COOPER NUCLEAR STATION
TABLE 3.1.1 (Page 2)
REACTOR PROTECTION SYSTEM INSTRUMENTATION REQUIREMENTS

Reactor Protection System Trip Function	Applicability Conditions Mode Switch Position: Shutdown	Startup	Refuel	Run	Trip Level Setting	Minimum Number of Operable Channels Per Trip Systems (1)	Action Required When Equipment Operability is Not Assured (1)
Main Steam Line Isolation Valve Closure MS-LMS-86 A,B,C, & D MS-LMS-80 A,B,C, & D		X(6)(9)		X(6)	≤ 10% of valve closure	4 4	A or C A or C
Turbine Control Valve Fast Closure TGF-63/OPC-1,2,3,4				X(4)	≥ 1000 psig turbine control fluid	2	A or B
Turbine Stop Valve Closure SVOS-1(1), SVOS-1(2) SVOS-2(1), SVOS-2(2)				X(4)	≤ 10% of valve Closure	2	A or B
Turbine First Stage Permissive MS-PS-14 A,B,C, & D		X(9)		X	≤ 30% first stage press.	2	A or B

NOTES FOR TABLE 3.1.1

1. There shall be two operable or tripped trip systems for each function. If the minimum number of operable instrument channels for a trip system cannot be met, the affected trip system shall be placed in the safe (tripped) condition, or the appropriate actions listed below shall be taken.

   A. Initiate insertion of operable rods and complete insertion of all operable rods within four hours.

   B. Reduce power to less than 30% of rated.

   C. Reduce power level to IRM range and place mode switch in the Startup position within 8 hours and depressurize to less than 1000 psig.

2. Permissible to bypass, with control rod block, for reactor protection system reset in refuel and shutdown positions of the reactor mode switch.

3. This note deleted.

4. Permissible to bypass when turbine first stage pressure is less than 30% of full load.

5. IRM's are bypassed when APRM's are onscale and the reactor mode switch is in the run position.

6. The design permits closure of any two lines without a full scram being initiated.

7. When the reactor is subcritical, fuel is in the vessel, and the reactor water temperature is less than 212°F, only the following trip functions need to be operable:

   a. Mode switch in shutdown.
   b. Manual scram.
   c. IRM high flux. 120/125 indicated scale.
   d. APRM (15%) high flux scram.

8. Not required to be operable when primary containment integrity is not required.

9. Not required while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MW(t).

10. Not required to be operable when the reactor pressure vessel head is not bolted to the vessel.

COOPER NUCLEAR STATION
TABLE 4.1.1 (Page 2)
REACTOR PROTECTION SYSTEM (SCRAM INSTRUMENTATION) FUNCTIONAL TESTS
MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTR. AND CONTROL CIRCUITS

Instrument Channel	Group (2)	Functional Test	Minimum Frequency (3)
High Water Level in Scram Discharge Volume CRD-LS-231 A & B CRD-LS-234 A & B CRD-LT-231 C & D CRD-LT-234 C & D	A	Trip Channel and Alarm	Once/3 Months
Main Steam Line Isolation Valve Closure MS-LMS-86 A,B,C, & D MS-LMS-80 A,B,C, & D	A	Trip Channel and Alarm	Once/Month (1)
Turbine Control Valve Fast Closure TCS-63/OPC -1,2,3,4	A	Trip Channel and Alarm	Once/Month (1)
Turbine First Stage Pressure Permissive MS-PS-14 A,B,C, & D	A	Trip Channel and Alarm	Once/3 Months
Turbine Stop Valve Closure SVOS-1 (1), SVOS-1 (2) SVOS-2 (1), SVOS-2 (2)	A	Trip Channel and Alarm	Once/Month (1)

NOTES FOR TABLE 4.1.1

1. Initially once per month until exposure (M as defined on Figure 4.1.1) is 2.0×10^5; thereafter, according to Figure 4.1.1 with an interval not less than one month nor more than three months after review and approval of the NRC. The compilation of instrument failure rate data may include data obtained from other boiling water reactors for which the same design instrument operates in an environment similar to that of CNS.

2. A description of the three groups is included in the Bases of this Specification.

3. Functional tests are not required when the systems are not required to be operable or are tripped. If reactor startups occur more frequently than once per week, the maximum functional test frequency need not exceed once per week.

   If tests are missed, they shall be performed prior to returning the systems to an operable status.

4. Deleted.

5. Test RPS channel after maintenance.

6. The water level in the reactor vessel will be perturbed and the corresponding level indicator changes will be monitored. This perturbation test will be performed every month after completion of the monthly functional test program.

COOPER NUCLEAR STATION

TABLE 4.1.2

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT CALIBRATION

MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Instrument Channel	Group (1)	Calibration Test (5)	Minimum Frequency (2)
IRM High Flux	C	Comparison to APRM on Controlled Shutdowns (7)	Note (4)
APRM High Flux			
Output Signal	B	Heat Balance	Once/Week
Flow Bias Signal	B	Internal Power and Flow Test with Standard Pressure Source (8)	Once/Refueling Outage
LPRM Signal	B	TIP System Traverse	Note (9)
High Reactor Pressure	A	Standard Pressure Source	Once/3 Months
High Drywell Pressure	A	Standard Pressure Source	Once/3 Months
Reactor Low Water Level	A	Pressure Standard	Once/3 Months
High Water Level in Scram Discharge Volume	A	Note (6)	Note (6)
Main Steam Line Isolation Valve Closure	A	Note (6)	Note (6)
Turbine First Stage Pressure Permissive	A	Standard Pressure Source	Once/6 Months
Turbine Control Valve Fast Closure	A	Standard Pressure Source	Once/3 Months
Turbine Stop Valve Closure	A	Note (6)	Note (6)

NOTES FOR TABLES 4.1.2

1. A description of three groups is included in the bases of this Specification.

2. Calibration tests are not required when the systems are not required to be operable or are tripped but are required prior to return to service.

3. Deleted.

4. Maximum frequency required is once per week.

5. Response time is not a part of the routine instrument channel test, but will be checked once per operating cycle. The response time measurement will be the time segment from the time the sensor contacts actuate to the time the scram solenoid valves deenergize.

6. Physical inspection and actuation of these position switches will be performed during the refueling outages.

7. On controlled shutdowns, the IRM reading 120/125 of full scale will be set equal to or less than 45% of rated Power. All range scales above that scale on which the most recent IRM calibration was performed will be mechanically blocked.

8. The Flow Bias Scram Calibration will consist of calibrating the sensors, flow converters and signal offset networks during operation. The instrumentation is an analog type with redundant flow signals that can be compared. The flow bias trip and upscale will be functionally tested according to table 4.1.1 to assure proper operation during the operating cycle. Refer to Bases of 4.1 for further explanation of calibration frequencies.

9. LPRM detectors shall be calibrated every six weeks of reactor power operation above 20% of rated power.

3.1 BASES (Cont'd.)

initiate the core standby cooling equipment. A high drywell pressure scram is provided at the same setting as the core standby cooling systems (CSCS) initiation to minimize the energy which must be accommodated during a loss of coolant accident and to prevent return to criticality. This instrumentation is a backup to the reactor vessel water level instrumentation.

A reactor mode switch is provided which actuates or bypasses the various scram functions appropriate to the particular plant operating status. Ref. paragraph VII.2.3.7 FSAR.

The manual scram function is active in all modes, thus providing for a manual means of rapidly inserting control rods during all modes of reactor operation.

The APRM (High flux in Start-Up or Refuel) system provides protection against excessive power levels and short reactor periods in the start-up and intermediate power ranges.

The IRM system provides protection

4.1 BASES (cont'd.)

2. The factor M is the exposure hours and is equal to the number of sensors in a group, n, times the elapsed time T (M = nT).

3. The accumulated number of unsafe failures is plotted as an ordinate against M as an abscissa on Figure 4.1.1.

4. After a trend is established, the appropriate monthly test interval to satisfy the goal will be the test interval to the left of the plotted points.

5. A test interval of 1 month will be used initially until a trend is established, which is based on system availability analysis and good engineering judgement plus operating experience.

Group (B) devices utilize an analog sensor followed by an amplifier and a bi-stable trip circuit. The sensor and amplifier are active components and a failure is almost always accompanied by an alarm and an indication of the source of trouble. In the event of failure, repair or substitution can start immediately. An "as-is" failure is one that "sticks" mid-scale and is not capable of going either up or down in response to an out-of-limits input. This type of failure for analog devices is a rare occurrence and is detectable by an operator who observes that one signal does not track the other three. For purpose of analysis, it is assumed that this rare failure will be detected within two hours.

The bi-stable trip circuit which is a part of the Group (B) devices can sustain unsafe failures which are

(6) Reliability of Engineered Safety Features as a Function of Testing Frequency, I.M. Jacobs, "Nuclear Safety", Vol. 9, No. 4, July-Aug. 1968, pp. 310-312.

COOPER NUCLEAR STATION
TABLE 3.2.A (Page 1)
PRIMARY CONTAINMENT AND REACTOR VESSEL ISOLATION INSTRUMENTATION

Instrument	Instrument I.D. No.	Setting Limit	Minimum Number of Operable Components Per Trip System (1)	Action Required When Component Operability is Not Assured (2)
Main Steam Line High Radiation	RMP-RM-251, A,B,C,&D	≤ 3 Times Full Power	2	E
Reactor Low Water Level	NBI-LIS-101, A,B,C,&D #1	≥+4.5 in. Indicated Level	2(4)	A or B
Reactor Low Low Low Water Level	NBI-LIS-57 A & B #1 NBI-LIS-58 A & B #1	≥-145.5 in. Indicated Level	2	A or B
Main Steam Line Leak Detection	MS-TS-121, A,B,C,&D 122, 123, 124, 143, 144, 145, 146, 147, 148, 149, [illegible]	≤ 200°F	2(6)	B
Main Steam Line High Flow	MS-dPIS-116 A,B,C,&D 117, 118, 119	≤ 150% of Rated Steam Flow	2(3)	B
Main Steam Line Low Pressure	MS-PS-134, A,B,C,&D	≥ 825 psig	2(5)	B
High Drywell Pressure	PC-PS-12, A,B,C,&D	≤ 2 psig	2(4)	A or B
High Reactor Pressure	RR-PS-128 A & B	≤ 75 psig	1	D
Main Condenser Low Vacuum	MS PS-103, A,B,C,&D	≥ 7" Hg (7)	2	A or B
Reactor Water Cleanup System High Flow	RWCU-dPIS-170 A & B	≤ 200% of System Flow	1	C

NOTES FOR TABLE 3.2.A

1. Whenever Primary Containment integrity is required there shall be two operable or tripped trip systems for each function.

2. If the minimum number of operable instrument channels per trip system requirement cannot be met by a trip system, that trip system shall be tripped. If the requirements cannot be met by both trip systems, the appropriate action listed below shall be taken.

   A. Initiate an orderly shutdown and have the reactor in a cold shutdown condition in 24 hours.

   B. Initiate an orderly load reduction and have the Main Steam Isolation Valves shut within 8 hours.

   C. Isolate the Reactor Water Cleanup System.

   D. Isolate the Shutdown Cooling mode of the RHR System.

   E. Isolate the Reactor Water Sample Valves.

3. Two required for each steam line.

4. These signals also start the Standby Gas Treatment System and initiate Secondary Containment isolation.

5. Not required in the refuel, shutdown, and startup/hot standby modes (interlocked with the mode switch).

6. Requires one channel from each physical location for each trip system.

7. Low vacuum isolation is bypassed when the turbine stop is not full open, manual bypass switches are in bypass and mode switch is not in RUN.

8. The instruments on this table produce primary containment and system isolations. The following listing groups the system signals and the system isolated.

   <u>Group 1</u>

   Isolation Signals:

   1. Reactor Low Low Low Water Level (≥-145.5 in.)
   2. Main Steam Line Low Pressure (≥825 psig in the RUN mode)
   3. Main Steam Line Leak Detection (≤200°F)
   4. Condenser Low Vacuum (≥7" Hg vacuum)
   5. Main Steam Line High Flow (≤150% of rated flow)

   Isolations:

   1. MSIV's
   2. Main Steam Line Drains

NOTES FOR TABLE 3.2.D

1. Action required when component operability is not assured.

   A. (1) If radiation level exceeds 1.0 ci/sec (prior to 30 min. delay line) for a period greater than 15 consecutive minutes, the off-gas isolation valve shall close and reactor shutdown shall be initiated immediately and the reactor placed in a cold shutdown condition within 24 hours.

   A. (2) Refer to Specification 3.21.A.2.

   B. A minimum of one instrument channel per trip system shall be operable when handling irradiated fuel inside secondary containment, and when moving loads inside secondary containment which have the potential to damage irradiated fuel. If this requirement cannot be met by a trip system, then that trip system shall be tripped. If this requirement cannot be met by both trip systems, then the following actions shall be taken:

   (1) Cease handling of irradiated fuel inside secondary containment and remove the load from over the irradiated fuel via the most direct path, or

   (2) Isolate secondary containment and start SBGT.

   C. During release of radioactive wastes, the effluent control monitor shall be set to alarm and automatically close the waste discharge valve prior to exceeding the limits of Specification 3.21.B.1.

   D. Refer to Section entitled "Additional Safety Related Plant Capabilities".

   E. Refer to Section 3.2.D.5 and the requirements for Primary Containment Isolation on high main steam line radiation, Table 3.2.A.

2. Trip settings to correspond to Specification 3.21.B.1.

3. Trip settings to correspond to Specification 3.21.C.6.a.

4. Minimum number of channels shall be one during mechanical vacuum pump operation.

COOPER NUCLEAR STATION
TABLE 4.2.A (Page 1)
PRIMARY CONTAINMENT AND REACTOR VESSEL ISOLATION SYSTEM
TEST AND CALIBRATION FREQUENCIES

Item	Item I.D. No.	Function Test Freq.	Calibration Freq.	Instrument Check
[illegible]				
Reactor Low Water Level	NBI-LIS-101, A,B,C,&D	Once/Month (1)	Once/3 Months	Once/Day
Reactor Low Low Water Level	NBI-LIS-57, A & B #2 NBI-LIS-58, A & B #2	Once/Month (1)	Once/3 Months	Once/Day
Reactor Low Low Low Water Level	NBI-LIS-57, A & B #1 NBI-LIS-58, A & B #1	Once/Month (1)	Once/3 Months	Once/Day
Main Steam Line High Radiation	RMP-RM-251, A,B,C,&D	Once/Month (1) (13)	Once/3 Months (1[illegible])	Once/Day
Main Steam Line Leak Detection	MS-TS-121, A,B,C,&D 122, 123, 124, 143, 144, 145, 146, 147, 148, 149, 150	Once/Month (1)	Once/Operating Cycle	None
Main Steam Line High Flow	MS-dPIS-116, A,B,C,&D 117 118 119	Once/Month (1) Once/Month (1) Once/Month (1) Once/Month (1)	Once/3 Months Once/3 Months Once/3 Months Once/3 Months	None None None None
Main Steam Line Low Press.	MS-PS-134, A,B,C,&D	Once/Month (1)	Once/3 Months	None
High Reactor Pressure	RR-PS-128, A & B	Once/Month (1)	Once/3 Months	None
Condenser Low Vacuum	MS-PS-103, A,B,C,&D	Once/Month (1)	Once/3 Months	None
Reactor Water C.U. High Flow	RWCU-dPIS-170, A & B	Once/Month (1)	Once/3 Months	None
Reactor Water C.U. High Space Temp.	RWCU-TS-150 A-D, 151, 152, 153, 154, 155, 156, 157, 158, 159, RWCU-TS-81, A,B,E,F RWCU-TS-81 C,D,G,H	Once/Month (1)	Once/Operating Cycle	None

PDR 6/13/92

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON NUCLEAR WASTE
WASHINGTON, D.C. 20555

January 21, 1992

MEMORANDUM FOR: James M. Taylor
Executive Director for Operations

FROM: Raymond F. Fraley
Executive Director, ACNW

SUBJECT: 38TH ACNW MEETING FOLLOW-UP ITEMS

Based on discussions regarding methods for improving implementation and follow-up of ACNW recommendations, a summary of "Actions, Agreements, Assignments, and Requests" made during each ACNW meeting is sent to your office following each meeting.

Attached is a summary of the "Actions, Agreements, Assignments, and Requests" made at the 38th ACNW meeting, December 18-19, 1991, that deal with requests made of the NRC staff or matters that are pertinent to NRC staff activities.

Attachment: As stated

cc: H. L. Thompson, EDO
J. L. Blaha, EDO
S. J. Chilk, SECY
E. J. Jordan, AEOD
R. M. Bernero, NMSS
T. E. Murley, NRR
E. S. Beckjord, RES
A. L. Eiss, NMSS
C. Abbate, NRR
W. Brown, OCM/IS
S. Bilhorn, OCM/KR
J. Kotra, OCM/JC
R. R. Boyle, OCM/FR
W. D. Travers, NRR
D. M. Crutchfield, NRR
P. Guinn, OCM/GP

SUMMARY OF ACTIONS, AGREEMENTS, ASSIGNMENTS, AND REQUESTS
38TH ACNW MEETING - DECEMBER 18-19, 1991

During its 38th meeting, December 18-19, 1991, the Advisory Committee on Nuclear Waste discussed several matters, completed and authorized the reports noted below.

REPORTS

- Program Plan for the Advisory Committee on Nuclear Waste (Report to Chairman Selin, dated December 23, 1991)

- Geologic Dating of Quaternary Volcanic Features and Materials (Report to Chairman Selin, dated December 24, 1991)

HIGHLIGHTS OF CERTAIN MATTERS CONSIDERED BY THE COMMITTEE

- Systems Analysis Approach to Reviewing the Overall High-Level Waste Program

  The Committee was briefed by Mr. Alex Radin on the report of the Monitored Retrievable Storage Review Commission. The Committee will continue to investigate the feasibility of using a systems analysis approach to review the overall high-level waste program, including the short and mid-range technical milestones for handling high-level waste, with the goal of developing its recommendations as to the scope of the review and the advisability of undertaking it.

- Meeting with the NRC Commissioners

  The Committee met with the Commissioners to discuss items of mutual interest. The principle topics of discussion were:

  - The reports to Commissioner Rogers on the NRC staff's performance assessment and computer modeling capabilities for HLW and LLW disposal facilities
  - The recent Working Group meeting on geologic dating
  - A status report on the feasibility of a systems analysis approach to reviewing the Overall High-Level Waste Program.

- Technical Position on the Identification of Fault Displacement and Seismic Hazards at a Geologic Repository

  The Committee began its review of the Technical Position on the Identification of Fault Displacement and Seismic Hazards at a Geologic Repository. A report is expected to be completed during the 39th ACNW meeting on January 15-17, 1992.

- Election of ACNW Officers

  The Committee reelected Dr. Dade W. Moeller and Dr. Martin J. Steindler to the positions of Chairman and Vice Chairman, respectively, for calendar year 1992.

- ACNW Future Activities

  - The Committee agreed to defer indefinitely the Working Group meeting (scheduled for January 15, 1992) to discuss the need for, and status of, proposed changes to 10 CFR Part 61.

  - The Committee agreed to extend the 39th ACNW meeting to provide adequate time to discuss the Committee's long range plans. The 39th ACNW meeting will be held January 15-17, 1992.

  - The members discussed a proposed agenda for the 44th ACNW meeting to be tentatively held on June 24-26, 1992, in Richland, Washington. The members recommended that a public meeting be held either at Pacific Northwest Laboratories or the DOE Richland Regional Operations Office as appropriate.

    Facility tours will be scheduled before and after the 44th ACNW meeting with representatives of the U.S. Department of Energy Hanford Facilities and the U.S. Ecology low-level waste disposal facility. Items of possible interest include:

    - Grouting Program for LLW
    - N-Reactor Decommissioning
    - Performance Assessment and Decontamination
    - Waste Tank Stabilization and Hydrogen-Control
    - Hydrology Modeling Capabilities

  - Dr. Pomeroy requested a meeting with the members of HLW NRC staff to discuss the use of "expert judgment" in performance assessment.

  - The Committee agreed to defer a status briefing on the Licensing Support System. The ACNW staff will provide information on the status of this work to the members.

  - The Committee agreed to invite Mr. Harold Denton to brief the Committee on SECY-91-364, International Standards, as it relates to nuclear waste.

- The Committee asked to be kept informed on the NRC and the Environmental Protection Agency's efforts to develop joint guidance on mixed waste testing and storage.

- The Committee agreed to invite Mr. Michael Mattia, Director of Risk Management, Institute of Scrap Recycling Industries, to brief the Committee on practice and procedures of the recycling industry in dealing with radioactive materials found in the recycling process.

- The Committee agreed to indefinitely defer further work on the impacts of the Clean Air Act on uranium mill tailings and the proposed revision of 40 CFR Part 61, Subparts I, T, and W.

Appendix A summarizes the items proposed for future meetings of the Committee and related Working Groups. This list includes items proposed by the Commissioners and NRC staff as well as ACNW members.

APPENDIX A. FUTURE SCHEDULE

39th ACNW Committee Meeting January 15-17, 1992

Systems Analysis Approach to Reviewing the Overall High-Level Waste Program - The Committee will continue deliberations to investigate the feasibility of a systems analysis approach to review the overall high-level waste programs, including the short and mid-range technical milestones for handling high-level waste, with the goal of developing its recommendations as to the scope of the review and the advisability of undertaking it.

Revision to NUREG-1200 - The Committee will review and comment on a proposed revision to NUREG-1200, Standard Review Plan for a Low-Level Waste Facility.

Staff Technical Position on the Identification of Fault Displacement and Seismic Hazards at a Geologic Repository - The Committee will complete its review and comment on the draft Staff Technical Position on the "Identification of Fault Displacement and Seismic Hazards at a Geologic Repository."

Presentation at the Low-Level Waste Forum Winter Meeting - The Committee will discuss a paper being prepared by the ACNW for presentation at the Low-Level Waste Forum Winter Meeting. The paper will be based on reports recently issued by the ACNW on various low-level radioactive waste topics

Working Group Meetings

Systems Analysis Approach to Reviewing the Overall High-Level Waste Program, February 19, 1992, 7920 Norfolk Avenue, Bethesda, MD (Larson) - The Working Group will continue to discuss the feasibility of a systems analysis approach to reviewing the overall high-level waste program, including the short- and mid-range technical milestones for handling high-level waste.

The Impact of Long-Term Climate Change in the Area of the Southern Basin and Range, April 21-22, 1992, 7920 Norfolk Avenue, Bethesda, MD (Gnugnoli) - The Working Group will discuss the historical evidence and the potential for climate changes in the Southern Basin and Range and their associated impacts on performance for the proposed high-level radioactive waste repository at Yucca Mountain.

<u>Residual Contamination Clean-up Criteria</u>, (Date to be determined), 7920 Norfolk Avenue, Bethesda, MD (Gnugnoli) - The Working Group will review the guidelines for radionuclide contamination limits for unrestricted use of sites and facilities that are or have been under NRC license, or were at one time under AEC license.

<u>Methods for Assessing the Presence of Natural Resources at the Proposed HLW Repository Site</u>, (Date to be determined), 7920 Norfolk Avenue, Bethesda, MD (Larson) - The Working Group will discuss methodologies for the assessment of the potential for natural resources at the proposed high-level waste repository site at Yucca Mountain. The relationship between natural resources and the potential for human intrusion will be emphasized.

PDR 5/13/92

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON NUCLEAR WASTE
WASHINGTON, D.C. 20555

February 14, 1992

MEMORANDUM FOR: James M. Taylor
Executive Director for Operations

FROM: Raymond F. Fraley
Executive Director, ACNW

SUBJECT: 39TH ACNW MEETING FOLLOW-UP ITEMS

Based on discussions regarding methods for improving implementation and follow-up of ACNW recommendations, a summary of "Actions, Agreements, Assignments, and Requests" made during each ACNW meeting is sent to your office following each meeting.

Attached is a summary of the "Actions, Agreements, Assignments, and Requests" made at the 39th ACNW meeting, January 15-17, 1992, that deal with requests made of the NRC staff or matters that are pertinent to NRC staff activities.

Attachment: As stated

cc: H. L. Thompson, EDO
J. L. Blaha, EDO
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W. D. Travers, NRR
D. M. Crutchfield, NRR

SUMMARY OF ACTIONS, AGREEMENTS, ASSIGNMENTS, AND REQUESTS
39TH ACNW MEETING - JANUARY 15-17, 1992

During its 39th meeting, January 15-17, 1992, the Advisory Committee on Nuclear Waste discussed several matters, and completed or authorized the report and memoranda noted below.

REPORT

* NRC Staff Technical Position on "The Identification of Fault Displacement and Seismic Hazards at a Geologic Repository" (Report to Chairman Selin, dated January 24, 1992)

MEMORANDA

* Standard Review Plan for the Review of a License Application for a Low-Level Radioactive Waste Facility (NUREG-1200) (Memorandum to Richard L. Bangart, Director, Division of Low-Level Waste Management and Decommissioning, NMSS, dated January 23, 1992)

* Summaries of the September 1991 EPRI Workshop on EPA's HLW Standards and of the December 1991 Society for Risk Analysis (SRA) Annual Meeting (Memorandum to Commissioner Rogers from Raymond Fraley, dated January 29, 1992)

* Document on International Perspectives on Low-Level Radioactive Waste Disposal (Memorandum to Commissioner Remick from Raymond Fraley, dated January 28, 1992)

HIGHLIGHTS OF CERTAIN MATTERS CONSIDERED BY THE COMMITTEE

1. Standard Review Plan for the Review of a License Application for a Low-Level Radioactive Waste Disposal Facility (NUREG-1200)

   The Committee reviewed and commented on a proposed revision to NUREG-1200, Standard Review Plan for a Low-Level Radioactive Waste Disposal Facility.

2. Systems Analysis Approach to Reviewing the Overall High-Level Waste Program

   The Committee continued discussions of the feasibility of applying a systems approach to the analysis of the overall high-level waste program, including the short and mid-range technical milestones for handling high-level waste, with the goal of reporting back recommendations as to the proposed scope of the review and the advisability of undertaking it.

3. <u>Staff Technical Position on the Identification of Fault Displacement and Seismic Hazards at a Geologic Repository</u>

The Committee completed its review and comments on the draft Staff Technical Position on the "Identification of Fault Displacement and Seismic Hazards at a Geologic Repository."

4. <u>Report on Meeting with the Director of the Division of Low-Level Waste Management and Decommissioning</u>

Dr. Moeller reported on a meeting he had with Mr. Richard Bangart, Director, LLWM, and Mr. Paul Lohaus on December 20, 1991. The meeting participants discussed the proposed revisions to 10 CFR Part 61 Licensing Requirements for Land Disposal of Radioactive Waste, low-level waste performance assessment, residual contamination limits, and several other items of mutual interest.

5. <u>Report on Meeting with the Chief, Geosciences and Systems Performance Branch, HWLM</u>

Dr. Pomeroy reported on his meeting with Ms. Margaret Federline, Chief, Geosciences and Systems Performance Branch, on January 14, 1992. Topics discussed included the use of expert judgment in high-level waste performance analysis, and the NRC's capabilities in performance assessment and computer modeling.

6. <u>DOE Study Plan</u>

The Committee discussed the status of the DOE Study Plans currently under review by the NRC staff. The Committee discussed the possibility of conducting an indepth Committee analysis of the DOE Study Plan for Characterization of Yucca Mountain Regional Surface-Water Runoff and Stream Flow. The Committee concluded that it would select a different study plan in the future for an indepth Committee analysis.

7. <u>ACNW Future Activities</u>

- The Committee agreed to indefinitely defer the Working Group meeting on the residual radioactive clean-up criteria levels for unrestricted use of contaminated sites that are or have been under NRC license. The staff members at NRC and EPA are working on the development of such criteria. The Committee will await the results of these efforts.

- The members discussed the [illegible] ACNW meeting, tentatively scheduled to be held on June [illegible], [illegible]

Washington. The ACNW staff will finalize the meeting dates and agenda with representatives of Pacific Northwest Laboratories and the Richland Regional DOE Operations Office.

Appendix A summarizes the items proposed for future meetings of the Committee and related Working Groups. This list includes items proposed by the Commissioners and NRC staff as well as ACNW members.

APPENDIX A. FUTURE SCHEDULE

40th ACNW Committee Meeting February 20-21, 1992 (Tentative Schedule)

Systems Analysis Approach to Reviewing the Overall High-Level Waste Program (Open) - The Committee will continue to consider the feasibility of using a systems analysis approach to review the short and mid-range technical milestones for handling spent nuclear power plant fuel, with the goal of developing recommendations as to the scope of the review and the advisability of undertaking it. The Committee will discuss the results of a February 19, 1992 working group meeting on this topic.

U.S. Environmental Protection Agency's high-level waste standards (40 CFR Part 191) (Open) - The Committee will be briefed by representatives of EPA on working draft #4 of 40 CFR Part 191.

Report on the EPRI Follow-on Meeting (Open) - The Committee will hear a report on the Electric Power Research Institute meeting, held February 4-6, 1992, on the U.S. Environmental Protection Agency's high-level waste standards (40 CFR Part 191).

Report on the Low-Level Waste Forum Winter Meeting (Open) - The Committee will hear a report on the Low-Level Waste Forum Winter Meeting held in San Diego, California, on January 29-31, 1992.

Report on the Meeting with Dr. David Morrison (Open) - Dr. Moeller will report on his meeting with Dr. David Morrison, Chairman, Nuclear Safety Research Review Committee.

Committee Activities (Open/Closed) - The Committee will discuss anticipated and proposed Committee activities, future meeting agenda, and organizational matters, as appropriate. The members will also discuss matters and specific issues that were not completed during previous meetings.

Working Group Meetings

Systems Analysis Approach to Reviewing the Overall High-Level Waste Program, February 19 (8:30 a.m. - 5:00 p.m., Room [illegible]) and 20 (tentative, [illegible] a.m. - 12:00 noon, Room [illegible]), 1992, 7920 Norfolk Avenue, Bethesda, MD (Larson) - The Working Group will continue to discuss the feasibility of a systems analysis approach to reviewing the overall high-level waste program, including the short- and mid-range technical milestones for handling high-level waste.

The Impact of Long-Term Climate Change in the Area of the Southern Basin and Range, May 26-27, 1992, 7920 Norfolk Avenue, Bethesda, MD (Gnugnoli) - The Working Group will discuss the historical evidence and the potential for climate changes in the Southern Basin and Range and their associated impacts on performance for the proposed high-level radioactive waste repository at Yucca Mountain.

Methods for Assessing the Presence of Natural Resources at the Proposed HLW Repository Site, July 29, 1992, 7920 Norfolk Avenue, Bethesda, MD (Larson) - The Working Group will discuss methodologies for the assessment of the potential for natural resources at the proposed high-level waste repository site at Yucca Mountain. The relationship between natural resources and the potential for human intrusion will be emphasized.

Commonwealth Edison
1400 Opus Place
Downers Grove, Illinois 60515

May 7, 1992

Dr. Thomas E. Murley, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attn: Document Control Desk

Subject: LaSalle County Station Units 1 and 2
In-Service Inspection Program
Submittal of Relief Request RI-24
NRC Docket Nos. 50-373 and 50-374

References: (a) M. Richter (CECo) letter to T. Murley (NRC), dated October 3, 1991; Structural Margin Evaluation for Reactor Pressure Vessel Head Studs.
(b) M. Richter (CECo) letter to T. Murley (NRC), dated December 26, 1991; Relief Request RI-21 for Unit 2 Reactor Vessel Head Closure Studs.

Dr. Murley:

Commonwealth Edison (CECo) is pursuing an enhanced inspection program for the reactor vessel head closure studs at its Boiling Water Reactors. This inspection program will allow CECo to make informed decisions on long-term inspection and potential replacement strategies for the studs.

To support implementation of the enhanced inspection program, code relief is requested with respect to ASME Section XI sample expansion requirements based on the results of the magnetic particle inspections performed on the removed studs. The attached relief request, RI-28, presents CECo's proposed alternate sample expansion and examination methodology.

The attached relief request is applicable to both Units 1 and 2, and it is requested that the relief extend through the remainder of the first 10 year inspection interval, which will be completed after each Unit's sixth refueling outage.

Commonwealth Edison requests approval of this relief request to support the upcoming Unit 1 refueling outage which is scheduled to begin on September 26, 1992.

Dr. Thomas E. Murley, Director
May 7, 1992
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Please contact this office should further information be required.

Respectfully,

JoAnn M. Shields
Nuclear Licensing Administrator

Attachment: Relief Request RI-24 for LaSalle County Station

cc: A. Bert Davis, Regional Administrator-RIII
B.L. Siegel, NRR Project Manager-LaSalle
D. Hills, Senior Resident Inspector-LaSalle
R.A. Hermann, NRR Technical Staff
J.A. Davis, NRR Technical Staff
K.D. Ward, Region III

RELIEF REQUEST NO. RI-24
FOR LASALLE COUNTY STATION UNITS 1 & 2

COMPONENT IDENTIFICATION

Code Class: 1

References: Table IWB-2500-1
Paragraph IWB-2430

Examination Category: B-G-1

Item Number: B6.20 (In Place)
B6.30 (When Removed)

Description: Reactor Vessel Closure Stud Examination Requirements

CODE REQUIREMENT

LaSalle County Station is committed to the 1980 Edition, Winter 1980 Addenda of ASME Section XI. Table IWB-2500-1 requires a volumetric examination of Reactor Vessel Closure Studs if left in place, or a surface and volumetric examination of Reactor Vessel Closure Studs when removed from the flange. Removal is not a requirement at any time.

IWB-2430 requires that additional examinations be performed during the current outage if examinations performed in accordance with Table IWB-2500-1 reveal indications exceeding the acceptance standards of Table IWB-3410-1. If indications exceeding the acceptance standards of Table IWB-3410-1 are found as a result of the additional examinations, IWB-2430 requires examinations to be further extended in the current outage to include "the remaining number of similar components within the same examination category...."

BASIS FOR RELIEF

Commonwealth Edison Company (CECO) discovered stress corrosion cracking (SCC) in two reactor vessel closure studs at Dresden Unit 2 in late 1988. CECO is currently analyzing the stud material microstructure and mechanical properties. CECO is also pursuing a proactive program of enhanced stud inspections which exceed the requirements of Section XI and the recommendations of General Electric Nuclear Energy (GE) Rapid Information Communication Services Information Letter (RICSIL) 055, Revision 1, Supplement 1, "Reactor

Pressure Vessel Head Stud Cracking," March 26, 1992. The CECO program is also intended to include some of the additional recommendations of Regulatory Guide 1.65.

GE RICSIL 055 recommends that enhanced end shot UT be performed on "at least five RPV head studs either during the next refueling outage or at the next available opportunity." However, for the remainder of the first 10 year ISI Inspection Interval which encompasses the next two scheduled refueling outages (fifth and sixth) for LaSalle County Station Units 1 & 2, CECO plans to perform enhanced end shot UT of all RPV closure head studs (68 in Unit 1, 76 in Unit 2). The enhanced end shot UT technique developed by CECO utilizes a 3/4" to 1" diameter transducer with a frequency of 3.5 MHz or 5 MHz; the sensitivity of the examination is maximized by setting the background noise level at about 5% full screen height. This technique reliably detects a 0.3" deep saw cut notch from the top end of a reactor vessel stud. Any indications found with the enhanced end shot UT technique will be sized with bore probe UT. The bore probe UT technique developed by CECO reliably detects a 0.1" deep saw cut notch.

At each refueling outage CECO also plans to remove, if practicable, approximately 1/6 of the total number of studs (12 in Unit 1, 13 in Unit 2) from the flange of the LaSalle Reactor Pressure vessel for a wet fluorescent MT. Studs which are normally removed each outage to allow for the installation of the Cattle Chute will be excluded from the sample because they are not exposed to the aqueous environment likely to cause pitting. Cracking is believed to occur when pitted studs are tensioned while still exposed to water at the end of a refueling outage.

There are several reasons for removing a sample of studs during the remaining two refueling outages in the first 10 year ISI Inspection Interval for surface examination:

To provide data on incipient stud cracking.

To allow for additional metallurgical evaluation of cracking mechanisms and potential embrittlement phenomena, if cracked studs are found and replaced.

To provide a correlation between enhanced end shot UT, bore probe UT, and MT results, if cracked studs are found.

This information is necessary to make informed decisions on long-term inspection/replacement strategies.

Code structural margins will be assured through the enhanced end shot UT of all studs, and bore probe UT sizing of all cracked studs. Enhanced end shot and bore probe UT results will be evaluated in accordance with "Fracture Mechanics Based Structural Margin Evaluation for Commonwealth Edison BWR Reactor Vessel Head Studs," GE Nuclear Energy Report GE-NE-523-93-0991, DRF 137-0010, September 1991 (submitted with a M.H. Richter (CECO) letter to T.E. Murley (NRC) dated October 3, 1991). The GE structural margin evaluation is based on conservative fracture mechanics methodology and actual fracture toughness testing of material from one of the low-toughness Dresden Unit II studs. If the end shot is found to be nonconservative, then an expanded sample with the more sensitive bore probe will be performed. This approach will assure that Code structural margins are maintained with out expanding the MT sample.

Results of the enhanced end shot UT, bore probe UT, and MT will be compared in order to benchmark the minimum detection limit of the enhanced end shot UT technique. The minimum detection limit of the enhanced end shot UT technique will be judged against a conservative, bounding maximum allowable flaw size (established by the GE structural margin evaluation) which would be acceptable in all studs at the same time. If the minimum flaw detection limit of the enhanced end shot UT is found to be greater than the maximum allowable flaw size, additional bore probe UT examinations will be performed in lieu of the Section XI required MT sample expansion.

Expanding the MT sample if unacceptable surface indications are found would greatly increase the critical path time and manrem burden during the outage. And, as other utilities have found, it may be impossible to remove the desired sample of studs, without damage, within the time constrains of a refueling outage. It is estimated that complete removal of all studs, assuming no stuck studs, would take [illegible] additional critical path days and expend [illegible] additional manrem.

The proposed program is highly proactive, in that Section XI only requires a normal sensitivity end shot

UT to be performed in place, and RICSIL 055 only recommends enhanced end shot UT of at least five studs. In accordance with Section XI, structural margin would still be assured by the enhanced end shot and bore probe UT. Yet much essential information could be gained by surface examination of a limited sample of studs. For these reasons, CECO requests relief from the MT sample expansion requirements of Section XI IWB-2430 for the remainder of the first 10 year ISI Inspection Interval for both LaSalle County Station Units 1 & 2.

PROPOSED ALTERNATE EXAMINATION

In lieu of the Code Requirement, at each refueling outage conducted in the applicable time period for LaSalle County Station Units 1 & 2 each LaSalle County Station stud will be examined in place using enhanced end shot UT. Any flaws detected with the enhanced end shot UT will be sized using bore probe UT.
If an MT examination of a sample of studs reveals indications which are found by bore probe UT to exceed the maximum allowable flaw size, and were not detected by the enhanced end shot UT, then sample expansion will proceed using bore probe UT in lieu of the Section XI required MT sample expansion.

APPLICABLE TIME PERIOD

This relief is requested for each refueling outage for LaSalle County Station Units 1 & 2, beginning with the fifth refueling outage for Unit 1 which is scheduled to begin September 26, 1992. It is also requested that the relief extend through the remainder of the first 10 year Inspection Interval for each Unit (1 & 2) which will be completed after that Unit's sixth refueling outage. The sixth refueling outage for LaSalle County Station Unit 1 is scheduled to end in May of 1994. The sixth refueling outage for LaSalle County Station Unit 2 is scheduled to end in May of 1995.