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June 5, 1970

Mr. Robert Tedesco
Division of Reactor Licensing
U. S. Atomic Energy Commission
Beth. 008
Washington, D. C. 20545

Dear Bob:

In response to the questions received over the telephone concerning our meeting next week with the ACRS on furnace sensitized stainless steel components, we are enclosing copies for distribution prior to the meeting. Because of the shortage of time, some answers may not be complete. These issues can be clarified at the meeting.

Very truly yours,

Byron

Byron Lee, Jr.
Assistant to the President

2514

COMMONWEALTH EDISON COMPANY

Response to ACRS Concerns

on

Furnace Sensitized Stainless Steel Components

NOT INDEXED

1 - Operational plans for immediate future and until the next refueling?

2 - Considering action taken on Unit 3 with regard to furnace sensitized stainless steel components, why is similar action not to be taken on Unit 2?

3 - Why is it considered acceptable to not perform inspection of furnace sensitized stainless steel components during period that Unit 2 will be down following completion of 100 hour run?

4 - Planned in-service inspection program for furnace sensitized stainless steel components?

5 - Documentation of material presented at the meeting regarding sensitized stainless steel in primary coolant system including safe end design, thermal sleeve, pipe hangers, stresses, piping arrangements, fill, flush and drain procedures, etc.?

6 - Details of considerations given to fixes (time requirements, etc.)?

7 - Identify estimated whole measured oxygen concentrations throughout reactor and in the region of safe ends.

8 - Discuss and compare the differences between Dresden 2 and Nine-Mile Point with respectto cleaning procedures (including PT, stresses, fill, flush and drain procedures, heat treatment, oxygen and chloride limits, layout, and materials) to the extent information is available.

9 - Power level or reactor history at which a loss of coolant accident will not result in fuel failures and hydrogen can be disposed of by purging without the release of fission products.

10 - The extent of reevaluation of piping stresses, independence of review, basic assumptions and field confirmation of as-built configuration.

11 - What is the primary leak detection sensitivity and potential for increased sensitivity in the nozzle region - basis for action (shutdown, etc.)?

12 - Discuss the extent of inspection that is planned at the first refueling outage.

13 - Would the failure of any furnace sensitized stainless steel reactor vessel internal component result in the inability to cool the core?

14 - Technical basis for relating surveillance and leakage sensitivity to crack propagation rates?

(1) Due to the recent problems which have been experienced during the Dresden 2 startup testing program, we are not able, at this time, to state what our operational plans are for the immediate future and until the next re-fueling outage.

Question #2

The decision to replace or overlay the furnace sensitized stainless steel safe ends on the Dresden 3 and Quad-Cities 1 and 2 vessels was not made for reasons of safety. The decision was based on long-term economics and unit availability.

In making the change of safe ends on the Dresden 3 and Quad-Cities vessels we assumed that sensitized safe end deterioration on other installations (under circumstances different than those that exist at Dresden) would place a burden of proof on all sensitized safe ends. This proof would result in more extensive inspections and plant outages (planned and forced) than would otherwise be necessary. The cost and electric service reliability consequences of additional outages and inspection was considered to be greater than for an immediate change out for Dresden 3 and Quad-Cities 1 and 2, but not for Dresden 2 because of its advanced stage of construction.

We are confident that proof will be generated that the Dresden 2 safe-ends will perform adequately for the life of the plant. There is no reason presently known to believe they will not perform adequately for the life of the unit or at least for many years (witness the successful operation of Dresden Unit 1 for ten years).

Improving technology will result in many changes in materials and designs in the future. These changes cannot and must not be inferred to mean our past materials or designs were inadequate or unsafe. If they are, the credibility of the entire industry will be in jeopardy.

3.

To date, we have not planned to perform nondestructive examinations of the furnace sensitized stainless steel safe-ends and other furnace sensitized stainless steel parts of the D2 reactor coolant pressure boundary system during the period that Unit 2 will be down following completion of the 100-hour run because (a) the numerous examinations performed before the unit started operating disclosed no problems, (b) the leak detection system which has been in service while the unit is in operation has given no indication of leaks other than in valve packing and pump shaft glands, and (c) the environment has not been hostile.

4.

The presently planned inservice examination of furnace sensitized stainless steel parts of the D2 pressure boundary system is as described in Table 4.6.1, pages 113 to 118, inclusive. This is similar to the 1970 ASME Boiler and Pressure Vessel Code, Section XI, "Inservice Inspection of Nuclear Reactor Coolant Systems" in many respects, particularly with regard to bimetallic welds and welds in stainless steel piping. While the total area of the sensitized stainless steel safe-ends is not specifically mentioned in Table 4.6.1, the fact that the safe-ends are, in general, 12 inches or less in length with a weld at each end, ensures that the sensitized material will be volumetrically, visually and dye-penetrantly examined on all accessible surfaces.

5

I. STEPS TAKEN AT DRESDEN-2 AS A RESULT OF RECENT FAILURE SENSITIZED STAINLESS STEEL COMPONENTS

A. STRESS ON SAFE-ENDS REVIEWED.

B. HANGER AND RESTRAINT DESIGNS REVIEWED.

C. INSPECT HANGERS AND RESTRAINTS DURING AND AFTER HEATUP.

D. RECHECKED RECORDS ON PT.

E. RECHECKED RECORDS ON U/T BASE LINE.

F. RAN CHARPY V-NOTCH TESTS ON DRESDEN-3 SAFE ENDS.

5/22/70

II. DIFFERENCES BETWEEN DRESDEN-2 AND NMP RE FACTORS POTENTIALLY CONTRIBUTING TO PIPE CRACKS

A. THERMAL SLEEVE DESIGN.

B. NOZZLE SIZE VS PIPE SIZE (CORE SPRAY).

C. DIFFERENT VESSEL MANUFACTURER.

D. CLEANLINESS AND CLEANING PROCEDURES THROUGHOUT VESSEL ERECTION PERIOD.

E. 316 SAFE-END AT DRESDEN VS 304 AT NMP.

5/22/70

III. DRESDEN PROCEDURES TO AVOID OR DETECT PROBLEM

A. COMPLETE PT INSIDE AND OUTSIDE OF ALL SAFE-ENDS AFTER VESSEL SET.

B. PT OF ALL ACCESSIBLE SAFE-END AREAS AFTER HYDRO (JULY, 1969).

C. U/T OF ALL SAFE-ENDS 4-INCH AND ABOVE AFTER HYDRO (JULY, 1969).

D. CORE SPRAY NOZZLE PT'D AND CLEANED BEFORE THERMAL SLEEVE INSTALLED.

E. SYSTEM FILLED AND HYDROED WITH 500 - 2000 PPM TSP TO SERVE AS INHIBITOR.

5/22/70

Exhibit IV

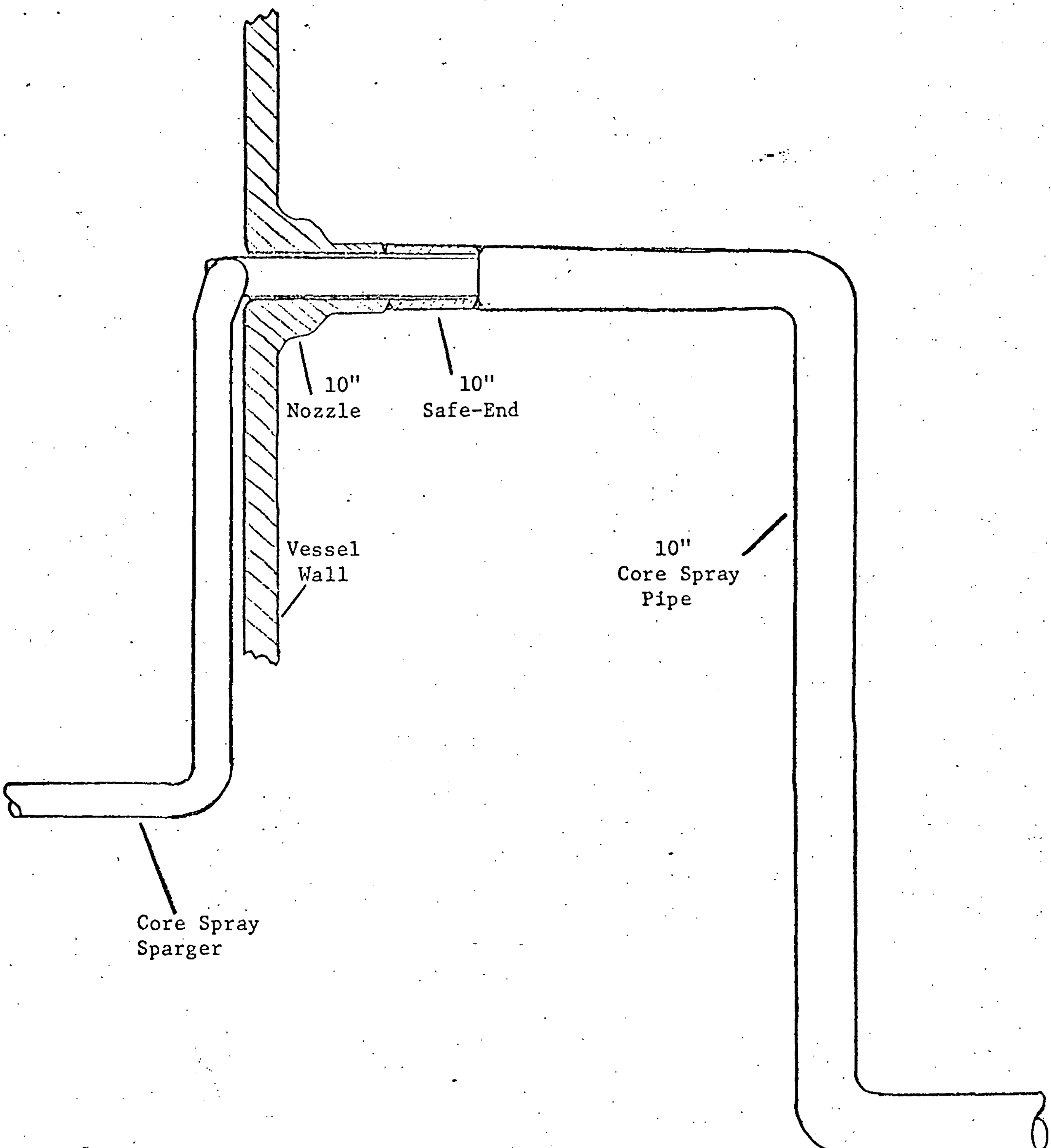
DRESDEN 2 CORE SPRAY SYSTEM

FLUSHING AND FILLING HISTORY

3-5-69 3-10-69	Flush Phase 3 vessel isolated.
6-22-69	Initial fill test pattern.
6-23-69	Filled vessel for hydro w/TSP - treated demineralized water - approx. 1000 ppm TSP.
6-23-69	Filled line for hydro to second isolation valve boundary with TSP - treated water.
7-3-69	Drained lines completely, backflushing from reactor to get rid of TSP.
9-7-69	Filled reactor for CRD tests and recirc. flow tests with water level above highpoint in line. No special filling.
10-30-69	Drained reactor for vibration instruments installation.
11-69	Filled for fueling.
2-16-70	Filled reactor for vessel operational hydro above line.
Prior to power generation	Testable check valve tests with reactor vessel filled.

There were four subsequent cooldowns which raised water level substantially above the high point of line.

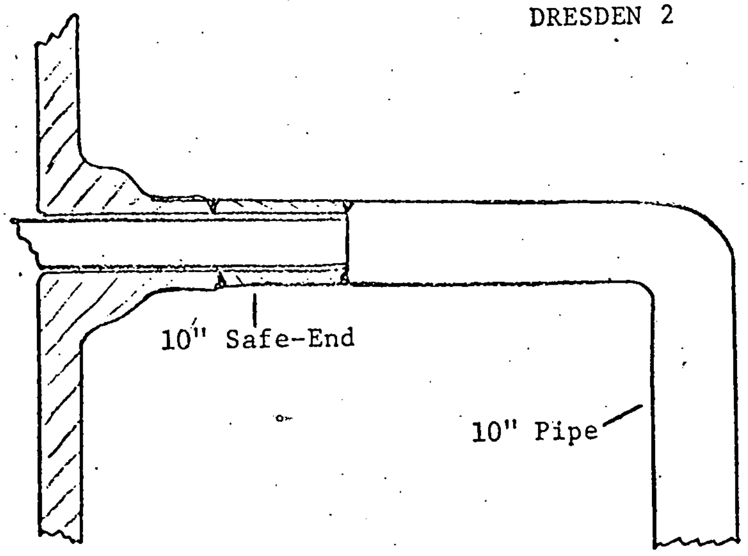
5/22/70



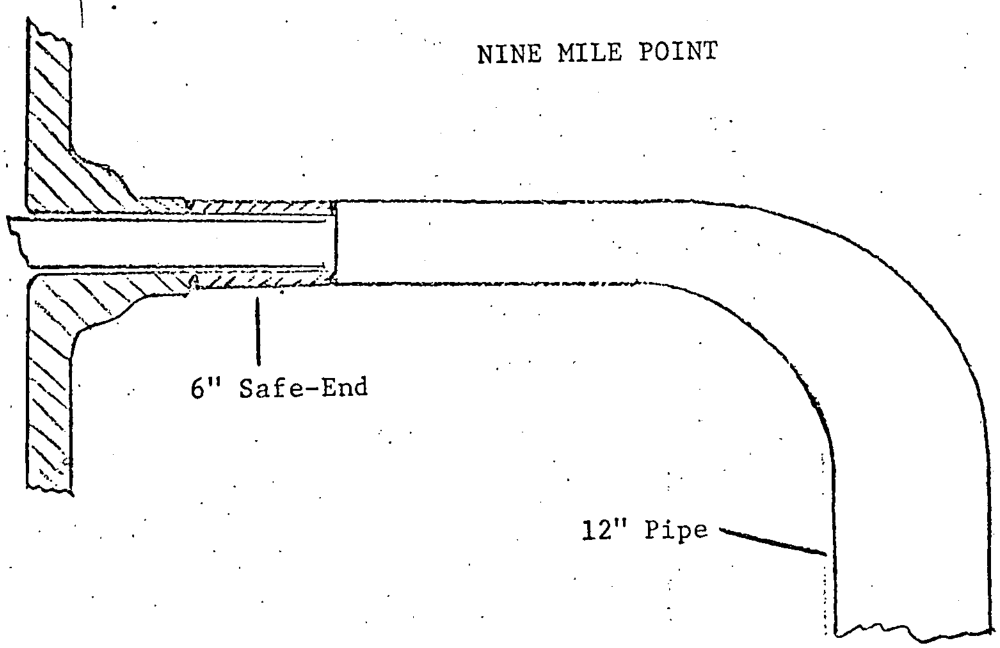
DRESDEN 2

CORE SPRAY CONFIGURATION

DRESDEN 2

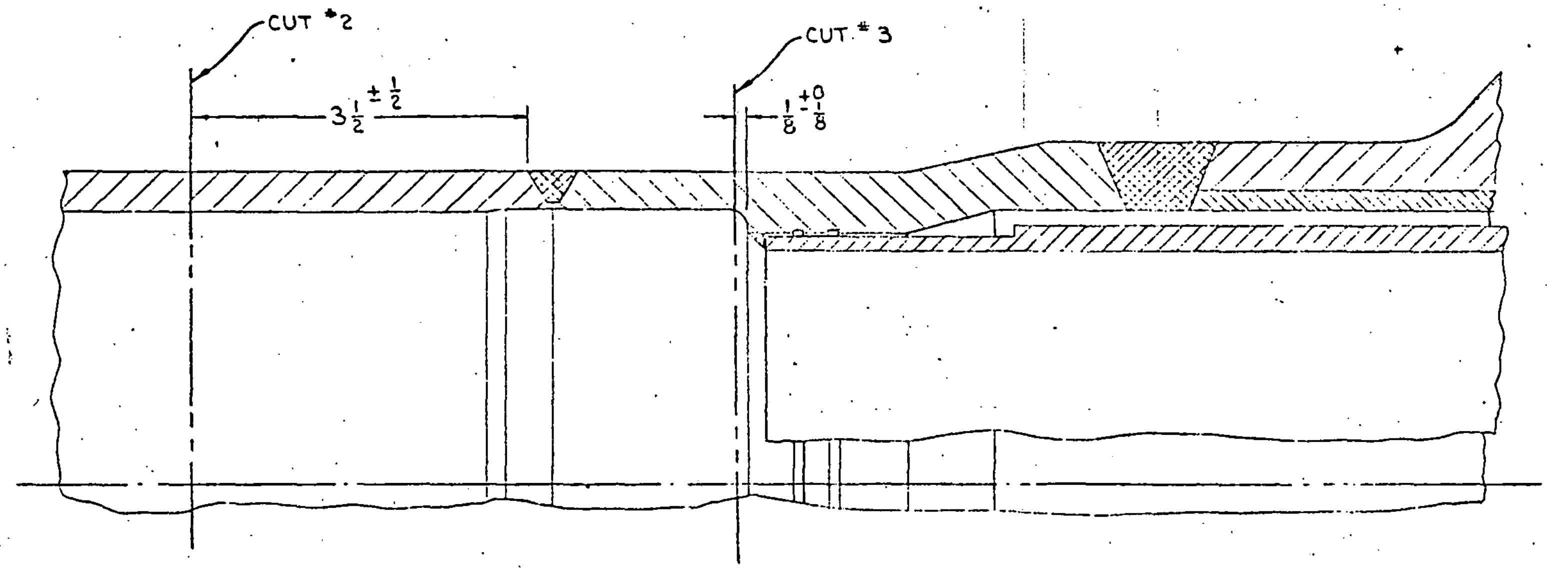


NINE MILE POINT



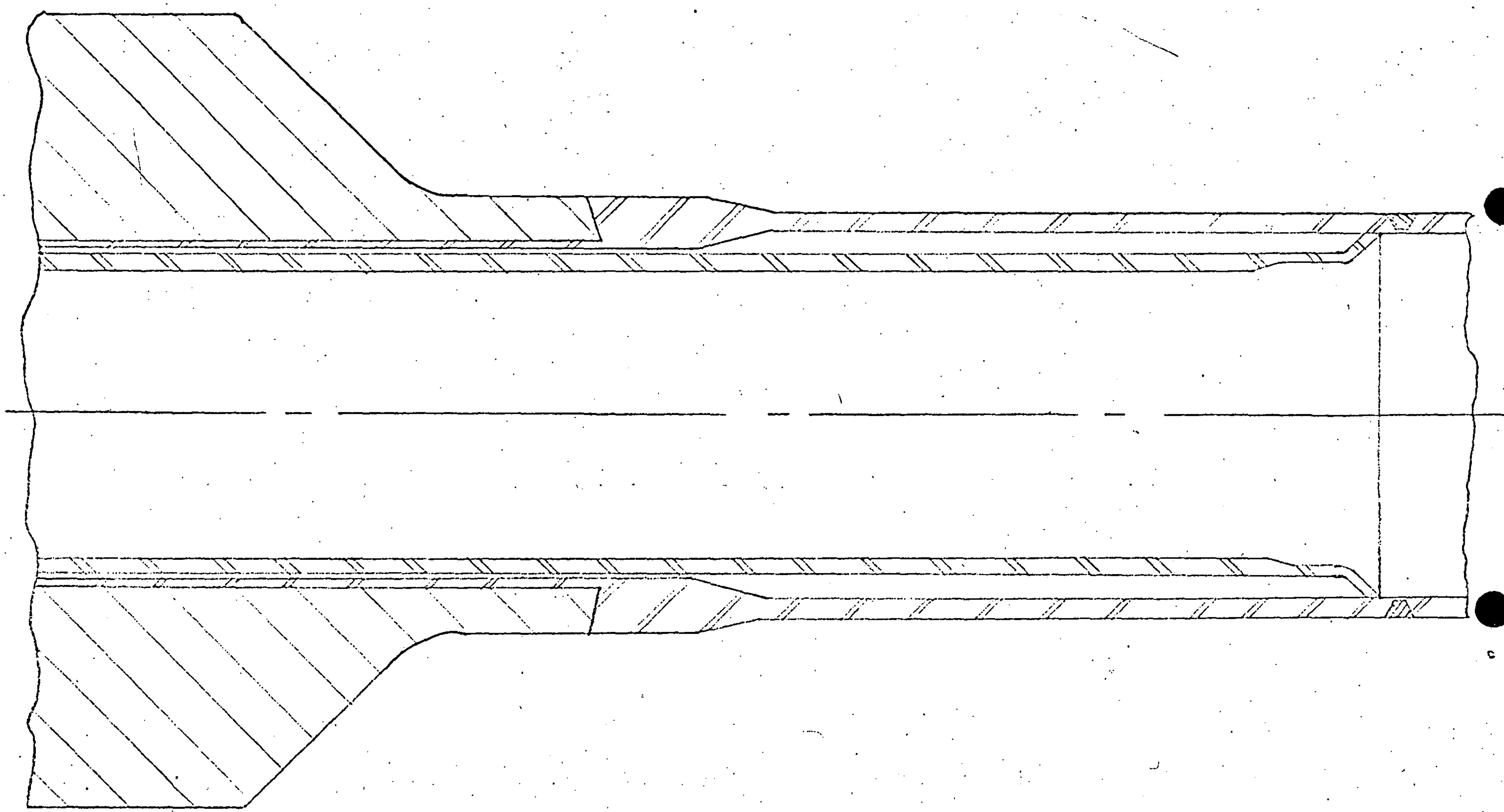
NINE MILE POINT

CORE SPRAY SAFE-END



DRESDEN 2

CORE SPRAY SAFE-END



SENSITIZED SAFE-ENDS

(B31.1 Piping Stresses, Psi)

Nozzle	Steady State Stresses		Expansion Stresses		Occasional Stresses	
	Weight Pressure	Allowable (Sh)	Thermal	Allowable (SA)	Weight Pressure, Seismic	Allowable (1.2Sh)
28" Recirc Outlet	5,850	16,000	2,800	27,400	7,200	19,200
12" Recric Inlet	4,620	16,000	5,850	27,400	6,800	19,200
14" Isolation Condenser	6,250	16,000	7,950	27,400	8,000	19,200
10" Core Spray	4,900	16,000	4,600	27,400	8,600	19,200
3" CRD Hyd. Return	2,900	14,450	8,300	27,000	6,600	17,300
Others with less than 2" lines	3,600	14,450	2,760	27,000	7,300	17,300

(6) Various fixes have been developed, discussed and studied by the vendors' and owners' representatives and their consultants. Many of the answers to other questions reflect these discussions and studies. We have not been able to determine that one nozzle safe-end has greater safety importance than any other. In the end, our discussions and studies have caused us to conclude that if a fix is required, the only fix that will provide the desired assurance is to replace all of the furnace sensitized safe-ends with material which is not sensitized. Our estimated schedule for this fix (replacement of all furnace sensitized safe-ends) is 41.8 weeks. This schedule period will also permit replacement of internal brackets if it is also required.

6/5/70

7 - Identify estimated whole measured oxygen concentrations throughout reactor and in the region of safe ends.

SUMMARY OF D-2 OXYGEN ANALYSIS TO 5/29/70

Abstract:

Oxygen measurements on samples of water in the D-2 reactor systems indicated the following concentration levels:

a. Reactor Recirculation Water: 220-270 ppb

b. Core Spray Header Water: 350-400 ppb

Intermixing of the core spray water with the bulk reactor water was confirmed by gamma spectrometry analysis of samples from the core spray header.

c. Primary Steam: 14-25 ppm

d. Condensate Demineralizer Effluent: 25-35 ppb

The sample of core spray header water was taken from the horizontal portion of the core spray line between the safe end and the elbow where the line turns down.

Summary of D-2 Oxygen Analysis to 5/29/70 (Cont.)

This summary of oxygen measurements performed at D-2 includes the results of analysis on the reactor recirculation water, (feedwater) water from core spray line, steam, and condensate demineralizer effluent.

With the exception of the feedwater analyses which were performed with Hayes Oxygen Analyzer, all measurements were taken using a Beckman Model 735, polarographic oxygen analyzer. The Hayes analyzer is calibrated against its internal reference system which consists of an electrolysis cell which generates known quantities of oxygen in solution while the Beckman analyzer was calibrated against the oxygen concentration of the normal atmosphere or as air saturated water.

The results of these measurements are as follows:

I. Core Spray Line

Analyses were performed on 5/29/70 following installation of a sample cooler on one leg of #1459B Core Spray Differential Pressure Meter. The plant was operating at 437 MW_{th} and over a four-hour period the oxygen concentration slowly decreased from 400 to 350 ppb. (Water was withdrawn at a flow rate of 250 cc/min for a period of five hours.) Measurements of the oxygen concentration of the reactor recirc water performed immediately prior to this time and at the same power level gave an oxygen concentration of 175 ppb. Gamma spectra measurements were also performed on samples of the core spray line water during the oxygen measurements which showed that the activities were similar to the reactor water.

II. Reactor Recirculation Water

The oxygen content of reactor water recirculation header has been recorded for periods of several weeks at power levels up to 75%. The observed oxygen levels generally fall in the range of 220 to 270 ppb with occasional short term periods with concentrations up to 300 ppb.

III. Primary Steam

Oxygen measurements were made on the primary steam on 5/17 at 75% power and 100% recirculation flow and at reactor water levels of 19 and 36 inches as part of the moisture carryon tests. The results indicated a constant radiolytic oxygen content of 14 ppm, independent of water level.

At the same time, recombination experiments on off-gas samples together with the observed off-gas flow indicate the oxygen measurements may be low by as much as a factor of two. Further experimental work to resolve this discrepancy is planned.

IV. Condensate Demineralizer Effluent

During normal operation the oxygen content of the condensate demineralizer effluent has ranged from 16 to 60 ppb. The higher levels are associated with periods of condensate make-up water addition and the most prevalent values are in the range of 25-35 ppb.

Short term increases in O_2 concentration up to 100 ppb are also observed when condensate demineralizer beds are placed in service.

JHH
6/5/70

8 - Discuss and compare the differences between Dresden 2 and Nine Mile Point with respect to cleaning procedures (including PT, stresses, fill, flush and drain procedures, heat treatment, oxygen and chloride limits, layout, and materials) to the extent information is available.

The basic difference between Dresden 2 and Nine Mile Point relative to the above items is as follows:

1. At the time that the Dresden system was erected and cleaned Project management was aware of the Oyster Creek sensitized stainless problem. Procedures were specifically developed for the Dresden 2 erection and cleaning to prevent the contamination of the sensitized stainless surfaces with chlorides and fluorides and to provide for thorough cleaning of these surfaces. The Nine Mile Point system was installed by the time the Oyster Creek problem had been evaluated.

2. The Dresden system was installed clean. Cleaning during installation was accomplished by wipe down with alcohol or acetone, not by the use of water flushing.

3. On initial fill of the vessel at the fabricators plant demineralized water was used for the Dresden 2 vessel whereas tap water was used for the Nine Mile Point vessel.

4. For the shop hydro expanding plugs were used to provide the seal at the nozzles on the Dresden vessel whereas caps were welded to the nozzles for the Nine Mile Point vessel hydro.

5. The stresses on sensitized safe end materials were reviewed and re-evaluated just prior to plant startup. The pipe hanger and restraint system was reviewed and inspected after the Nine Mile Point core

spray safe end failure occurred. In addition, restraints and hangers were inspected during heatup and after full temperature was reached. The design stress and hanger and restraint condition was not reviewed in this detail at Nine Mile Point before startup.

6. Other differences are indicated on the copies of flip charts provided in answer to Question #5.

RJA
6/3/70

(9) Give the reactor power level or operating history at which a loss of coolant accident will not result in fuel failures and hydrogen can be disposed of by purging without significant release of fission products.

Four assumptions must be made for this analysis:

1) The reactor core consists of new fuel.
2) As the power level is reduced, the peak linear heat generation rate is reduced proportionally.
3) Core cooling systems consist of one core spray and two LPCI's.
4) Significant release is meant to mean within 10 CFR 20.

The power level as a function of time is shown below. The temperature listed is the maximum temperature which will occur for the maximum size break. This temperature also represents the minimum temperature required for perforation at each power level/reactor history.

Power	Length of Time at Power	LOCA Temperature Maximum
100%	2.5 months	2000°F
90%	5.5 months	1870°F
80%	14 months	1730°F
70%	48 months	1590°F

The power level can be below 70% for an indefinite amount of time without resulting in clad perforations as a result of a loss of coolant accident.

The fission product release without fuel perforation would consist of the normal off-gas release except that as the gases are released they are filtered by the Standby Gas Treatment System. The holdup time for fission product release could also be increased. Thus, the purging of the hydrogen would not result in a significant release of fission products.

(10) The extent of re-evaluation of piping stresses, independence of review, basic assumptions and field confirmation of as-built configuration.

The criterion utilized for the design of the piping systems connecting to the vessel is the USAS B31.1 piping code. This code requires consideration of operating, thermal and seismic loadings and defines the method of stress combinations and allowables associated with each combination. These piping systems were analyzed and met the code prior to issuance of the operating license in December, 1969. As these systems analyses were completed by the Architect-Engineer, Sargent and Lundy Engineers, the analyses were submitted to the General Electric Company and reviewed by the APED piping unit in San Jose.

As a result of the Nine Mile Point incident, the Dresden 2 piping analyses were completely reviewed by both S&L and GE personnel. The review consisted of an analytical and configuration verification. The analytical review, by members of both companies, consisted of insuring stresses due to the various type loadings were considered. The loads specifically are operating loads of internal pressure and dead weight, seismic and thermal. These loads acting upon the piping systems result in resultant forces and moments on the vessel safe ends. The review checked the configuration utilized in the computer model to assure that the analyzed systems were in accordance with the as-built piping arrangements. In addition, the input data for the computer calculations were reviewed to assure that such items as pipe size and schedule, temperatures, vessel movements and support locations had been properly selected. The results of this review assured that the correct inputs were used and that the resulting output forces and moments were correct. The individual systems considered were also checked to assure that the output piping deflections and forces were those used by the piping hanger designer to design the hangers. The forces and moments were then combined by the GE company to determine stress values at the safe ends. These resulting stresses are shown in the attached Table I. The values are very close to the stress values obtained previously by Sargent & Lundy. In addition to the safe ends listed, the other vessel nozzles that do not have sensitized stainless safe ends were checked in a similar manner. These other systems are the main steam and the feedwater systems.

The stress values shown are computed at the minimum wall position where the piping joins the safe end. The bi-metallic weld where the vessel nozzle and the safe end meet has a thicker section and, therefore, lower stresses than shown. Although not required by B31.1, a direct addition of the thermal, seismic, weight, and pressure stresses results in values that are less than yield.

In addition to the analytical checks, a number of field reviews of the actual hardware was performed for all piping inside the drywell. These reviews consisted of checks of hanger positions and settings both in the hot and cold piping conditions. The checks were to assure that expected movements as predicted by the analytical programs were actually occurring and that the systems were not blocked or restrained in any unusual manner. These checks verified the calculations and hanger settings. The checks were made by the piping hanger contractor, Bergen-Patterson, and at various stages by GE field personnel, GE design personnel, S&L personnel and C.E.Co. site personnel. The inspections resulted in one minor hanger adjustment and removal of one hanger locking pin.

It is therefore concluded that the Dresden 2 plant is safe to operate from the stress standpoint because the stresses are low, the values have been checked and the physical arrangement and movements are as predicted.

TABLE I

SENSITIZED SAFE-ENDS

(B31.1 Piping Stresses, Psi)

Nozzle	Steady State Stresses		Expansion Stresses		Occasional Stresses	
	Weight Pressure	Allowable (Sh)	Thermal	Allowable (SA)	Weight Pressure, Seismic	Allowable (1.2Sh)
28" Recirc Outlet	5,850	16,000	2,800	27,400	7,200	19,200
12" Recric Inlet	4,620	16,000	5,850	27,400	6,800	19,200
14" Isolation Condenser	6,250	16,000	7,950	27,400	8,000	19,200
10" Core Spray	4,900	16,000	4,600	27,400	8,600	19,200
3" CRD Hyd. Return	2,900	14,450	8,300	27,000	6,600	17,300
Others with less than 2" lines	3,600	14,450	2,760	27,000	7,300	17,300

(11) LEAK DETECTION SYSTEM

The sensitivity of detecting leakage of primary system water into the drywell air is such that a leakage of from 1 to 2 ml per minute can be detected in about 10 minutes by the continuous recording air particulate monitor. Larger leakage or change in leakage rate can be detected more rapidly.

At present, the soluble activity in reactor water after 8 hours decay is about 5×10^{-2} uci/ml. The volume of air in the containment is about $5 \times 10^{+9}$ cc. Each ml of reactor water dispersed uniformly in the containment results in an increase in airborne particulate activity of about 1×10^{-11} uci/cc. In 10 minutes with a 1 ml/min leak the airborne activity level in the containment reaches 1×10^{-10} uci/cc. The air particulate monitor will show a rise of about 200 cpm when the air being sampled is 1×10^{-10} uci/cc.

The continuous recording air particulate sampler is a fixed filter type. Each day the filter paper is removed and counted in a proportional counter. The proportional counter can easily determine 5×10^{-6} uci of activity on the filter paper. The volume of air sampled over a 24 hour period is about 3.4×10^{-8} cc. Therefore, the minimum sensitivity of the daily composite is about 2×10^{-14} uci/cc. The normal environmental background levels at Dresden are from 7×10^{-13} uci/cc to 3×10^{-12} uci/cc.

If the continuous recording air particulate monitor is taken out of service for maintenance, a continuous portable air sampler is substituted. On the 24 hour composite from this sampler counted in the counting room, a sensitivity of about 5×10^{-14} uci/cc is possible.

The 22 drywell sample points are arranged to locate the area in which a leak occurs. Each area is sampled once a week. The sample is obtained over a period of one hour by a portable air sampler with a capacity of 2 cfm. These samples are counted in the counting room. The minimum detectable level of this system is about 1×10^{-12} uci/cc of airborne particulate activity.

On the basis of the 24 hour composite samples and the one hour grab samples, a leak of 1 ml./min of reactor water to the containment will easily be detected even if less than 1% of the activity becomes airborne and is picked up on the particulate filters.

At present, 3 of the containment air sample lines are located 120° apart in the area between the reactor vessel and the biological shield. These samples are adequate for good monitoring of all vessel nozzle areas. Whenever a reactor water or steam leak exists within the containment, the air particulate activity will rise on all samples taken from the containment because of the air circulation system design. Normally, the activity level in the vicinity of a leak is 3 to 5 times higher than the average of other areas within the containment.

The normal back ground level of airborne radioactivity within the containment when the plant is operating is about 1×10^{-10} uci/cc. Samples of individual areas in Unit 2 containment indicates that there is a slight seepage of reactor water into the containment from packing gland on valves and pumps. Dresden Unit 1 experience shows that slight leakage through valve and pump packing is to be expected.

The normal practice at Dresden Station is to maintain the airborne particulate radioactivity level within all areas of the plant as low as possible. When a rise of greater than 1×10^{-8} uci/cc in airborne particulate activity is found on the 24 hour composite sample from the containment, a complete set of samples will be taken to determine the area in which the activity level is highest which would indicate the locations of the leak. When the leak area is located, the plant load is reduced (Usually at night) so that entry can be made into the containment. When the leak is located, it is repaired or valved out if possible. If it is not possible to locate the leak, an evaluation will be made and appropriate actions taken.

(12) Table 4.6.1, pages 113 to 118, inclusive, of Appendix A to Operating License DPR-19, Docket No. 50-237 describes in considerable detail the extent of the examinations and inspections that are planned throughout the service life of Dresden 2 reactor coolant pressure boundary materials. Categories F (1 and (2 on page 114 and J on page 116 relate to the safe-end to nozzle welds and the pipe to safe-end welds and category M on page 118 relates to the internals and integrally welded internal supports of the reactor vessel. Responsive to the interest in the condition of furnace sensitized safe-ends we are now seriously considering examining one of the two 28-inch recirculating outlet nozzle safe-ends, one of the ten 12-inch recirculating inlet nozzles and one of the two 10-inch core spray inlets during the first refueling outage. We will, of course, visually examine all internal attachments whose failure may adversely affect core integrity during the first refueling outage as specified on page 118 for these category M parts.

(13) Would the failure of any furnace sensitized stainless steel reactor vessel internal component result in the inability to cool the core?

No. Failure of any furnace sensitized stainless steel component within or exterior to the reactor pressure vessel will not result in the inability to cool the reactor core.

(14) Ten year's experience with Dresden 1 during which cracks have been initiated and propagated to the extent that leaks have occurred and the leaks have been detected by the leak detection system and found by plant personnel plus the finding of reportable flaw indications as well as flaw indications that required removal, during the nondestructive examinations of our inservice inspections which are a part of our surveillance program, has caused us to relate surveillance and leakage sensitivity to crack propagation rates. This relationship has been expressed in our surveillance program for Dresden 2 reactor coolant pressure boundary materials which is detailed in Table 4.6.1, pages 113 to 118, inclusive, of Appendix A to Operating License DPR-19, Docket No. 50-237.