Martin, Region I Administrator
M. L. Boyle, NRC Project Manager, Millstone Unit No. 1
W. J. Raymond, Senior Resident Inspector, Millstone Unit Nos. 1, 2, and 3

Docket No. 50-245
B13651

Attachment 1

Millstone Nuclear Power Station, Unit No. 1
Integrated Safety Assessment Program

List of ISAP Topics

November 1990

Attachment 1
B13651\Page 1
November 30, 1990

Millstone Nuclear Power Station, Unit No. 1
Integrated Safety Assessment Program
List of ISAP Topics

Topic Number	Closed[1]	Title
1.01	9-29-89	Gas Turbine Generator Start Logic Modifications
1.02		Tornado Missile Protection
1.03	11-9-88	Containment Isolation--Appendix A Modifications
1.04	11-9-88	RWCU Pressure Interlock
1.05	11-9-88	Ventilation System Modifications
1.06		Seismic Qualification of Safety-Related Piping
1.07		Control Room Design Review
1.08	11-9-88	Safety Parameter Display System
1.09		Regulatory Guide 1.97 Instrumentation
1.10	9-29-89	Emergency Response Facilities Instrumentation
1.11	11-9-88	Post Accident Hydrogen Monitor
1.12	11-9-88	Control Room Habitability
1.13	11-9-88	BWR Vessel Water Level Instrumentation
1.14		Appendix J Modifications
1.15	11-9-88	FSAR Update
1.16	11-9-88	10CFR50, Appendix R
1.17	11-9-88	Replacement of Motor-Operated Valves
1.18	11-9-88	ATWS
1.19		Integrated Structural Analysis
1.20	11-9-88	MOV Interlocks
1.21	11-9-88	Fault Transfers
1.22		Electrical Isolation
1.23/	11-30-90	Grid Separation Procedures/Degraded
1.25	11-30-90	Grid Voltage Procedures
1.24	11-9-88	Emergency Power
1.26	11-9-88	Equipment Classification/Vendor Interface (GL 83-28, Item 2.1)
1.27	11-9-88	Post-Maintenance Testing (GL 83-28, Items 3.1.1 and 3.1.2.)
1.28	11-9-88	Post-Maintenance Testing Technical Specification Changes (GL 83-28, Item 3.1.3)
1.29	11-9-88	Response to GL 81-34
1.30	11-9-88	Post-Trip Review Data and Information (GL 83-28, Item 1.2)
1.31		Equipment Classification/Vendor Interface (GL 83-28, Item 2.2)
1.32	11-9-88	Post-Maintenance Test Procedures (GL 83-28, Items 3.2.1 and 3.2.2)

Topic Number	Closed	Title
1.33	11-9-88	Post-Maintenance Testing Technical Specification Changes (GL 83-28, Item 3.?.3)
1.34	9-29-89	Reactor Trip System Testing (GL 83-28, Items 4.5.2 and 4.5.3)
1.35	11-9-88	Reactor Trip System Functional Testing (GL 83-28, Item 4.5.1)
1.36		Technical Specifications Covered by GL 83-36
1.37	11-9-88	Technical Specification Changes to address 10CFR50.72 and 10CFR50.73
1.38	11-9-88	Expand Quality Assurance List
1.39	11-9-88	Radiation Protection Plans
1.40	9-29-89	Bolting Degradation or Failure
1.41	11-9-88	Flooding of Compartments by Backflow
1.42	11-9-88	Main Steam Line Leakage Control System
1.43		Water Hammer
1.44	11-9-88	Asymmetric Blowdown Loads on Reactor Systems
1.45	11-9-88	Systems Interactions
1.46	11-9-88	Determination of SRV Pool Dynamic Loads
1.47	9-29-89	Containment Emergency Sump Performance
1.48	9-29-89	Safety Factor for Penetration X-10A
1.49	11-9-88	Reactor Vessel Surveillance Program
1.50	9-29-89	Isolation Condenser Start-Up/Makeup Failures
1.51	9-29-89	Failure to Restore Main Condenser
1.52		SRV Failure - Setpoint Drift
1.10C		Fire Barrier Penetration Seal Program Upgrade
1.101		Fire Detection System Code Compliance
1.102		Fire Suppression System Code Compliance
1.103	9-29-89	Standby Gas Treatment System Redundancy
1.104		Pump Flow Rate Instrumentation
1.105	9-29-89	Valves LP 15 A/B, 16 A/B, & CU 2/3
1.106		Station Blackout
1.107		Drywell Spray Flow Indication
1.108	11-30-90	Torus Vacuum Breakers
1.109	9-29-89	IGSCC Countermeasures
1.110		Vital Area Ventilation
1.111		Motor Operated Valve Testing, GL 89-10
1.112		Service Water System Evaluation, GL 89-13
1.113		Hardened Wetwell Vent, GL 89-16
1.114		Individual Plant Examinations, GL 88-20
1.115	11-30-90	Reactor Water Level Reference Leg Break, GL 89-11
1.116		Remote Wide-Range Yarways
1.117		Safety Parameter Display System Upgrade
1.118		Plant Heating Steam System
1.119		Resolution of Unresolved Safety Issue A-46
2.01	11-9-88	LPCI Remotely Operated Valves LP-50A and B
2.02	11-9-88	Drywell Humidity Instrumentation

Topic Number	Closed	Title
2.03	11-9-88	Process Computer Replacement
2.04	11-9-88	High Steam Flow Setpoint Increase
2.05	9-29-89	Hydrogen Water Chemistry Study
2.06		Condenser Retube
2.07	11-9-88	Sodium Hypochlorite System
2.08		Extraction Steam Piping
2.09	9-29-89	Upgrading of P&IDs
2.10		Drywell Ventilation Systems
2.11	9-29-89	Stud Tensioners
2.12	9-29-89	Reactor Vessel Head Stand Relocation
2.13	11-9-88	Turbine Water Induction Modifications
2.14	11-9-88	Evaluation and Implementation of NUREG-0577
2.15	4-30-90	Torque Switch Evaluations for MOVs
2.16	11-9-88	Reactor Protection Trip System
2.17	4-30-90	4.16-kV, 480-V, and 125 VDC Plant Distribution Protection Study
2.18	4-30-90	Spent Fuel Pool Storage Racks/Transportation Cask
2.19	11-9-88	DC System Review
2.20	11-9-88	RWCU System Isolation Setpoint Reduction
2.21	11-9-88	480-V Load Center Replacement of Oil-Filled Breakers
2.22	11-9-88	Control Rod Drive System Water Hammer Analysis
2.23	9-29-89	Instrument, Service, and Breathing Air Improvements
2.24	11-9-88	Off-Site Power Systems
2.25	11-9-88	Drywell Temperature Monitoring System
2.26	11-9-88	Reliability Equipment
2.27	11-9-88	Spare Recirculation Pump Motor
2.28	4-30-90	Long-Term Cooling Study
2.29	11-9-88	FWCI Assessment Study
2.30		MSIV Closure Test Frequency
2.31	11-9-88	LPCI Lube Oil Cooler Test Frequency
2.32	11-9-88	Primary Containment Pumpback System
2.33	9-29-89	RBCCW Leak Rate Testing
2.100	11-9-88	Emergency Gas Turbine Generator Reliability Study
2.101	9-29-89	Shutdown Cooling Discharge Valve Replacement
2.102	9-29-89	Main Transformer Replacement
2.103		Loss of 125-VDC Power Study
2.104	11-30-90	Feedwater Nozzle Leakage Monitoring System
2.105	4-30-90	480-Volt Motor Soft Start
2.106		Diesel Air Start System Upgrade
2.107		Class 1E Generator Antimotoring Protection
2.108		Generator Antimotoring Protection While Shutdown
2.109	11-9-88	ECCS Keepfill System
2.110	11-9-88	Head Spray Line Removal
2.111	11-9-88	(Project Dropped)
2.112	9-29-89	Gas Turbine Governor Control System Replacement

Topic Number	Closed	Title
2.113	9-29-89	Recirculation Pump Vibration Monitoring
2.114	11-30-90	RBCCW System Class Boundaries
2.115		Replacement of LPC-4A & 4B Valves
2.116		Replacement of Reactor Water Cleanup Valves
2.117		Environmental Qualification of IC Local Control Stations
2.118		Replacement of the Normal Station Service Transformer
2.119	4-30-90	(Topic Deleted)
2.120		Atmospheric Control System Containment Isolation Valves
2.121		Feedwater Venturi Replacement
2.122		RSST Transfer Trip Scheme Replacement
2.123		Chemistry Laboratory HVAC System
2.124		Replacement of IC System Valves and CU-28

---

(1) Date refers to the periodic ISAP/IIS submittal providing proposed justification for closure of topic.

Docket No. 50-245
B13651

Attachment 2

Millstone Nuclear Power Station, Unit No. 1
Integrated Safety Assessment Program

Updates on Existing ISAP Topics

November 1990

Millstone Nuclear Power Station, Unit No. 1
Integrated Safety Assessment Program
Updates on Existing ISAP Topics

Topic 1.02--Tornado Missile Protection

This topic was addressed in detail in our previous ISAP submittals. As reflected therein, the proposed project under this topic has a positive public safety benefit. The project remains on schedule, with completion anticipated by the end of 1990.

Topic 1.06--Seismic Qualification of Safety-Related Piping

As discussed in our April 30, 1990 submittal, this topic addresses the completion of the remaining 186 pipe support modifications. Current plans are to complete 41 modifications during the 1991 refueling outage. These are specifically related to operational transient concerns on the main steam and turbine bypass steam piping.

An additional 46 modifications are planned for the 1993 refueling outage. The 1993 support/restraint modifications will concentrate on containment piping penetrations to assure containment integrity during a seismic event.

The remaining 99 modifications, if warranted, are to be accomplished during subsequent refueling outages. These are being reviewed to determine whether alternate design bases can be used or some modifications can be eliminated. The modifications mainly concern equipment and other supports/restraints not associated with containment integrity. Additional information on these modifications will be forwarded in a future ISAP/IIS update report.

Topic 1.07--Control Room Design Review

As discussed in the April 30, 1990 status report, this topic addresses the resolution of Human Engineering Discrepancies (HEDs) as identified by the Control Room Design Review (CRDR). In the previously submitted CRDR Implementation Plan[1] and the CRDR Summary Report,[2] NNECO identified a substantial number of HEDs. The ARM evaluation and larger ISAP prioritization process has been utilized, including input from the Staff, to schedule implementation of HED resolution.

The ISAP review has been completed for Millstone Unit No. 1 and all HED work packages have been scheduled for resolution. As detailed in our April 30, 1990 status report and further discussed with the Staff at the June 13, 1990 meeting, all HEDs will be resolved over the next three refueling outages. The Staff agreed to receive documentation of any future changes to the schedule, including justification for the deletion of any HEDs, as part of the periodic ISAP/IIS updates.

By letter dated July 10, 1990,[3] the NRC forwarded its review of the CRDR. The Staff concluded, as stated in the safety evaluation, that the CRDR for Millstone Unit No. 1 meets the criteria in Section 18.1, of NUREG-0800, Rev. 0, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," and meets the CRDR provisions of Supplement 1 to NUREG 0737, "Clarification of TMI Action Plan Requirements."

The following topics remain scheduled for correction during the 1991 refueling outage:

Topic 1.07.3	Inadequate/Incorrect Instrument Scales
Topic 1.07.7	Control Panel Mimics for EOP Related Systems
Topic 1.07.8	Control Panel Mimics for non-EOP Related Systems
Topic 1.07.14	Secondary Containment delta-P Indication
Topic 1.07.16	Control Panel Relabelling, Control Switch Handle Replacement for Pumps and Valves, and CRP-905 Rod Display
Topic 1.07.18	Addition of APR 120 Second Timer Bypass Switch

Topics 1.07.3, 1.07.7, 1.07.8 and 1.07.16 pertain primarily to main control board (MCB) layout, mimics, and scales, the correction of which will impact approximately 400 discrete devices.

Topic 1.07.14 presently scheduled for correction during the 1991 outage, may be rescheduled as a result of further review of design and implementation benefits. That topic involves installation of parameter indication for use by the operator as an entry condition to the emergency operating procedures (EOPs). This potential modification is being reviewed in detail to determine whether a suitable design can be developed to measure the delta-P to the required accuracy (¼" of water vacuum) without spurious alarming. The modification will be implemented as scheduled if a suitable design is determined.

Topic 1.07.18 addresses installation of the 120-second APR Timer Bypass Switch. The condition under which this APR timer override would be used is currently under evaluation and will be resolved prior to implementation. This may involve a deviation from the BWR Owners' Group Emergency Procedure Guidelines.

As discussed in our April 30, 1990 status report, the following topics remain scheduled for completion during the 1993 refueling outage:

Topic 1.07.3	Inadequate/Incorrect Instrument Scales (also addressed in 1991 outage)
Topic 1.07.5	Addition of Process Computer Points
Topic 1.07.6	Addition of New Equipment
Topic 1.07.7	Control Panel Mimics for EOP Related Systems (also addressed in 1991 outage)
Topic 1.07.8	Control Panel Mimics for non-EOP Related Systems (also addressed in 1991 outage)
Topic 1.07.12	Feedwater Controls and Reactor Water Level Instrumentation
Topic 1.07.15	Multi-Point Recorder Replacement

The following topics remain scheduled for the 1995 refueling outage:

Topic 1.07.1	Relocation of Containment Parameter Indication
Topic 1.07.2	Control Room Environment
Topic 1.07.4	Annunciator Review
Topic 1.07.5	Addition of Process Computer Points (also addressed in 1993 outage)
Topic 1.07.6	Addition of New Equipment (also addressed in 1993 outage)
Topic 1.07.9	Addition of Motor Operated Valves LP-50A & B, MW-92A, B, C, D, and MW-91A & B

Topic 1.07.11 Replacement of the Reactor Mode Switch
Topic 1.07.13 Removal of Unused Components
Topic 1.07.17 Modification of Existing Plant Controls

Topic 1.07.10, as discussed in the April 30, 1990 submittal, examined the installation of bypass switches to eliminate the need to install individual bypass jumpers during severely degraded emergency conditions. The following items were not identified during the original CRDR since the EOP 590 procedures were not developed. The proposed installation of bypass switches listed below would enhance the response time of the operators during degraded emergency conditions, by precluding the need to install selected bypass jumpers.

- RWCU Bypass Switch to defeat Group 5 and SLC isolation signals (EOP 590.3)
- Group 1 Isolation Bypass Switch for Reactor Low Low Level (EOP 590.14)
- Group 4 Isolation Signal Bypass Switch (EOP 590.15)
- RWCU Isolation Signal Bypass Switch (EOP 590.16)
- Reactor Building and Control Room HVAC Isolation Signal Bypass Switch (EOP 590.17)
- Group 1 Bypass except Steam Line Hi Rad (EOP 590.19)
- Group 1 Isolation Bypass Switch (EOP 590.20)
- ATWS Logic Trip Bypass Switch (EOP 590.22)
- Drywell Cooler Fan Interlock Switch (EOP 590.25)
- Reactor Protection System Logic Bypass Switch

The functional capability of this topic was achieved by fabrication of jumpers and cabinet labelling rather than the design modification of switch installation. Therefore, NNECO has accomplished the intent of this topic and accordingly considers Topic 1.07.10 closed.

The majority of the HEDs scheduled for correction during the 1991 outage will be completed on schedule. However, to ensure the most effective means of implementing the corrections, consideration must be given to both the stated ISAP schedule and to providing integrated modification packages with the maximum human factors benefit. Therefore, some of the 1991 corrections may be deferred and some of the 1993/1995 corrections moved forward, thus providing an integrated change during each outage, ensuring maximum operator benefit.

On June 13, 1990, members of NNECO met with the NRC Staff to discuss CRDR issues. In response to that meeting, NNECO submitted a June 29, 1990 letter[(4)] which supplied information in addition to the April 30, 1990 submittal. Specifically, Attachment 4 in the April 30, 1990 submittal was totally superseded by Attachment 2 in the June 29, 1990 letter. Also, during the June 13, 1990 meeting, a group of 47 non-ISAP HFDs were presented and discussed. The status of this group remains the same (i.e. 14 are complete, 27 are in progress, and the remaining 6 will be accomplished no later than the 1995 outage.)

Topic 1.14--Appendix J Modifications

As discussed in our previous ISAP submittals, NNECO determined that modifications to permit Type-C testing of Drywell Sump Drain Valves SS-3, SS-4, SS-13, and SS-14, contained in Penetrations X-18 and X-19, were warranted. These modifications place very high in the ISAP ranking, and accordingly, continue to be scheduled for implementation during the 1991 refueling outage.

With respect to all remaining penetrations encompassed by this topic, NNECO continues to believe that exemptions are warranted. In a letter dated November 8, 1990,[5] NNECO provided the Staff with answers to several questions relating to our April 29, 1988 submittal, along with additional information which we believed was relevant to the Staff's review of the submittal. Pending final NRC review and approval of the requested exemptions, NNECO considers this portion of the topic to be resolved. However, as stated in the November 8, 1990 letter, NNECO will evaluate reverse direction testing concerns for valves LP-14 A & B and notify the Staff of our proposed course of action before April 30, 1991.

Topic 1.19--Integrated Structural Analysis

This topic was addressed in our previous ISAP submittals. As discussed therein, NNECO had submitted information to resolve this issue. NNECO believes this ISAP topic is open pending NRC Staff review and concurrence with the previously submitted information.

Topic 1.22--Electrical Isolation

This ISAP topic encompassed a review of electrical isolation provisions at Millstone Unit No. 1 against the criteria of 10CFR50.55a(h) and IEEE Std. 279-1971. As noted in the previous ISAP submittals, NNECO provided the Staff with information to resolve this issue. Pending final Staff review and approval of NNECO's submittal on this issue, NNECO considers this ISAP topic to be resolved.

Topic 1.23--Grid Separation Procedures
Topic 1.25--Degraded Grid Voltage Procedures

These topics were addressed in detail in our previous ISAP submittals. NNECO also responded to the Staff's May 31, 1989 letter (which provided the NRC's position on the undervoltage protection scheme at Millstone Unit No. 1) by letter dated July 20, 1990.[6] In the letter, NNECO stated that implementation of a split-logic design was not justified and that the present design provided adequate protection. We noted that we would evaluate a split-logic design change as part of the ISAP process, provided the Staff could demonstrate a sufficient safety benefit from such a design, given the modifications already completed under this ISAP topic.

Subsequently, NNECO met with the Staff on October 16, 1990 to discuss the issues concerning a split-logic design. As a follow-up to the meeting, NNECO provided a letter[7] documenting several of the key points discussed at the meeting. NNECO reiterated its belief that implementation of a split-logic design at this time was not justified and that the existing design provides adequate protection. We further stated that while NNECO believes that additional modifications are not required to meet the appropriate regulation or criteria, we would be willing to upgrade the relays in the fast transfer scheme to Class 1E, provided it resolves the Staff's concerns with the existing logic design. NNECO is currently waiting for the Staff to respond to the October 30, 1990 letter. However, independent of the split-logic issue, NNECO considers this topic resolved and therefore proposes closure.

Topic 1.31--Equipment Classification/Vendor Interface (GL 83-28, Item 2.2)

By submittals dated September 25, 1987[8] and December 18, 1987,[9] NNECO provided information to the NRC Staff addressing the remaining issues under this topic. The NRC Staff has reevaluated its position with respect to this issue via Generic Letter (GL) 90-03, "Relaxation of Staff Position in Generic Letter 83-28, Item 2.2 Part 2, 'Vendor Interface for Safety-Related Components,'" issued on March 20, 1990. NNECO responded to GL 90-03 in a letter dated September 24, 1990,[10] and is currently awaiting NRC concurrence with the stated positions.

Topic 1.36--Technical Specifications Covered by GL 83-36

As previously discussed in the September 29, 1989 and April 30, 1990 submittals, this topic addresses proposed changes in plant Technical Specifications to reflect applicable TMI Action Plan Items identified in Generic Letter 83-36. NNECO submitted proposed license amendment requests required for the remaining items (i.e., postaccident sampling, noble gas effluent monitors, sampling and analysis of plant effluents, containment high-range radiation monitor, containment pressure monitor and containment water level monitor) via four letters dated August 1, 1989. NNECO considers this topic resolved pending NRC Staff review and concurrence.

Topic 1.52--SRV Failure - Setpoint Drift

As discussed in the September 29, 1989 and April 30, 1990 submittals, SRV setpoint drift is no longer a significant contributor to the risk of core melt at Millstone Unit No. 1. The topic is being kept open to address recommendations from the BWR Owners' Group (BWROG) which is currently addressing this issue. Several alternatives/recommended courses of action are being evaluated by the BWROG and are expected to be issued to the industry in early 1991.

Topic 1.100--Fire Barrier Penetration Seal Program Upgrade

This topic was discussed in our previous ISAP submittals. In a letter dated September 28, 1988,[11] NNECO provided a detailed description of the penetration seal program upgrade initiations and a revised schedule for completion of the program. The nonoutage walkdowns have been completed and the outage walkdowns continue to be scheduled for completion during the 1991 refueling outage. The program is scheduled to be complete by August, 1991 consistent with our previous submittals.

Topic 1.101--Fire Detection System Code Compliance

This project involves certain modifications to fire detection systems to ensure both their compliance with the intent of the NFPA codes and their ability to function in accordance with their design purpose. NNECO remains committed to assessing the potential modifications, which resulted from a previous ISAP evaluation, but as discussed in our September 29, 1989 and April 30, 1990 submittals, the proposed modifications continue to place extremely low in the ISAP ranking and therefore continue to be a candidate for cancellation. Additional information, which may result from this review, will be provided in a future ISAP/IIS submittal.

Topic 1.102--Fire Suppression System Code Compliance

This project involves modifications to the fire suppression systems to ensure their compliance with the intent of the NFPA codes and their ability to function in accordance with their design purpose. NNECO remains committed to assessing the potential modifications, which resulted from a previous ISAP evaluation, but as discussed in the September 29, 1989 and April 30, 1990 submittals, the proposed modifications continue to result in a very low ISAP ranking and are therefore still a candidate for cancellation. Additional information, which may result from this review, will be provided in a future ISAP/IIS submittal.

Topic 1.106--Station Blackout

This topic addresses the specific modifications which are being proposed to resolve the station blackout (SBO) issue for Millstone Unit No. 1. The Staff issued its Safety Evaluation of SBO response for Millstone Unit No. 1 by letter dated August 29, 1990. NNECO provided its response to the Staff's safety evaluation in a letter dated October 10, 1990.[12] In the letter, NNECO proposed its course of action to resolve the crosstie weatherization issue and also proposed its schedule for modification and testing of the control and protective circuitry associated with the circuit breakers that will be utilized for the SBO crosstie. NNECO stated that the plant modifications would be evaluated within ISAP and subsequently scheduled in the IIS, upon Staff concurrence with the proposed action plan.

Topic 1.107--Drywell Spray Flow Indication

This topic was discussed in detail in our April 30, 1990 submittal. The current scope of the project involves installation of redundant and environmentally qualified instruments to measure the drywell spray flow rate. As stated, the proposed modification ranks very low due to negligible attribute scores. Accordingly, if accomplished, it will be scheduled in the IIS for no sooner than a 1995 refueling outage completion date.

Topic 1.108--Torus Vacuum Breakers

This topic was discussed in detail in our September 29, 1989 submittal and also discussed in the April 30, 1990 submittal. As stated, a project was not proposed to resolve the concern that valves AC-3A & B are designed to fail open on loss of instrument air or 120V Instrument AC. However, an analysis was performed to determine the maximum possible benefit that could be derived from resolving this issue. The project received a very low ARM score and overall ranking. Given the low safety impact and low ARM ranking, NNECO considers this topic closed.

Topic 1.111--Motor Operated Valve Testing, Generic Letter 89-10

This ISAP topic addresses Generic Letter (GL) 89-10, "Safety-Related Motor Operated Valve Testing and Surveillance," issued by the NRC Staff in a letter dated June 28, 1989.(13)

By letter dated December 15, 1989,(14) NNECO responded to GL 89-10, certifying that detailed programs are being developed to address safety-related motor-operated valve (MOV) testing and surveillance at Millstone Unit No. 1. The program developed for Millstone Unit No. 1 will encompass the guidance as detailed in the Generic Letter.

The Staff requested that the program for Millstone Unit No. 1 be defined by the next refueling outage, subsequent to issuance of the Generic Letter.

As proposed, this program will be defined by May 1991 and completed prior to startup from the 1995 refueling outage. This is reflected in the IIS. This schedule may change if, while conducting the program there is a substantial change in scope of work from that initially projected. Should the program reveal potential modifications, an ARM evaluation will be performed. These resulting modifications will be scheduled according to their rank and reported in a future ISAP/IIS submittal. By letter dated October 25, 1990, GL 89-10, Supplement 3, "Consideration of the Results of NRC Sponsored Tests of Motor-Operated Valves" was issued. It is our understanding that Supplement 3 was issued as a result of the Staff's review of NRC-sponsored test results and MOV data provided by the BWR Owners' Group, which indicated that deficiencies may exist in certain MOVs. Several actions were requested. NNECO is evaluating the requests and currently plans to provide its response within 30 days of receipt (November 9, 1990).

Topic 1.112--Service Water System Evaluation, Generic Letter 89-13

This ISAP topic addresses Generic Letter (GL) 89-13, "Service Water System Problems Affecting Safety-Related Equipment," issued by the NRC Staff in a letter dated July 18, 1989.(15)

By letter dated January 25, 1990,(16) NNECO responded to GL 89-13, by providing a response to each of the five (5) recommended actions and additional information in response to Item V which addressed training. In response to the recommended actions and clarification provided by the Staff in the NRC sponsored workshops on GL 89-13, system reviews and analyses are being performed.

In a letter dated June 1, 1990,[17] NNECO proposed an alternate schedule for completion of Item 4 of GL 89-13 for Millstone Unit No. 1. Item 4 requires each licensee to confirm that the Service Water System will perform its intended function in accordance with the licensing basis of the plant and was to be completed prior to restart from the 1991 refueling outage (per GL 89-13 schedule). NNECO proposed that Millstone Unit No. 1's review be completed prior to startup from the Cycle 14 refueling outage in 1993. The alternate schedule is based on the experience gained from the work performed on the Haddam Neck Plant Service Water system, the complexity of each unit's particular system design and the existing personnel resources available to complete this review. As stated in previous ISAP submittals, recommended actions involving hardware modifications will be evaluated within the ARM. Modifications will be scheduled in the IIS, depending on their ranking. Subsequent information depending on their ranking will be reported in a future ISAP/IIS report.

Topic 1.114--Individual Plant Examinations, Generic Letter 88-20

By letter dated November 23, 1988,[18] the Staff transmitted Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities--10CFR50.54(f)." By letter dated July 27, 1989,[19] NNECO outlined the general approach which would be utilized in responding to Generic Letter (GL) 88-20, and summarized key points of past involvement in the field of probabilistic risk assessment (PRA), integrated safety assessment (ISA) and accident management (AM). By letter dated October 31, 1989,[20] NNECO provided the Staff with additional information pertaining to the particular methods and approaches expected to be utilized for the remaining portions of the IPE for the four Northeast Utilities (NU) plants. Specifically, the summary report schedules were reiterated, with the Millstone Unit No. 1 report expected to be submitted in mid-to-late 1991. The NRC Staff responded by letter dated January 9, 1990,[21] transmitting their acceptance of NNECO's approach, methodology and schedule.

NNECO recognizes the IPE as an analytical process. As such, potential modifications, which are generated from the plant specific analysis, will likely result in project assignments (PAs). These PAs will then be individually evaluated within the ISAP process. Information gathered as a result of these evaluations will be provided in a future ISAP/IIS update.

Topic 2.08--Extraction Steam Piping

As discussed in our September 29, 1989 and April 30, 1990 submittals, the extraction steam lines to the high pressure feedwater heaters continue to be scheduled for replacement during the 1993 (Cycle 14) refueling outage.

Topic 2.10--Drywell Ventilation Systems

This topic was discussed in our previous submittals. Currently, two of the three remaining coolers are scheduled for coil replacement installation during the 1991 refueling outage. The third coil is not anticipated to be replaced, as it was previously replaced in 1984. The installation of the variable speed fan motor drives for the five (5) already replaced cooling coils, was not completed during the 1989 outage because of resource constraints. The variable speed drive modification was initiated in 1989 and all construction work is currently scheduled for completion in the 1991 refueling outage.

Topic 2.30 MSIV Closure Test Frequency

This topic was last discussed in the November 9, 1988 submittal. NNECO stated that this topic involved only a technical specification change which was being evaluated as a routine licensing matter. The topic involved no hardware modifications or significant resource commitments and was therefore closed.

The original safety evaluation concluded that the positive safety benefit due to a reduction in the frequency of MSIV closure-induced transients would outweigh the negative safety benefit due to the decreased assurance of the operability of the reactor protection system logic. (At that time, inadvertent full closure of an MSIV during the 10% closure test would cause a scram due to high steam flow in the other 3 lines.) Since origination of the change, plant operating procedures have been revised to require a power reduction to about 65% prior to the conduct of the tests, to preclude a scram if MSIV over travel occurs. As such, there would no longer be a reduction in the frequency of MSIV closure-induced transients resulting from the technical specification change. Therefore, NNECO has determined that the change is not justified at this time, but will be reevaluated following completion of the Millstone Unit No. 1 IPE.

Topic 2.103--Loss of 125-VDC Power Study

This ISAP topic addressed the replacement of existing battery chargers with battery eliminating type, self-regulating chargers so as to provide reliable continuous DC power in the event of a loss of battery. The topic was discussed in detail in the September 29, 1989 submittal and briefly mentioned in the April 30, 1990 submittal.

As reflected therein, the project received a moderate ranking and remains scheduled in the IIS for the 1995 (Cycle 15) refueling outage.

Topic 2.104--Feedwater Nozzle Leakage Monitoring System

This topic was addressed in detail in our previous ISAP submittals. This project had previously received the highest ISAP value and relative ranking. This was a nonoutage project and was completed during August 1990. Implementation of this project may justify an extension of the inspection intervals for the nozzles and spargers. That extension would reduce the man-rem cost that the inspections involve. NNECO will address any extensions that it believes are justified as a routine licensing matter. Therefore, NNECO considers this topic closed.

Topic 2.106--Diesel Air Start System Upgrade

As discussed in our September 29, 1989 and April 30, 1990 submittals, current plans are to perform a visual boroscope inspection of the air receiver every refueling outage to verify internal cleanliness. Additional information, which may result from this inspection, will be provided in a future ISAP/IIS submittal. Accordingly, no modifications are currently planned.

Topic 2.107--Class 1E Generator Antimotoring Protection

This topic was addressed in detail in our submittal dated September 29, 1989, and briefly discussed in our April 30, 1990 submittal. This project received a high ARM value and ranking based largely on the relatively low remaining cost. The implementation of this project will avoid 2.05 hours of lost full power operation per year and installation continues to be scheduled for the 1995 (Cycle 15) refueling outage.

Topic 2.108--Generator Antimotoring Protection While Shutdown

This topic was addressed in detail in our submittal dated September 29, 1989 and briefly discussed in our April 30, 1990 submittal.

This project received a high ARM value and relative ranking based upon its positive Economic Performance score and its relatively low remaining cost. It continues to be scheduled for implementation during the 1995 (Cycle 15) refueling outage.

Topic 2.114--Reactor Building Closed Cooling Water System Class Boundaries

This topic addresses an engineering study that considers design and installation of modifications which may be necessary to ensure that the reactor building closed cooling water (RBCCW) system will have adequate class boundary breaks. A preliminary assessment of the scope of the project has concluded that no modifications are required and the project assignment is in the process of being closed. Therefore, NNECO considers this ISAP study closed.

Topic 2.115--Replacement of LPC-4A & B Valves

This topic was discussed in detail in our April 30, 1990 submittal. As stated, the project involves the replacement of the valves and motor-operators for LPC-4A & B, in an attempt to provide better throttling characteristics for the Emergency Service Water system. The project ranked high due to the anticipated impact on unit operation resulting from corrective maintenance. Therefore, the project continues to be scheduled for completion during the 1991 refueling outage.

Topic 2.116--Replacement of Reactor Water Cleanup Valves

This topic was discussed in detail in our April 30, 1990 submittal. As stated, the project involves the replacement of valve CU-2 (inside the drywell) and its motor-operator, and the installation of a new motor-operated valve in the RWCU inlet piping immediately outside the drywell. In addition, block valves would be installed on each side of CU-2A. The modification has been rescheduled in the IIS to be completed during the 1993 refueling outage because the delivery date and the testing of the new valves could not be scheduled and completed to support a 1991 outage completion.

Topic 2.117--Environmental Qualification of IC Local Control Stations

As discussed in the April 30, 1990 status report, the modifications addressed under this topic are required because of EEQ concerns that a break in the reactor building may expose local control panels 2254 and 2255 to a harsh environment.

Although the overall ISAP ranking for this project is medium, the need to resolve the EEQ concern elevated the project priority. Therefore, this project remains scheduled in the IIS to be completed during the 1991 refueling outage.

Topic 2.118--Replacement of the Normal Station Service Transformer

As discussed in the April 30, 1990 status report, the Normal Station Services Transformer (NSST) at Millstone Unit No. 1 had exhibited signs of deterioration for approximately the past seven years. During August 1989, the electrical load had to be shifted to the reserve station service transformer when the NSST showed signs of internal arcing. Inspection results indicated a failed "A" Phase no load tap changer and overheating in the low voltage windings as indicated by discoloration and brittleness of the insulation. This project involves the repair and reinstallation of the NSST.

The high public safety attribute score and the extremely high economic performance score, combined with a minimal net remaining project cost, resulted in a high overall ISAP ranking. The NSST has been repaired and returned to the site. As previously scheduled in the IIS, it will be reinstalled during the next outage of sufficient duration.

Topic 2.120--Atmospheric Control System Containment Isolation Valves

This topic addresses a current engineering study which considers possible modifications to existing valves or the procurement of new valves to mitigate high valve failure rate during local leak rate testing.

This engineering study has not been completed. Any forthcoming potential modifications or replacements will be evaluated in a subsequent ARM process cycle. This information will then be included in a future ISAP/IIS update report.

References

(1) E. J. Mroczka letter to U.S. Nuclear Regulatory Commission, "Supplement 1 to NUREG-0737 Control Room Design Review Implementation Plan," dated March 2, 1987.

(2) E. J. Mroczka letter to U.S. Nuclear Regulatory Commission," Supplement to NUREG-0737, Control Room Design Review Summary Report," dated October 30, 1989.

(3) M. L. Boyle letter to E. J. Mroczka, "Detailed Control Room Design Review--Millstone Nuclear Power Station, Unit No. 1 (TAC No. 56138)," dated July 10, 1990.

(4) E. J. Mroczka letter to U.S. Nuclear Regulatory Commission, "Millstone Unit No. 1 Response to NRC Staff Additional Information Request on Control Room Design Review," dated June 29, 1990.

(5) E. J. Mroczka letter to U.S. Nuclear Regulatory Commission, "Request for Exemption from Appendix J Requirements--Additional Information," dated November 8, 1990.

(6) E. J. Mroczka letter to U.S. Nuclear Regulatory Commission, "Degraded Grid Undervoltage Protection," dated July 20, 1990.

(7) E. J. Mroczka letter to U.S. Nuclear Regulatory Commission, "Loss of Normal Power Logic," dated October 30, 1990.

(8) E. J. Mroczka letter to U.S. Nuclear Regulatory Commission, "Generic etter 83-28, Item 2.2," dated September 25, 1987.

(9) E. J. Mroczka letter to U.S. Nuclear Regulatory Commission, "Generic Letter 83-28, Item 2.2," dated December 18, 1987.

(10) E. J. Mroczka letter to U.S. Nuclear Regulatory Commission, "Response to Generic Letter 90-03--Vendor Interface for Safety-Related Components," dated September 24, 1990.

(11) E. J. Mroczka letter to U.S. Nuclear Regulatory Commission, "Fire Barrier Penetration Seal Program Upgrade," dated September 28, 1988.

(12) E. J. Mroczka letter to Dr. T. E. Murley, "Response to Safety Evaluation for Station Blackout (TAC No. 68566)," dated October 10, 1990.

(13) James G. Partlow letter to All Licensees of Operating Nuclear Power Plants and Holders of Construction Permits for Nuclear Power Plants, "Safety-Related Motor-Operated Valve Testing and Surveillance (Generic Letter 89-10)," dated June 28, 1989.

(14) E. J. Mroczka letter to U.S. Nuclear Regulatory Commission, "Haddam Neck Plant, Millstone Nuclear Power Station, Unit Nos. 1, 2, and 3, Generic Letter 89-10, Safety-Related Motor-Operated Valve Testing and Surveillance," dated December 15, 1989.

(15) J. G. Partlow to All holders of Operating Licenses or Construction Permits for Nuclear Power Plants, "Service Water System Problems Affecting Safety-Related Equipment (Generic Letter 89-13)," dated July 18, 1989.

(16) E. J. Mroczka letter to U.S. Nuclear Regulatory Commission, "Haddam Neck Plant, Millstone Nuclear Power Station, Unit Nos. 1, 2, and 3, Service Water System--Generic Letter (GL) 89-13," dated January 25, 1990.

(17) E. J. Mroczka letter to U.S. Nuclear Regulatory Commission, "Service Water System--Generic Letter (GL) 89-13," dated June 1, 1990.

(18) D. M. Crutchfield letter to All Licensees Holding Operating Licenses and Construction Permits for Nuclear Power Reactor Facilities, "Individual Plant Examination for Severe Accident Vulnerabilities--10CFR50.54(f), Generic Letter 88-20," dated November 23, 1988.

(19) E. J. Mroczka letter to U.S. Nuclear Regulatory Commission, "Haddam Neck Plant, Millstone Unit Nos. 1, 2, and 3, Response to Generic Letter 88-20, Individual Plant Examinations for Severe Accident Vulnerabilities," dated July 27, 1989.

(20) E. J. Mroczka letter to U.S. Nuclear Regulatory Commission, "Haddam Neck Plant, Millstone Unit Nos. 1, 2, and 3, Response to Generic Letter 88-20, Supplement 1, Individual Plant Examinations for Severe Accident Vulnerabilities," dated October 31, 1989.

(21) J. F. Stolz letter to E. J. Mroczka, "Review of 60-Day Response to Generic Letter 88-20, Individual Plant Examinations (IPE) (TAC Nos. 74417/74432/74434)," dated January 9, 1990.

Docket No. 50-245
B13651

Attachment 3

Millstone Nuclear Power Station, Unit No. 1
Integrated Safety Assessment Program

Evaluation of New ISAP Topics or
Reevaluation of Existing Topics

November 1990

Millstone Nuclear Power Station, Unit No. 1
Integrated Safety Assessment Program
Evaluation of New ISAP Topics or
Reevaluation of Existing Topics

Topic 1.09--Regulatory Guide 1.97 Instrumentation

As discussed in our submittal dated April 30, 1990, NNECO determined that reviews of Main Feedwater Flow and cooling water flow to Engineered Safety Feature (ESF) system components were not performed as part of the initial Topic 1.09. As proposed, this topic was reopened and these issues have been evaluated to determine which, if any, modifications would provide significant benefit, thus meriting implementation. The results of these reviews are provided under sub-topic 1.09.1 and 1.09.2, respectively. Additionally, information pertaining to environmentally and seismically qualified neutron monitoring instrumentation, not available at the time of the original evaluation submitted in 1986, may now be available. Information on this issue is provided as Topic 1.09.3. These issues consist of modifying/upgrading/installing instruments to comply with the guidance of Regulatory Guide 1.97.

Topic 1.09.1--Main Feedwater Flow

It was unclear whether the current main feedwater flow indication met the guidelines of Regulatory Guide 1.97. Specifically, the environmental qualification of the instrumentation was in question. The transmitters which provide both control and indication of main feedwater flow, FT-644A and B, are environmentally qualified for the environment that they will experience, and on the EQ master list. Their associated cables are also qualified. As such, the replacement of the instrumentation is not needed, and would not provide any benefit. Therefore, any proposed modifications would not provide a measurable improvement in public risk. Accordingly, there are no additional modifications required and NNECO considers this subtopic closed.

Topic 1.09.2--Cooling Water Flow to Engineered Safety Feature (ESF) Components

It was unclear whether the current ESF system components flow indication met the guidelines of Regulatory Guide 1.97. Specifically, the environmental qualification of the emergency service water (ESW) system flow instrumentation was in question. The transmitters which provide indication of ESW flow to the LPCI heat exchangers, FT-1542A and B, are environmentally qualified for the environment that they will experience, and on the EQ master list. Their associated cables are also qualified. As such, the replacement of the instrumentation is not needed, and would not provide any benefit. Therefore, the proposed modifications would not provide a measurable improvement in public risk. Accordingly, there are no modifications required and NNECO considers this subtopic closed.

Topic 1.09.3--Neutron Monitoring

I. Introduction

Environmentally and seismically qualified neutron monitoring instrumentation was not available when the system was originally installed or at the time of the original evaluation submitted in 1986. Qualified equipment may now be available. Neutron monitoring systems have been tested to demonstrate their qualification for use in some Boiling Water Reactors (BWR). These tests have not been determined to be applicable to Millstone Unit No. 1. One BWR operating utility has significant concerns about the qualification testing that has been performed. This topic consisted of an evaluation to determine the benefit of replacing the existing equipment with potentially qualified equipment which meets Regulatory Guide 1.97, as requested by the NRC Staff in a letter dated February 13, 1990.[1]

II. Evaluation

A. Public Safety

This proposed project involves environmentally qualified instruments with redundant channels and separate power sources for neutron flux measurement. In the current design, the instruments to measure neutron flux are not environmentally qualified. There is a concern that the operator may lose the ability to determine core power following a transient. This could occur due to the harsh environment caused by a loss of coolant accident (LOCA) in the drywell or reactor building, or due to loss of power for the instrumentation.

In the Millstone Unit No. 1 design, each of the control rods has nonenvironmentally qualified position indications, including full-in, full-out lights. Following a reactor trip, the operator can infer subcriticality of the reactor by reviewing the rod position. If all control rods are inserted, subcriticality is ensured and none of the subsequent operator actions require knowledge of the core power level.

If at least two adjacent rods remain withdrawn, an instrument reading is needed to ensure core subcriticality. With failure of neutron flux measurement, the operator will not be able to determine the core power and therefore will classify the event as an anticipated transient without scram (ATWS). None of the subsequent operator actions described in ATWS mitigating procedures, however, require knowledge of core power level. For this low probability scenario, loss of neutron flux monitoring will be only an inconvenience. The loss will not prevent the operator from bringing the plant down to safe shutdown. This proposed modification would provide an insignificant reduction in public risk. Therefore, this issue has a public safety benefit of $150/year.

B. Economic Performance

The current monitoring system has not affected plant availability in the past. Assuming a potentially new system will have the same reliability, installation of such instruments will not impact plant availability. Therefore, the effect of this modification on plant availability is negligible.

C. Personnel Safety

Replacement of existing neutron monitoring equipment for post accident conditions will not directly affect personnel safety. The use of the existing neutron monitoring system or a potentially new system, environmentally qualified for post accident indication, would have very little affect on the safety of plant personnel.

D. Personnel Productivity

No effect is anticipated. Therefore, benefit is presumed near zero ($0/year).

III. Conclusion

This project received a very low overall ranking due to the small public safety benefit and very high project cost. Although the NRC Staff requested the inclusion of installation of qualified neutron flux monitoring instrumentation into this IIS, in the previously referenced letter,[1] there are no modifications scheduled. NNECO considers this topic resolved pending NRC Staff review and concurrence.

Topic 1.43--Water Hammer

I. Introduction

This topic was previously addressed in detail in our submittal dated November 9, 1988 and briefly mentioned in the September 29, 1989 and April 30, 1990 reports. It was also recently discussed in NNECO's response to Generic Letter 89-19 "Request for Action Related to Resolution of Unresolved Safety Issue A-47, 'Safety Implication of Control Systems in LWR Nuclear Power Plants.'"[2]

In its response, NNECO stated that it believed that the existing vessel overfill protection system provided adequate protection. Nevertheless, since the Millstone Unit No. 1 Probabilistic Safety Study had identified the loss of 120 V vital AC as contributing approximately 10% of the CMF, NNECO proposed to evaluate several actions to correct this situation. The key problems with this event are considerable loss of instrumentation (e.g., 5 out of 6 level indicators fail low), misleading alarms, and potentially extensive damage to plant equipment due to loss of both feedwater control and the high-water-level feedwater pump trip following a manual reactor scram.

As stated in the April 30, 1990 report, the current project scope includes implementation of procedural changes and modification of existing plant annunciators (alarms) to provide prompt recognition of a loss of vital AC. This portion of the topic has been evaluated during this ARM evaluation cycle. In addition, this topic includes the evaluation of relocation and/or repowering of two of the four existing vessel level instruments, which will be performed in a future ISAP/IIS update report.

II. Evaluation

A. Public Safety

Following a loss of the 120 V vital AC bus, the control room operators could fail to diagnose the event due to the presence of misleading indications, and the reactor vessel could overfill causing damage to the main steam lines, the isolation condenser, and other equipment, precluding their use for decay heat removal.

With the current plant configuration, following a loss of the vital AC bus, 5 of 6 reactor vessel level indicators, which are powered by vital AC, would fail low. In addition, the alarm "Vital AC on Alternate Supply" would annunciate, which would not clearly indicate to the operator that the vital AC bus was lost. It is highly probable that the operator would promptly manually scram the reactor after suddenly observing the vessel level indicators drop. Since the loss of vital AC would also result in a loss of feedwater control and the vessel high level feed pump trip, the failure to manually trip the feedwater pumps would be assumed to lead to vessel overfill and water hammer in the main steam lines and the isolation condenser. It is assumed that the operator would most likely not realize that the feedwater pumps should be tripped, since he would think that the vessel level was low.

Following the addition of a vital AC undervoltage alarm, DC-powered vessel level indication on CRP 905, the enhancement of operating procedures addressing loss of vital AC events, and specific training on this issue, the ability of the operator to correctly diagnose the event would be greatly improved. As such, the assumed probability of failing to trip the feedwater pumps would be significantly reduced.

Based on a CMF reduction of 6.1E-6/yr and a Man-Rem savings of approximately 188 Man-Rem over the remaining life of the plant, this project was assigned a score of $10,950/year.

B. Economic Performance

Loss of the 120 V vital AC will likely initiate the following chain of events:

- Upon loss of vital AC, the feedwater regulating valves will be locked in their "as is" position (assumed to be open).
- There will be a considerable loss of key level instrumentation (5 of 6 indicators fail low).

- o A misleading indication will show that vital AC is powered by the "alternate supply" when, in fact, it is unavailable.
- o The plant will be manually tripped; however, feedwater will continue to flow past the locked open regulating valves.
- o With loss of both the feedwater control and the high-level feed pump trip, extensive damage will likely occur due to a severe water hammer in the main steam lines.

It is likely that the operators will successfully mitigate a core damage event, however, not before extensive damage to plant equipment occurs (main steam line supports, turbine, safety relief valves). Because an event of this type has not occurred, the downtime estimate is 2-5 weeks (3.5 week average). Therefore, the yearly probability of a loss of vital AC power event is 3.54 E-03. As such, the economic benefit of implementing effective modifications is 2.1 hours of full power operation per year, which equates to $27,880 per year based on 1990 replacement power costs.

C. Personnel Safety

The proposed modifications will not affect personnel safety.

D. Personnel Productivity

The proposed modifications will not affect personnel productivity.

III. Conclusion

The proposed changes consist of the addition of an undervoltage alarm for the vital AC bus as well as an additional DC powered ATWS level indication on the main control board and upgraded procedural guidance. These steps will reduce the contribution to core melt frequency from over 10 percent to approximately 3 percent for the Loss of Vital AC event. This project received a very high overall ranking due to its benefits in public safety and economic performance. Accordingly, this project is scheduled in the IIS for completion during the 1991 refueling outage.

Topic 1.104--Pump Flow Rate Instrumentation

I. Introduction

This topic was discussed in detail in our submittal dated September 29, 1989 and briefly mentioned in the April 30, 1990 submittal. This topic originally encompassed evaluation of flow measuring devices for the following pumps:

- o Condensate Pumps (A, B, C)
- o Condensate Booster Pumps (A, B, C)
- o Service Water Pumps (A, B, C, D)
- o Reactor Building Closed Cooling Water Pumps (A, B)
- o Turbine Building Secondary Closed Cooling Water Pumps (A, B)
- o Diesel Generator Fuel Oil Transfer Pumps (A, B)

- o Gas Turbine Generator Fuel Forwarding Pumps (A, B)
- o Reactor Feedwater Seal Injection Pumps (A, B, C)

As stated in the April 30, 1990 report, this topic was expanded to include Shutdown Cooling Pumps A & B. The Service Water System pumps have been evaluated under ISAP Topic 1.104.1, and are discussed below. The remaining pumps will be evaluated after the 1991 refueling outage and the results of the evaluation will be provided in a future ISAP/IIS update report.

Topic 1.104.1--Service Water Pump Flow Rate Instrumentation

I. Introduction

This ISAP topic addresses the installation of instrumentation to monitor flow through portions of the Service Water System. The objective of installing the new instrumentation is to provide indication of pump performance during IST pump testing and a secondary purpose is to monitor for RBCCW or TBSCCW heat exchanger plugging.

The proposed modification includes sensors which will monitor the flow of Service Water via a 24" line providing flow to the RBCCW heat exchangers. The second sensor will be located on a 16" line which provides Service Water flow to the TBSCCW heat exchangers. To determine Service Water pump flow, the unmonitored Service Water lines will be isolated and the pumps run separately. The summation of the two sensor readings will provide the flow of the pump which is being tested. Because two or more Service Water pumps must be operating under normal plant conditions, individual pump flow testing can only be accomplished during cold shutdown/refueling outage conditions.

II. Evaluation

A. Public Safety

In order to accurately determine flow from a Service Water pump, it would be necessary to: secure the three remaining Service Water pumps; isolate flow to the noninstrumented lines; and add the separately monitored flow to the TBSCCW and RBCCW heat exchangers. As this can only be accomplished with the turbine off line, the ability to accurately monitor individual pump flow and/or total Service Water flow during normal operating conditions will still not exist. Furthermore, installation of the flow sensing devices immediately downstream of the pumps is not feasible due to the layout of the piping. Therefore, the proposed modifications will not provide individual pump flow indication or total Service Water flow indication locally or in the control room.

However, a potential operational benefit would be the ability to locally monitor Service Water flow to the RBCCW and TBSCCW heat exchangers. Operators could have a more direct means of detecting probable mussel fouling of the heat exchangers. However, since increases in the closed cooling water temperatures have provided this indication in the past, no measurable positive safety benefits resulting from this project can be credited.

Therefore, the proposed modifications would not provide a measurable improvement in public risk. As such, this project was assigned a public risk score of $0/year.

B. Economic Performance

The measuring of flow rates of these pumps may potentially detect some flow rate changes and, depending on the degree of deviation, indicate a need for corrective action. A deviation could either be the result of an actual pump problem or faulty instrumentation. If faulty instrumentation were the cause, it would be quickly determined and corrected. If an actual pump problem were the cause, it would be detected by other monitoring equipment and result in a pump shutdown, regardless of the flow instrumentation data. Implementation of this project will have a negligible impact on plant availability.

C. Personnel Safety

The areas designated for the installation of the local instrumentation are low radiation areas so the affects on radiation exposure are negligible. The local instrumentation will be installed at elevated locations on the Service Water piping which increases the occurrence of accident falls and personnel injury. This aspect of the new instrumentation will be minimized by following the OSHA requirement for the installation of staging and platforms. Based on this assessment, the potential for personnel safety is a negative $-1,800 dollars/year.

D. Personnel Productivity

The new Service Water flow instrumentation will require additional component calibration, maintenance, and surveillances. The instrumentation itself cannot be used during normal operation to monitor flow rates and evaluate system performance on a continuing basis. Based on other similar equipment, a nonconservative estimate of 80 man-hours per year will be required to maintain all aspects of this modification. Therefore, the impact on personnel productivity is $-2,560/year.

II. Conclusion

This project received a very low overall ranking due to the negative personnel safety and personnel productivity attribute scores. However, the installation of annubar flow devices during the 1991 outage will provide NNECO with an indication of the relative benefits of installation of similar flow measuring devices. Accordingly, this project is scheduled in the IIS for a 1991 refueling outage completion date. Other flow indicators will be evaluated for installation after the 1991 refueling outage.

Topic 1.110--Vital Area Ventilation

I. Introduction

An NRC-sponsored regulatory effectiveness review (RER) was conducted at the Millstone Station for Units 1, 2, and 3, during June 7-15, 1988. The purpose of the RER program is to assess the effectiveness of safeguards against radiological sabotage at operating nuclear power plants relative to those design basis threats contained in 10CFR73. As a result of the RER audit, the NRC identified areas which, if corrected, could result in an enhancement to the overall security program. One area was the "hardening" of vital barrier penetrations which include heating/ventilation air ducts. One ducting system was initially identified at Millstone Unit No. 1. (The specific area may not be disclosed due to its classification as safeguards information.) In addition to addressing the deficiency of this one penetration, NNECO was committed to review other vital area ventilation ducting to ensure that no other similar potential areas exist. A thorough evaluation of all of the vital area ventilation ducting has since been completed. This review identified that no modifications are required on any vital area ducting other than the one duct penetration identified during the RER audit. This modification is scheduled for late 1991.

The vital area duct work under scrutiny in this project is low velocity duct work. Installation of security barriers in these ducts represents an approximately 12-14 percent reduction in total flow area. Typically, the low velocity duct work flow area may be restricted by as much as 50 percent before any significant decreases in fan performance can be observed. Therefore, adequate ventilation for the vital area and the associated components should be maintained. To insure adequate ventilation is maintained, air flow measurements will be obtained before and after installation to determine the effect on the system. If necessary, the ventilation system(s) will be rebalanced to offset any system effect induced by the security grating installations.

II. Evaluation

A. Public Safety

The modifications would not have any measurable impact on public safety, since the assumed probability of plant sabotage following access through the ventilation openings is already considered relatively low. Air flow testing and any necessary rebalancing of the ventilation systems following the installation of the security barriers would ensure that sufficient heating and cooling capacity is maintained.

The proposed modifications would not provide a significant improvement in public risk. As such, this project was assigned a public risk score of $0/year.

B. Economic Performance

Based on the nature of the proposed modifications, these vital area modifications are not expected to have an impact on plant performance or availability.

C. Personnel Safety

The modifications will not directly affect personnel safet . They have been proposed to address security concerns and will not improve normal plant safety or safety following an accident condition. Therefore, the affects on personnel safety are negligible.

D. Personnel Productivity

The areas designated for modification will not impact normal plant operation or events following an accident condition. Therefore, the affects on personnel productivity by the proposed modifications will be negligible.

III. Conclusion

This study received a very low overall ranking due to the negligible attribute scores. However, there are modifications scheduled at this time due to regulatory sensitivity of this issue. The proposed modification is scheduled for implementation in late 1991.

Topic 1.113--Hardened Wetwell Vent, Generic Letter 89-16

By letter dated June 15, 1990,(3) the NRC Staff provided its backfit analysis for installation of a hardened wetwell vent at Millstone Unit No. 1. The Staff indicated that the results of their analyses demonstrated that backfitting the plant with hardened vent capability was warranted. In addition, the Staff stated that we should reconsider our decision not to install the hardened wetwell vent.

On July 24, 1990, NNECO and other members of the Boiling Water Reactor Owners' Group (BWROG) met with the Staff to discuss installation of the hardened wetwell vent at isolation condenser plants. Subsequently, on August 7, 1990, a telephone conversation took place between NNECO and Mr. W. T. Russell and other members of the Staff. NNECO was notified that the Staff had considered the information presented at the July 24, BWROG/NRC meeting, but had decided to proceed with the imposition of the backfit to install a hardened wetwell vent at Millstone Unit No. 1.

NNECO conducted an analysis of the benefits of a hardened wetwell vent using the latest available deterministic and probabilistic risk assessment (PRA) information for Millstone Unit No. 1 and reported the results in a letter dated August 8, 1990.(4) We also included pertinent plant unique information to better allow the Staff to fully understand the circumstances that exist at Millstone Unit No. 1. We concluded that the incremental benefit of installing a hardened wetwell vent to satisfy the basic design objective of <u>preventing</u> a core-melt event as a consequence of a transient with subsequent loss of decay heat removal capability (TW sequences) is not sufficient to warrant an immediate decision. We stated that until the Individual Plant Examination

(IPE) (back-end analysis) is completed in late-1990, the potential benefits for mitigation of core-melt scenarios could not be determined. It is not unreasonable to believe that the IPE may identify modifications that positively affect a number of accident scenarios, including TW sequences. Such modifications could demonstrate significantly higher safety benefits than installing hardened vent capability. Therefore, NNECO believed that evaluating the hardened wetwell vent with other IPE-related modifications in an integrated environment, such as in ISAP, was most appropriate.

In a letter dated August 20, 1990,(5) the Staff stated that it was their belief that the results of the NRC's backfit analysis for installation of a hardened wetwell vent remained valid, based in part on a preliminary review of our August 8, 1990 letter. Accordingly, the Staff stated their plan to issue an Order directing NNECO to install the hardened wetwell vent, unless a commitment was made to install the vent at Millstone Unit No. 1, within two weeks of the August 20, 1990 letter.

NNECO responded to the Staff's letter and on September 4, 1990(6) advised the Staff that NNECO committed to installation of a hardened vent at Millstone Unit No. 1, as the Staff had requested. We stated that we would proceed with the initial design and engineering of a hardened vent, to support installation during the 1993 refueling outage, currently scheduled to commence in February 1993.

In accordance with the discussion between our organizations on August 31, 1990, we reiterate that our IPE effort remains on schedule, such that additional insights regarding the safety benefits to be realized by installing a hardened wetwell vent, will be available by the end of 1990. If the IPE effort demonstrates a compelling reason for this commitment to be reconsidered, we will promptly bring the appropriate information to the attention of the Staff, outlining what we believe is the preferred course of action. Unless and until such time that the NRC concurs with some other course of action, we will proceed with installation of a hardened vent during the 1993 refueling outage.

Subsequent to our September 4, 1990 letter, the Staff responded to NNECO stating that our plans and schedule for installing the hardened wetwell vent were acceptable. In addition, the Staff stated that their conclusions, on the merits of the hardened vent, remain valid.

While NNECO continues to believe that its analysis demonstrates that the NRC's Backfit Analysis overstated the benefits of a hardened wetwell vent, and that the IPE, in conjunction with ISAP, would be the more appropriate environment to evaluate and schedule proposed containment modifications, NNECO is currently proceeding with its initial engineering to support installation and testing during the 1993 refueling outage.

As discussed above, should NNECO identify a compelling reason to reconsider the commitment to install the hardened vent, we will notify the Staff accordingly.

Consistent with past practice of evaluating all plant modifications, the hardened wetwell vent modifications have been evaluated under this topic.

I. Introduction

The standby gas treatment system (SGTS) is an emergency system with two redundant trains designed to filter and exhaust higher than normal airborne radioactivity from the reactor building and/or the drywell to the ventilation stack. The proposed hardened wetwell vent would provide SGTS bypass capabilities during severe accident conditions where there is a loss of long-term decay heat removal and high drywell pressure is present.

The design of the proposed hardened vent system involves the installation of a 12-inch bypass line, including one motor-operated valve and downstream blowout disc in parallel with the SGTS filter units. Circuit changes are needed to open existing, normally closed, containment isolation valves 1-AC-10 and 1-AC-11. Control circuits for SGTS unit isolation valve number 1-SG-2A, and B, and 1-SG-4A and B, are to be modified to ensure the valves close during hardened vent operation. These SGTS valves are now designed to open during design basis events.

The conceptual design also includes installation of a 12-inch, motor-operated bypass line around the existing SGTS filter units, and installation of new bypass piping and motor-operated valves parallel with containment isolation valves 1-AC-10 and 11. Motor-operated valves would also be installed at the SGTS filter train inlet and outlet, one upstream of 1-SG-2A and B and one downstream of 1-SG-5. This design keeps the hardened vent function separate and independent from the present SGTS functions. In total, five (5) DC motor-operated valves, two (2) blowout discs, and necessary piping and controls would be added. The valves would be remotely operated from the control room.

II. Evaluation

A. Public Safety

The benefit of installing a hardened vent has been evaluated in comparison to maintaining the existing "soft" vent capability. The existing vent is expected to reduce all sequences involving loss of long-term decay heat removal except sequences caused by ATWS or human error from 2.22E-5 year to 2.91E-6 year. This equates to a 1.93E-5/year reduction. The proposed hardened vent would, at most, further reduce these sequences to 2.01E-6/year, which equates to an additional 9E-7/year reduction, a public risk benefit of $2,000/year.

B. Economic Performance

From a system demand perspective, since this hardened vent system is normally idle and is only to be used during severe accident conditions, there is no impact on plant availability.

From a maintainability perspective, the system itself cannot be tested due to the permanent isolation via the blowout discs. However, the individual MOVs would likely be subject to surveillance testing under the IST program.

Under normal circumstances, the system will be in an idle state during both plant operation and outages. System operability can only be verified via valve testing and surveillance. If a valve failure is discovered, it is expected that repairs can be made with no impact on plant availability, assuming a minimum seven-day allowance for restoration and average valve reliability characteristics.

Finally, maintainability of other systems and equipment would not be measurably affected since it is assumed that maintainability is always considered during all plant changes.

Implementation of this proposed change and its impact on plant availability is highly dependent on the reliability of the newly installed valves and their subsequent testing and surveillance requirements. Assuming that the valve's reliability falls within the industry average, testing and surveillance activities are not excessive and there is an allowance of no less than seven days to repair a failed valve, the addition of the hardened wetwell vent is not expected to have a measurable impact on plant availability.

C. Personnel Safety

The hardened vent capability could provide some personnel safety (industrial and radiation) protection against elevated temperatures and radiation levels in the reactor building resulting from containment venting while implementing emergency procedures or in response to severely degraded plant conditions. The potential estimated saving under these severe conditions is three man-rem, to plant personnel. However, the use of the proposed hardened vent system, during routine plant operation, would be extremely unlikely. Since this evaluation considers the impact on safety between the preimplementation and the postimplementation situation for the "day-to-day activities associated with routine operation," the personnel safety benefit would be negligible.

D. Personnel Productivity

The additional maintenance, administration, training, and surveillance required for a system designed for use only during degraded conditions (beyond the design basis) provides no benefit during plant operation. A quantitative analysis of the affect on personnel productivity cannot be achieved without the identification of specific components or equipment. However, based on other similar equipment, 100 man-hours per year will be required to maintain all aspects of this modification for a personnel productivity score of $-3,200/year.

III. Conclusion

Overall, this project received a low ARM value and relative ranking. However, the regulatory sensitivity of this issue has significantly increased project priority. Accordingly, this project has been scheduled in the IIS for implementation in the 1993 refueling outage.

Topic 1.115--Reactor Water Level Reference Leg Break, Generic Letter 89-11

I. Introduction

This topic was initiated due to the Staff concern regarding the possibility of a reactor water level instrument sensing line break coupled with an additional independent single failure of a component in a control or protection system on all BWR plants. This was presented to all BWR owners via Generic Issue (GI) 101. All BWR plant designs were placed into one of five categories. The NRC then analyzed each of the five categories for the probability of occurrence for this event. The reactor water level measurement systems in BWRs consist of three main components. The upper portion of the sensing line is the reference water leg and is connected to a condensing chamber and to the reactor vessel steam space. The variable water leg is connected to the reactor below the expected normal range of water level (lower portion of sensing line). The actual water level is then sensed via a differential pressure transmitter located between the reference and variable water legs. These differential pressure sensors are used as input into the protection and control systems. The NRC postulated that a break or leak in the reference leg could cause the reference water leg to decrease, thus, causing an indication of a false high reactor water level. This false high indication would cause the trip of the reactor feed pumps, and in addition, may also prevent the automatic operation of emergency safety systems. This problem could also confuse an operator's ability to assess the actual reactor vessel water level.

After the NRC Staff completed their analysis of this issue, they concluded that all BWR designs, in conjunction with operator training and procedures, provided adequate protection in the event of an instrument line break in any of the reactor vessel water level instrument systems. Therefore, the NRC Staff did not propose any corrective action.

As requested in Generic Letter 89-11, NUREG/CR-5112 was reviewed. Several differences between the NUREG and the actual Millstone Unit No. 1 designs were identified. NNECO concluded that a turbine trip would not occur from a reactor vessel water level instrument line break, contrary to the NRC Staff analysis.

The Millstone Unit No. 1 plant response to this break combined with a single failure of a reactor level indicator switch LIS-263-57A (or B) or LIS-263-58A (or B), depending on which reference leg is broken, was evaluated. In this postulated scenario, the reactor vessel level would decrease. However, due to the false high level indication, the Main Steam Isolation Valves (MSIVs) would not close at low-low level and the Feedwater Coolant Injection (FWCI) system would not initiate. However, based on 10-minute operator action, the Automatic Depressurization System (ADS) and the Low Pressure Coolant Injection (LPCI) would result in a scenario that is bounded by the small break LOCA Analysis. This break is bounded by the small break LOCA analysis because the postulated break is 0.02 ft.^{2} versus 0.01 ft.^{2} postulated in the LOCA analysis. Therefore, pressurization of the drywell will be faster than the condition analyzed and will actuate a reactor scram on high drywell pressure (less than 22 seconds). One can postulate that a 1-inch instrument line in the reactor vessel level control system could also break. Since the size of this line is less than 0.01 ft.², it would result in a longer time to reactor scram. These smaller breaks are not required to be postulated as part of the current plant licensing basis. Nonetheless, NNECO evaluated the scenario of a break or leak in the RPV instrument reference leg sensing line of 1 inch or less along with a single active failure of LIS-263-57A (or B) or LIS-263-58A (or B). This scenario assumes that the turbine control system is able to maintain Reactor Pressure Vessel (RPV) pressure within the normal operating limits and the drywell coolers are still operating which will not allow the drywell pressure to increase (i.e., no trip due to increased drywell pressure). This ISAP topic is a direct result of this postulated scenario.

II. Evaluation

A. Public Safety

The public safety benefit of the proposed project was quantified by assuming that the core melt sequences that result from the lack of a scram due to the failure of a single level switch to operate following a break in the other reference leg would be eliminated by the modifications.

With the existing plant design, the rupture of one of the two reference legs (the initiating event), followed by the failure of one of the two narrow range low level switches (LIS-263-57A or B, or LIS-263-58A or B, depending on which reference leg is broken), would be expected to result in the following:

- The break is small enough so that the turbine control system maintains reactor pressure within the normal operating range;
- The reactor feedwater pumps would trip on the perceived high water level;
- Reactor water level would decrease due to no high pressure makeup, but a reactor scram would not occur at low level (+8 inches) due to the failure of the level switch;
- The MSIVs would not get a closure signal at low level, since this signal is also dependent on the failed level switch;
- The automatic start of FWCI would be prevented due to the false high water level trip signal;
- Upon vessel level decreasing to low-low level, other safety functions that are not dependent on the failed low level switch, such as LPCI, Core Spray, IC, and an ATWS scram signal would be expected to activate.

While the proposed project entails the installation of four additional low level transmitters to alleviate the single failure issue (and reduce the CMF by 1.8E-8/yr.), the majority of the core melt frequency resulting from a reference leg break would still remain.

The loss of feedwater due to the false high level signal would still occur, since the high level feedwater pump trip logic is not being modified. If the proposed change is implemented, the remaining CMF due to a reference leg break would be 2.7E-6/yr.

Based on a CMF reduction of 1.8E-8/yr and Man-Rem savings of less than one Man-Rem over the remaining life of the plant, this project was assigned a score of $50/year.

B. <u>Economic Performance</u>

This proposed change utilizes the existing sensing lines with the addition of four new level transmitters, two added on each sensing line leg. However, the addition of this instrumentation would potentially create both an advantage and a disadvantage. The advantage is this potential modification would ensure the activation of the scram logic in the proposed scenario as well as ensure scram activation of this system under the analyzed scenarios (i.e., instrument line break of greater than 1 inch with the active single failure of LIS-263-58A(B)). The disadvantage is that by adding the additional instrumentation, the chances for an inadvertent plant trip due to a reactor scram is increased (i.e., one failure in every 1,841 years (current design) versus one failure in every 456 years (proposed design)). It should be noted that even with this "order of magnitude" change, the probability still remains relatively small. According to plant records, to date there has not been an impact on plant availability due to these components.

The proposed additional level sensing instruments will be designed to exist in the containment atmosphere. The location of these instruments will not inhibit their maintenance, nor other equipment in the area. Therefore, the reliability and maintainability of this system would not be affected.

With the addition of this instrumentation, the control logic associated with the Reactor Protection System (RPS) system would require modification. However, this logic change would be expected to be minor. Operator training on the logic changes would also be required.

Since the existing Technical Specification Limiting Condition for Operation (LCO) would not be modified, nor would new Technical Specifications be required, this proposed change would provide Millstone Unit No. 1 with more latitude during component failure events because the reactor scram logic would have additional redundancy. Therefore, since the probability of multiple failure due to instrument malfunction remains negligible, and the existing Technical Specifications and instrument maintainability would remain unchanged, the effect of this issue on plant availability would be negligible.

C. Personnel Safety

The addition of four new reactor vessel level transmitters and corresponding logic modifications to the RPS system would prevent a single active failure from impacting the design basis of the RPS system. These modifications would result in placing the RPV water level scram logic from a "one out of two" taken twice logic scheme to a "one out of four" taken twice logic scheme. The four additional water level transmitters and associated wiring changes would be required to meet the current standards for Category I, seismic qualification, and EEQ qualifications.

Since the proposed scenario is outside the Millstone Unit No. 1 design basis, and the ATWS system will ultimately cause a reactor scram signal to be generated, the impact on personnel safety is minimal. The required use of the additional level instrumentation logic would be extremely rare, therefore no savings in man-rem or personnel safety would be expected.

D. Personnel Productivity

The additional calibration, maintenance, administration, training, and surveillance required for a system designed for use only during degraded conditions (beyond the design basis) provides no benefit during plant operation. A quantitative analysis of the affect on personnel productivity cannot be achieved without the identification of specific components or equipment. However, based on other similar equipment, a nonconservative estimate of 160 man-hours per year would be required to maintain all aspects of this modification. Therefore, the impact on personnel productivity is estimated to be approximately four personnel work-weeks which equates to $-5,120/year.

III. Conclusion

This project received the lowest ARM value and relative ranking. This is a result of a negative personnel productivity score and two negligible attribute scores coupled with a moderate project cost. Accordingly, there are no modifications scheduled, and NNECO considers this topic closed, relative to the scope of the Generic Letter. However, based on insights gained from review of this issue, we are considering further study beyond the scope of the Generic Letter.

Topic 1.116--Remote Wide-Range Yarways

I. Introduction

The wide-range Yarway reactor vessel level transmitters and their associated cables were not designed and installed as environmentally qualified, and therefore do not meet the guidance of Regulatory Guide 1.97.

Since the existing transmitters do not accurately account for variations in temperature and pressure, the control room operators are required to convert indicated level using a group of curves to determine the actual vessel level in several sections of the EOPs. Both of the transmitters are powered by the 120-volt Vital AC bus. Following a loss of the 120-volt Vital AC bus, the control room operators could fail to diagnose the event, due to the presence of misleading indications, including the apparent rapid loss of vessel level. The reactor vessel could overfill, potentially causing damage to the main steam lines, the isolation condenser, or other equipment. This could preclude their use for decay heat removal.

This topic will evaluate the replacement of the two wide-range Yarway level transmitters (LITS-236-73A and B) and their associated cables with environmentally qualified components. The new transmitters would automatically compensate for changes in temperature and pressure, and provide correct indication of vessel level on new displays. Independent AC power sources would be provided for each of the channels.

II. Evaluation

A. Public Safety

The wide-range Yarway level transmitters provide permissives to open the drywell spray valves when core level is greater than two-thirds. Since these interlocks can be easily overridden by the operator, the significance of this function to public safety is minimal.

The major function of the wide-range Yarways is indication of vessel level. Several of the dominant core melt sequences include the cognitive operator error of failing to realize the need to stabilize or restore vessel level.

Since the existing transmitters are operable, their one-for-one replacement with environmentally qualified transmitters and cables would only provide a benefit for scenarios in which a LOCA in the reactor building disables the transmitters, and scenarios in which high radiation levels in the drywell disable the transmitters. (The transmitters and all exposed cables are outside the drywell.) Because in general, the Millstone Unit No. 1 PSS assumes core damage for most unisolated LOCAs outside the drywell, the replacement of the transmitters for strictly EQ reasons would not provide much benefit for the CMF.

The degree of automatic temperature/pressure compensation of the new transmitters has not yet been finalized, but it is assumed that the final design would simplify the determination of actual vessel level and preclude the need for some of the interpolation of graphs while using the EOPs. The human error probability (HEP) associated with determining vessel level could therefore be reduced. In addition, the change from both transmitters being powered by Vital AC to one powered by Vital AC and the other from an independent source would permit the reduction of the HEP associated with determining vessel level and taking appropriate action following a loss of the Vital AC bus.

Based on a CMF reduction of 1.7E-5/yr and a Man-Rem savings of approximately 516 Man-Rem over the remaining life of the plant, this project has a public safety impact of $30,020/year.

It should be noted that much of the benefit of this project is the same benefit that would be gained from implementing ISAP Topic 1.07.12, "CRDR: Feedwater Control System and Reactor Water Level Instrumentation," and ISAP Topic 1.43, "Loss of Vital AC."

B. Economic Performance

During normal plant operation the RPV level is at +30". During and following an accident, the operators' major concern is to ensure that the water level in the reactor is above the top of the active fuel (-127.5"). The trip set points are set at water levels of +48" and +8". At +48" the feedwater system and turbine are tripped and at +8" the reactor will trip.

During normal plant operation, adequate redundancy presently exists because the presence of a harsh environment is not a realistic expectation. At conditions where the RPV level either exceeds +48" or falls below +8", the plant will not be in a power producing mode, thus, there cannot be any impact on plant availability. Therefore, the potential loss of RPV level indication will affect only those scenarios in which water level is considerably above or below the normal value when the plant is off-line. Thus, no plant availability impact is expected.

Portions of this modification may be implemented during all six plant operational modes. The existing system will be maintained in service as required by plant Technical Specifications. Turnover of the replacement system will be accomplished so as to minimize impact upon the refueling outage schedule and to fully comply with the Technical Specification requirements associated with these channels. Therefore, the effect of this issue on plant availability would be negligible.

C. Personnel Safety

The proposed modification will not affect personnel safety.

D. Personnel Productivity

The proposed modification will not affect personnel productivity.

III. Conclusion

Overall, this project received a high ARM value and relative ranking. The substantial public safety benefit and the environmental qualification concern necessitate the timely replacement of the two wide-range Yarway level transmitters. Accordingly, this modification has been scheduled in the IIS to be completed during the 1991 refueling outage.

Topic 1.117--Safety Parameter Display System Upgrade

I. Introduction

Although the Safety Parameter Display System (SPDS) is not required to support normal operation, several changes are needed to bring the system into conformance with revision 4 of the Emergency Procedure Guidelines (EPGs) and the recently revised Emergency Operating Procedures (EOPs). This topic will evaluate the addition of secondary containment parameters and the improvement of several trending displays.

II. Evaluation

A. Public Safety

The greatest potential benefit of this project is considered to be the ability of the SPDS to trend various parameters and perform computations and interpolations for the operators. These functions could be of value during rapidly progressing scenarios, in which the operators have many changing parameters for which they need to remain cognizant.

As an estimate of the benefit of this project, a five percent reduction in the probability of error of the cognitive decisions that the operators would have to make in the 20 minutes following the initiation of a core melt sequences was imposed. The cognizant error probabilities in which the operator has more than 20 minutes to make decisions were not reduced.

Based on a CMF reduction of 1.2E-6/yr and a Man-Rem savings of approximately 37 Man-Rem over the remaining life of the plant, this project has a public safety impact of $2,155/year.

B. Economic Performance

The function of the SPDS is to provide the control room operators with a concise display of critical plant parameters during post-accident conditions. Because it is used primarily during post-accident operation there is no impact on normal plan operation. In addition, since it is expected that there will be no future Technical Specifications requirements on SPDS operability during normal plant operation there is also no link to plant availability. The implementation of the SPDS update has no direct relationship to the power generation process and, therefore, the effect of this issue on plant availability would be negligible.

C. Personnel Safety

The addition of new and modified SPDS displays necessary to support Revision 4 of the EPGs will have no affect on personnel safety.

D. Personnel Productivity

The SPDS computer points would have very little affect on personnel productivity since the displays are seldom used during normal plant operation. The SPDS displays are designed to be used during accident conditions to enhance the operators ability to implement the EOP's. Therefore, the SPDS displays and process computer points will not directly affect personnel productivity.

III. Conclusion

This project received a medium overall ranking due to the public safety benefit. This upgrade has not been scheduled in the IIS at this time. Information will be provided in a future ISAP/IIS update.

Topic 1.118--Plant Heating Steam System

This topic involves the engineering analysis and modifications necessary for restoration of heating steam system, which was isolated earlier this year when qualification issues arose. Modifications include the installation of rerouted heating steam supply piping and valves external to the HVAC room; the replacement of steam coils with electric coils; the removal of the 4" steam supply piping and valves from the switchgear room to the turbine deck; and the establishment of heating steam supply to the reactor building external to the HVAC room.

These modifications were not identified in time for inclusion in this cycle of ISAP ARM review. However, in recognition of Environmental Qualification (EQ), High Energy Line Break (HELB) and Pipe Whip concerns, this project requires immediate action. As such, these modifications are currently being implemented in a phase approach with all modifications scheduled for completion during the 1991 (Cycle 13) refueling outage.

Topic 1.119--Resolution of Unresolved Safety Issue A-46

This topic encompasses NNECO's plant-specific response to USI A-46, "Seismic Qualification of Equipment in Operating Plants." NNECO has addressed this issue in conjunction with the Seismic Qualification Utility Group (SQUG) and reported details in the previous ISAP reports.

At this time, no activities related to this topic are scheduled in the IIS. When the GIP received final approval from the NRC Staff and all issues are resolved, NNECO will define a schedule for implementation and will include it in a future IIS. This position was discussed in NNECO's letter dated January 22, 1990.[7]

Topic 2.06--Condenser Retube

I. Introduction

This topic was discussed in our previous ISAP submittals. Information obtained during the 1989 refueling outage showed some severe inlet waterbox end erosion. Modifications to repair the inlet ends have been evaluated and a representative sample of the eroded tubes were successfully sleeved during March 1990. Plans are to sleeve 16,000-18,000 tubes during the upcoming 1991 refueling outage. Six to nine-inch sleeves will be inserted into the inlet end of the tubes to repair existing erosion. This is expected to improve condenser performance for the next cycle.

This topic addresses a one-for-one replacement of the existing 70/30 copper-nickel condenser tubes with new tubes of an undetermined material. Condenser tube leakage introduces impurities into the feedwater system which, in turn, cause corrosion problems in the reactor coolant system, turbine blades, and rotors. To prevent the corrosion problem and maintain feedwater purity, Millstone Unit No. 1 takes prompt action to repair any condenser leaks consistent with ALARA considerations and detection capabilities. This frequent attention has heavily impacted plant availability through plant deratings. During the period of January 1976 to June 1990, Millstone Unit No. 1 experienced an estimated total loss of 450,648 MWh due to condenser tube leakage and other maintenance. This loss equates to 689 hours of full power operation and amounts to over $9 million using 1990 replacement power costs.

II. Evaluation

A. Public Safety

The public safety impact of this proposed project was evaluated by calculating the increase in the CMF that would be applicable if the condenser modifications are not implemented, and assuming that this increase would be avoided. It is assumed that the degraded condenser would impact the CMF in two ways: Chloride intrusion into the feedwater system would necessitate a manual reactor scram if conductivity levels were too high; Additional plugging of faulty condenser tubes would most likely impact the full load reject capability following a loss of electric load, since the ability to maintain condenser vacuum would be decreased.

Based on a CMF reduction of 1.18E-5/yr and a Man-Rem savings of approximately 579 Man-Rem over the remaining life of the plant, this project was assigned a score of $31,900/year.

B. Economic Performance

The project plan is to replace the present 70/30 copper nickel tubes with a yet unspecified tube material. Titanium is one material being considered due to its advantages, such as its ability for erosion resistance and its ability to be cleaned easily with abrasive material. In contrast, copper nickel is susceptible to both erosion and corrosion which has been found to be a major cause of condenser tube failures. The disadvantage of the titanium is that it has a lower heat transfer capability than copper nickel and it is more susceptible to microorganism fouling. The present capability of the condenser allows for up to 12 percent tube failures (plugging) before any reduction in cycle efficiency can be noticed. Analysis shows that No. 22 BWG titanium tubes produce the same condensing capability as that of the present copper nickel condenser when 10 percent of its tubes are plugged. Therefore, to get the same condenser capability as originally designed, one has to either change the tube sheets to allow for an increase in the number of titanium tubes or increase the overall heat transfer coefficient of the condensers by reducing the allowable fouling, using thinner tube walls and changing tube pattern.

Condenser retube modifications will certainly reduce the present power losses due to tube leakage for some years to come. The length of time before noticing any condenser inefficiency or tube failure is greatly dependent on the design implemented and the performance of the required maintenance applicable to the new condenser tube material.

If the condenser tubes are not replaced, it is expected that plant operation would be continually impacted to an increasing degree. Short-term repairs would probably keep the condenser in operation, however, the frequency and duration of downpowers and outages would increase with time. Because it is impossible to ascertain the future effect of short-term repairs, it will be assumed that if the condenser tubes are not replaced, the condenser related plant production impact will be equal to that of the last 14.5 years. This was estimated to be a 0.69 percent equivalent capacity factor loss.

Accordingly, with this capacity factor loss and assuming replacement of the condenser tubes have an effectiveness of approximately 90 percent, the condenser tube replacement of 70/30 copper nickel tubes will eliminate condenser tube failures, and therefore, increase the equivalent availability by 42.4 hours of full power operation per year. This benefit is based on the assumption that all the necessary design modifications will be implemented in order to prevent tube vibration and galvanic corrosion.

C. <u>Personnel Safety</u>

Condenser modification will result in a significant reduction in the frequency of condenser tube leak identification. The reduction in leak detection activities could result in the decrease in frequency of a serious accident by 300 person-hours per year and a potential 5 man-rem per year savings in radiation exposure.

The condenser tube leaks result in an abnormal depletion of condensate demineralizer resin due to the inleakage of salt water. The net affect of the tube leakage is an increase in resin replacement and resin removal. Although there is a potential for an increase in personnel injury during these evolutions, the seriousness of the types of injuries are minor. The radiological savings for a reduction in resin handling equate to approximately 5 man-rem per year. Therefore, the proposed project has a personnel safety benefit of $70,000/year.

D. Personnel Productivity

Replacement of the condenser tubes, repair of the water boxes, and upgrading of the cathodic protection system will greatly enhance personnel productivity at Millstone Unit No. 1. An estimated savings for maintenance activities, surveillance activities, and corrective maintenance associated with activities directly attributable to condenser tube leaks could result in an estimated savings of 2500 person-hours per year, which equates to a personnel productivity benefit of $80,000/year.

III. Conclusion

This project received a moderate ARM value and relative ranking based mainly on the remaining project cost of approximately $25 million. A portion of the tubes are scheduled for sleeving during the 1991 (Cycle 13) refueling outage. Retubing is tentatively scheduled for the 1993 refueling outage in the IIS.

Topic 2.121--Feedwater Venturi Replacement

I. Introduction

This topic involves the evaluation of the design modifications necessary to replace the two existing feedwater flow nozzles with two new venturi flow elements. The primary function of the existing feedwater flow nozzles is to measure the flow in the main feedwater lines going into the reactor vessel. This flow signal is also utilized for other indication and process inputs.

The feedwater flow nozzles require the assignment of an adjustment factor to compensate for degraded performance. The degraded performance inhibits core power operation within the licensed core power limits. Presently, the adjustment factors are conservative, thereby limiting core power operation to values lower than the licensed limit.

Initially, only one of two feedwater flow elements required the adjustment factor but, the second feedwater flow element recently experienced performance degradation and now requires an adjustment factor.

Although this approach has permitted the unit to operate near its licensed power limits, it could generate an additional 2.0 to 3.0 MWe if the conservatism in the feedwater flow measurement is eliminated.

II. Evaluation

A. Public Safety

Degradation in the performance of the feedwater flow nozzles has necessitated the application of conservative adjustment factors to ensure core power is maintained within licensed limits. The proposed modification will allow the plant to operate closer to the thermal limits, but will not have any significant measurable benefits to the core melt frequency or public safety.

Therefore, the proposed modifications would not provide a significant improvement in public risk. As such, this project was assigned a public risk score of $0/year.

B. Economic Performance

The implementation of this proposed modification will have zero impact on plant reliability and maintainability. However, the impact on plant capacity and equivalent availability is estimated as 2.5 MWe per hour and 24.11 hours per year, respectively. The economic impact due to an increase in capacity is $320,090/year.

C. Personnel Safety

Since the replacement flow venturis are almost an exact replacement item (i.e. no change in location or orientation), no changes in personnel safety are affected.

D. Personnel Productivity

The replacement feedwater venturi's will require no new additional component calibration, maintenance, or surveillances. The instrumentation is used during normal operation to monitor feedwater flow rates required for reactor power measurements. Therefore, the overall value of this project on a personnel productivity basis will be a negligible positive change.

III. Conclusion

This project received a high ARM valve and relative ranking based on its large impact on plant capacity and equivalent availability. Accordingly, it is scheduled in the IIS to be implemented in the 1991 (Cycle 13) outage.

Topic 2.122--Reserve Station Service Transformer Transfer Trip Scheme Replacement

This new topic encompasses the engineering, procurement and installation of a replacement transfer trip scheme for the existing out-dated Millstone Unit No. 1 RSST. The existing system is approximately twenty-five (25) years old, and has on occasion, been unreliable. Furthermore, replacement parts are unavailable as this equipment is no longer manufactured. Used parts were installed in 1988 when the transmitters and receivers at the plant and switchyard required replacement.

This project was initiated well into this ISAP ARM review cycle. As a result, an evaluation has not been performed. However, based on reliability and performance concerns, this project requires immediate attention and has therefore been scheduled for implementation during the 1991 refueling outage.

Topic 2.123--Chemistry Laboratory HVAC System

This new topic involves engineering evaluation and potential modifications to install an HVAC system in the Millstone Unit No. 1 Chemistry Laboratory.

A project assignment has been initiated to evaluate design and equipment requirements. Once the project assignment detailed review is complete, any forthcoming potential modifications will be reviewed within the ISAP process. This evaluation and associated information will be submitted in a future ISAP/IIS update report.

Topic 2.124--Replacement of IC System Valves and CU-28

This new topic involves engineering evaluation and potential modifications for the replacement of five motor operated valves. The use of parallel disc gate valves or smaller valves, which will allow for the reuse of existing power cables, is being considered. Installation of a new valve downstream of 1-IC-4 may allow 1-IC-4 to remain in its current location.

A project assignment has been initiated to evaluate design criteria and regulatory requirements. Once the project assignment detailed review is complete, any forthcoming potential modifications will be reviewed within the ISAP process. This evaluation and associated information will be submitted in a future ISAP/IIS update report.

<u>References</u>

(1) M. L. Boyle letter to E. J. Mroczka, "Postaccident Neutron Flux Monitoring for BWRs - (ISAP Topic 1.09) Millstone Nuclear Power Station, Unit No. 1," dated February 13, 1990.

(2) E. J. Mroczka letter to U.S. Nuclear Regulatory Commission, "Response to Generic Letter 89-19--Request for Action Related to Resolution of Unresolved Safety Issue A-47," dated March 27, 1990.

(3) T. E. Murley letter to E. J. Mroczka, "Staff's Backfit Analyses for Millstone Unit No. 1 Regarding Installation of a Hardened Wetwell Vent (Generic Letter 89-16) (TAC No. 74872)," dated June 15, 1990.

(4) E. J. Mroczka letter to U.S. Nuclear Regulatory Commission, "Response to Staff's Backfit Analysis--Hardened Wetwell Vent," dated August 8, 1990.

(5) J. G. Partlow letter to E. J. Mroczka, "Requested Delay of Decision to Install Hardened Wetwell Vent for Millstone Nuclear Power Station, Unit No. 1 (TAC No. 74872)," dated August 20, 1990.

(6) E. J. Mroczka letter to U.S. Nuclear Regulatory Commission, "Hardened Wetwell Vent," dated September 4, 1990.

(7) E. J. Mroczka letter to U.S. Nuclear Regulatory Commission, "Haddam Neck Plant, Millstone Unit Nos. 1 and 2, Resolution of Unresolved Safety Issue A-46," dated January 22, 1990.

Docket No. 50-245
B13651

Attachment 4

Millstone Nuclear Power Station, Unit No. 1
Integrated Safety Assessment Program

Integrated Implementation Schedule

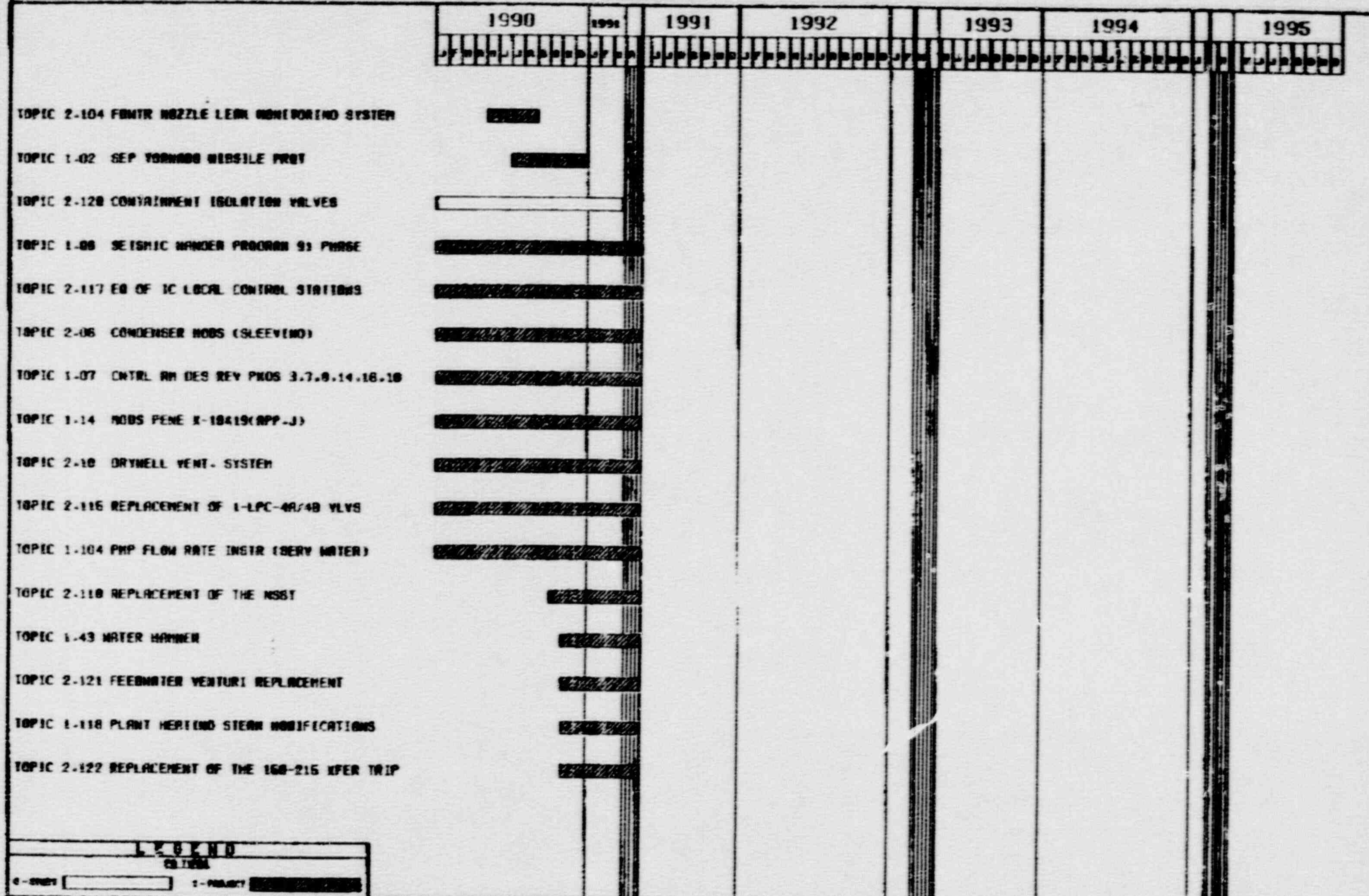
November 1990

NORTHEAST NUCLEAR ENERGY COMPANY

MILLSTONE UNIT NO. 1

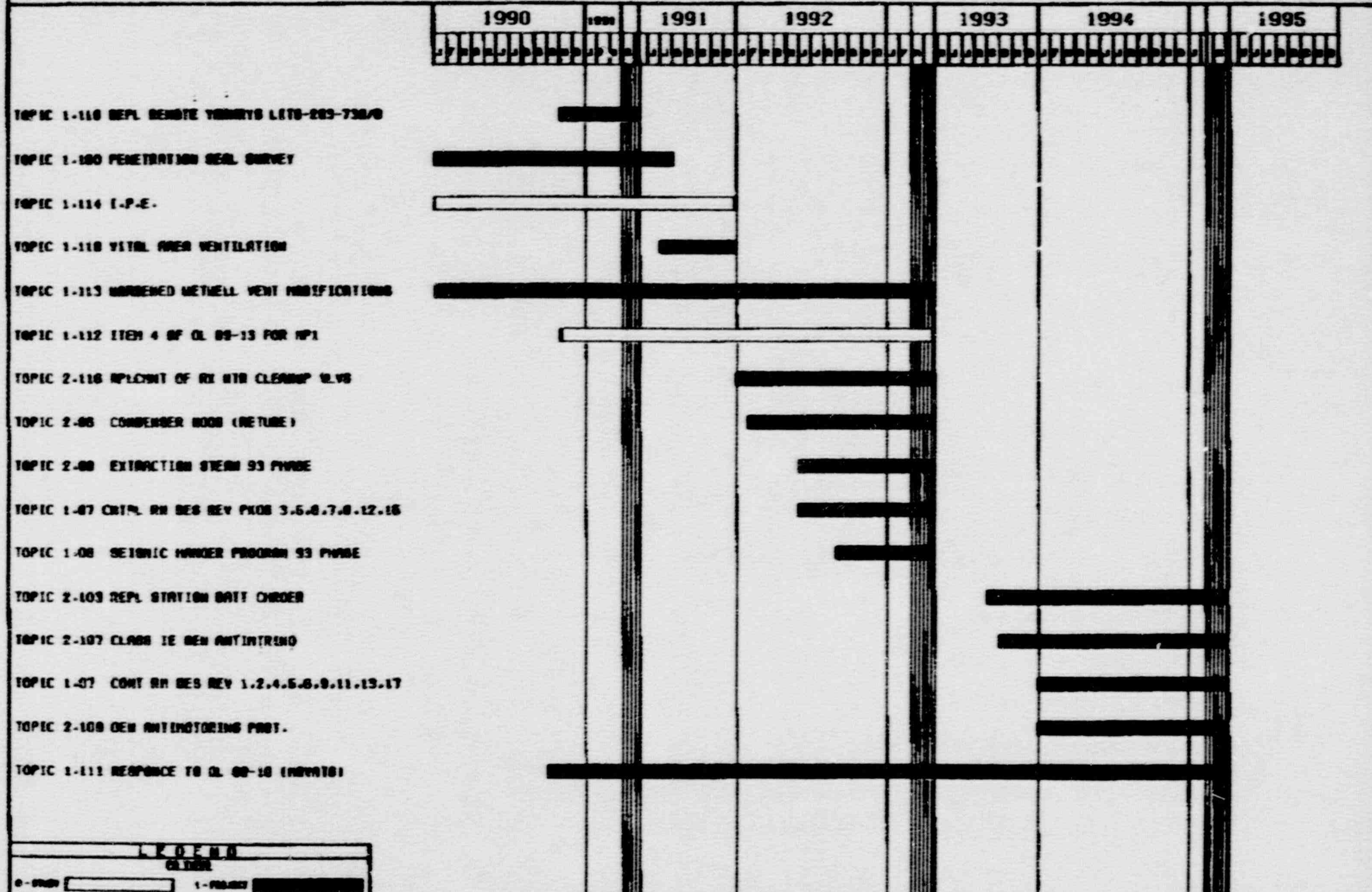
INTEGRATED IMPLEMENTATION SCHEDULE

SORT BY EARLY FINNISH



NORTHEAST NUCLEAR ENERGY COMPANY
MILLSTONE UNIT NO. 1
INTEGRATED IMPLEMENTATION SCHEDULE

SORT BY EARLY FINISH



Attachment 5

Millstone Nuclear Power Station, Unit No. 1
Integrated Safety Assessment Program

Summary Table of ISAP ARM Scores

November 1990

11/30/90

MILLSTONE UNIT NO. 1 ARM RANKINGS

IAP #	Title	PA #	Overall Rank	Public Safety	Economic Performance	Personnel Safety	Personnel Productivity	Remaining Project Cost	Total Value	Rank Value	Public Man-Rem	Occ O/A Man-Rem	Occ Inst Man-Rem	Net Man-Rem
1.14	APPENDIX J MODS-SUMP DRAIN VALVES	88-013	1	1011	200000	-103	0	90,000	141287.49	156.99	20.4	-2	-1	17.4
1.43	WATER HAMMER	None	2	10950	27880	0	0	80,000	34955.50	43.69	188	0	0	188
2.108	GEN. ANTIMO. PRO. WHILE SHUT DOWN	85-120	3	0	103546	0	0	168,000	72482.20	43.14	0	0	0	0
2.107	CLASS 1E GEN. ANTIMOTORING PROT	85-019	4	0	29897	0	0	50,000	20927.90	41.86	0	0	0	0
2.08	EXTRACTION STEAM PIPING REPLACEMENT	83-078	5	2977	1299000	460	1400	2,416,000	914883.97	37.87	55.5	15	-15	55.5
2.118	REPLACEMENT OF THE NSST	89-048	6	7000	258086	0	0	544000	190530.20	35.02	0	0	0	0
1.07.16	CONTROL ROOM DESIGN REVIEW (pkg 16)	81-047	7	6100	700	0	2400	40,000	10411.00	26.03	116	0	0	116
1.116	REMOTE WIDE-RANGE YARWAYS	90-074	8	30020	0	0	0	250,000	42328.20	16.93	516	0	-3	513
2.115	REPLACEMENT OF 1-LPC-4A/4B VALVES	89-034	9	0	111968	0	0	613,000	78377.60	12.79	0	-1	-10	-11
2.121	FEEDWATER VENTURI REPLACEMENT	90-012	10	0	320090	0	0	1,800,000	224063.00	12.45	0	0	-5	-5
1.02	WIND AND TORNADO LOADINGS/MISSILES	83-160	11	21659	0	-6	-1400	298000	29761.15	9.99	414	0	0	414
1.07.18	CONTROL ROOM DESIGN REVIEW (pkg 18)	81-047	12	0	0	0	1200	10,000	660.00	6.60	0	0	0	0
1.07.17	CONTROL ROOM DESIGN REVIEW (pkg 17)	81-047	13	0	700	0	0	10,000	490.00	4.90	0	0	0	0
1.07.12	CONTROL ROOM DESIGN REVIEW (pkg 12)	81-047	14	2250	700	0	0	100,000	3662.50	3.66	40.4	0	0	40.4
1.07.08	CONTROL ROOM DESIGN REVIEW (pkg 8)	81-047	15	0	700	0	2400	50,000	1810.00	3.62	0	0	0	0
1.07.03	CONTROL ROOM DESIGN REVIEW (pkg 3)	81-047	16	850	700	0	0	50,000	1688.50	3.38	16.4	0	0	16.4
1.07.07	CONTROL ROOM DESIGN REVIEW (pkg 7)	81-047	17	0	0	0	2400	50,000	1320.00	2.64	0	0	0	0
2.06	CONDENSER RETUBE	81-105	18	31900	562902	70000	80000	25,000,000	576810.40	2.31	579	10	-5	584
2.117	EQ OF IC LOCAL CONTROL STATIONS	89-042	19	1300	0	0	0	83,000	1833.00	2.21	25	0	-1	24
2.103	MUSOER 1-81-REPLACE BATT CHARGERS	88-029	20	7600	0	0	0	587,000	10716.00	1.83	149	0	0	149
2.116	RPLCMNT OF Rx WATER CLEANUP VALVES	89-035	21	2150	0	8000	2500	1,151,000	1512[illegible].50	1.31	45.3	160	-50	155.3
1.117	SAFETY PARAMETER DISPLAY SYSTEM	89-079	22	2155	0	0	0	254,000	30[illegible].55	1.20	37	0	0	37
1.100	PENETRATION SEAL PROGRAM UPGRADE	83-029	23	9091	0	118	2800	1,390,000	14516.43	1.04	200	12	-10	202
1.07.15	CONTROL ROOM DESIGN REVIEW (pkg 15)	81-047	24	0	700	0	3600	250,000	2470.00	0.99	0	0	0	0
1.07.14	CONTROL ROOM DESIGN REVIEW (pkg 14)	81-047	25	0	0	0	2400	180,000	1320.00	0.73	0	0	0	0
1.07.09	CONTROL ROOM DESIGN REVIEW (pkg 9)	81-047	26	2750	0	0	-3600	1,000,000	1897.50	0.19	53.6	-2	-40	11.6
1.07.06	CONTROL ROOM DESIGN REVIEW (pkg 6)	81-047	27	330	700	0	1200	1,000,000	1615.30	0.16	6.7	0	0	6.7
1.07.02	CONTROL ROOM DESIGN REVIEW (pkg 2)	81-047	28	100	700	0	1200	1,200,000	1291.00	0.11	2	0	0	2
2.10	DRYWELL VENTILATION SYSTEM	83-017	29	1250	0	0	-1400	1,008,000	992.50	0.10	1	0	0	1
1.113	HARDENED WETWELL VENT, GL 89-16	None	30	2000	0	0	-3200	1,500,000	1060.00	0.07	42	0	-1	41
1.07.05	CONTROL ROOM DESIGN REVIEW (pkg 5)	81-047	31	125	0	0	0	480,000	176.25	0.04	2.2	0	0	2.2
1.09.3	RG 1.97- NEUTRON MONITORING	90-003	32	150	0	0	0	700,000	211.50	0.03	0	0	0	0
1.110	VITAL AREA VENTILATION	88-024	33	0	0	0	0	162,000	0.00	0.00	0	0	0	0
1.06.01	SEISMIC QUAL. OF SAFETY-REL. PPG 1991	79-162	34	0	0	0	0	4,006,000	0.00	0.00	0	0	-2	-2
1.06.02	SEISMIC QUAL. OF SAFETY-REL. PPG 1993	79-162	35	0	0	0	0	513,000	0.00	0.00	0	0	-4	-4
1.06.03	SEISMIC QUAL. OF SAFETY-REL. PPG 1995	79-162	36	0	0	0	0	100	0.00	0.00	0	0	-4	-4
1.07.01	CONTROL ROOM DESIGN REVIEW (pkg 1)	81-047	37	0	0	0	0	22,000	0.00	0.00	0	0	0	0
1.07.04	CONTROL ROOM DESIGN REVIEW (pkg 4)	81-047	38	0	0	0	0	50,000	0.00	0.00	0	0	0	0
1.07.13	CONTROL ROOM DESIGN REVIEW (pkg 13)	81-047	39	0	0	0	0	14,000	0.00	0.00	0	0	0	0
1.107	DRYWELL SPRAY INSTRUMENTATION	None	40	0	0	0	0	49,999	0.00	0.00	0	-2	-10	-12
1.102	FIRE SUPPRESSION SYSTEMS-NFPA COMPL.	87-073	41	0	0	0	0	1,366,000	0.00	0.00	0	0	-4	-4
1.101	FIRE DETECTION SYSTEMS-NFPA COMPL.	87-072	42	0	0	-2000	-1400	1,401,000	-3450.00	-0.25	0	-2	-10	-12
1.104.1	PUMP FLOW RATE INSTR.-SERVICE WATER	88-007	43	0	0	-1800	-2560	386,000	-3820.00	-0.99	0	0	0	0
1.07.11	CONTROL ROOM DESIGN REVIEW (pkg 11)	81-047	44	270	0	0	-1200	25,000	-279.30	-1.12	5.2	0	0	5.2
1.115	Rx WATER LEVEL REF. LEG BREAK; GL 89-11	None	45	50	0	0	-5120	100,000	-2745.50	-2.75	0.85	0	-3	-2.15