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September 26, 1979

Dr. Harold Denton, Director
Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, DC 20555

RE: 44 Fed. Reg. 50925
Abnormal Occurrence Event

Dear Harold:

Reference is made to the description of the abnormal occurrence event at Oyster Creek Nuclear Generating Station on 2 May 1979 contained in the Federal Register on 30 August 1979.

The last several paragraphs (page 50927) of the description on the generic implications of the event imply that the event could occur only at several BWRs of older design. This may be true. However, not mentioned is the importance of all operators being aware of which parameter (level, pressure, power level, etc.) is being measured by specific instrumentation in their plant. It is my understanding that the triple low level alarm was initially considered to be spurious because it was inconsistent with other level indications available to the operators. However, different instruments were indicating different water level conditions, i.e. in the annulus and in the core.

I had occasion to talk to a number of operators at another BWR several months after the Oyster Creek abnormal occurrence. Most vaguely remembered hearing or reading about the OCNGS event but considered that it was not applicable to their plant because it was of the jet pump design (one could readily draw this conclusion from the event description). All had overlooked the importance of knowing which of the level instruments in their own plant read what level (annulus, core, other, etc.). In fact, only one individual knew which of various water level instruments indicated annulus level and which indicated core level.

My point is this, it is important for operators to know which instrument reads what specific parameter. Knowledge of this might help them better understand instrument indications and plant status under abnormal plant conditions. The abnormal occurrence event description in the Federal Register did not make such a point, and thus, I believe the basic significance of the event was probably lost by operators at most other plants.

Sincerely,

Forrest

Forrest J. Remick
Assistant Vice President of
Research & Graduate Studies

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cc: Paul F. Collins, Chief
Operator Licensing Branch

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days, will be held at the Federal Judicial Center, 1520 H Street, N.W.

The meeting will deal with the work and role of courts of limited jurisdiction, trends in the volume and mix of tort cases in federal and state courts, and reform efforts aimed at increasing access to courts for resolving disputes. The meeting will be open to the public, and minutes of the proceedings will be made available upon request.

Additional information may be obtained from Mr. C. Ronald Ellington, Office for Improvements in the Administration of Justice, United States Department of Justice, Washington, D.C. 20530. Telephone (202) 633-3593.

Harry A. Scarr, Ph.D.,

Administrator, Federal Justice Research Program.

[FR Doc. 79-27046 Filed 8-29-79; 8:45 am]

BILLING CODE 4410-01-M

NUCLEAR REGULATORY COMMISSION

Abnormal Occurrence Event; Damage to New Fuel Assemblies

Section 208 of the Energy Reorganization Act of 1974, as amended, requires the NRC to disseminate information on abnormal occurrences (i.e., unscheduled incidents or events which the Commission determines are significant from the standpoint of public health and safety). The following incident was determined to be an abnormal occurrence using the criteria published in the **Federal Register** on February 24, 1977 (42 FR 10950). Appendix A (Example I.C.2) of the Policy Statement notes that a substantiated case of actual or attempted * * * sabotage of a facility can be considered an abnormal occurrence. The following description of the event also contains the remedial actions taken.

Date and Place—On May 7, 1979, the NRC Resident Inspector at the Surry Power Station was notified by the licensee (Virginia Electric and Power Company—VEPCO) that while conducting inspections of new fue r Unit 2 it was found that 62 of 64 assemblies were coated with a white crystalline substance. Surry Units 1 and 2 are pressurized water nuclear power plants located in Surry County, Virginia.

Nature and Probable Consequences—On May 7, 1979, while conducting routine inspections of new fuel, the licensee discovered that a foreign substance had been poured onto 62 of the 64 new fuel assemblies stored in the Fuel Building, a vital area which contains both new and spent fuel. An analysis of the substance determined it to be sodium hydroxide. As a result of this analysis and the uncertainty of the extent of damage, the licensee is returning all the assemblies to the vendor for refurbishment. The licensee determined that there were no indications of damage to the spent fuel, nor was there evidence of unauthorized individuals gaining access to the vital area.

Fuel as the Surry site is stored in the Fuel Building, an area which is locked and alarmed, and to which access is controlled by the use of specially coded access cards. Authorized individuals, who are permitted access to the Fuel Building using the specially coded access cards, are afforded unimpeded access to both the new and spent fuel.

Since normally conducted inspections by the licensee detected the damage to the new fuel, there is little chance that these assemblies—damaged in this way—would have been used in the reactor. While the actual consequences of this incident had no effect on the public health and safety, the incident did represent a potential threat in that it occurred within a vital area where sabotage to both new fuel and spent fuel was possible.

Cause or Causes—The cause was an alleged criminal act. On May 7, 1979, the licensee notified the FBI of the damage to the new fuel. The FBI conducted an investigation which culminated in two plant workers surrendering to Surry County authorities on June 19, 1979. A grand jury hearing was held in Surry, Virginia on July 24, 1979; trial is scheduled for October 10–12, 1979. The two workers, under advice from their attorney, have refused to describe the details of the safety issues which reportably motivated them to commit the acts.

Actions Taken to Prevent Recurrence Licensee—As a result of the incident, and to assist the FBI in its investigation, the licensee considerably reduced the number of people permitted access to the Fuel Building and stationed a security guard inside the Fuel Building to verify access authorization. These were prompt temporary actions. The licensee has completed a thorough review of their access control program, and are now more selective in determining whether unescorted access should be provided. The licensee has made the Superintendent of Administrative Services responsible for coordinating corrective actions, and to ensure that weaknesses are corrected even if noted by someone not normally responsible for that particular professional discipline. These actions are consistent with the NRC IE Bulletin described below. Similar measures were also instituted at VEPCO's North Anna Power Station.

NRC—An NRC IE Security Inspector was dispatched to the site on May 8, 1979. Additionally, the Region II Senior Investigator, the Region II Security Section Chief and a Health Physics Inspector were onsite to assist the NRC Resident Inspector and to provide onsite assistance to the FBI. NRC IE Security Inspectors have examined the corrective measures taken by the licensee.

NRC IE Information Notice No. 79-12, "Attempted Damage to New Fuel Assemblies" was issued on May 11, 1979, to alert all NRC licensees who store new fuel assemblies of this problem.

NRC IE Bulletin No. 79-16, "Vital Area Access Controls" was issued on July 26, 1979 to require specific actions by the licensees, including a report by September 9, 1979 of actions taken and planned.

Dated at Washington, D.C., this 22d day of August 1979.

For the Nuclear Regulatory Commission.

Samuel J. Chilk,

Secretary of the Commission.

[FR Doc. 79-27163 Filed 8-29-79; 8:45 am]

BILLING CODE 7590-01-M

Abnormal Occurrence Event; Indication of Low Water Level in a Boiling Water Reactor

Section 208 of the Energy Reorganization Act of 1974, as amended, requires the NRC to disseminate information on abnormal occurrences (i.e., unscheduled incidents or events which the Commission determines are significant from the standpoint of public health and safety). The following incident was determined to be an abnormal occurrence using the criteria published in the **Federal Register** on February 24, 1977 (42 FR 10950). Appendix A (Example II.A.1) of the Policy Statement notes that exceeding a safety limit of license Technical Specifications, (10 CFR 50.36(c)) can be considered an abnormal occurrence. The following description of the event also contains the remedial actions taken.

Date and Place—On May 2, 1979, the NRC was notified by the licensee (Jersey Central Power and Light Company) of an event at their Oyster Creek facility. The Oyster Creek Nuclear Plant utilizes a boiling water reactor and is located in Ocean County, New Jersey.

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Nature and Probable Consequences

Summary

A loss of feedwater transient at the Oyster Creek facility on May 2, 1979, resulted in a significant reduction in water inventory above the reactor core area as measured by one set of water level instruments (triple-low level), while the remaining two sets of level instrumentation in the reactor annulus indicated water levels above any protective feature setpoint (Figure 1). The water level measured within the core shroud area fell below the triple-low level setpoint, a safety limit, of 5-feet, 6-inches above the top of the fuel. Subsequent analyses by the licensee have conservatively determined that the minimum water level over the top of the fuel was 1 to 1½ feet. Coolant sample analyses and offgas release rates support the conclusion that no fuel damage occurred.

Sequence of Events

Oyster Creek is a non-jet pump BWR [1] with a licensed power of 1930 MWt. Immediately prior to the transient, the reactor was operating at 98% power with the reactor vessel water level at 13-feet, 4-inches above the top of the fuel. The "D" reactor recirculation loop was out-of-service because of a recirculation pump seal cooler problem and the "S" startup transformer was out-of-service for inspection of the associated 4160 volt cabling.

The initiating event was a false high reactor pressure scram. The pressure spike that led to the scram signal was generated by the way an instrument technician was performing surveillance testing on isolation condenser pressure switches. The signal resulted in a simultaneous reactor scram and the tripping of all operating recirculation pumps. The tripping of all operating recirculation pumps is a safeguard to mitigate the consequences of anticipated-transients-without-scram events.

Thirteen seconds after the reactor scram, the turbine tripped at the low load setpoint. The turbine trip initiated a transfer of power from the auxiliary transformers to the startup transformers. Because one startup transformer "SB" was out of service, two feedpumps and two condensate pumps (pumps 1B and 1C) on the associated 4160v bus (2B) lost power. The third feedpump (1A) tripped due to low suction pressure during the feedwater transient. An immediate attempt to restart the 1A feedwater pump, powered by the live 4160v bus (1A), was unsuccessful because of failure of an auxiliary oil pump to start. The lube oil pump is interlocked in the feedpump start sequence. This was the only equipment failure during the transient.

Subsequent to the reactor scram, reactor water inventory initially decreased due to steam flow through the turbine bypass valves to the main condenser. This loss together with the void collapse associated with decreased temperature of the core inlet coolant and the subsequent loss of feed flow, resulted in a rapid reactor water level reduction to the low water level alarm setpoint of 11-feet, 5-inches above the top of the fuel at 13.6 seconds. The operator manually initiated closure of all main steam line isolation valves (MSIV) at about 43 seconds into the transient to conserve water. The minimum indicated water level in the annulus was 9-feet, 8-inches above the top of the fuel (the low-low setpoint is 7-feet, 2-inches above the top of the fuel).

After closure of the MSIV, an isolation condenser was manually placed in service for core decay heat removal. The isolation condenser was condensing steam from the core and returning the condensate to the reactor annulus through a connection to a recirculation loop pump suction line (Figure 1). At approximately a minute and a quarter after the reactor scram, the discharge valves in "A" and "E" recirculation loops were closed in accordance with a Standing Order that was in effect. Closing the "A" and "E" loop discharge valves had been necessary to prevent inadvertent stopping of the isolation condenser due to forced flow from operating recirculation pumps being sensed as if it were an isolation-condenser line break. (This Standing Order was no longer appropriate since an ATWS modification had been made that tripped the recirculation pumps coincident with high-pressure or low-level scrams. The necessary procedure change had not been performed following the plant modification.) At the same time, the "B" and "C" loop discharge valves were apparently closed, in anticipation of restarting the recirculation pumps. The "D" loop discharge valve had been closed prior to the event because the associated pump was out of service.

The reactor triple low water level (5-feet, 6-inches above the top of the fuel) setpoint was reached at 172 seconds into the transient. The triple low level setpoint activates one of the permissives on the automatic depressurization system and alarms in the control room to alert the operator.

The triple-low level in the core shroud area resulted from the restriction of the flow path between the annulus and the core region by closure of all recirculation pump discharge valves. With the recirculation pump discharge valves closed (discharge piping is over 2 feet in diameter), the only flow path back to the core region was via the 2-inch bypass lines around the discharge valves. The effect of this flow restriction was to reduce the water level in the core region and to increase the level in the reactor annulus area.

Reactor pressure was controlled by intermittent manual operation of the two isolation condensers. At about half an hour into the transient, a recirculation pump was started. The operator tripped the recirculation pump within 2 minutes when he noted a rapid decrease in the annulus level (the recirculation pumps take suction from the annulus). About 5 minutes later, a feedpump was started. At about 40 minutes into the event, a primary recirculation pump and a reactor feedpump were restarted for continued cooldown of the reactor. From about 40 minutes onward, the water level within the core region was normal. The plant attained cold shutdown condition within 9 hours.

Review of the occurrence by the licensee and NRC established that although the water level in the core shroud area went below the triple low level setpoint, the core remained covered and consequently no fuel damage would be expected.

Cause or Causes—A spurious reactor high pressure scram was initiated by a pressure spike on the reactor high pressure scram switches, caused by an instrument technician while performing surveillance testing. However, the fundamental causes of the resulting sequence of events were:

(1) The operators essentially isolated the reactor annulus and core region from each other by shutting all recirculation pump discharge valves.

(2) Notes or Cautions against closing all suction and discharge valves in the recirculation loops were not adequate.

Actions Taken To Prevent Recurrence

Licensee—The licensee performed a thorough evaluation of the event to determine whether any fuel damage had occurred and developed follow up actions. As a result of the evaluation and discussions with the NRC, the licensee took the following significant actions:

[1] The non-jet pump BWR is of an older design. The newer designs incorporate jet pumps within the reactor pressure vessel to improve the coolant recirculation system performance. The jet pump concept reduced the number of external coolant recirculation loops to two.

1. The triple-low level was established as a Safety Limit for all modes of reactor operation.[2]

2. A requirement was added to the Technical Specifications that the suction and discharge valves in at least two recirculation loops be open at all times. The procedures were changed to implement this requirement.

3. Operator training sessions were held, the event was thoroughly discussed and the revised procedures reviewed.

NRC—Following notification from the licensee of the event, an NRC inspector was dispatched to the site. Additional NRC personnel arrived at the site on May 3, 1979 to review the situation and determine the status of the plant. Fact finding by the NRC was supplemented by information obtained from the licensee, the reactor vendor (General Electric) and fuel supplier (Exxon). A safety evaluation report (SER) of the event was prepared which discusses the minimum water level experienced in the reactor vessel and the fuel conditions. The following three requirements were added to the Technical Specifications:

1. The triple-low level was made a Safety Limit for all mode-switch positions.

2. At least two recirculation loop discharge and suction valves must remain in the full open position.

3. The time duration of the low-low level signal was required to be not greater than that used in the safety analysis for the limiting loss-of-inventory transient.

The NRC staff also recommended that the licensee consider the surveillance program and level instrument improvements.

It was concluded that no evidence of fuel damage was apparent, and that the facility could be safely returned to operation.

Based on the satisfactory actions taken by the licensee, on May 30, 1979 the NRC authorized the licensee to resume operation.

The possible generic implications of the Oyster Creek event have been considered. Nine Mile Point Unit 1 (operated by Niagara Mohawk Power Corporation and located in Oswego County, New York) and LaCrosse (operated by Dairyland Power Cooperative and located in Monroe County, Wisconsin) are the only reactors presently operating which are susceptible to a similar event. Immediate requirements similar to those which were required for Oyster Creek (Technical Specification changes 1 and 2) were implemented at these facilities prior to their start-up (they were both in a shutdown condition at the time of the Oyster Creek event). The third requirement will be implemented as soon as practicable.

Two other plants (Dresden Unit 1 and Big Rock Point), which are presently in extended shutdowns, would also be susceptible to a similar event. However, it is planned to impose appropriate requirements on these two plants prior to their startup.

In addition, on May 29, 1979 the NRC issued IE Information Notice No. 79-13, detailing this event, to all holders of operating licenses and construction permits.

Dated at Washington, D.C. this 22nd day of August 1979.

For the Nuclear Regulatory Commission.

Samuel J. Chilk,

Secretary of the Commission.

BILLING CODE 7590-01-M

[2] At the time of the event, the licensee's technical specifications defined the triple low water level as a Safety Limit when the reactor mode switch was in the "SHUTDOWN" mode only. A limiting safety system setting was also associated with the double low water level when the mode switch was in the "RUN" position. Even though the mode switch had been placed in the "REFUEL" position by the operator shortly after initiation of the transient, the event was regarded by the licensee as if a Safety Limit had been violated.

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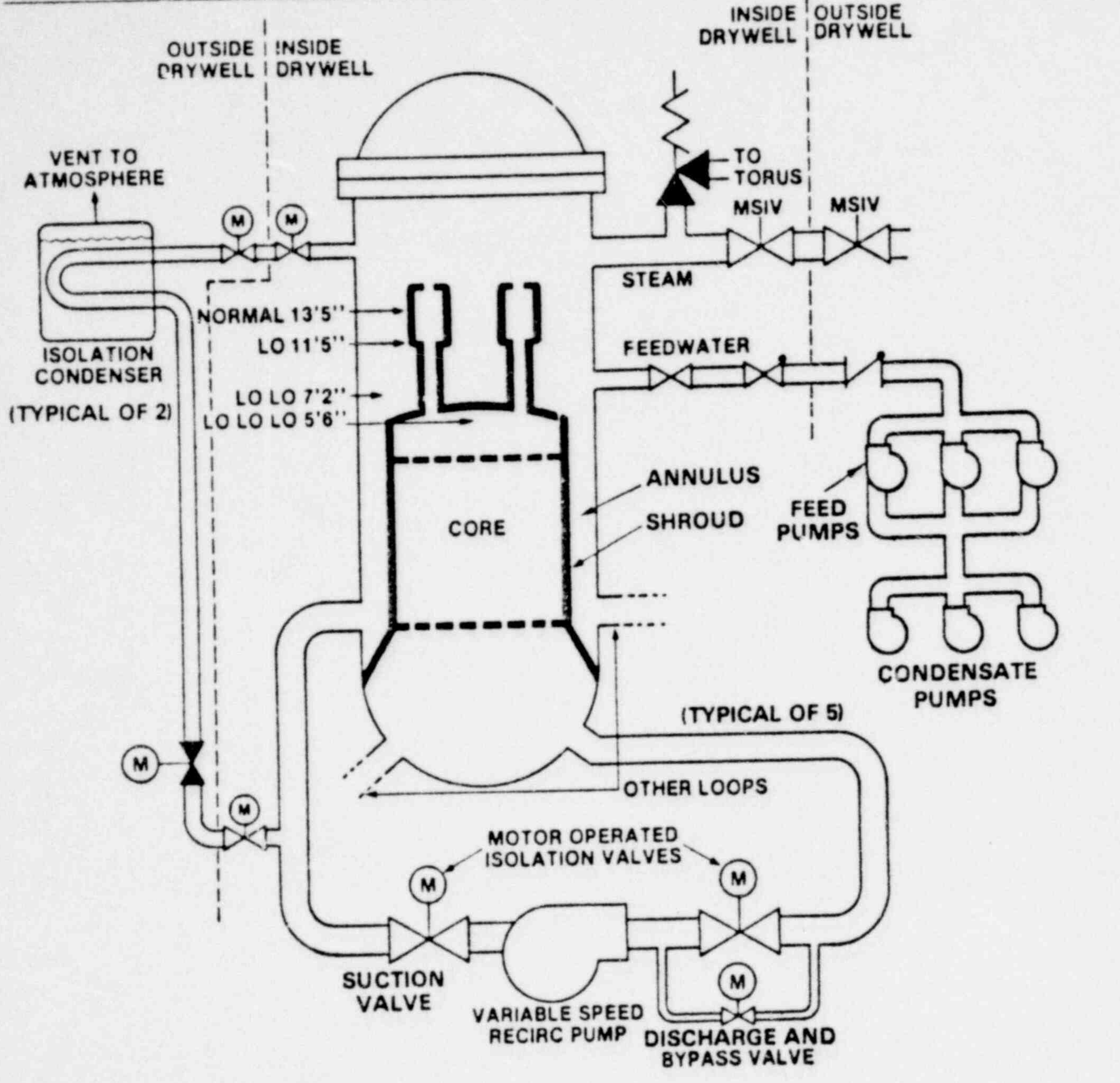

Figure 1. RECIRCULATION, STEAM AND ISOLATION CONDENSER SCHEMATIC

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