JUN 20 1985

50-395

MEMORANDUM FOR: C. J. Heltemes, Jr., Director
Office for Analysis and Evaluation
of Operational Data

FROM: Harold R. Denton, Director
Office of Nuclear Reactor Regulation

SUBJECT: ABNORMAL OCCURRENCE REPORT TO CONGRESS
FOR FIRST QUARTER CY 1985

We have reviewed your memorandum of June 4, 1985, forwarding the final draft of the Abnormal Occurrence (AO) Report to Congress for the First Quarter CY 1985. There is one proposed abnormal occurrence for commercial nuclear power plants, Premature Criticality During Start-up (V. C. Summer).

We agree that the unexpected criticality event at Summer should be classified as an Abnormal Occurrence, primarily because of the significant personnel errors involved. However, we note that the event is an Anticipated Operational Occurrence with consequences bounded by the uncontrolled rod cluster control assembly bank withdrawal transient, analyzed and discussed in Chapter 15 of the licensee's Final Safety Analysis Report.

Since your draft included additional discussion which we had not reviewed previously, we have enclosed marked-up pages of your draft report which include editorial comments. We have discussed our comments with Mr. Paul Bobe of your staff.

Original Signed by
H. R. Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosure:
As stated

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REPORT
TO Congress

REPORT TO CONGRESS ON ABNORMAL OCCURRENCES

JANUARY-MARCH 1985

NUCLEAR POWER PLANTS

The NRC is reviewing events reported at the nuclear power plants licensed to operate during the first calendar quarter of 1985. As of the date of this report, the NRC had determined that the following was an abnormal occurrence.

85-1 Premature Criticality During Startup

The following information pertaining to this event is also being reported concurrently in the Federal Register. Appendix A (see Example 9 of "For All Licensees") of this report notes that an accidental criticality can be considered an abnormal occurrence. (For a reactor, an "accidental criticality" can be defined as a criticality which is achieved when the approach to criticality is not being properly controlled by the plant operators.) In addition, general criterion 3 notes that major deficiencies in use of licensed facilities can be considered an abnormal occurrence.

Date and Place - On February 28, 1985, during a plant startup at about 1:30 p.m., the Virgil C. Summer Nuclear Station Unit 1 experienced an unanticipated transient which resulted in a high flux positive rate trip (automatic shutdown). The plant, which is operated by South Carolina Electric and Gas Company (the licensee), utilizes a Westinghouse-designed pressurized water reactor. The plant is located in Fairfield County, South Carolina.

Nature and Probable Consequences - During a nuclear power plant startup, control rods are withdrawn in a predetermined sequence to achieve criticality. In order to avoid rapid increases in power, three barriers of defense are used, i.e., personnel performance, procedural control, and reactor instrumentation to automatically scram (trip) the reactor. For the February 28, 1985, event, the first two barriers failed. Consequently, attaining criticality was not recognized and rod withdrawal was continued until the startup rate approached, by later estimates, 16 to 17 decades per minute (dpm).

At about six percent power, a reactor trip occurred on the high flux positive rate trip. The plant responded as designed to the reactor protection system actuation. The positive rate trip is derived from an increase of five percent of rated thermal power (RTP) within a two-second period. The limiting safety system setting is 6.3 percent of rated thermal power increase in two seconds. To obtain a positive rate trip during startup requires a reactivity insertion rate much greater than usually encountered. Since this was an uncommon occurrence, NRC Region II management directed that a special inspection be conducted of the circumstances associated with the event.

Both the licensee's and the NRC's investigations concluded that the event was caused by both personnel error and procedure deficiencies as discussed below in the sequence leading up to the positive rate trip.

Prior to a startup, a critical control rod bank position is estimated. This is done by first calculating reference critical data (RCD) to determine samarium and xenon reactivity effects. This data is then used to calculate estimated critical conditions (ECC). Poison concentrations and reactivities are corrected for buildup and decay from shutdown to the estimated time of criticality. Changes in control rod positions, boron concentration, and temperature are also taken into consideration. The calculation is considered acceptable by the licensee if the actual critical rod position is within 50 steps of predicted, otherwise an investigation of the cause of the error is required.

The reactor startup on February 28, 1985, at about 1:30 p.m. was preceded by a startup that same day at 6:30 a.m. The reactor was critical for approximately three hours prior to shutdown. The RCD was based on data taken for the brief period of criticality rather than data for equilibrium conditions from the previous power history. Therefore, when the ECC was calculated for the reactor startup at 1:30 p.m., the incorrect values of reactivity worth of poisons in the core were used. Additionally, the value used for control rod worth in the ECC calculation was based on middle of life (MOL) rod worth curves instead of beginning of life (BOL) rod worth curves. The station curve book provides rod worth curves for three times during core life; beginning, middle, and end of life. The reactor is presently between the BOL and MOL in Cycle 2, and the BOL curve would more accurately reflect rod worth. These two factors contributed to the miscalculation of the estimated critical condition by 128 control rod steps. The ECC predicted criticality at 168 steps on the Bank D control rods, while the actual critical rod height was later determined to be at 40 steps on Bank D.

Under the direct supervision of the shift supervisor, the control rods were withdrawn by an operator trainee with no previous reactor or simulator experience at this facility in withdrawing rods. The shift supervisor, believing the reactor would go critical at about 168 steps on Bank ~~B~~ D, instructed the trainee to withdraw the bank to 100 steps. This position, had the ECC been correctly calculated, would have left the reactor subcritical, even allowing for the 50 step margin of error discussed previously.

However, the trainee was not instructed in the need to anticipate criticality any time rods were being withdrawn or to closely monitor the available instrumentation for indication of criticality. Neither did the shift supervisor provide the necessary attentiveness or monitoring himself. Consequently, attaining criticality at 40 steps on Bank D was not recognized and rod withdrawal was continued until the reactor scrammed on the high flux positive rate trip. This occurred when Bank D reached 76 steps.

Two other licensed operators were on duty in the control room at the time. The operator at the controls was engaged in other startup-related activities on another part of the control board. The control room supervisor, a licensed senior operator, was at his assigned station, which afforded a good overview of the control room; however, his view of instrumentation important to this event was blocked by the shift supervisor and the trainee.

The actual safety consequences of the event were minimal. It is estimated that even if the positive rate trip had not occurred (failure of instrumentation or failure of the rods to scram), power in the core for Bank ~~B~~ D at 76 steps would have peaked at about 32% RTP due to the Doppler effect. In addition, if rod

motion for Bank D had continued to 100 steps (the shift supervisor's instructions to the trainee), and a positive rate trip had not occurred, the power peak is estimated to be about 43% RTP (again due to the Doppler effect).

However, the event is significant because it represented an unnecessary challenge to the reactor protection system, and because the reactor was not being properly controlled during plant startup.

Cause or Causes - The cause was primarily due to the failure of the shift supervisor (who was responsible for the trainee's actions) to be fully aware of plant status, to closely monitor instrumentation and to anticipate criticality whenever rods were being withdrawn as required by station procedures.

Contributing to, but not justifying the failure to monitor and anticipate criticality, was a calculated estimated critical position which was in error by more than 125 rod steps. The error in estimated critical position resulted, primarily, from procedural inadequacies.

Actions Taken to Prevent Recurrence

Licensee - The shift supervisor was removed from duty until the licensee completed an evaluation of the event, its causes, and the supervisor's capability to continue licensed operator duties. The supervisor was given formal counseling for failure to maintain an awareness of plant conditions during reactor startup. The supervisor resumed licensed operator duties on March 13, 1985.

Since there were deficiencies in the methods of estimating critical rod position for reactor startups, procedures used for the calculation of ECCs were revised to provide improved guidance for data usage and limitations for determination of core conditions for reactor startups. The station control rod curve book was also revised to clearly label burnup dependent curves with the appropriate burnup windows. This will provide a more accurate means of selecting the appropriate curves for ECC calculations.

NRC - As mentioned previously, an inspector from NRC Region II performed a special inspection from March 4-8, 1985, of the circumstances associated with the event. The inspection consisted of selected examinations of procedures and representative records, interviews with licensee personnel and observation of activities in progress.

As a result of the inspection, two violations of NRC requirements were identified: (1) failure of the shift supervisor to closely monitor instrumentation and anticipate criticality whenever rods were being withdrawn, and (2) inadequate procedures to estimate critical rod position within reasonable limits when the reactor was operated on an intermittent schedule at varying power levels. A notice of these violations, together with the inspection report, were forwarded to the licensee on April 3, 1985 (Ref. 1).

This incident is closed for purposes of this report.

material since an NRC reviewer might have made the requested changes if the errors had not been recognized.

(4) On June 23, 1984, the licensee provided additional proposed changes to the technical specifications, which as stated in (3) above, were "intended, in general, to enhance clarity or provide consistency with the plant design and operation." In one instance, the basis for a proposed change was false. The statement was material since an NRC reviewer might have made the requested change if the error had not been recognized.

(5) On August 5, 1984, the licensee certified in a letter that the technical specifications submitted to the NRC were accurate up to that time. However, the statement was false as reflected in an August 14, 1984, letter requesting additional changes to correct an error in the August 5, 1984, submittal. The statement was material since the NRC might have issued a license with erroneous technical specifications, had the licensee not subsequently corrected the error.

Numerous inspections involving these matters were conducted by the NRC and also several management meetings and Enforcement Conferences were held with the licensee. Written commitments were made by the licensee as a result of these meetings and inspection reports.

On March 21, 1985, the NRC forwarded to the licensee a Notice of Violation and Proposed Imposition of Civil Penalties in the amount of $125,000 (Ref. C-2). As stated in the forwarding letter, the primary responsibility for ensuring that the license contains appropriate technical specifications clearly rests with the licensee. The licensee's failure to fulfill its obligation to thoroughly know and understand the technical specifications which are a part of its license cannot be excused. The alleged material false statements listed in the Notice are indicative of a failure to exercise responsibility to ensure the accuracy and completeness of each and every submittal of information made or required to be made as part of the licensing process.

For the five alleged material false statements, the NRC considered proposing a civil penalty of $250,000 for these violations. However, in recognition of the fact that the informality of the NRC's process for review of technical specifications contributed to the problem, the proposed penalty was mitigated by 50%.

The NRC will closely monitor the licensee's corrective actions. Failure to carry them out satisfactorily could lead to further enforcement actions.

This event received considerable attention by Congress, the media and the public.

2. <u>Failure of Tendon Anchor Heads in Containment Post-Tensioning System</u>

On January 28, 1985, when Farley Unit 2 was shut down for refueling, inspection of the Unit 2 reactor containment building disclosed that a tendon anchor head had failed. Farley Unit 2 is operated by Alabama Power Company and is located in Houston County, Alabama. The plant utilizes a Westinghouse-designed pressurized water reactor which is housed in a post-tensioned concrete containment building.

The purpose of the tendons is to provide reinforcement to the concrete containment building by application of a compressive stress (i.e., post-tensioning force) to the concrete. When internal pressures are applied to the post-tensioned concrete, they are offset by the previously applied compressive stress. Failure of the tendon anchor head releases the post-tensioning force in the tendon.

The problem was discovered by a licensee employee who was conducting a pre-integrated leak rate test (an Appendix J, 10 CFR §50 requirement) walkdown of the exterior of the Unit 2 containment building. The employee noticed that a grease can (cap) covering the top of a vertical tendon was deformed. Inspection of the lower grease can on the same tendon disclosed that the lower grease can also was damaged. Removal of the lower grease can disclosed that the field anchor head had broken into seven pieces. In addition, numerous broken wires from the 170-wire tendon were found. Inspection of another tendon disclosed that the field anchor head on this tendon was cracked and separated into two pieces. The anchor heads and the tendons were supplied by INRYCO Inc., a subsidiary of the Inland Steel Company. Review of the tendon fabrication and installation records disclosed that the field anchor heads from both of these tendons had the same fabrication lot control number (i.e., lot control number HV).

Further review of the installation records disclosed that 47 other Unit 2 tendons had field anchor heads from lot control number HV. There were no anchor heads from lot control number HV installed in the Unit 1 containment. Based on manufacturing records, INRYCO concluded that there are no other anchor heads from lot control number HV installed at any other post-tensioned nuclear facility. In order to determine the cause of this problem, the utility implemented an extensive inspection and testing program. The inspection and testing program included visual inspection and replacement of the remaining 47 HV anchor heads, inspection of 55 randomly selected anchor heads from the non-HV lots, and performance of laboratory testing on the two failed and four other HV anchor heads.

The laboratory testing included chemical and physical properties, scanning, electron microscopy, as well as load testing. The testing was conducted at the Inland Steel Laboratory and at Battelle National Laboratory. Based on preliminary test results, available from both laboratories on February 24, 1985, the utility concluded that the failed anchor heads were not related to a specific lot control number. The primary failure mechanism was identified as hydrogen stress cracking, the cause of which was attributed to the presence of moisture around the anchor heads.

As a result, the licensee modified and expanded the inspection program to inspect all 130 vertical tendon anchor heads and all below-ground horizontal tendon anchor heads. Additionally, magnetic particle tests (MPT) were performed on the 24 HV lot anchor heads that had been removed from tendons. Eight of the 24 were found to have cracks when subjected to MPT. During the expanded visual inspection program, a third field anchor from a vertical tendon was found to be broken into five pieces. This anchor head was from lot control number HP, a different lot control number than the two previously identified failed anchor heads.

At a meeting held at the NRC offices in Bethesda, Maryland, on March 1, 1985, the licensee outlined a detailed program to resolve the tendon anchor head failures on Unit 2 and committed to perform an inspection of tendon anchor heads installed in the Unit 1 containment building.

An NRC Region II inspector, with extensive experience in tendon installation and tendon inservice inspection activities, performed detailed inspections of the utility's repair program and the activities to identify the cause of the problem. No violations or deviations were identified. In addition, test specimens were obtained by the NRC from two failed anchor heads (one each from HV and HP), and two non-failed HV anchor heads, for independent confirmatory laboratory testing. This testing was conducted at the Brookhaven National Laboratory (BNL) of Long Island, New York. The testing program was developed by NRC and BNL personnel. Based on the results of the testing, which were completed in early April 1985, BNL concluded that the tendon anchor head failures were caused by hydrogen stress cracking, which agreed with the results of the testing performed for the utility at Inland Steel and Battelle Labs.

The NRC issued Inspection and Enforcement Information Notice No. 85-10 on February 6, 1985, to all nuclear power reactor licensees to inform them of this event (Ref. C-3). The Notice also informed licensees of previous anchor head failures which occurred during construction of the Bellefonte and Byron facilities.

On March 8, 1985, the NRC issued Supplement 1 to Information Notice No. 85-10 to all nuclear power reactor licensees (Ref. C-4). This supplement updated information provided to the licensees on February 6, 1985, by identifying the cause of the tendon anchor head failures, presenting results of inspections performed by Alabama Power Company through March 1, 1985, and advising the licensees that the presence of moisture or free water during tendon surveillance activities should be considered as evidence of an abnormality and further action may be required.

The NRC has established a task force for a long-term program to identify and address the potential generic implications of the Farley Unit 2 event.

This event is not considered reportable as an abnormal occurrence since the problem involved [redacted] a [redacted] reduction in the degree of protection of the public health or safety. The integrity of a post-tensioned concrete containment structure is based on a highly redundant system of several hundred tendons. Bechtel Corporation (the containment structural designer) verified that containment integrity is maintained with as many as eight vertical tendons detensioned for the 40-year design life. More safety margin exists earlier in containment lifetime.

3. Recent Emergency Diesel Generator Failures

During the past several months. there have been a number of engine failures of emergency diesel generators (EDGs), involving various reactor plants and makes of EDGs (see Table 1).

Because the various problems have occurred in engines that are far from the end of their normal design lives, there is some feeling both in the industry and the NRC that testing requirements may have aggravated certain weaknesses and led to premature failures. In order to avoid potential failures, several options have been suggested, including (1) licensees' adoption of NRC's (Generic Letter 84-15; see Reference C-5) and manufacturers' recommendations on testing procedures to minimize stress and wear; (2) improved preventive maintenance, more frequent inspections, and improved operating practices by licensees; and (3) NRC