Commonwealth Edison
1400 Opus Place
Downers Grove, Illinois 60515

#7
3/19

March 16, 1992

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

Attention: Document Control Desk

Subject: Revised Response to Bulletin 88-08, Thermal Stresses in Piping Connected to Reactor Coolant Systems.

Byron Units 1 and 2,
NRC Docket Numbers 50-454 and 50-455

Braidwood Units 1 and 2,
NRC Docket Numbers 50-456 and 50-457

References:
(1) Bulletin 88-08, Thermal Stresses in Piping Connected to Reactor Coolant Systems, dated June 22, 1988.

(2) Bulletin 88-08, Supplement 3, dated April 11, 1989.

(3) A. Hsia (NRC) to T. Kovach (CECo) letter dated December 11, 1991.

(4) "Evaluation Criteria for Responses to NRC Bulletin 88-08, Action 3 and Supplement 3" provided in Reference (5)

(5) Teleconference between NRC/NRR, CECo and Westinghouse on Thursday February 13, 1992.

(6) M.H. Richter to NRC letter dated July 17, 1989

Dear Dr. Murley:

The purpose of this letter is to provide a revised response to Bulletin 88-08 for Byron and Braidwood Stations. Because of clarifications provided by NRC staff in Reference (3) and in the Reference (5) phone call, Commonwealth Edison (CECo) is revising the Byron and Braidwood Bulletin 88-08 programs to include temperature monitoring provisions. Also, based on CECo's review of the Reference (4) criteria recently provided by the NRC, a previously identified location is no longer considered susceptible to thermal fatigue cracking and is no longer in the scope of Bulletin 88-08 actions for Byron and Braidwood Stations.

Ltr Encl JE16
Change NRC PDR 1 1 NP
NSIC 1 1 NP

After completion of the temperature monitoring installation, the Byron and Braidwood Bulletin 88-08 programs will provide for the monitoring of potentially susceptible locations. A discussion of; Bulletin requirements, affected areas, the analyses performed, as well as the monitoring and examination programs is contained in Attachment (1). Attachment (2) is a discussion of industry efforts for final resolution of Bulletin 88-08.

Enclosed in Attachment 3 are:

1. 1 Copy of WCAP-1287, "Evaluation of Thermal Stratification for the Byron and Braidwood Units 1 and 2 Residual Heat Removal Lines" (Proprietary).

2. 1 Copy of WCAP-12388, "Evaluation of Thermal Stratification for the Byron and Braidwood Units 1 and 2 Residual Heat Removal Lines" (Non-Priority).

Also enclosed in Attachment 3 are a Westinghouse authorization letter, CAW-92-277, accompanying affidavit, Proprietary Information Notice, and Copyright Notice.

Enclosed in Attachment 4 are:

3. 1 Copy of WCAP-12425, "Evaluation of Byron and Braidwood Units 1 and 2 Auxiliary Spray Lines per NRC Bulletin 88-08" (Proprietary).

4. 1 Copy of WCAP-13245, "Evaluation of Byron and Braidwood Units 1 and 2 Auxiliary Spray Lines per NRC Bulletin 88-08" (Non-Proprietary).

Also enclosed in Attachment 4 are a Westinghouse authorization letter, CAW-92-278, accompanying affidavit, Proprietary Information Notice, and Copyright Notice.

As Items 1 and 3 contain information proprietary to Westinghouse Electric Corporation, it is supported by an affidavit signed by Westinghouse, the owner of the information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.790 of the Commission's regulations.

Accordingly, it is respectfully requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.790 of the Commission's regulations.

/scl:1619:2

Correspondence with respect to the copyright or proprietary aspects of the items listed above or the supporting Westinghouse Affidavit should reference CAW-92-277 and/or CAW-92-278 and should be addressed to Nicholas J. Liparulo, Manager of Nuclear Safety and Regulatory Activities, Westinghouse Electric Corporation, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

To the best of my knowledge and belief, the statements contained in this document are true and correct. In some respect these statements are not based on my personal knowledge, but on information furnished by other CECo employees, contractor employees, and consultants. Such information has been reviewed in accordance with company practice, and I believe it to be reliable.

If there are any questions or comments, please contact me at (708) 515-7292.

Sincerely,

D. J. Chrzanowski

David J. Chrzanowski
Nuclear Licensing Administrator Generic Issues

cc: A. Bert Davis, Regional Administrator-RIII (w/o Attachments 3 and 4)
R. Pulsifer, Project Manager-NRR/PDIII-2
A. Hsia, Project Manager-NRR/PDIII-2 (w/o Attachments 3 and 4)
R. Elliott, Project Engineer-NRR/PDIII-2 (w/o Attachments 3 and 4)
S. DuPont, Senior Resident Inspector (Braidwood) (w/o Attachments 3 and 4)
W. Kropp, Senior Resident Inspector (Byron) (w/o Attachments 3 and 4)

State of Ill County of Cook
Signed before me on this 16th day
of March, 19 92 by D.J.C.
Notary Public

" OFFICIAL SEAL "
SANDRA C. LARA
NOTARY PUBLIC, STATE OF ILLINOIS
MY COMMISSION EXPIRES 6/25/94

/scl:1619:3

ATTACHMENT 1

BULLETIN 88-08 REQUIREMENTS

AND

COMMONWEALTH EDISON PROGRAM DESCRIPTION

Background

Reference (1) requested that licensees take actions to assure that certain reactor coolant system (RCS) lines will not be subjected to unacceptable thermal stresses. The first of these actions requested licensees review the systems connected to the RCS to determine if unisolable sections of piping could be subjected to stresses from temperature stratification or temperature oscillations. If these connecting lines could not be subjected to this type of condition, no additional actions were required. Action (2) of the Bulletin requested licensees to nondestructively examine the welds, heat-affected zones, and high stress areas of those lines identified in Action (1) to assure that there were no existing flaws. Finally, Action (3) requested licensees to plan and implement a program to provide continuing assurance that the piping, identified in Action (1), will not be subjected to combined cyclic and static thermal stresses that could result in fatigue failure.

Reference (2) notified licensees of another instance of thermally induced fatigue cracking and subsequent RCS leakage. This incident at the Genkai Plant in Japan involved the Residual Heat Removal (RHR) line. The supplement did not require any additional actions but requested that the RHR lines be evaluated for thermal fatigue cracking susceptibility. CECo responded to this Supplement in Reference (6) and included the RHR pump suction lines of Byron and Braidwood into the Bulletin 88-08 program.

Affected Locations

Originally the CECo Bulletin 88-08 program had identified, and previous submittals had discussed, three system connections as part of Bulletin 88-08 Action (1). These connections were; safety injection charging line to RCS (four lines per unit), the auxiliary pressurizer spray (aux spray) line to main pressurizer spray line (one line per unit), and the RHR pump suction line to RCS (two locations per unit).

However, after reviewing Reference (4), CECo has determined that the aux spray line to main spray line is configured in a way that precludes unacceptable thermal stresses from developing. That is, the aux spray check valves are located approximately 50 to 52 pipe diameters from the main spray nozzle at all Byron and Braidwood units (see Figures 1 through 4). This is much greater than the 25 pipe diameter distance defined in the NRC evaluation criteria section 3.1(D)(b) as exempting lines from Bulletin consideration. At this distance any cooler water forced upward from the lower section trap by potential check valve leakage would be warmed to main spray temperature before reaching the nozzle.

CECo Bulletin 88-08 Engineering Analyses

As explained in the Reference (5) conversation, the CECo Bulletin 88-08 program is not simply an ISI inspection based program. It is a program based on engineering analyses that concluded that for two of the Byron and Braidwood locations it is unlikely that the thermal fatigue phenomenon described in the Bulletin could occur. However, rather than remove the aux spray lines and RHR lines from consideration, CECo elected to perform temporary monitoring and commission additional analyses to determine the impact of worst case leakage.

To address the continuing assurance provision of Bulletin 88-08 for the auxiliary spray line, CECo had instrumented the Byron Units 1 and 2 auxiliary spray lines with surface-mounted temperature sensors to detect adverse temperature distributions. A review of the monitoring data indicated that the temperatures were steady, with no cycling observed. The temperatures were also within the expected range, and it was therefore concluded that inleakage of cold fluid from the auxiliary spray line into the main spray piping was not occurring. To ensure the integrity of the piping over the design life of the plant, analysis was performed assuring valve leakage and associated stress cycling. A conservative transient was developed, based upon the experience of the Farely safety injection piping failure. Transient stresses were calculated and used as input to a fatigue analysis, using approved ASME Section XI methodology, to determine an acceptable period of operation between inservice inspection intervals.

The following conservatisms were assumed in the overall approach:

- Isolation valve leaks continuously.
- Top-to-bottom of pipe temperature difference is 300°F, based on comparison of maximum potential temperature difference with Farley safety injection.
- Cyclic period of 7.3 minutes, based on heat transfer calculations.
- Fluid temperature transient is instantaneous, i.e. step change in time.
- Initial crack size of 10% of the wall thickness.
- Final crack size limited to 60% of the wall thickness.

The results of this analysis was that 39 months of power operation is an acceptable period for inservice inspection intervals to provide continuing assurance of the pressurizer auxiliary spray piping integrity, and therefore monitoring is not necessary. This information is documented in Westinghouse Report WCAP-12425, October 1989 (Attachment 4).

To address the continuing assurance provision of the Bulletin 88-08 for the residual heat removal (RHR) suction piping, analysis was performed, and is documented in Westinghouse Report WCAP-12387 (Attachment 3). A comparison was first made to the Genkai RHR configuration which support the conclusion that a Genkai-type transient is unlikely to occur at Byron and Braidwood. Specifically, transients are not expected since turbulent penetration of hot RCS water is expected to extend nearly to the isolation valve.

Evaluations were carried out, however, assuming valve leakage and associated thermal transients, based on the experience of the Genkai piping failure. Transient stresses were calculated and used as input to a fatigue analysis. Since ASME fatigue usage factor requirements could potentially be exceeded under the assumed conservative transient loadings, fatigue crack growth calculations were performed to determine an acceptable inservice inspection interval. The following conservatisms were assumed in the overall approach:

- Entire length of horizontal piping stratified.
- Most stiffly supported configuration of the eight locations at Byron and Braidwood was analyzed.
- Maximum moment stress and maximum through wall stress assumed to occur at any location in the horizontal piping.
- Cyclic period parametrically investigated from 10 to 60 minutes.
- Initial flaw size of 15% of the wall thickness.
- Final flaw size of 60% of the wall thickness.

This conservative analysis concluded that 4.1 years of continuous valve leakage and transient loading would result in a crack of 60% of the wall thickness. Inservice inspection was therefore recommended at every other refueling outage, or about three years of operation.

CECo recognized that much was unknown about the Farley, Tihange and Genkai events and therefore had chosen to supplement the analyses with an enhanced, augmented ultrasonic examination program. This examination exceeds ASME Section XI requirements in several aspects. First, the frequency of examination is every other refueling. Second, the examination uses IGSCC techniques and EPRI qualified examiners. Also, in the case of the aux spray location, a mock-up of the branch connection was developed to optimize the inspection.

However, CECo understands the NRC's concerns regarding fatigue crack initiation and is now proposing a temperature monitoring program for the RHR location. The monitoring program will replace the interim examination program.

Proposed and Existing Monitoring Programs

The proposed monitoring program for the RHR lines will follow the guidelines and exceedance criteria established in Reference (4). Prior to the installation of the monitoring equipment at Byron and Braidwood, RCS integrity at the RHR connection will be assured by the existing analyses and examinations described above. If after monitoring the RHR location it is determined that turbulent penetration (discussed previously) is in fact occurring, CECo will discuss with the NRC the option of discontinuing temperature monitoring on RHR.

The aux spray locations, as stated previously, will no longer be considered as candidate locations for a Bulletin 88-08 monitoring program. The aux spray location will, however, remain in the Bulletin 88-08 ultrasonic examination program until the results of the EPRI program, described in Attachment (2), are available to evaluate this location.

CECo proposes to maintain the ongoing leakage monitoring program for the four (per unit) safety injection locations as an alternative to temperature monitoring. This leakage monitoring program meets the requirements of Action (3) of the Bulletin which states; "Plan and implement a program to provide continuing assurance that unisolable sections of all piping connected to the RCS will not be subjected to combined cyclic and static thermal and other stresses that could cause fatigue failure during the remaining life of the unit." The CECo leakage monitoring program, by assuring leak tight integrity of the isolation valves, prevents thermal stresses from developing thereby preventing any fatigue crack initiation. A single line drawing, to help explain the monitoring program, is shown in Figure (5). Although the drawing is listed as a Braidwood Unit 1 drawing it is typical of the Byron Units 1/ 2 and Braidwood Unit 2 configurations.

The leak test procedure which monitors any leakage past the isolation valves was specifically developed for Bulletin 88-08. This procedure is part of an overall Technical Specification surveillance that also monitors the leakage by the check valves in this portion of the Safety Injection system.

Simply described the procedure records the leakage past isolation valves SI8801A-1 and SI8801B-2 by monitoring leakage through test connection SI044. First the piping downstream of check valve SI8815 and the isolation valves is depressurized. Next, the charging pumps are started and are run against the closed SI8801A-1 and SI8801B-2 valves. It is then that any leakage is accurately measured using a graduated cylinder and stopwatch through the test connection SI044. Leakage past these isolation valves is required to be reported and reviewed by Engineering for impact on the Bulletin 88-08 program. To date, no leakage has been detected passed these isolation valves at Byron or Braidwood. This monitoring program has assured that unacceptable thermal stresses have not been induced into the down stream piping or RCS nozzle connection.

This isolation valve leakage monitoring is performed;

a. At least once per 18 months;

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 72 hours or more and if leakage testing has not been performed in the previous 9 months;

c. Prior to returning the valve to service following maintenance; repair, or replacement work on the valve;

d. Within 24 hours following valve actuation due to automatic or manual action or flow through the valve, except for valves 1RH8701A and B and 1RH8702A and B.

This monitoring frequency encompasses all occasions when these isolation valves may have a potential to leak, that is after start-up, after stroking or after maintenance.

Industry Program

Commonwealth Edison is closely following the industry's efforts in optimizing a Bulletin 88-08 program. The ultimate solution to this issue will rely on the results of EPRI/TASCS program described in Attachment (2). CECo plans to review the EPRI results and take appropriate actions when these results become available.

Conclusions

To summarize, all locations potentially susceptible to thermal fatigue cracking at Byron and Braidwood are or will be monitored. The monitoring program for the safety injection lines, four per unit, is an ongoing leakage monitoring program and consists of verifying that the isolation valves upstream of the check valves do not leak. Assuring that these valves do not leak satisfies the Bulletin requirement that these sections of pipe are not subjected to thermal stresses that could cause fatigue failure.

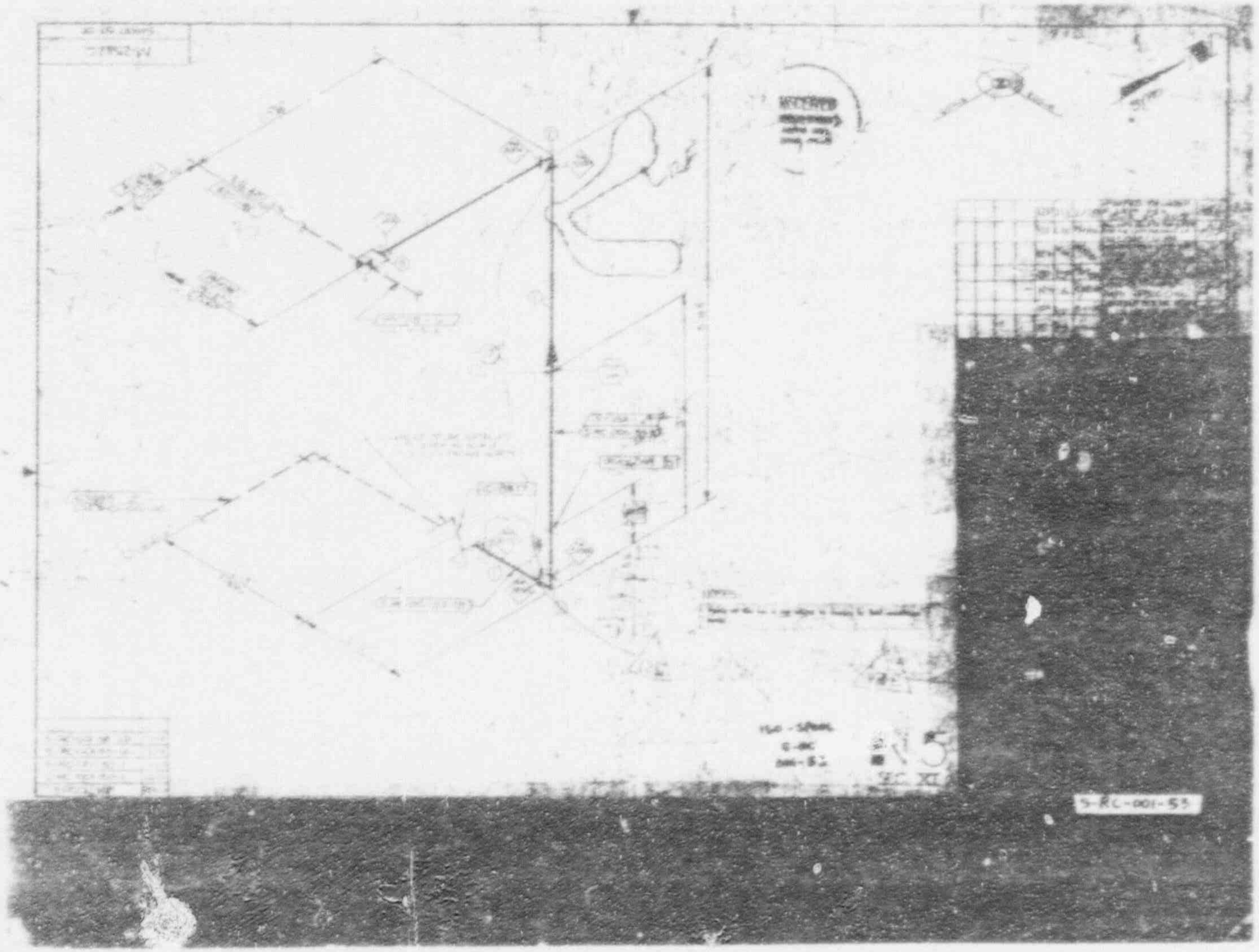
The temperature monitoring equipment for the RHR lines will be installed at Byron and Braidwood as follows:

Unit	Outage
Braidwood Unit 1	A1RO4 Spring 1994
Braidwood Unit 2	A2RO3 Spring 1993
Byron Unit 1	B1RO5 Spring 1993
Byron Unit 2	B2RO4 Fall 1993

The current Bulletin 88-08 ultrasonic examinations of the RHR high stress locations will continue until the monitoring equipment is in place. Completion of the RHR monitoring installation will satisfy the requirements of Bulletin 88-08 for Byron and Braidwood Stations.

The auxiliary spray line, because of the location of the check valve relative to the main spray connection, is no longer part of the Bulletin 88-08 monitoring program. However the high stress locations on these lines will continue to be ultrasonically examined every other refueling outage until the EPRI TASCS program is complete. At that time, CECo will determine if any locations should be added to or deleted from the Bulletin 88-08 monitoring and examination program.

Figure 1
Byron Unit 1
Aux Spray Piping



SC

Figure 2
Byron Unit 2
Aux Spray Piping

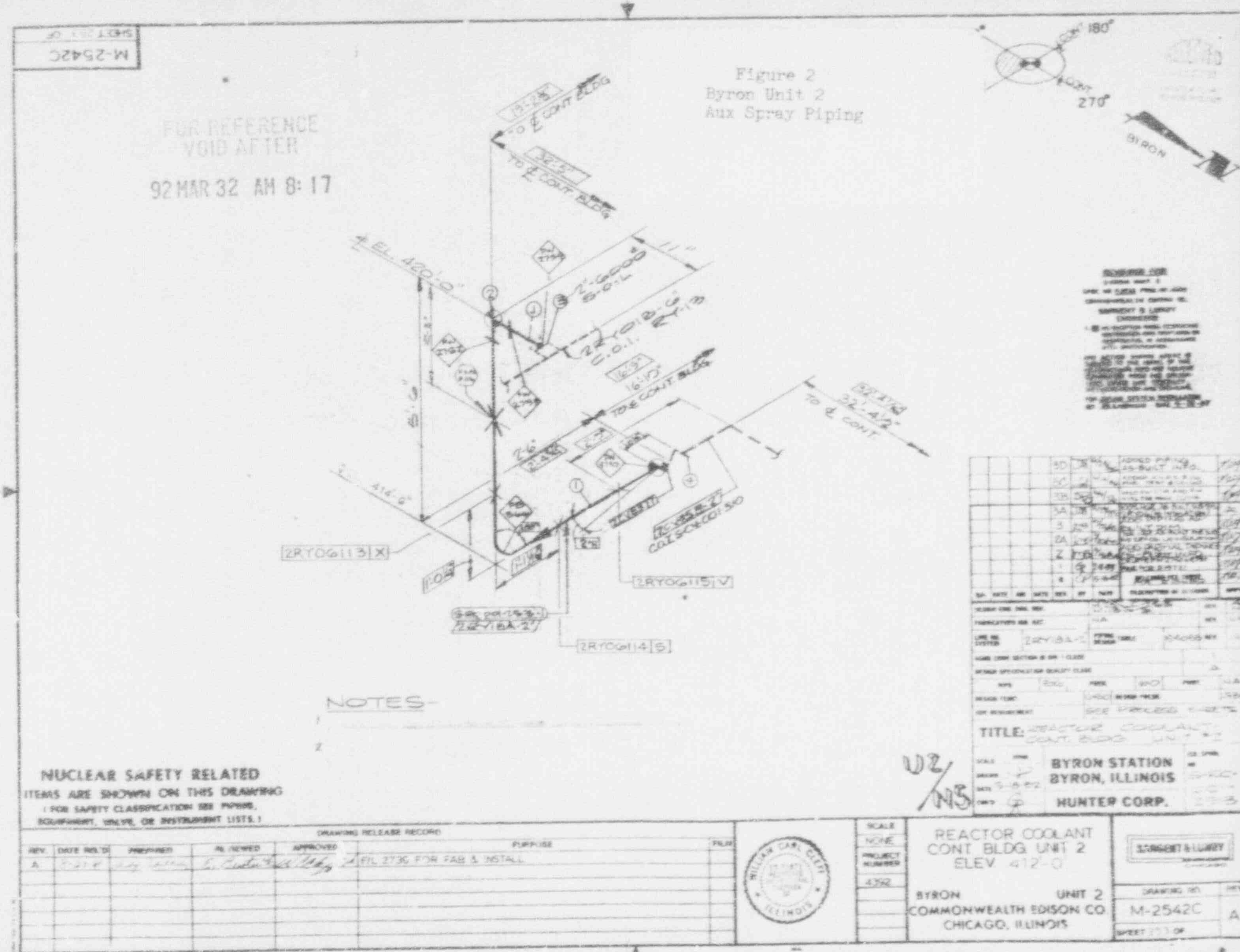
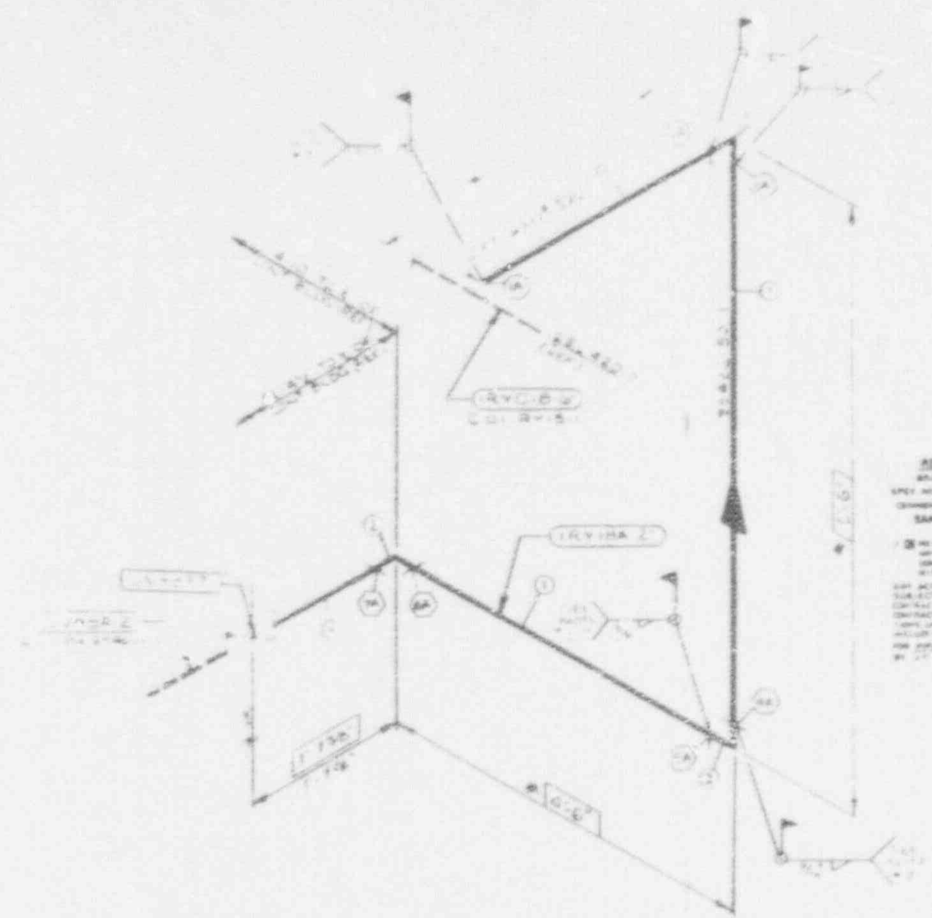


Figure 3
Braidwood Unit 1
Aux Spray Piping



NOTES:

1. FABRICATION AND INSTALLATION DATA AND INSTALLATION TOLERANCE PER PROCEDURES GCP B21. GENERAL NOTES PER ...
2. ◯ DENOTES FIELD WELD
3. INSULATION: TYPE B
4. ▭ DIMS. ARE "AS-CONST" FOR PIPING CONFIGURATION SHOWN.
5. ● DENOTES "AS-CONST" DIM. SAME AS DESIGN DIM.
6. REFER TO FCO* 7031 PER REV.A
7. ... WELD NO. ... WELD LEG SIZE PER ...
8. BCN * 27736 DOES NOT REQUIRE REWORK TO THIS DWG AND DOES NOT VOID BWO * 545

REVIEWED FOR
SARGENT & LUNDY

I S I
SURFACE
EXAM

PHILLIPS, GETSCHOW CO

REACTOR COOLANT SYSTEM

N/A

DRAWING RELEASE RECORD

REV	DATE REL'D	PREPARED	REVIEWED	APPROVED	PURPOSE	FILM

SCALE

PROJECT NUMBER

A.S.M.E.
SECTION XI

DOC. STA. *2

SARGENT & LUNDY

DRAWING NO. | REV

Figure 4
Braidwood Unit 2
Aux Spray Piping

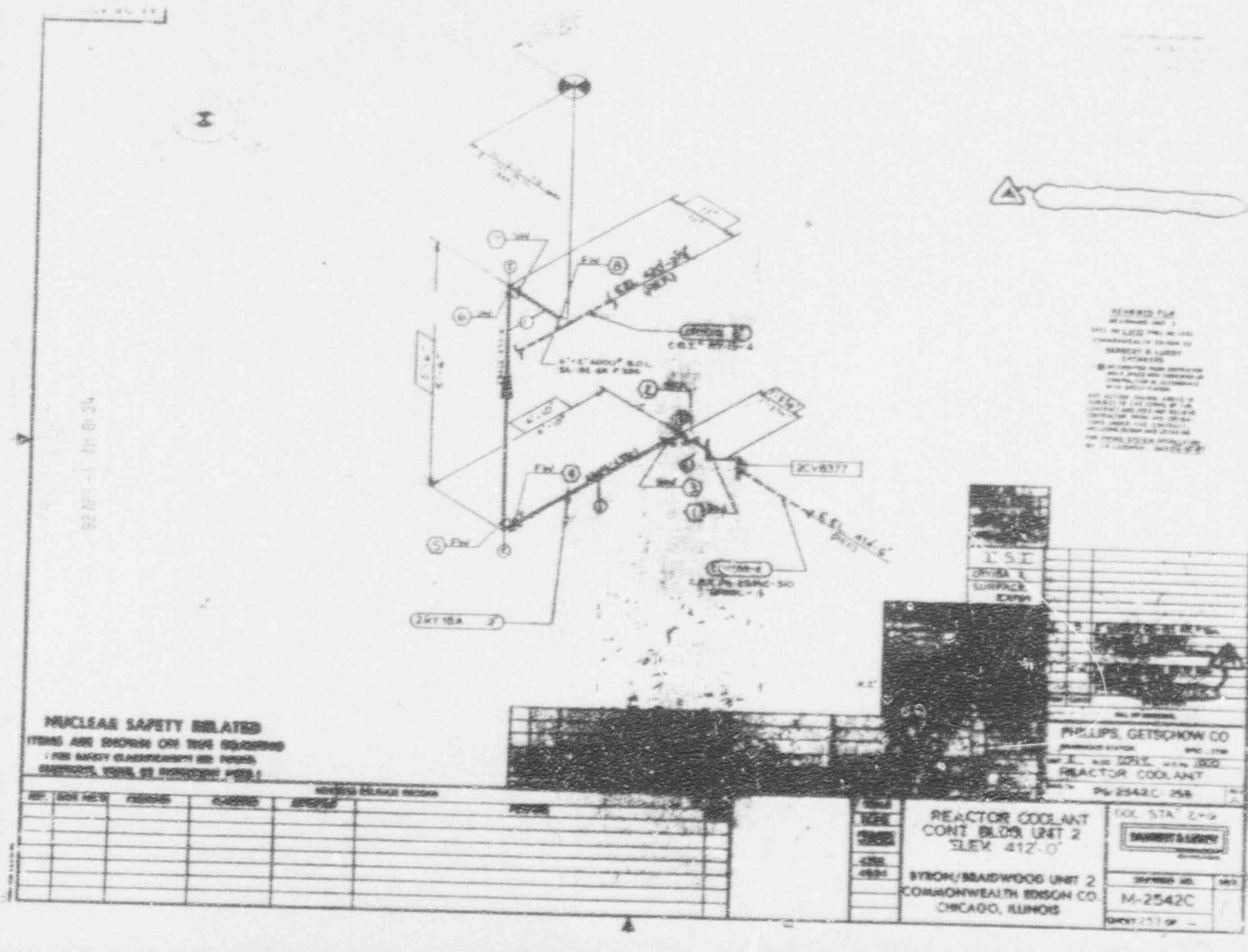
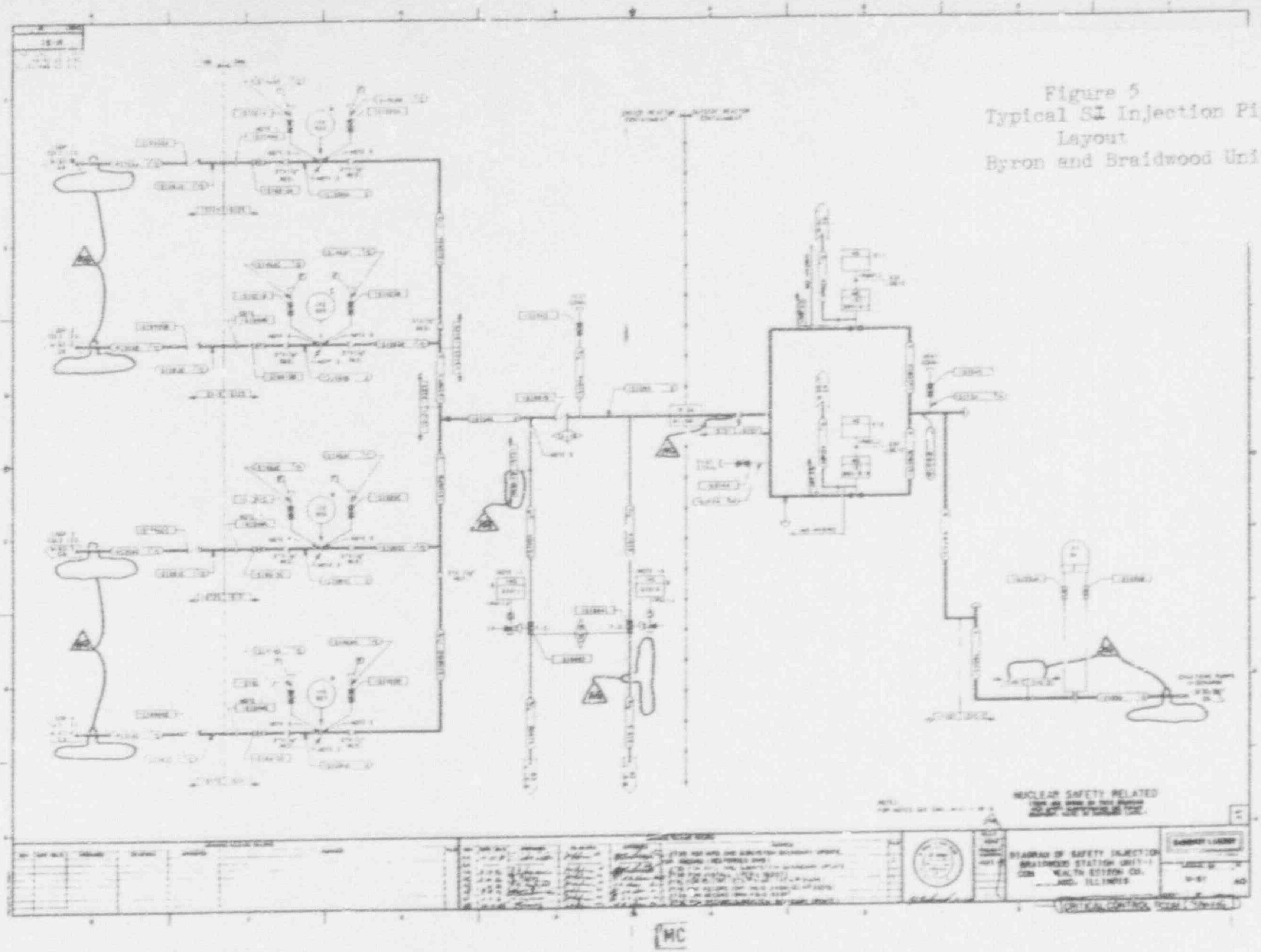


Figure 5
Typical SI Injection Piping
Layout
Byron and Braidwood Units



ATTACHMENT 2

INDUSTRY EFFORTS FOR BULLETIN 88-08 RESOLUTION

EPRI TASCS PROGRAM

The EPRI TASCS (Thermal Stratification, Cycling and Striping) program has been developed to provide industry with the tools needed to evaluate the impact of thermal stratification issues in piping systems.

The need for this program was determined following two years of investigation into the existing methods and data to evaluate TASCS phenomena. It was concluded that the existing methods were overly conservative and resulted in more monitoring and inspections that are actually required. This is because most evaluations assumed loadings similar to those which caused the pipe cracks discussed in NRC Bulletin 88-08. Therefore, this program was established to develop more realistic conservative loading for the TASCS phenomena.

The subtasks of the program are as follows:

1. Task 1: Develop Preliminary Categorization and Screening Methodology. The objectives are: To determine an approach for defining, without extensive analysis, lines affected by TASCS phenomena; to determine parameters and their ranges to consider in developing modes to evaluate TASCS; where possible, establish limits below which thermal fatigue due to TASCS will not occur; and to determine conditions needing experimental support within the project scope.

2. Task 2: Define TASCS Analytical Approach. The objectives are: To define a methodology for evaluating susceptible lines that have "failed" the screening criteria; to define preliminary inputs for thermal stress and fatigue analysis; and to determine where new correlations and evaluation methodology are required as inputs to later tasks.

3. Task 3: Define Testing and Analysis Program. The objective is to define the scope of testing and analysis based on available data and input from Tasks 1 and 2.

4. Task 4: Conduct Testing and Data Acquisition Analysis. The objective is to conduct testing, data acquisition and analysis as specified in Task 3.

5. Task 5: Develop Correlations/Models. The objective is to develop correlations, models, etc. for screening criteria and evaluation methodology for use by utility operators and design engineers.

6. Task 6: Develop Final Screening and Categorization Methodologies. The objective is to finalize the screening and categorization methodology developed in Task 1 using the correlations and methods developed in Task 5.

7. Task 7: Guidance Manual and Evaluation Tools. The objective is to develop guidance manual for determining systems susceptible to TASCS (from Task 6) and for providing engineering tools necessary to evaluate thermal fatigue impact on systems determined to be potentially susceptible.

This program is scheduled to be completed in May 1993, at which time the guidance manual and screening criteria will be available for utility use. Workshops are also planned to provide training to the plant operators and engineers who will utilize this methodology.

Applicability to NRC Bulletin 88-08

The TASCS program is directed by EPRI and a utility advisory committee consisting of representatives of the major nuclear owner's groups (Westinghouse, B&W, CE and GE). Specific priority has been given to addressing issues related to NRC Bulletin 88-08. Therefore testing and analysis has an emphasis on leakage type flows, the potential for this flow to stratify, and the interactions which occur when the leakage flow mixes. This program will therefore produce tools directly suitable to evaluating the issues associated with the auxiliary spray line and its interaction with the pressurizer spray line.

ATTACHMENT 3

1 copy of WCAP 12387 (Proprietary)

1 copy of WCAP 12388 (Non-Proprietary)

and Westinghouse authorization letter, CAW-92-277, accompanying affidavit, Proprietary Information Notice, and Copyright Notice.