DRESDEN NUCLEAR POWER STATION

UNIT 3

SPECIAL REPORT

OF

INCIDENT OF MAY 4, 1972

REPORT ON DRESDEN 3 INCIDENT OF MAY 4, 1972

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SUMMARY

On May 4, 1972, following a spurious reactor scram, a Main Steam Line isolation valve closure initiated a reactor pressure transient. An isolation condenser condensate return valve failed to open during the transient and resulted in a reactor pressure increase to approximately 1110 psig. At this pressure an electromatic relief valve operated. A safety valve lifted momentarily at approximately the same time the relief valve operated. This resulted in a release of primary steam to and pressurization of the drywell. The transient was terminated approximately 21 minutes after the scram.

During the transient, the maximum drywell pressure reached was 2.5 psig. The maximum and minimum reactor pressures were 1108 and 825 psig respectively. The reactor water level reached a minimum of plus one inch and a maximum of 54.5 inches reactor water level as indicated on the recorder. The maximum drywell temperature was 180°F.

No uncontrolled release of radioactivity was made to the environment as a result of the incident. The radioactivity released during this shutdown period was less than would have been released during normal operation. During post incident recovery, both the primary system and the primary containment were maintained in an isolated condition until analysis of reactor water and containment atmosphere could be made.

Damage was insignificant, being limited to piping insulation.

The following report contains a sequence of events, damage report, corrective actions and conclusions.

A. CHRONOLOGICAL DISCUSSION

A complete description of the events is presented below in chronological order with discussion where pertinent. Response of significant parameters following the scram is presented in Figure 1.

Item

1. On May 4, 1972, Dresden Unit 3 was operating in a base load condition at 802 MWe and 2300 MWt. Other significant parametric values prior to the scram included:

Steam flow	9.0×10^6 lb/hr
Feedwater flow	9.0×10^6 lb/hr
Reactor pressure	990 psig
Reactor water level	plus 28 inches

Equipment in service included "3A" and "3B" reactor feedwater pumps ("3C" pump in standby). The reactor containment was inerted with nitrogen and at a normal pressure of approximately 0.25 psig. Chimney off-gas release rate was approximately 6,900 u Ci/sec.

2. At 09:01:21 41/120, a "Reactor D Lo Level" trip alarm annunciated, and at 09:01:21 44/120 the reactor scram was initiated. At 09:01:21 58/120 the "Reactor D Lo Level" trip reset. Ordinarily an alarm of this type indicates a trip of only half of the reactor protection system, yet this is the only alarm which annunciated prior to the scram.

3. The reactor scram was followed by a normal post-scram reactor water level transient. The reactor water level began decreasing immediately due to collapse of voids, tripping the reactor low level scram relays at 09:01:25, and causing the two operating feedwater pumps to increase their output to about 10.6×10^6 lb/hr (the feedwater pumps did not go into the "runout" mode. "Runout" is initiated at a flow of 6×10^6 lb/hr for one pump and 12×10^6 lb/hr for two pumps.)

4. At approximately 15 seconds after the scram, the operator moved the reactor mode switch from "run" to "refuel" as required by operating procedure.

5. The reactor water level reached a low point of about plus one half inch on the recorder and began increasing rapidly. At time 34 sec., "Reactor A Low Level" and "Reactor B Low Level" trips reset, at 35 sec. the operator tripped "B" reactor feed pump, at 35.5 sec., "Reactor C Low Level" and "Reactor D Low Level" trips reset, and at 37 sec. "A" reactor feed pump was manually tripped by the operator. The two reactor feed pumps were

tripped on increasing level at about 20 to 25 inches.

6. The reactor water level continued to increase at a diminishing rate as a result of water added by recirculation pump run-back to minimum flow, continued input by the feedwater pumps as they slowed down, and control rod drive system input. As the level reached 48 inches, a group I isolation was initiated and the main steam isolation valves closed at 5 minutes 2 seconds. The reactor water level peaked at 54.5 inches and gradually started to decrease.

7. The reactor pressure decreased rapidly following the scram due to shrinkage and, initially, continued output of steam to the turbine. The pressure decreased to 828 psig before turning around at time one minute due to addition of decay heat. The main steam line isolation valve closure at 850 psig did not occur because the reactor mode switch had been taken out of the "run" mode.

8. The reactor pressure continued to increase from the decay heat input until it reached about 1060 psig at 10 minutes when an attempt was made to initiate the isolation condenser. However, the isolation condenser return valve, 3-1301-3 would not open electrically from the control room and an operator was dispatched to open the valve locally.

9. When the reactor pressure reached 1100 psig, an order was given by the shift supervisor to open an electromatic relief valve. Before this could be done manually, 3A electromatic operated automatically and relieved the reactor pressure to about 1035 psig at time 14 min. The pressure then began increasing again but was maintained below 1100 psig by manually operating the electromatic relief valves.

10. At approximately the same time 3A electromatic relief valve automatically operated (at time 14 minutes) a "Containment High Pressure" alarm annunciated and the drywell pressure increased to about 2½ psig. This alarm initiated the proper ECCS functions; core spray pumps started, low pressure coolant injection pumps started, and the Unit 3 and Unit 2/3 diesel generators started as designed. It was later determined that drywell pressurization was caused by safety valve 3-203-4A lifting momentarily and prematurely. The set point was 1210 psig and the maximum reactor pressure reached was approximately 1108 psig.

11. The high drywell pressure alarms reset (at time 23 minutes) when the drywell pressure reached approximately 1½ psig. This was within nine minutes after annunciating. It continued to decrease slowly.

12. At time 25 minutes the isolation condenser return valve was manually opened and the isolation condenser was used to reduce reactor pressure in an orderly manner.

13. A reactor water sample was taken at time one hour and 40 minutes. Results of the sample were normal and blowdown of the reactor via the cleanup system commenced.

14. The Drywell pressure was reduced to normal 40 minutes (at 0941) after the scram. The Drywell was sampled at 1025. Results were within specifications for venting and the drywell was purged. Drywell entry was made at 2258, May 4, 1972.

B. DAMAGE ASSESSMENT

When safety valve 3-203-4A lifted, steam was ejected into the drywell. This steam struck the line from electromatic relief valve 3A to the torus, line number 3-3019-A-8" and also the base of electromatic relief valve 3E. Upon striking the base of the relief valve, a piece of insulation was torn loose. The insulation was replaced and line 3-3019-A-8" was visually inspected and found to be undamaged. A sketch of the safety and relief valve arrangement is presented in Figure 2.

C. OPERATIONS ASSESSMENT

Operator response was in accordance with operating procedures throughout the transient and post transient recovery. Water level was maintained "on scale" well below the main steam lines and reactor pressure was never greater than 1110 psig.

After containment pressure had decreased to less than 2 psig, the primary containment isolation automatically reset, and a sample of drywell atmosphere was collected and analyzed. Based on the results of the analysis, the drywell purged according to established operating procedures in preparation for entry and damage evaluation.

The ECCS systems which were initiated during the transient were not secured until the high drywell pressure initiation signal had cleared.

The primary system was kept isolated until a sample of reactor water could be checked. It was necessary to place a jumper on the recirculation sample valve so that a sample could be taken.

D. CORRECTIVE ACTIONS AND CONCLUSIONS

1. Following the plant shutdown, investigation of the isolation condenser valve 3-1301-3 disclosed that the "B" auxiliary contact on the closing circuit for the valve motor operator circuit breaker was not consistently closing properly. This contact is in the electrical interlock circuit for the open contactor. The auxiliary contact arm was adjusted and proper operation of the valve was verified. The torque setting on the valve had been set at one to close and one to open. The torque setting was reset to three for opening to allow this valve to open against greater resistance.

   The last surveillance performed on this valve prior to the scram was on 1/15/72 when the valve operated satisfactorily. Subsequent to the startup following the scram, the valve was operated electrically three times and performed properly.

2. Safety valve 3-203-4A which lifted was serial number BK7155. The valve had originally been set at a steam pressure lift point of 1210 psig. After the incident, a nitrogen test was conducted on the valve and it lifted approximately 75 psig lower than previously set. The valve was replaced with valve BK7163 which had been set with steam and nitrogen at the manufacturer's plant located in Alexandria, Louisiana and was verified with nitrogen at 1140 psig prior to installation.

3. The 3A electromatic relief valve was tested and found to have a lifting pressure of 1108 psig and a reset of 1035 psig, indicating that the set point had drifted from its last setting of 1133 psig trip and 1052 reset made 6/28/71. All electromatic relief valves were checked and recalibrated to proper relieving pressures.

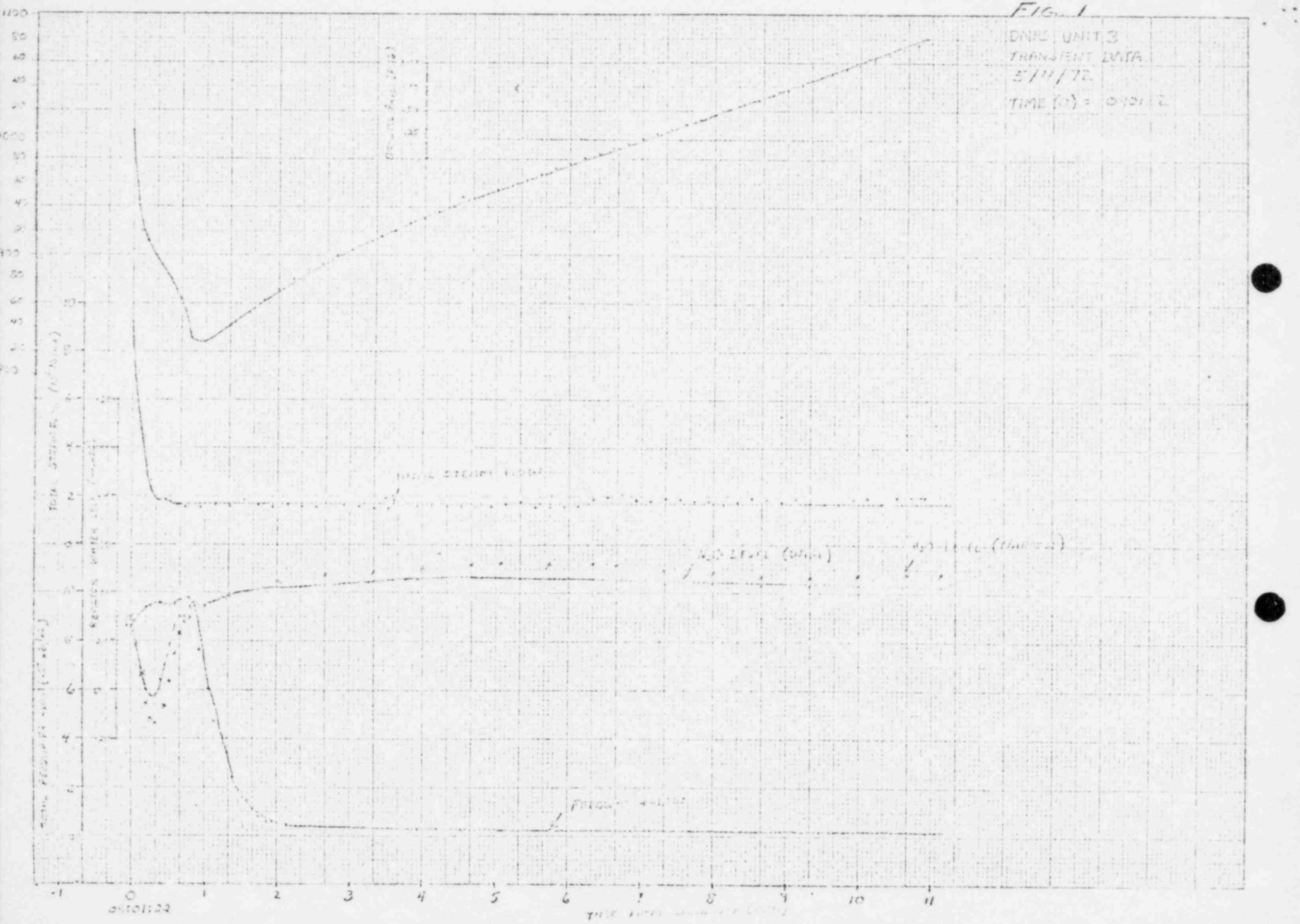
4. An investigation into the initiating cause of the scram was conducted. At the time of the scram, the only indication of malfunction was a spurious "Reactor "D" Low Level" trip. No other reactor protection system trips were annunciated. Following the shutdown a reactor protection system functional check was conducted to determine if all computer and annunciator inputs were operable. No abnormalities were found. The cause of the scram is not known.

5. A test program is presently in progress on Dresden 2 to tune up and optimize the performance of the feedwater control system. It is anticipated that, when control system improvements have been completed, the feedwater system will control level, automatically without action, during transients.

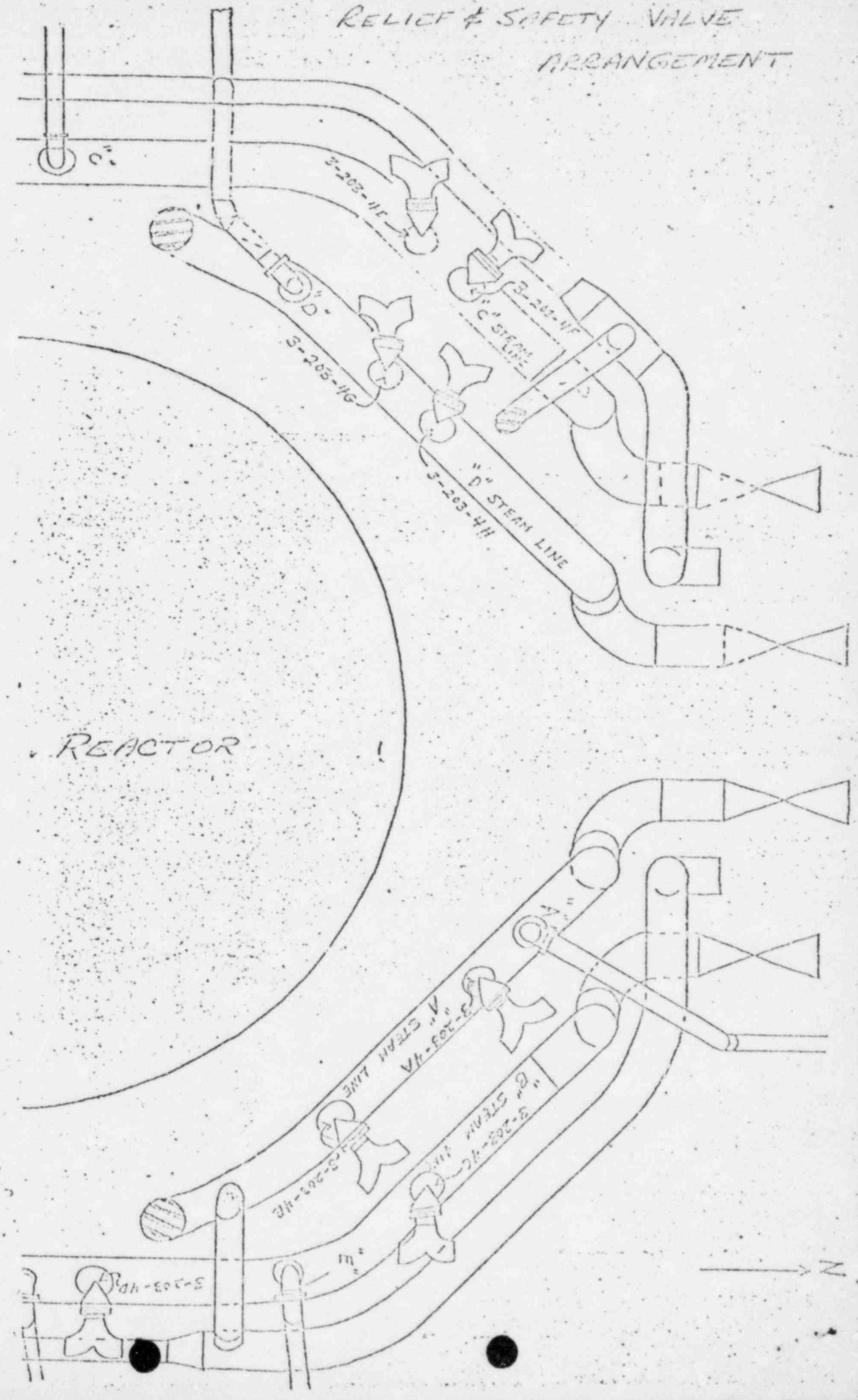
6. A program of investigation into the cause of spurious safety valve operation is continuing.

7. The transient was reviewed by the Station Review Board and the Nuclear Review Board. Following their review, startup was approved and the plant returned to service.

KEUFFEL & ESSER CO.



RELIEF & SAFETY VALVE ARRANGEMENT



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June 3, 1972

50-249

Mr. Edward J. Bloch, Acting Director
Directorate of Licensing
U.S. Atomic Energy Commission
Washington, D.C. 20545

Subject: License DPR-25, Dresden Nuclear Power Station Unit #3.

Reference: Letter to Dr. Peter A. Morris from W. P. Worden dated May 12, 1972 regarding subject information.

Dear Mr. Bloch:

Enclosed for your information is a special report on the Dresden Unit 3 Incident of May 4, 1972, which resulted in pressurization of the primary containment to approximately 2½ psig.

Sincerely,

W. P. Worden

W. P. Worden
Superintendent
Dresden Station

WPW:sds
Enc.

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TRW
J.K.
RM

50-249 incident

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