LICENSEE EVENT REPORT (LER)

Form Rev 2.0

Facility Name (1)	Docket Number (2)	Page (3)
Dresden Nuclear Power Station, Unit 2	0 5 0 0 0 2 3 7	1 of 2 7

Title (4)

Heat Damage to Upper Elevation Drywell Components Due to Closed Ventilation Hatches

Event Date (5)			LER Number (6)			Report Date (7)			Other Facilities Involved (8)	
Month	Day	Year	Year	Sequential Number	Revision Number	Month	Day	Year	Facility Names	Docket Number(s)
									N/A	0 5 0 0 0
1 1	1 4	8 8	8 8	0 2 2	0 0	1 2	1 3	8 8	N/A	0 5 0 0 0

OPERATING MODE (9)	N
POWER LEVEL (10)	0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR
(Check one or more of the following) (11)

___ 20.402(b)	___ 20.405(c)	___ 50.73(a)(2)(iv)	___ 73.71(b)
___ 20.405(a)(1)(i)	___ 50.36(c)(1)	___ 50.73(a)(2)(v)	___ 73.71(c)
___ 20.405(a)(1)(ii)	___ 50.36(c)(2)	___ 50.73(a)(2)(vii)	X OTHER
___ 20.405(a)(1)(iii)	___ 50.73(a)(2)(i)	___ 50.73(a)(2)(viii)(A)	(Voluntary Report)
___ 20.405(a)(1)(iv)	___ 50.73(a)(2)(ii)	___ 50.73(a)(2)(viii)(B)	
___ 20.405(a)(1)(v)	___ 50.73(a)(2)(iii)	___ 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

Name	TELEPHONE NUMBER	
	AREA CODE	
Richard H. Johnson, Plant Performance Monitoring Group Leader Ext. 2674	8 1 5	9 4 2 - 2 9 2 0

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS
X	V B	M 0	J 1 2 7	Y	X	A D	L 0 V	B 3 1 4	N
X	A D	L 0 V	B 3 1 4	N					

SUPPLEMENTAL REPORT EXPECTED (14)		Expected Submission Date (15)	Month	Day	Year
X Yes (If yes, complete EXPECTED SUBMISSION DATE)	NO		0 1	2 3	8 9

ABSTRACT (Limit to 1400 spaces, i.e, approximately fifteen single-space typewritten lines) (16)

On November 14, 1988, at 1500 hours, with Unit 2 shutdown for a scheduled refueling outage, the fourth and fifth drywell elevations were determined from various indications to have experienced excessive temperatures during the previous operating cycle. The conditions indicative of high temperatures included degraded electrical cables, peeling and discolored paint, and degraded valve operator components. The apparent cause of this event was attributed to closure of four ventilation hatches in the drywell refueling bulkhead (the ceiling of the fourth drywell elevation) at the end of the previous refueling outage, thereby preventing the ventilation supply from reaching the reactor head area. The root cause of these hatches being inadvertently left closed has been attributed to procedure deficiency. Procedure revisions will ensure that these hatches are left open during power operation. Other corrective actions will include the necessary repairs to damaged cable and valves. An enhanced drywell temperature monitoring system will also be provided. Safety analysis to date has shown minimal safety significance of this event, and this Licensee Event Report (LER) is being submitted voluntarily. There have been no previous occurrences of long-term elevated drywell temperatures at Dresden Nuclear Power Station. An event which occurred on June 5, 1970 documented some short-term elevated drywell temperatures which had occurred on Dresden Unit 2.

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PLANT AND SYSTEM IDENTIFICATION:

General Electric - Boiling Water Reactor - 2527 MWt rated core thermal power.

Nuclear Tracking System (NTS) code numbers are identified in the text as (XXX-XXX-XX-XXXXX).

EVENT IDENTIFICATION:

Heat Damage to Upper Elevation Drywell Components Due to Ventilation Supply Hatches Being Left Closed.

A. CONDITIONS PRIOR TO EVENT:

Unit: 2 Event Date: November 14, 1988 Event Time: 1500 hours

Reactor Mode: N Mode Name: Shutdown Power Level: 0%

Reactor Coolant System (RCS) Pressure: 0 psig

B. DESCRIPTION OF EVENT:

1. Identification of Problem

On November 14, 1988, at 1500 hours, with Unit 2 in a refueling outage, the fourth and fifth drywell elevations were determined from various indications to have experienced excessive temperatures during the previous operating cycle. The conditions indicative of high temperatures included degraded electrical cables, peeling and discolored paint, and degraded valve operator components.

Dresden Unit 2 was shutdown on October 29, 1988, for a scheduled refueling outage. The first indication of heat damage was peeling and discolored paint observed by Mechanical Maintenance Department personnel on October 30, 1988, while removing shield plugs from the top of the reactor well. See Figure 1.

The initial indication of the apparent cause of the heat damage to the paint was seen on October 31, 1988, when the drywell head [NH] was removed and the four ventilation hatches in the refueling bulkhead were found closed. It should be noted that the two manway hatches in the refueling bulkhead were found open. Figure 2 shows the location of six hatches in the refueling bulkhead (the ceiling of the fourth elevation). Normal ventilation to the reactor head area is through the two supply hatches and then downward through the two return hatches and two manway hatches. All six hatches should have been open. Figure 3 shows drywell cooling system temperature data from various thermocouples located in the drywell. Figures 4 and 5 show details of air flow to the fourth drywell elevation and to the fifth drywell elevation (reactor head area).

Another early indication of a problem was when the Reactor Head Vent Valves A02-220-46 and A02-220-47 did not open during shutdown for the refueling outage. The difficulty in opening these valves was later attributed to excessive temperatures at the location of the valves (the fourth drywell elevation). Work requests were submitted to repair the reactor head vent valves.

Additional indications of heat damage were seen during an initial inspection of the third and fourth drywell elevations by Operating, Maintenance, and Radiation Protection Department personnel. Large amounts of red dust were noted in some areas; sealtite jacket damage, and leakage of grease from valve operators were also noted. Peeling and discolored paint was observed in the vicinity of the reactor bulkhead separating the fourth and fifth drywell elevations. Although past steam leaks have resulted in localized iron oxide deposits similar to the red dust noted, the quantity and wide distribution of red dust had not been seen before.

Based on the indications given above, the existance of excessive upper drywell elevation temperatures during the previous Unit 2 operating cycle were suspected by Mechanical Maintenance Department personnel. Inspections of the two Environmentally Qualified (EQ) valves on the fourth drywell elevation were initiated to determine if temperatures had been high enough to cause significant degradation. The inspection results for these two valves, Isolation Condenser [BL] inlet steam isolation valve MO2-1301-1 and High Pressure Coolant Injection (HPCI) [BJ] steam supply inboard isolation valve MO2-2301-4, are summarized in Tables 1 and 2. External cables were found cracked and brittle and the grease in the valve operators was low, but serviceable. The lower main bearing of the valve operator for MO2-2301-4 was found broken.

In response to the indications of heat damage, a multidisciplinary group of Station, Commonwealth Edison Corporate, and Architect Engineer personnel initiated an expanded investigation of drywell conditions on November 10, 1988. On November 14, 1988, following a review of the preliminary findings, Station management decided to prepare this voluntary report to be submitted as a Licensee Event Report.

2. Relevant Events During the Past Operating Cycle

a. Refueling Bulkhead Hatches

In order to load or unload fuel, the drywell head must be removed, and the reactor well flooded. Under these conditions, the fifth drywell elevation (reactor head area) becomes part of the reactor well; therefore, penetrations in the refueling bulkhead floor of the fifth drywell elevation (i.e., ventilation hatches, manway hatches, and hydrogen sensing lines) must be sealed watertight prior to flooding the reactor well. Refueling outage 10 for Dresden Unit 2 began on November 29, 1986, and ended on May 2, 1987. Installation of the drywell head [NH] began March 24, 1987, and was completed on March 27, 1987, utilizing Dresden Maintenance Procedure (DMP) 1600-5, Revision 2, Drywell Head Replacement and Installation of Shield Blocks. Step 3 of this revision states "OPEN all required ventilation openings. Remove pipe caps from the two hydrogen sensing lines 2(3)-2401-AB-1/2"HB and 2(3)-2401-BC-1/2"HB." As discussed in Section C, Apparent Cause of Event, this step was only partially completed. The two manway hatches were opened and the pipe caps were removed, but the four ventilation hatches were not opened. The fact that these ventilation hatches had been left closed appears to be the apparent cause of the excessive temperatures on the fourth and fifth drywell elevations.

As inspection of the drywell [NH] must be conducted prior to reestablishing Primary Containment whenever Primary Containment is broken and personnel have entered the drywell. This inspection is documented using Dresden Operating Surveillance (DOS) 1600-10, Pre-Startup Drywell Inspection Plan. Step 7 of DOS 1600-10, Revision 4, states "VERIFY that the hatch doors to the reactor head area are OPEN." Table 3 lists the entries for this step of DOS 1600-10, Revision 4, for the surveillance completed following refueling outage 10 (conducted April 19, 1987) and subsequent drywell entries made during the course of the operating cycle.

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As discussed in Section C, Apparent Cause of Event, this step was only partially completed at the end of refueling outage 10, since only the manway hatches were verified to be open. The apparent cause of the ventilation hatches not being verified open is that DOS 1600-10, Revision 4, Step 7, did not explicitly include the ventilation hatches.

b. Drywell Temperature Indications

During the last Unit 2 operating cycle, drywell temperatures were monitored using thermocouples [IM] on the lower drywell elevations. Area temperature indications for the third, fourth, and fifth drywell elevations were not available. Figure 3 shows temperature data from six thermocouples for the weeks of March 27, 1987 through October 20, 1988. High temperature readings in the north main steam line relief valve area of the second drywell elevation (point T10) were recorded in June 1987, with a peak value of 206 degrees F recorded on June 12, 1987. These high readings were attributed to the installation of a new temperature element for point T10 at a different location. This change was made during refueling outage 10 under Modification M12-2-86-12. The new temperature element is located near the Main Steam Lines (MSLs). Other drywell area thermocouples replaced and relocated under this modification showed both increases and decreases of similar magnitude when compared with readings obtained during the previous operating cycle.

In general, peak temperatures for 1988 were approximately equal to those for 1987. One exception was the temperature reading in the south section of the second elevation (point T9). This reading showed a step increase of approximately 22 degrees F for the week of December 17, 1987, and remained high throughout 1988. The cause of this step increase was attributed to 2A drywell cooler [VB] (one of seven) being taken out of service due to failure of blower motor 2-5788-A on December 15, 1987. The failure of the blower motor was caused by water leaking from a pipe union on the bottom head drain line.

c. Observation of Red Dust on Fourth Drywell Elevation

During a radiation survey of the fourth drywell elevation on August 26, 1987, red dust was noted in the vicinity of Standby Liquid Control [BR] inboard isolation valve 2-1101-1 but not in other areas. Figure 2 shows the floor plan of the fourth drywell elevation and includes major steam lines. Dresden Unit 2 was shutdown for a scheduled dual unit maintenance outage from May 14, 1988 to May 29, 1988. During the dual unit outage, a radiation survey of the fourth drywell elevation on May 18, 1988, noted red dust between valves M02-1301-1 and M02-2301-4. As discussed in Section C, Apparent Cause of Event, the major source of the red dust was apparently the excessive temperatures in the upper drywell elevations.

d. Operation and Maintenance of Valves on the Fourth Drywell Elevation

Although degradation of valves M02-1301-1 and M02-2301-4 was found following Unit 2 shutdown as previously discussed, the operating history and maintenance work performed during the previous operating cycle did not give earlier indications of problems with these valves. Similarly, degradation of valves A02-220-46 and A02-220-47 was also not identified prior to Unit shutdown.

Valves M02-1301-1 and M02-2301-4 are included in Dresden Operating Surveillance (DOS) 1600-1, Quarterly Valve Timing. Tables 4 and 5 give the valve timing results since January 1987 for these valves. No significant changes in stroke times were seen during the operating cycle other than those attributable to timing the valve utilizing the "switch-to-light" method rather than the "light-to-light" method for valve M02-1301-1. Valve M02-2301-4 appears to have a stroke time on August 29, 1988, that was approximately 0.7 seconds longer than the "switch-to-light" times measured previously on October 13, 1987. All valve timings were well below the alert limits given in Tables 4 and 5.

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Valve MO2-1301-1 was cycled monthly during the performance of Dresden Instrument Surveillance (DIS) 1300-2, Unit 2 Isolation Condenser Steam/Condensate Line High Flow Calibration. During part of the operating cycle, this surveillance was performed using Special Procedure (SP) 87-9-151. Valve MO2-1301-1 was last closed at approximately 1000 hours on October 30, 1988, when it was routinely taken out-of-service in order to support the current refueling outage.

Valve MO2-2301-4 was cycled twice monthly, at approximately two week intervals, during the performance of DIS 2300-1, HPCI Steam Line High Flow Isolation Master Trip Unit Calibration. Valve MO2-2301-4 was also cycled at least four times per month, at approximately one week intervals, during the performance of DIS 2300-3, HPCI Turbine Permissive (Reactor Pressure Greater than 90 psig) Master Trip Unit Calibration. Valve MO2-2301-4 was also cycled monthly during the performance of DOS 2300-1, HPCI MO Valves and Pump Operability - Monthly. Valve MO2-2301-4 was last closed at approximately 0430 hours on October 30, 1988, when it was routinely taken out-of-service in order to support the current refueling outage.

Reactor Head Vent Valves AO2-220-46 and AO2-220-47 did not open during shutdown for the refueling outage. Work requests were submitted to repair the reactor head vent valves.

Maintenance histories for valves MO2-1301-1, MO2-2301-4, AO2-220-46 and AO2-220-47 from January 1987 through October 1988 are summarized in Tables 6 through 9.

C. APPARENT CAUSE OF EVENT:

1. Root Cause

The apparent cause of the excessive temperatures on the fourth and fifth drywell elevations was attributed to lack of forced ventilation to the fifth elevation (the reactor head area) that resulted from the ventilation hatches being closed. The root cause of the ventilation hatches being left closed is attributed to deficiencies in two procedures, DMP 1600-5, Revision 2, Drywell Head Replacement and Installation of Shield Blocks, and DOS 1600-10, Revision 4, Pre-Startup Drywell Inspection Plan. If either procedure had given adequate guidance on reactor head area ventilation requirements, this event could have been avoided.

As stated earlier, normal ventilation to the reactor head area is through the two supply hatches and then downward through the two return hatches and two manway hatches. This is illustrated in Figures 4 and 5. The restricted ventilation supply to the reactor head area is believed to have allowed local temperatures to approach that of the reactor vessel head. Heat would then be carried upward to the concrete shield plugs by a combination of thermal conduction through the drywell head, thermal radiation, and natural convection. Heat would similarly be carried downward to the fourth drywell elevation. The normal downward air flow from the reactor head area is a major air supply to the fourth drywell elevation. Hot air issuing upward from between the reactor vessel and the biological shield which ends at the fourth elevation is a second source of heat to the fourth drywell elevation. The restriction of normal air flow within the fourth drywell elevation is also believed to have contributed to excessive local temperatures.

DMP 1600-5, Revision 2, was in effect during the installation of the Unit 2 drywell head. Step 3 of this revision states "OPEN all required ventilation openings. Remove pipe caps from the two hydrogen sensing lines 2(3)-2401-AB-1/2"HB and 2(3)-2401-BC-1/2"HB." This step did not clearly identify which hatches were required to be open. Only the manway hatches were found open. The pipe caps were removed prior to installation of the drywell head. This is evidenced by the installation of pipe caps in order to flood the reactor well for fuel unloading in November 1988. In contrast, DMP 1600-5, Revision 3, Step 7 states "Open the four ventilation and the two manway hatches and wire in the open position". This requirement is also found in DMP 1600-5, Revision 4, which is presently in effect. Although Revision 3 of DMP 1600-5 was initiated in late 1986, On-Site Review was not completed and Revision 3 was not issued for use until March 27, 1987.

DOS 1600-10, Revision 4, was in effect during the past Unit 2 operating cycle. Step 7 of this revision states "VERIFY that the hatch doors to the reactor head area are OPEN". In total, seven different Shift Supervisors performed the DOS 1600-10 inspections listed in Table 3. Four of the individuals did not inspect the hatches to the reactor head area and either marked Step 7 "N/A" or initialed that step. Reasons given were that climbing to the upper drywell elevations is prohibited for drywell entries at power and that there was no need to check the hatches at the end of a short outage if no work had been done on the fourth elevation. Each of the three Shift Supervisors that had inspected the manway hatches stated that Step 7 of DOS 1600-10 had been interpreted as applying only to the manway hatches. Consequently, the ventilation hatches were not checked either during the initial inspection at the end of refueling outage 10 or during subsequent inspections during the operating cycle. The apparent cause of the ventilation hatches not being verified open is that DOS 1600-10, Revision 4, Step 7, did not explicitly include the ventilation hatches. It should be noted that entry to the reactor head area to inspect the ventilation hatches at the end of a short outage would only be possible if the reactor vessel flange had been cooled down.

2. <u>Contributing Causes</u>

As stated above, Revision 3 of DMP 1600-5, Drywell Head Replacement and Installation of Shield Blocks, was not approved and issued for use until after installation of the drywell head. Had Revision 3 of DMP 1600-5 been approved and issued for use earlier, the refueling bulkhead ventilation hatches may have been properly opened. Therefore, the long interval between initial preparation of this revision and approval and issuance of this revision is judged to be a contributing cause of this event.

The ventilation return hatches in the refueling bulkhead were found closed. This may have been a contributing cause of this event, but the open manway hatches have a larger area than the supply hatches. Therefore, the open manway hatches should have provided adequate ventilation return had the supply hatches been open.

A drywell ventilation walkdown found instances of degraded dampers and ductwork. An inspection door for a duct was found open (approximately two square feet) and a hole (approximately three square feet) was found in the drywell basement. The hole and open inspection door would have allowed a bypass of air from the upper portion of the drywell ventilation system. In addition, recirculation of air would have occurred through the 2A drywell cooler after the blower motor failed on December 15, 1987. Furthermore, taking the 2A drywell cooler out-of-service further reduced the cooling capacity of the drywell ventilation system. An analysis is in progress to determine if any of these conditions contributed significantly to the excessive temperatures in the upper drywell elevations. Preliminary results indicate that the degraded dampers and ductwork in addition to loss of the 2A drywell cooler were contributing causes, but that none were the root cause of the excessive temperatures.

Known steam and water leaks had occurred inside the drywell during the early part of the past operating cycle. These were subsequently repaired. Based on the volume of water pumped from the drywell sumps, however, the magnitude of the leakage was not unusually high. Therefore, the steam and water leaks may have been a contributing cause, but do not appear to be the root cause of the excessive temperatures. An analysis will be conducted to determine the equivalent steam leak size necessary to cause the excessive temperatures.

A walkdown of reactor vessel and steam line insulation noted some instances of gaps, misalignments, and missing pieces. An analysis is in progress to determine if any of these conditions contributed significantly to the excessive temperatures in the upper drywell elevations. Preliminary results indicate that the condition of the insulation may have been a contributing cause, but was not the root cause of the excessive temperatures.

3. Cause of the Red Dust

The red dust was investigated to determine if it was a result of the excessive temperatures or due to some other cause. Review of preliminary analysis results concluded that the red dust is primarily iron oxide. The possibility of a steam leak being the source of the iron oxide was investigated, but radiation readings indicated that the bulk of the iron oxide had not been carried out of the piping by a steam leak. Also, based on the volume of water pumped from the drywell sumps, the magnitude of the leakage was not unduly high.

The lower part of the inside of the drywell head was found to be corroded. The underside of the manway hatches and ventilation hatches and parts of the refueling bulkhead were also found heavily corroded. It was concluded that the red dust in the vicinity of valve M02-1301-1 and M02-2301-4 was mainly rust from the refueling bulkhead and rust from the reactor head area through the south manway. This conclusion is supported by areas of red dust found on ledges immediately below the south manway. The red dust near valve 2-1101-1 is believed to be a combination of iron oxide from a steam leak and rust from the refueling bulkhead and drywell head.

The high amount of rust formation is attributed to excessive temperatures which caused paint on the inside of drywell head and on the refueling bulkhead between the fourth and fifth drywell elevations to blister and peel, thus exposing the steel. Tests have shown that a rapid temperature rise to approximately 385 degrees F will cause similar peeling of the paint used on the drywell head. Further testing will be performed to determine if long-term exposure of the paint to lower temperatures would also cause peeling. The paint used on the refueling bulkhead will also be tested in an attempt to establish a lower limit on the temperature at which paint peeling could have occurred.

4. Cause of the Failure of the Reactor Head Vent Valves

Inspection of valves A02-220-46 and A02-220-47 revealed that the operator diaphragms were slightly hardened when compared to new diaphragms, and a tear was present in the A02-220-47 operator diaphragm. Heat is believed to be the cause of the apparent degradation of the diaphragms. The condition of the diaphragms has not been clearly established as the cause of the valve failure. Further testing and inspection during repair of the valves is expected to identify the actual failure mechanism. Failure of the reactor head vent valves is not necessarily indicative of unusually high upper elevation temperatures; these operator diaphragms have needed replacement in the past as discussed in Section G, Component Failure Data.

D. SAFETY ANALYSIS OF EVENT:

1. Significance for Environmentally Qualified (EQ) Equipment

The safety significance of this event with respect to operability of EQ equipment is deemed minimal because no EQ equipment is located on the fifth elevation of the drywell and since EQ equipment on the fourth elevation of the drywell was tested and operated satisfactorily during the operating cycle and following Unit shutdown.

However, excessive temperatures may have prematurely aged EQ equipment. Grease samples from valve operators MO2-1301-1 and MO2-2301-4 were analyzed by the grease manufacturer and were determined to have experienced temperatures in the range of 225 to 250 degrees F. Further analysis of grease samples will be performed, including samples from EQ valve operators from lower drywell elevations. Upon determination of the temperature profile within the drywell, the impact of excessive temperatures on EQ equipment lifetimes will be assessed to determine if lifetime reduction occurred or equipment replacement is warranted.

2. <u>Significance of Apparent Degradation of Reactor Head Vent Valves</u>

The reactor head vent valves are located on the fourth elevation of the drywell and did not open during Unit shutdown. However, these valves are not EQ. The apparent degradation of these valves is concluded to have minimal safety significance.

3. <u>Significance for Primary Containment Integrity</u>

a. <u>Steel Containment Vessel and Drywell Head</u>

Possible material effects of excessive temperatures on the steel containment including the drywell head were investigated. The material used is ASME SA212, Grade B. This material is no longer available, but is very similar to SA516 Grade 70. The physical properties of these steel materials will not change unless the materials are exposed to temperatures in excess of approximately 650 degrees F. The reactor head area temperature could not exceed the main steam temperature of approximately 540 degrees F. Therefore, the excessive upper drywell elevation temperatures will have no short-term or long-term effect on the integrity of the steel.

Possible thinning of the drywell head due to rusting was also investigated. Thickness measurements were conducted using the ultrasonic method. Measurements were performed at 106 locations. A review of the original shop drawings and the standards to which the mill plates were purchased concluded that the actual plate thickness is within the tolerance band of the nominal plate thickness.

Excessive upper drywell elevation temperatures may have caused thermal stresses on the steel containment and drywell head. Upon determination of the temperature profile within the drywell, structural calculations will be performed to determine if any stresses exceeded code limits.

b. <u>Seal Between Steel Containment Vessel and Drywell Head</u>

The hold-down bolts connecting the drywell head to the steel containment are made from SA320, Grade L7 steel. During the fabrication these bolts were quenched at temperatures of 850 degrees F or more. The physical properties of these bolts will not change unless they are exposed to temperatures in excess of approximately 850 degrees F. Therefore, the excessive upper drywell elevation temperatures will have no short-term or long-term effect on the integrity of the hold-down bolts.

Tests of the drywell head flange double tongue and groove seal design were carried out at the Idaho National Engineering Laboratory under Nuclear Regulatory Commission sponsorship. Tests of this seal design used the same gasket material as used for Dresden Units 2 and 3. The test report (NUREG/CR-4944, Containment Penetration Elastomer Seal Leak Rate Tests, July 1987) concluded that the failure temperatures for this seal design were in excess of 700 degrees F in a nitrogen atmosphere. The report included tests using gasket material that had been thermally aged at 300 degrees F for 168 hours. The report concluded that the effects of thermal aging on failure temperatures were "nearly insignificant". The safety significance of the excessive upper drywell elevation temperature is therefore deemed minimal with respect to the integrity of the drywell head flange seal.

4. <u>Significance for Loss of Coolant Accident (LOCA) Analyses</u>

Initial drywell temperatures are assumed in the LOCA analysis for maximum drywell pressure and temperature. If the bounding assumptions for initial drywell temperatures are found to be exceeded upon determination of the temperature profile within the drywell, then an analysis for the drywell under LOCA conditions would be performed and discussed in the planned supplemental report. Preliminary review of LOCA analyses by fuel vendors indicates that the drywell temperature is not a parameter used in nuclear fuel reload safety analyses.

5. <u>Significance for Continued Operation of Dresden Unit 3</u>

The applicability of this problem to Dresden Unit 3 was investigated. The Unit 3 drywell pre-startup inspection at the end of the most recent refueling outage was performed by the Unit Operating Engineer who was accompanied by an NRC Resident Inspector. The personnel involved stated that they remembered climbing into the drywell head area to visually verify that the ventilation hatches were open. In addition, DMP 1600-5, Revision 3, Drywell Head Replacement and Installation of Shield Blocks, was in effect when the Unit 3 drywell head was last installed. The conclusion that this excessive temperature problem was not applicable to Unit 3 was later confirmed by an inspection of the upper elevations of the Unit 3 drywell on November 28, 1988. The ventilation and manway hatches were found open and no indications of excessive temperatures were noted.

E. <u>CORRECTIVE ACTIONS</u>:

The immediate corrective action was the formation of the task force to further investigate the problem and to recommend repairs. A walkdown of steam line insulation was performed. A walkdown of the drywell ventilation system was also conducted. A sample of the red dust was analyzed. Ultrasonic thickness measurements of the drywell head were performed.

Inspection results for valve MO2-1301-1 are summarized in Table 1; this valve will be overhauled prior to Unit 2 startup (237-200-88-13701). Inspection results for valve MO2-2301-4 are summarized in Table 2; this valve will be overhauled prior to Unit 2 startup (237-200-88-13702). Further analysis of grease samples from valve operators on lower drywell elevations will be performed prior to Unit 2 startup (237-200-88-13703). Valves AO2-220-46 and AO2-220-47 will be repaired as necessary prior to Unit 2 startup (237-200-88-13704).

Snubbers 2-1403-30 and 2-1404-31 were removed from the third drywell elevation and tested in order to verify operability. Although these snubbers performed satisfactorily in the operability test, one of the snubbers will be disassembled and inspected for degradation of any parts. New snubbers will be installed prior to Unit 2 startup (237-200-88-13705).

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A study was initiated to determine the temperature profile within the drywell based on available thermocouple readings and an analytical model (237-200-88-13706). If the temperature profile exceeds the bounding assumptions in the Final Safety Analysis Report, then the temperature profile will be used to determine if any code stress limits for the steel containment and drywell head were exceeded (237-200-88-13707). If the temperature profile exceeds the bounding assumptions in the Final Safety Analysis Report, then analysis of the drywell under LOCA conditions will be performed and nuclear fuel safety analyses will be reviewed to determine any safety significance (237-200-88-13708). If the temperature profile indicates that temperature assumptions for any EQ equipment were exceeded, then that equipment will be analyzed to determine the need to reduce the lifetime of that equipment or to repair or replace that equipment (237-200-88-13709). These corrective actions related to the temperature profile will be completed prior to Unit 2 startup, and any additional analyses or corrective actions will be discussed in the planned supplemental report.

Inspection results for electrical cables are summarized in Table 10. Other inspection results for lighting cables, junction boxes, bellows leak-off switches, and thermocouples are summarized in Table 11. Repairs are expected to be needed primarily on the fourth drywell elevation. Necessary repairs of Safety-Related cables will be made prior to Unit 2 startup and will be based on cable testing and inspection and on EQ considerations (237-200-88-13710). Other necessary cable repairs will be performed, but some (e.g., lighting cables) may be scheduled for the next Unit 2 refueling outage (237-200-88-13711).

Analysis of the effects of the as-found condition of the drywell ventilation system will be completed prior to Unit 2 startup (237-200-88-13712). The 2A drywell cooler blower motor 2-5788-A will be repaired or replaced prior to Unit 2 startup (237-200-88-13713). A drywell ventilation line up procedure will be prepared and dampers will be properly positioned prior to Unit 2 startup (237-200-88-13714). The hole found in the drywell basement ventilation duct will be repaired prior to Unit 2 startup under Work Request 80376 (237-200-88-13715).

Analysis of the effects of degraded reactor vessel and steam line insulation will be completed prior to Unit 2 startup (237-200-88-13716). Reactor vessel and steam line insulation will be repaired as necessary prior to Unit 2 startup (237-200-88-13717).

Further analysis of the red dust will be performed prior to Unit 2 startup (237-200-88-13718).

The drywell head, refueling bulkhead, and steel containment will be repainted as necessary; some or all of the repainting may be scheduled for the next Unit 2 refueling outage (237-200-88-13719).

The root cause of the ventilation hatches being improperly left closed was addressed by revising DOS 1600-10 on December 6, 1988, to clearly specify the required verification that the two manway hatches, two ventilation supply hatches, and two ventilation return hatches in the refueling bulkhead are open. In addition, the Drywell Head Installation Checklist in DMP 1600-5 will be revised prior to Unit 2 startup to include a sign-off that the ventilation and manway hatches were left wired open (237-200-88-13720).

An enhanced drywell temperature monitoring system will be provided to monitor temperatures in the Unit 2 upper drywell elevations prior to Unit 2 startup (237-200-88-13721).

An enhanced drywell temperature monitoring system will be provided to monitor temperatures in the Unit 3 upper drywell elevations. Part of this system should be available to begin recording some upper elevation temperature indications later in the current operating cycle. Work on the enhanced system will be completed during the next Unit 3 refueling outage (237-200-88-13722).

A supplemental report will be submitted to the Nuclear Regulatory Commission at least two weeks prior to Unit 2 startup. The supplemental report will summarize the results of the task force investigation and will summarize repair work either completed or scheduled as a result of this event (237-200-88-13723).

F. PREVIOUS OCCURRENCES:

There have been no previous occurrences of long-term elevated drywell temperatures at Dresden Nuclear Power Station.

A previous occurrence in which Dresden Unit 2 had experienced short-term elevated drywell temperatures was reported in "Special Report of Incident of June 5, 1970" and "Supplementary Information to Special Report of Incident of June 5, 1970" on Docket 050237. During that event the drywell experienced a transient approximately one hour in duration that was conservatively calculated to peak at 320 degrees F and 20 psig. Because the drywell design conditions were 281 degrees F at 62 psig maximum pressure, these reports include discussion of analyses of the drywell for a maximum temperature of 340 degrees F at low pressure conditions.

G. COMPONENT FAILURE DATA:

The 2A drywell cooler blower motor 2-5788-A failure was attributed to an electrical fault caused by water leaking into the motor. A search of the Nuclear Plant Reliability Data System (NPRDS) data base revealed no previous failures at Dresden Nuclear Power Station. An industry-wide search of drywell cooler blower motors revealed 72 reported failures, six of which were attributed to grounds or shorts.

Reactor head vent valves are not reportable to NPRDS. A review of the maintenance history for valve A02-220-46, however, revealed that the operator diaphragm had been replaced under work requests with completion dates of June 25, 1984, and December 12, 1986. A review of the maintenance history for valve A02-220-47 revealed that the operator diaphragm had been replaced under work requests with completion dates of June 25, 1984, May 30, 1986, and December 12, 1986.

Manufacturer	Nomenclature	Model Number	Mfg. Part Number
Joy Mfg. Co.	2-5788-A Drywell Cooler Blower Motor	D-100-60	N/A
Blaw-Knox Co.	A02-220-46 Reactor Head Vent Valve	5K365YK242	N/A
Blaw-Knox Co.	A02-220-47 Reactor Head Vent Valve	5K365YK242	N/A

TABLE 1

SUMMARY OF INSPECTION RESULTS
ISOLATION CONDENSER VALVE M02-1301-1
INLET STEAM ISOLATION VALVE

Work Request	Results
79811	Valve operator was disassembled. The grease appeared to have broken down and was oil-like. The spring pack bearing was very loose and worn. All parts appeared to have been blackened. The worm had brass carry over discoloration. The paint was found blistered. The gaskets were hard and baked on. The quad rings and O-rings remained soft and appeared to be in good shape.
79812	Weepage of valve operator lubricant noted. The sealtite jacket was damaged. T-drains were filled with oil. Debris and oil noted in limit switch compartment cover; and compartment seal needed replacing. Insulation degradation noted for field control wiring, internal jumpers, field power cables, and motor leads. Geared limit switch showed signs of dirt and corrosion. The external wire from the geared limit switch had degraded and needed to be replaced.
79935	Cables 32556 and 32558: Cable jackets burned and cracked at junction boxes 1 and 2. Cable jackets looked good at junction box 3. Some oxidation or corrosion noted at terminals.

TABLE 2

SUMMARY OF INSPECTION RESULTS
HIGH PRESSURE COOLANT INJECTION VALVE MO2-2301-4
STEAM SUPPLY INBOARD ISOLATION

Work Request	Results
79809	Valve operator was disassembled. The grease was black and had a burnt smell. The grease was oil-like with some small grease spots. The race of the lower main bearing had broken and the bearing was found in pieces. The quad rings and O-rings were flattened in spots, apparently from heat. All bearings were bad. The main worm shaft was worn and slightly twisted. The worm had brass attached to it. The worm gear was worn. The center hub was worn.
79810	Sealtite damaged. T-drains had oil in drain holes. Dirt and oil noted in limit switch compartment cover; compartment seal needed replacement. Excessive grease leakage observed for limit switch compartment. Insulation degradation noted for field control wiring, internal jumpers, and field power cables. Geared limit switch showed signs of dirt, excessive wear, and corrosion. Geared limit switch grease showed evidence of contamination and was described as medium hard, which was unacceptable. The external wires of the geared limit switch wires and lugs were damaged. The torque switch showed signs of excessive heat. The motor showed some grease leakage, and the pinion gear was loose.
79934	Cables 22628 and 22630: Cable jackets cracked at junction box 1, apparently from excessive heat. Some oxidation or corrosion noted at terminals.

TABLE 3

INSPECTION OF DRESDEN UNIT 2 DRYWELL BULKHEAD HATCHES
DURING THE PAST OPERATING CYCLE USING DOS 1600-10, STEP 7

DATE	RESULTS
04-19-87	Initialed
07-19-87	Initialed
08-27-87	Initialed
09-21-87	Marked N/A
09-22-87	Marked N/A
12-21-87	Marked N/A
12-24-87	Marked N/A
03-19-88	Initialed
04-14-88	Initialed with note "never went above 1st landing"
04-27-88	Marked N/A
05-27-88	Marked N/A

TABLE 4

VALVE TIMING RESULTS
M02-1301-1
JANUARY 1987 THROUGH OCTOBER 1988
(ALERT LIMITS = 26.5 SECONDS)

DATE	MEASURED STROKE TIME (SECONDS)
04-06-87	20.25
07-04-87	20.38
10-13-87	20.25, 21.25*, 21.24*, 21.25*
01-11-88	20.13
03-08-88	20.25
05-11-88	20.25
08-28-88	21.2*

*Times obtained using "switch-to-light" rather than "light-to-light" method.

TABLE 5

VALVE TIMING RESULTS
M02-2301-4
JANUARY 1987 THROUGH OCTOBER 1988
(ALERT LIMIT = 21.2 SECONDS)

DATE	MEASURED STROKE TIME (SECONDS)
04-06-87	15.31
07-04-87	15.06
08-27-87	15.25
10-13-87	15.25, 16.95*, 17.16*, 16.85*
01-11-88	15.25
03-08-88	15.25
05-11-88	15.38
08-29-88	17.7*

*Times obtained using "switch-to-light" rather than "light-to-light" method.

TABLE 6

MAINTENANCE PERFORMED
MO2-1301-1
JANUARY 1987 THROUGH OCTOBER 1988

Work Request	Completion Date	Description of Work
61137	01-27-87	Inspected motor.
60995	02-24-87	Disassesmbled valve operator; found grease hard, dark, and black; found quad rings and O-rings very hard and brittle; installed new gaskets, quad rings and O-rings; removed old grease and reassembled using Mobilux EPO.
62160	03-09-87	Drilled two 1/4" holes in the bottom of the MO 2-1301 junction box.
56911	03-12-87	Performed EQ surveillance (inspection and maintenance).
61006	03-14-87	Disconnected and reconnected.
63331	03-26-87	Adjusted packing to eliminate leakage.
63628	04-05-87	Removed and reinstalled cover for inspection.
66662	07-19-87	Removed two rings of packing - found in good shape; put in new rings and adjusted.

TABLE 7

MAINTENANCE PERFORMED
M02-2301-4
JANUARY 1987 THROUGH OCTOBER 1988

Work Request	Completion Date	Description of Work
60994	02-24-87	Disassembled valve operator; found grease very firm and quad rings hard; reassembled using new gaskets, quad rings and O-rings; installed Mobilux EPO grease.
56932	03-12-87	Performed EQ surveillance.
61007	03-12-87	Disconnected and reconnected.
53043	03-20-87	Adjusted packing.
59110	03-26-87	Tested using Special Procedure 87-2-32 in response to I.E. Bulletin 85-03.
63002	03-14-87	Cut and capped packing leak-off line.
63343	03-26-87	Adjusted packing to eliminate leakage.
63975	04-18-87	Preventive maintenance - no problem found.
64682	05-04-87	Valve cycled - jumpered installed.
64727	05-06-87	Wiring installed.
68333	08-27-87	Adjusted packing.

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

TABLE 8

MAINTENANCE PERFORMED
A02-220-46
JANUARY 1987 THROUGH OCTOBER 1988

Work Request	Completion Date	Description of Work
54876	02-24-87	Replaced broken sealtite.
63206	03-21-87	Installed new air regulator.
63339	03-24-87	Adjusted packing.
63853	04-11-87	Tightened fittings at solenoid to eliminate leakage.
64148	04-17-87	Adjusted packing and lubricated stem. Valve stroked several times to check operation.

TABLE 9

MAINTENANCE PERFORMED
A02-220-47
JANUARY 1987 THROUGH OCTOBER 1988

Work Request	Completion Date	Description of Work
63207	03-21-87	Installed new regulator.
63344	03-24-87	Adjusted packing.
63888	04-13-87	Replaced and adjusted limit switch.

TABLE 10

INSPECTION RESULTS
FOR ELECTRICAL CABLES

Work Request	Results
80025	Cable #26327 for head seal leak-off valve A02-220-52 on the third drywell elevation appeared good. Gaskets on junction boxes and valves need to be replaced.
80026	Cable #26325 for head seal leak-off valve A02-220-51 on the third drywell elevation appeared good except for evidence of water, moisture, or oil on the cable.
80027	Cable #21013 for Core Spray [BM] manual valve 2-1402-6A limit switches on the third drywell elevation had discolored insulation but no evidence of surface tracking. The terminal showed evidence of surface tracking. The terminal showed evidence of oxidation or corrosion. Sealtite was brittle and cracked. The gaskets on the limit switches were brittle and cracked.
80028	Cable #21015 for Core Spray manual valve 2-1402-6B limit switches on the third drywell elevation had discolored insulation but no evidence of surface cracking. The terminal showed evidence of oxidation or corrosion. Sealtite was brittle and cracked. The gaskets on the limit switches were brittle and cracked.
80031	Cable #26339 for Standby Liquid Control manual valve 2-1101-1 limit switches on the fourth drywell elevation was dark, hard, brittle, and discolored. There was no jacket on the insulation. A limit switch cover gasket was broken. A wire nut cover appeared burnt.
80032	Cable #26334 for head vent valve A02-220-46 on the fourth drywell elevation had discolored insulation. The insulation of terminal lugs appeared burnt off. A carrier block was cracked.
80033	Cable #26337 for head vent valve A02-220-47 on the fourth drywell elevation had dark and cracked insulation. A limit switch cover gasket was brittle and black. A wire nut boot and a tape splice appeared melted.
80037	Cable #22615 for Core Spray check valve A02-1402-9A on the fourth drywell elevation had degraded insulation and discolored lugs. Dirt and rust were noted in junction box 2RB47 – the cover was off and hanging by one screw. A gasket for a fitting on the solenoid was missing. A test switch was cracked. There was no cable jacket.
80038	Cable #20993 for Core Spray check valve A02-1402-9B on the fourth drywell elevation had had a hardened jacket with kinks, twists, and sharp bends. Terminal lug insulation was discolored. Dirt and rust were noted in junction box 2RB47 – the cover was off and hanging by one screw. A gasket for a fitting on the solenoid was missing. A test switch was slow in reacting after being pushed in.
79935	Cables #32556 and #32558 for Isolation Condenser valve M02-1301-1: Cable jackets burned and cracked at junction boxes 1 and 2. Jackets looked good at junction box 3. Some oxidation or corrosion noted at terminals.
79934	Cables #22628 and #22630 for HPCI valve M02-2301-4: Cable jackets cracked at junction box 1, apparently from excessive heat. Some oxidation or corrosion noted at terminals.

TABLE 11

INSPECTION RESULTS
FOR LIGHTING CABLES, JUNCTION BOXES,
BELLOWS LEAK-OFF SWITCHES, AND THERMOCOUPLES

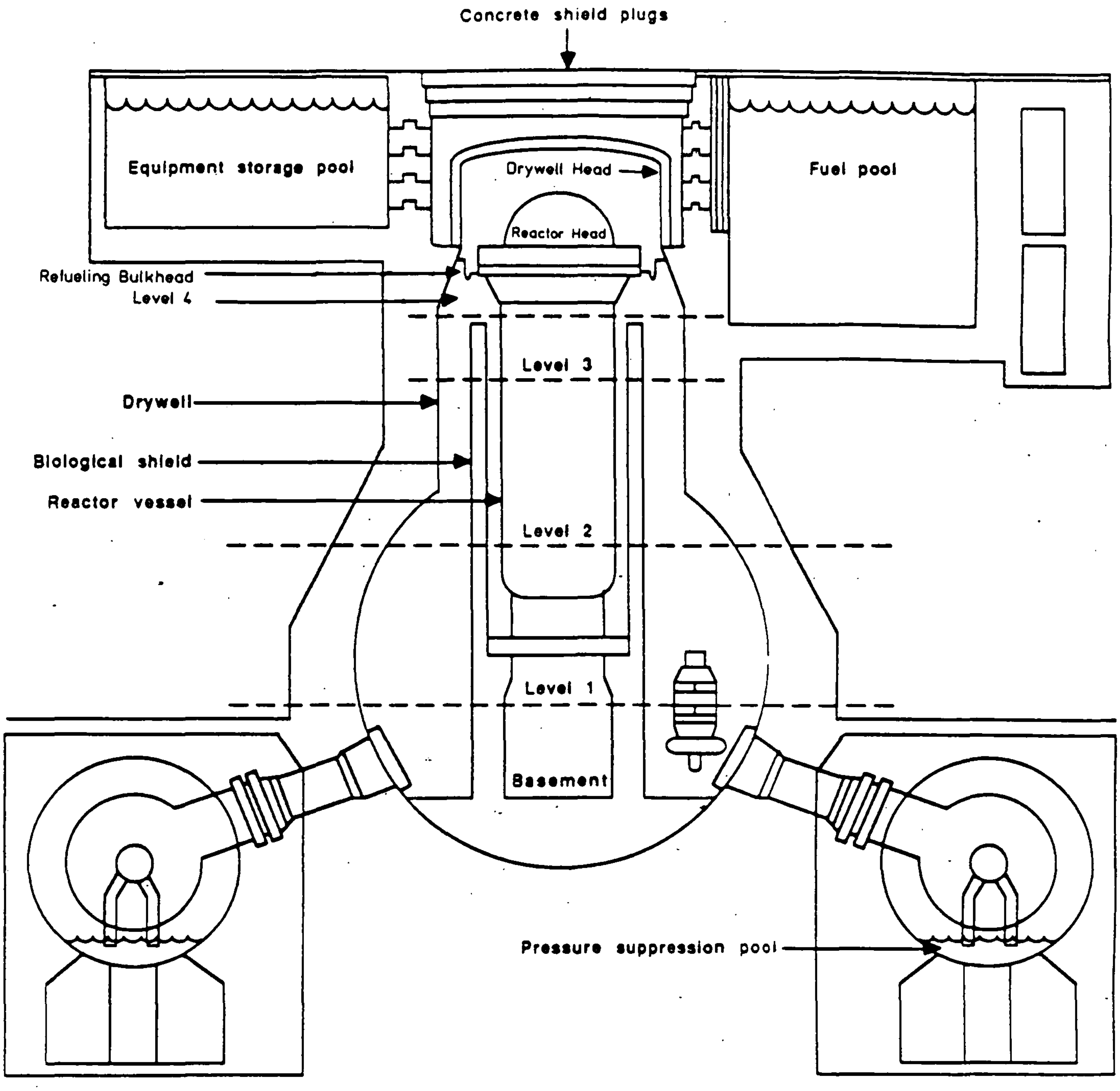
Work Request	Results
80029	Power cables for lighting were inspected at five locations in the vicinity of the fourth drywell elevation. By the fourth elevation ceiling, cracked, discolored, and hardened insulation was seen. At receptacles above the entry ladder and near penetration X108B, discolored insulation was seen. At a T-fitting just below the entry hatch, discolored and hardened insulation was seen. No degradation to insulation was seen at a landing below the fourth elevation.
80066	Nine junction boxes on the fourth drywell elevation were inspected. Some of the boxes had a white powder or a black residue inside. Sealtite was missing or brittle and peeling. Thermocouple connector caps were brittle and discolored.
80067	Bellows leak-off level switch 2-1901-102 on the fourth drywell elevation was inspected. No degradation was observed.
80068	Bellows leak-off level switch 2-261-19 on the fourth drywell elevation was inspected. No degradation was observed.

80069

Thermocouple	Location	Results
2-2390A	2nd Elev.	No degradation seen.
2-5741-15B	3rd Elev.	Cable jacket and insulation degraded.
2-1301-1	3rd Elev.	No degradation seen.
2-5741-13A	4th Elev. (588')	Cable jacket and insulation degraded.
2-5741-15A	4th Elev. (575')	No degradation seen.
2-5741-15C	4th Elev. (575')	No degradation of insulation seen. Cable jacket degraded.
2-5741-14B	4th Elev. (589')	Cable jacket and insulation degraded.
2-5741-13B	4th Elev. (588')	Cable jacket and insulation degraded.
2-263-66-A1	4th Elev.	No degradation seen.
2-263-67-A1	4th Elev.	No degradation seen.
2-263-66-B1	4th Elev.	No degradation seen.
2-263-67-A2	4th Elev.	No degradation seen.
2-263-66-A2	4th Elev.	No degradation seen.
2-263-66-B2	4th Elev.	No degradation seen.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) Year	Sequential Number	Revision Number	Page (3)
Dresden Nuclear Power Station	0 5 0 0 0 2 3 7	8 8	- 0 2 2	- 0 0	2 3 OF 2 7

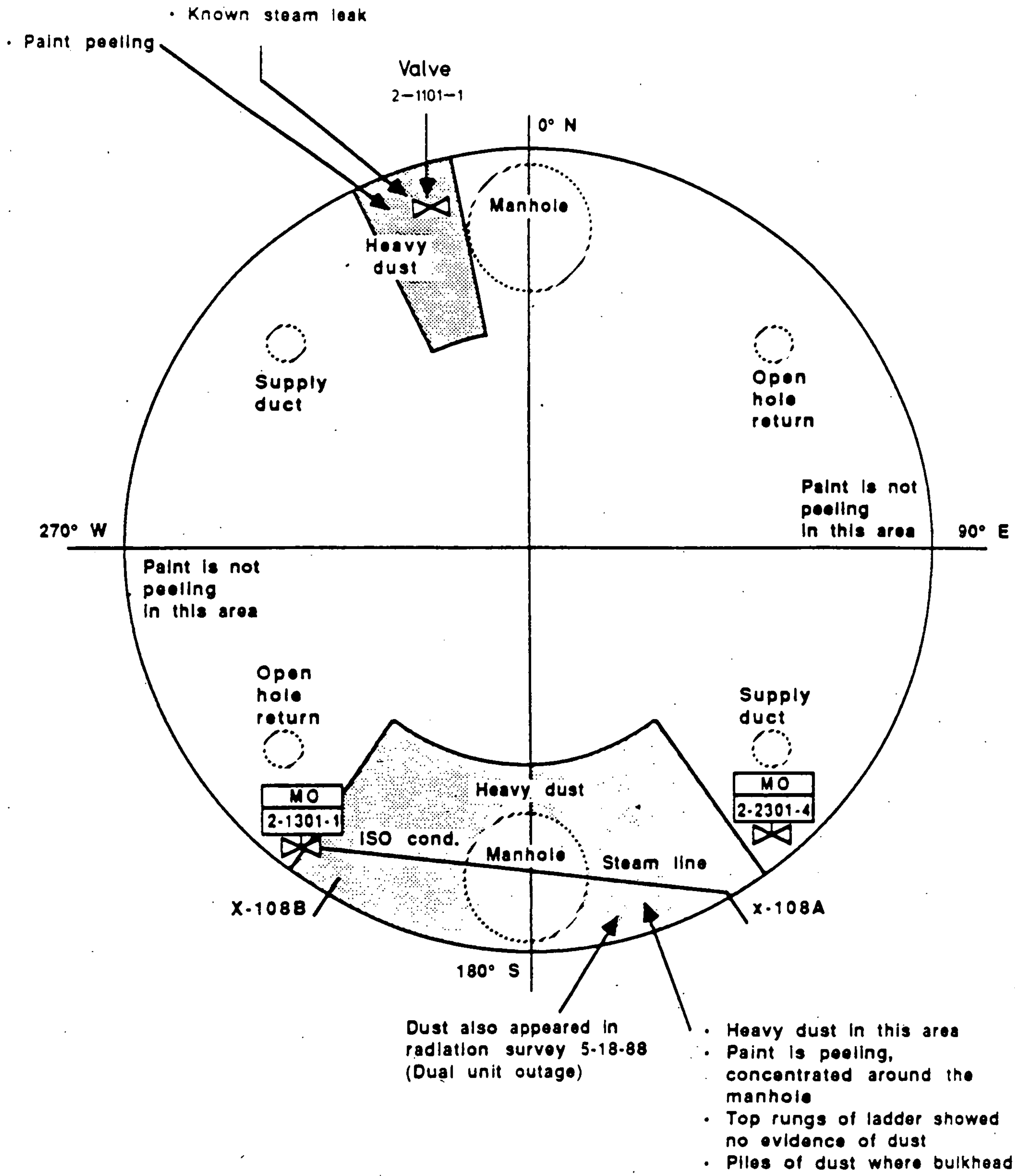
TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]



DRESDEN UNIT 2 DRYWELL
FIGURE 1

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) Year	Sequential Number	Revision Number	Page (3)
Dresden Nuclear Power Station	0 5 0 0 0 2 3 7	8 8 -	0 2 2 -	0 0	2 4 OF 2 7

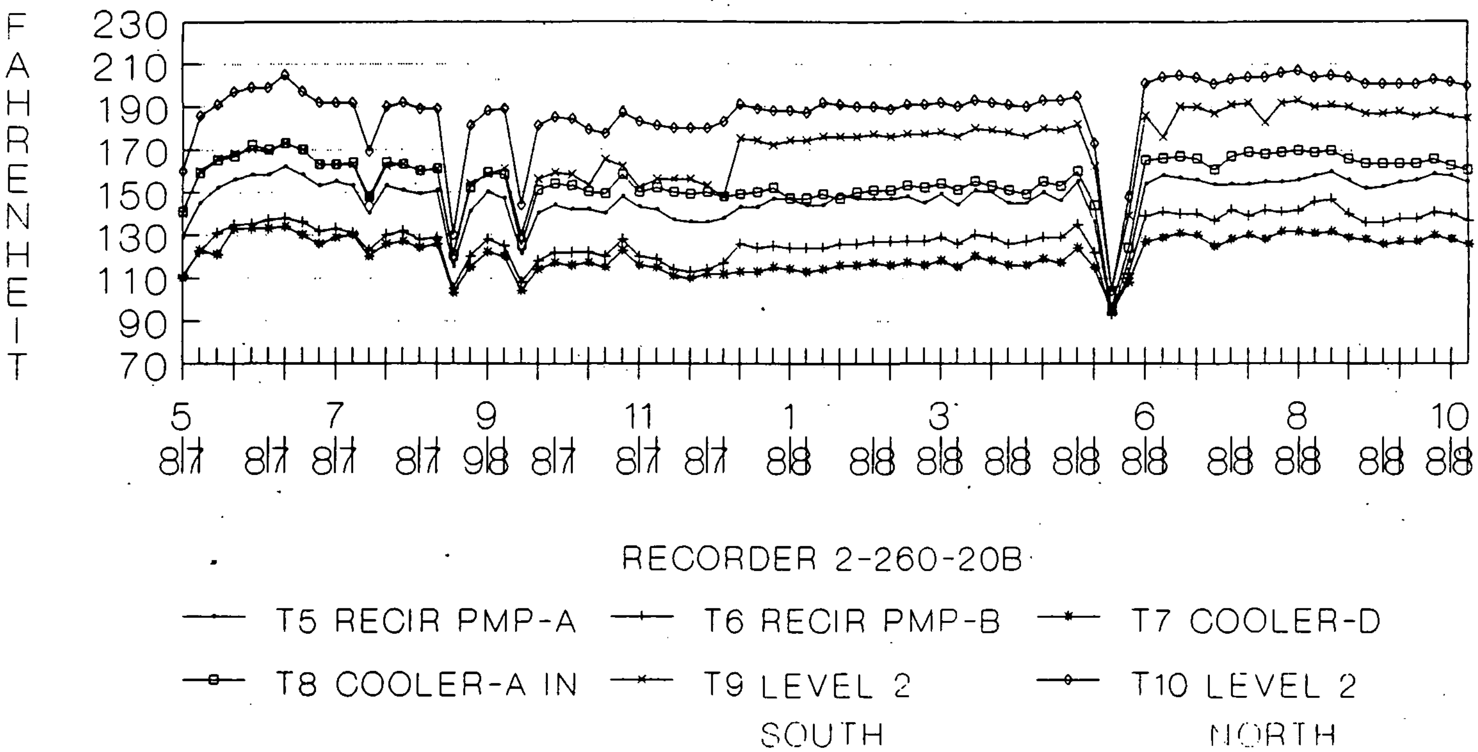
TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]



DRESDEN UNIT 2 DRYWELL
FOURTH ELEVATION
FIGURE 2

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			Page (3)
		Year	Sequential Number	Revision Number	
Dresden Nuclear Power Station	0 5 0 0 0 2 3 7	8 8	- 0 2 2	- 0 0	2 5 OF 2 7

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]



DRESDEN UNIT 2 DRYWELL COOLING SYSTEM
TEMPERATURE DATA
FIGURE 3

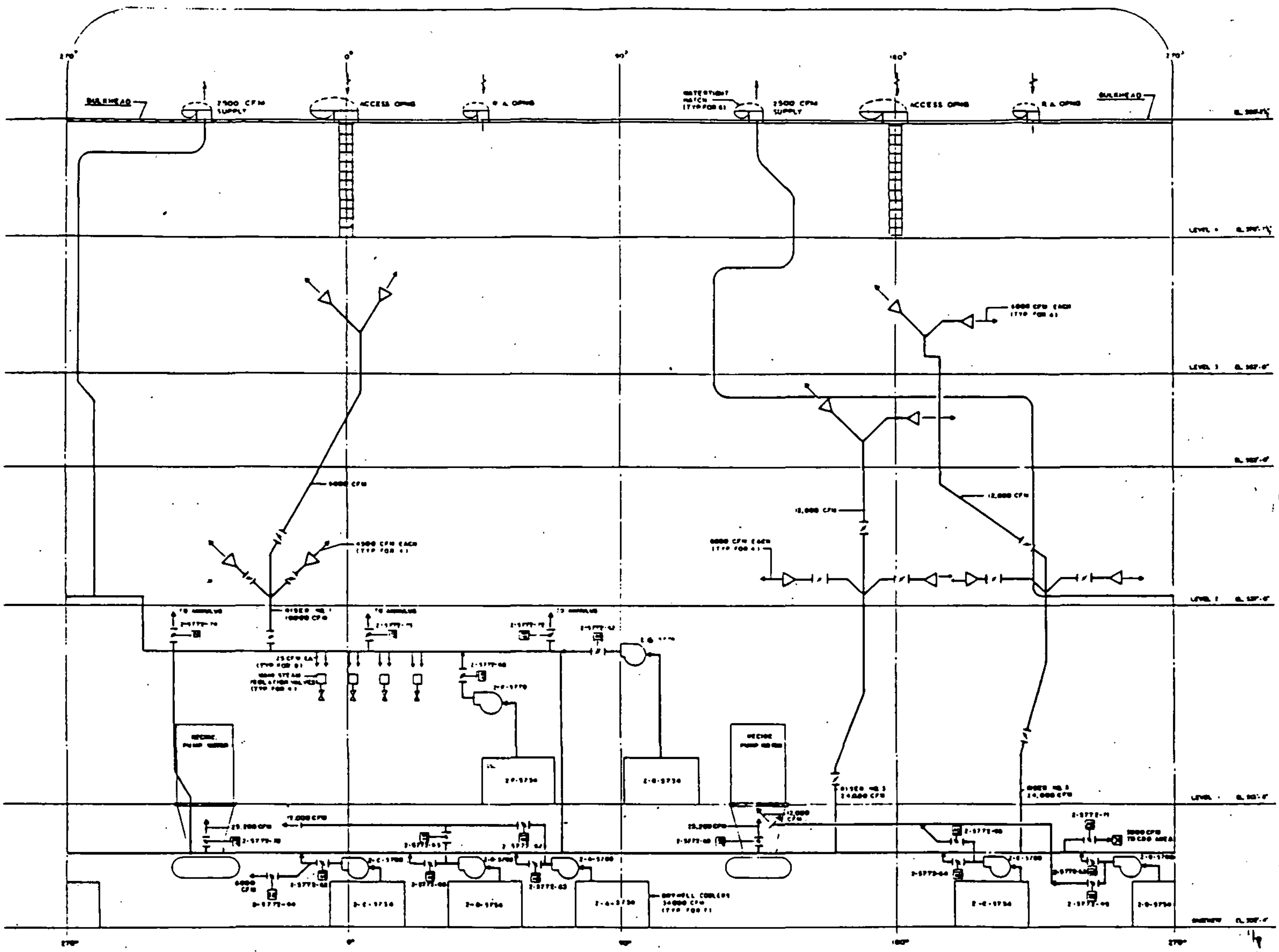


DIAGRAM OF DRESDEN UNIT 2
DRYWELL COOLING SYSTEM
FIGURE 4

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION Form Rev 2.0

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) Year	Sequential Number	Revision Number	Page (3)
Dresden Nuclear Power Station	0 5 0 0 0 2 3 7	8 8	- 0 2 2	- 0 0	2 7 OF 2 7

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

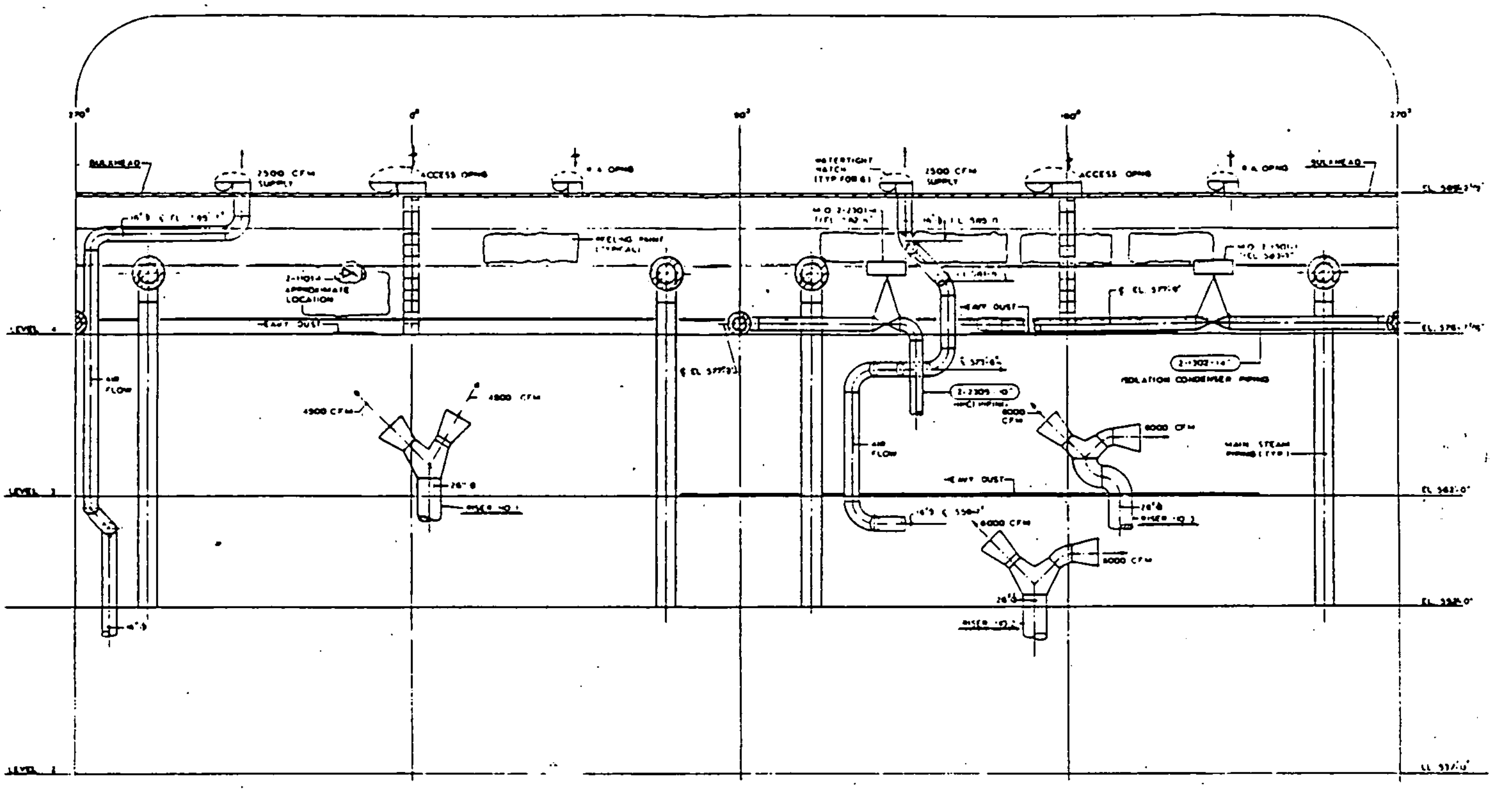


DIAGRAM OF DRESDEN UNIT 2
DRYWELL COOLING SYSTEM
UPPER ELEVATIONS
FIGURE 5

Commonwealth Edison
Dresden Nuclear Power Station
R.R. #1
Morris, Illinois 60450
Telephone 815/942-2920

December 13, 1988

EDE LTR #88-937

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Licensee Event Report (LER) #88-022-0, Docket #050237 is being submitted as a voluntary LER as specified by NUREG 1022, Supplement No. 1, Section II.19.1. The initial review of this event determined that none of the 10CFR50.73 reporting requirements are applicable, although studies currently underway and discussed in the LER may determine that one or more of the reporting requirements do apply. The results of these studies will be submitted as a supplemental report to this LER.

Lawrence F. Gerner for
E.D. Eenigenburg
Station Manager
Dresden Nuclear Power Station

EDE/ade

Enclosure

cc: A. Bert Davis, Regional Administrator, Region III
File/NRC
File/Numerical

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