RESTART

SAFETY EVALUATION REPORT

BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

U.S. NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF

JERSEY CENTRAL POWER AND LIGHT COMPANY

OYSTER CREEK NUCLEAR GENERATING STATION

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I. INTRODUCTION

On May 2, 1979 the Oyster Creek Nuclear Generating Station experienced a sequence of events which caused the indicated reactor water level to fall below the "low-low-low" (triple-low) alarm setpoint. The triple-low level is sensed within the core shroud and corresponds to an elevation as low as 4'8" above the top of the active fuel. This water level corresponds approximately to the lower limit for direct instrument indication. Although the reactor had already scrammed when water level fell below the triple-low alarm setpoint, a question of adequacy of core cooling during the event was raised.

Region I of the Office of Inspection and Enforcement was notified by the licensee (Jersey Central Power & Light) on the day of the event. Inspectors from Region I went to the site. The Office of Nuclear Reactor Regulation was also notified. A team from ONRR went to the site May 3, 1979, to gain first-hand information. Investigations continued for several days thereafter by the technical staff of the licensee, the reactor vendor, licensee consultants, and the NRC. The investigations focused upon the contributing causes of the event and an assessment of the core condition.

On May 9, 1979, licensee representatives and its consultants met with the NRC staff to discuss the event, their analysis of the core condition, the corrective actions necessary to prevent reoccurrence and lessons learned. The licensee submitted a report pursuant to 10 CFR 50.36 (c)(1) for staff review by letter[1] dated May 12, 1979. Letters[2,3,4] from the licensee dated May 17, 1979 and May 19, 1979 forwarded additional information and requested authorization to restart the reactor.

This safety evaluation addresses the condition of the Oyster Creek core following the event of May 2, 1979 and the changes to the plant design and operation necessary to prevent recurrence. The report describes our review of the core condition, the licensing-basis loss of coolant inventory transient, the Technical Specification changes, the operating procedures, special startup tests, and other considerations.

A summary of the May 2nd event follows.

The Oyster Creek reactor was operating normally at approximately 98% power, with one of its five reactor coolant system loops and one of its two startup transformers out-of-service, when a simultaneous reactor trip and ATWS recirculation pump trip occurred. The cause of these trips was a momentary spurious high RCS pressure signal caused by routine surveillance testing of the isolation condenser initiation pressure switches.

As a result of the reactor scram and recirculation pump trip, reactor power, steam flow, pressure, water level, recirculation flow and turbine generator output began decreasing. At about 13 seconds into the transient the turbine-generator tripped at the low load trip point. This subsequently caused all three reactor feed pump motors to trip, because the backup electric power source supplied by the one available startup transformer was not capable of providing power to condensate and feedwater pumps sufficient to retain even partial feedwater supply to the reactor vessel. The reactor operator attempted at this time to restart a feedwater pump but was unsuccessful. Reactor water level continued to drop since the steam flow exiting the reactor was only being replaced by makeup from a single control rod drive pump. Recognizing the continued inventory loss from the reactor, the operator started a second control rod drive pump at 31 seconds into the event and initiated manual reactor isolation at 43 seconds. With the reactor fully isolated from the main condenser the operator manually closed the discharge valves of recirculation loops "A" and "E" which take return condensate from the two isolation condensers. The operator manually placed into service one of the isolation condensers at this time. It is believed that the operator also initiated closure of the "B" and "C" loop discharge valves about this time as a first step in starting one or both the associated reactor coolant pumps which had tripped at the start of the event. Additionally, as indicated previously, one loop (loop D) was already isolated and out-of-service, with its discharge valve closed prior to the event. All of the discharge valve bypass lines were open prior to and throughout the event however. As the discharge valves moved to the full-closed position the reactor vessel water inventory distribution continued to shift away from the core region toward the downcomer (annulus). At 172 seconds, the reactor low-low-low water level instrument trip point was reached. All discharge valves were fully closed at 186 seconds. Heat was removed from the system subsequently by alternately manually actuating and stopping

the isolation condenser. Reactor pressure and annulus water level increases and decreases were noted during this period and were caused by the intermittent isolation condenser operation. At approximately 32 minutes the operator started the C recirculation loop pump. The pump was shutdown and the discharged valve reclosed, however, when the operator observed water level in the annulus quickly dropping. At about 37 minutes one feedwater pump was restarted causing water level in the annulus to rapidly rise to 13'8" above the top of the core. At 39 minutes a recirculation pump was placed in service and the triple-low water level in the core region was observed to be cleared. At this time steps were initiated to bring the plant to a cold shutdown condition.

II. EVALUATION

II.A. Oyster Creek Core Condition

As part of our evaluation, we have reviewed calculations provided by the licensee of the minimum water level which could have existed over the Oyster Creek core on May 2, 1979. Additionally, we have reviewed the radioactivity and chemistry analyses of the plant provided by the licensee.

II.A.1 Minimum Water Level Over the Core

a) Reason for Level Calculations

Water level in the annulus was recorded during the event. However, due to partial isolation of the annulus from the core (discussed in Section I and in the attached Appendix), the minimum recorded level in the annulus did not correspond to the minimum level reached in the core region during the event.

The instrumentation that monitors water level in the core region is not recorded as a function of time. Rather, the core region level instrument provides visible and audible signals in the control room when core water level decreases below the alarm setpoint low-low-low level. The lowest alarm setting possible for the core region level instrument is 4 ft-8 in (56 in.) above the core, which is the elevation above the core of that instrument's pressure tap. On May 2, the setting for the low-low-low level alarm was 10" above that minimum or 5 ft-6 in (66 in) above the core. The time when that low-low-low level signal was received during the May 2 event was recorded (172 seconds after scram)[1] and this single point (level and time) represents the only direct core region water level measurement recorded during the incident.

Except for the first few seconds following scram, a sufficient condition to demonstrate lack of core damage is that the water level remained above the top of the core. Since the minimum incore water level was not measured, the calculations were performed to determine whether or not the core uncovered during the May 2 event.

b) Calculations for the First Few Seconds After Scram

Reactor scram caused a rapid power decrease for the first few seconds following the May 2 reactor trip. However, the recirculation pumps had also tripped simultaneously with the scram, so reactor flow was also decreasing. Transient Minimum Critical Power Ratio (MCPR) calculations were performed by Exxon Nuclear Company using their Plant Transient Simulator Code.[5]* Results of those calculations indicated that MCPR values increased from the steady state MCPR that existed prior to scram. Thus, acceptable cooling was maintained in the core during the initial rapid power and flow decrease period.[2] Physically, this means that the heat being transferred to the reactor coolant (a combination of stored heat and power being produced) decreased more rapidly than the coolant's ability to remove that heat was decreasing.

c) Minimum Level Calculations

Following the rapid power and flow decrease transient discussed above, a sufficient, but not necessary, condition to demonstrate lack of core damage is that the water level remained above the top of the core. Since the minimum water level above the core was not measured and/or recorded calculations were performed to conservatively determine the minimum level reached during the May 2 event.

Minimum water level calculations were independently performed by the General Electric Company (GE),[1] and the Exxon Nuclear Company (ENC).[2] The Nuclear Regulatory Commission (NRC) staff performed preliminary calculations in preparation for evaluating the other calculations. All of the calculations indicated that the core did not uncover.

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*The PTS model has previously been applied to Oyster Creek plant to determine MCPR values during transients.

Each of the above groups independently performed the same basic type of "boiloff" calculation. In addition, ENC performed a "mass inventory" calculation. All calculations were initiated by modeling the system that existed at Oyster Creek 172 seconds into the May 2 event. The initialized conditions are that the Main Steam Line Isolation Valves were closed, and all steam produced in the core went to the Isolation Condenser where it was condensed and returned to the annulus. Flow passed between the annulus and the core only through the five small (2" diameter) bypass pipes described below. The only change to the system inventory came from mass addition into the core region from the control rod drive (CRD) pumps. Other methods and conditions common to all of the water level calculations are described below.

1) The single measured water level inside the shroud, low-low-low level (66 inches above the core) at 172 seconds following scram, was used in calculating the "initial" (i.e., t=172 seconds) water inventory. The "initial" in-shroud water inventory was in turn used in the calculations of inventory at times later than 172 seconds. The calculated in-shroud inventories were then used to infer water levels above core at later times, the final result desired.

   Errors in calculating changes in the void content or distribution in the various regions inside the shroud after the 172 second calculation-initiation time would affect the final calculated water levels above the core. However, any bias in void content would tend to propagate through the calculations in such a manner as to "cancel", i.e., not affect the water level vs time calculations. This is because the initial inventory included effects of a calculated void content and distribution; the total amount of water that must be present in the core and upper plenum in order to hold 10" of water above the low-low-low level measurement tap (i.e., the low-low-low level alarm point) is dependent on the void content of the various regions below that tap. Stated differently, less voids below the measurement tap would allow water that was previously above the tap (and therefore measured) to drop to levels below the tap and no longer be measured. Thus, the void content in regions below the tap is important in determining the time at which the core level drops below the low-low-low level point. However, that time was taken from the actual low-low-low level measurement, thus automatically taking into account the correct, actual void content and distribution without regard to whether or not that void content and distribution was correctly predicted. As long as no major errors are made in predicting changes in void content due to changes in the parameters which affect void formation (after calculation-initiation time) then no significant errors in minimum calculated water level will be introduced. Since valves were opened or closed in the recircula-

tion flow path, no large recirculation flow temperature changes occurred, no large core power changes occurred, etc., large changes from the initial void content and distribution would not be expected, and were in fact not predicted by the calculations. Therefore, whatever voids were "holding the level up" when the initial (low-low-low) level measurement was made, would continue to "hold the level up" roughly the same amount during later calculated times. Thus, small errors in calculating changes in the void content would slightly affect later water level calculations, but errors in the understanding of absolute values of void content and distribution would "cancel out" of the calculations.[6]

In addition to the above, we believe no significant errors are present in the absolute values of void content and distribution. GE has compared the calculated void fractions with values from proprietary data which was taken over a mass flux and void fraction range which covers the values of mass flux and void fraction predicted by these calculations, with good agreement.[1] Also, ENC has provided a "maximum uncertainty in void fraction" sensitivity study showing that effects on minimum calculated water level due to errors in void fraction would be only 5 inches level decrease.[6]

Due to the acceptable agreement of calculated vs measured absolute void content, plus the lack of sensitivity of the calculated water level results to absolute values of void content and distribution, we find the treatment of void content in the level calculations to be acceptable.

2) Annulus-level-versus-time data were used to determine the initial (t=172 sec) inventory in the annulus and the pressure differential (head) available to drive water from the annulus region into the core region through the five 2" diameter recirculation-pump-discharge-valve bypass-valves and associated piping. Temperatures in the annular region were measured throughout the transient and remained subcooled; therefore, the void considerations discussed above for the core region are of no concern for the annular region.

3) Plant data were used to calculate flow resistance in the above mentioned 2" lines, i.e., the calculations assumed actual pipe size, thickness, material, roughness, length, number and type of bends, entrance and exit shape, and valve size and type in calculating flow resistance.

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4) Recirculation flow (flow from the annulus to the core) was calculated using the driving head and flow resistances determined as described in 2) and 3) above. All main recirculation pump discharge valves were assumed to be closed at the 172 second initial time.

5) Recirculation flow temperature was taken from measurements. Mixing of the cooler recirculation flow with warmer water in the lower plenum was not assumed. The cooler recirculation flow was assumed to stratify in the bottom of the lower plenum. If mixing in the lower plenum had been assumed, it would not have resulted in contraction and lowered level due to void collapse in the lower plenum (the lower plenum remained subcooled - i.e., there were no voids there to be collapsed. The stratification assumption therefore does not result in nonconservatism in the water level calculation. The stratification assumption does conservatively maximize the time before the cooler recirculation flow reaches the core (maximizes time before inventory losses from the core due to steaming would be reduced due to the cooler recirculated liquid reaching the hotter core region).

6) Steaming to the isolation condenser was assumed as that required to remove all heat produced by the following heat sources: a) best estimate decay heat as a function of time; and b) a conservative value of stored heat due to temperatures above saturation in the fuel, core internals, and coolant inventory. These values are functions of the decreasing saturation temperature (determined by the measured and recorded core saturation pressure as a function of time).

   The isolation condenser capability was evident from the measured pressure decreases when the isolation condensers were operating (pressure would increase if all steam were not condensed).

7) The assumed flow into the core region through the Control Rod Drive (CRD) pumps varied among the several groups' calculations from 40 gpm to 130 gpm. NRC staff calculation performed for 130 gpm and 100 gpm (65 and 50 gpm each). sey Central Power and Light believes that the two-pump total CRD flow directly to the core region is at least 60 gpm, and their best estimate is over 100 gpm.

40 gpm, CRD flow) probably contain conservatism resulting from assuming 20 to 60 gpm less CRD flow than actually existed.

Results of all calculations performed utilizing the above methods indicated a minimum two phase mixture water level above the core between 1.0 and 3.5 feet above the top of the active fuel (i.e., individual calculations from the groups were within that range). Time-of-occurrence of the minimum level varied from 7.5 to 32 minutes after scram. The NRC staff calculation indicated the most margin (3.2 feet) and the earliest minimum level time (7.5 minutes after scram). The other groups' calculations conservatively included lowering of core water level due to lower plenum contraction caused by recirculation flow, which the NRC calculations had neglected. Condensate from the isolation condenser flowing from the annulus through the five 2" lines into the hotter lower plenum (i.e., the recirculation flow) would cause the lower level due to volume shrinkage in the lower plenum.

The other groups' calculations also conservatively included a larger amount of core-intervals and fuel-stored-heat being removed (by steaming) from the core than did the NRC staff calculations. Approximate correction of the NRC staff calculations for these effects results in reasonable agreement with the other calculations (i.e., within the range of the other results). The staff's calculations were preliminary in preparation for evaluating the licensee's calculations.

To further alleviate any potential concerns regarding the role of void calculations and assumptions on the minimum calculated water level, GE performed a calculation which removed the "credit" for voids at times after t=172 seconds but kept the penalty for voids at t=172 sec. That is, in calculating initial core water inventory, the measured level at t=172 seconds was corrected (reduced) for calculated void content present at that time. The calculation then started with this artificially reduced inventory; reduced (collapsed) water level vs time was calculated assuming no swelling in the core or upper plenum (residence time of voids in the core of zero, or all steaming occurs at the upper surface are equivalent conservative assumptions). This calculation, even with these conservative assumptions, predicted a minimum collapsed level of 1.67 feet above the core.

Also, additional calculations performed by Exxon Nuclear Company utilized a different basic method (a mass inventory allocation process) to distribute the available mass through the system depending upon known volumes along with measured levels and measured thermo-dynamic conditions. These calculations shared dependence with the other calculations on void distribution and initial inventory distribution. However, they did not share dependence on heat transfer and steam production calculations since inventory in the core region was inferred by tracking all other regions (with available, recorded measurements taken during the May 2 incident) and subtracting the sum of the masses in the other regions from the initial total system mass (which was constant except for an assumed 40 gpm CRD flow). Results of those calculations were in agreement with the first set of calculations described, predicting a minimum two phase level of 1.62 ft. above the core at 12 to 32 minutes after scram.

a) Conclusions

On the basis of the acceptable MCPR calculations reported above in Section (b) and on the basis of agreement of all (independent) calculations reported in Section (c) that the water level remained above the core, and the conservatisms described that are present in the methods used, we conclude that the two phase mixture water level did not drop below the top of the core during the May 2 event and no fuel damage occurred.

II.A.2 Primary Coolant and Off-Gas Analyses

The licensee and the staff have examined the radiochemical records for empirical evidence of core damage. The primary coolant sample analyses, from before and for several days after the transient, showed no unusual increase in the concentrations of radionuclides. The Iodine-131 concentration went up by a factor of two at shutdown but iodine spikes of that magnitude at shutdown are normal due to reactor system depressurization.

The readings from the stack and steam air ejector radioactivity monitors are continuously recorded on a strip chart. The strip chart around the time of the incident showed no unusual increase in the release of airborne radioactivity. There were spikes in the stack reading at shutdown and when the mechanical vacuum pumps were started. But again, off-gas increases are normal at shutdown and when mechanical vacuum pumps are started. Thus, there is no indication from either the primary coolant analyses or the off-gas rates

that there was any abnormal release of fission products from the fuel due to the transient. Therefore, we agree with the licensee's conclusion that the radiochemical records provide evidence that the core was not damaged as a result of the event.

II.B Licensing Basis Loss of Coolant Inventory Transient

The licensee has submitted [3] an analysis of the most severe postulated loss of reactor coolant inventory transient at the Oyster Creek Nuclear Generating Station. The purpose of the analysis is to show that with the revised Technical Specifications water will not fall below the low-low-low level. Our evaluation of the licensee's bounding analysis is provided in the following sections.

II B.1 Low-Low-Low Water Level Safety Limit

At the time of the May 2, 1979 event, paragraph 2.1.D of the Oyster Creek Technical Specifications defined a water level of 4'-8" above the top of the normal active fuel zone to be a fuel cladding integrity safety limit. Technical Specification 2.1.2 states: "Whenever the Reactor is in the shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than 4' 8" above the top of the normal active fuel zone." The purpose of this limit was to assure adequate margin for decay heat cooling of the fuel during periods when the reactor is shutdown and corresponds to the lowest reactor vessel water level that can be monitored. As a result of the event of May 2, 1979, however, it was recognized by both the licensee and us that the low-low-low water level safety limit is applicable to all operating modes. We and the licensee agree that a water level above the core that can be monitored is an appropriate basis for concluding that significant fuel failure does not occur.

Accordingly, the licensee has proposed,[4] that the subject Technical Specification definition be changed to make clear that the low-low-low water level (4' - 8" above the top of the active fuel zone) is a safety limit applicable to all modes of operation including transient conditions. Low-low-low water level thus becomes the safety limit applicable to the licensing basis loss of coolant inventory transient. This is acceptable.

II.B.2 Bounding Event Description

In order to assure that Oyster Creek will not violate the triple-low water level safety limit during any anticipated operational occurence, the licensee has analyzed the postulated transient event which results in the largest loss of reactor coolant system inventory. For Oyster Creek, the licensee states[3] that a total loss of feedwater (LOFW) starting from hot full power, results in the most severe reduction in reactor vessel water levels.

For the LOFW event, feedwater flow to the reactor vessel rapidly falls to zero. Thus, with full power reactor heat generation continuing, steam flowing from the reactor to the turbine causes reactor water level to decrease rapidly. For Oyster Creek, a reactor scram occurs when water level in the annulus reaches the "low" water level set point which corresponds to a height of 11'-5" above the top of the active fuel region. The reactor scram causes a further rapid drop in water level in the annulus as the reduction in heat generation rate results in a marked decrease in the core void content. Steam generation continues caused by core decay heat and stored heat effects. This steam continues to exit the reactor thereby causing a continuing water level decrease in the vessel. When water level in the annulus reaches the "low-low" water level setpoint, corresponding to 7'-2" above the top of the active fuel region, main steam line isolation valve closure will automatically initiate to terminate inventory loss from the reactor vessel. Inventory loss is completely terminated when the MSIVs are fully closed. For Oyster Creek, the low-low water level also initiates an ATWS pump trip. A small water level swell occurred in the vessel annulus due to the reduced core flow and the increased core voiding. Additionally, low-low water level in the annulus corresponds to the isolation condenser initiation set point. Thus, after a short time delay the drain valves of the isolation condenser would start to open to remove core decay heat and stored heat from the isolated reactor vessel. Since the isolation condenser piping system normally is filled with liquid water, some inventory makeup can also be supplied to the reactor vessel when the system actuates. With the isolation condensers actuated the reactor system would depressurize and cool down. Continued depressurization cooldown, and shrinkage of the contained inventory would occur until core spray flow would restore the decreasing water levels in the reactor vessel.

We have compared the LOFW event to other transient events postulated for Oyster Creek. We agree that the LOFW transient described above will result in the largest inventory loss from the reactor vessel should it occur. However, the distribution of this inventory within the reactor vessel (e.g. downcomer and core regions) is dependent on reactor coolant pump condition (running or not running) and recirculation loop discharge valve and bypass valve positions. For analysis purposes the tripping of all recirculation pumps at low-low water level, conservatively accounts for the shifting of reactor vessel inventory away from the core region and towards the annulus region. The effect of recirculation loop valve positions is explicitly accounted for in the analysis assumptions (see Section II.B.4) and in the plant Technical Specifications (see Section III).

II.B.3 Codes and Methods

The calculational methods which were used to determine the minimum water level over the core in the event the limiting loss of coolant inventory consists of two parts. The first part utilizes the Exxon Nuclear Company PTSBWR2 plant transient analysis code[5] to calculate reactor vessel inventory and water levels for the first 125 seconds of the transient. The second part utilizes a degenerative (special) case of calculational methods discussed in Section II.A.1, herein, to extrapolate to the minimum water level over the core during the cooldown-depressurization phase (when the isolation condensors are operating) until core water level recovery occurs as a result of core spray system flow initiation. Additionally, the second part includes methods to assess the effect of discharge valve position on the steady-state water level in the core region.

The PTSBWR2 Code which has been used in connection with previously accepted Oyster Creek plant transient simulations for core reload applications, was modified to model the automatic initiation and heat removal characteristics of the isolation condensers. The addition of this model thereby enables simulation of the steam condensing (heat removal) function of the isolation condensors subsequent to reactor vessel isolation. Conservatively, no credit for the inventory associated with the subcooled water normally stored in the isolation condenser was included in the revised version of the PTSBWR2 Code. Both isolation condensors were modeled in the analysis. A time delay from the time of low-low water level to initiation of the isolation condenser drain valve opening was also included.

The effects of discharge valve position on steady-state water level in the core region was evaluated with a hydraulic analysis of the recirculation lines. This analysis modeled the recirculation line geometry with standard fluid mechanics methods. The methods included the geometric pressure loss coefficients which includes a factor for the fixed rotor recirculation pump. The pressure loss coefficient for the recirculation pump was based on in situ tests. The other pressure loss coefficients involve standard methods and are adequate. The analysis assumed differential driving heads between annulus and core regions which are within the range of values assumed for the overall analyses. The methods were used to calculate the natural circulation flow from the downcomer region to the core region.

II.B.4 Assumptions

The licensee's bounding analysis assumptions(3) which can significantly affect the calculated ractor coolant system inventory lost during the transient have been evaluated, together with the assumptions which can adversely affect the calculated distribution of inventory between the vessel annulus and the core region. Collectively these assumptions should result in a conservative prediction of the minimum water level over the core during the transient.

Inventory Loss Assumptions

The analysis was performed assuming an initial full power level of 1930 MWt. This power level, in conjunction with the assumed low reactor water level scram, will maximize the rate of inventory lost from the reactor vessel up until the complete closure of the main steam line isolation valves. The total reactor coolant inventory lost from the reactor vessel up to the time of MSIV closure has been conservatively modeled. The analysis assumes feedwater flow into the reactor vessel falls to zero in 3.5 seconds with MSIV closure initiated on low-low water level in the annulus. Additionally, the MSIV closure time is the maximum (10 seconds) permitted by the present Oyster Creek Technical Specifications. To prevent additional reactor coolant inventory loss which might otherwise occur due to system repressurization (after MSIV closure) the analysis takes credit for the heat removal and pressure control

functions of the Oyster Creek isolation condensers. The analysis assumes automatic actuation and operation of both isolation condensers for heat removal and pressure control, although no credit has been taken for the subcooled inventory of water normally stored in the isolation condenser piping. The analysis results are based on automatic initiation of the isolation condensers caused by the decreasing water level in the annulus being sustained at or below the low-low water level setpoint for 10 seconds. After the 10 second time delay, the isolation condenser drain valves are assumed to open fully in 20 seconds. Additionally, the analysis conservatively takes no credit for the small source of inventory makeup associated with control rod drive flow.

In summary, it is assumed that after MSIV closure no reactor coolant inventory loss or makeup occurs until core spray flow terminates the decrease of core water level.

Inventory Distribution Assumptions

The actuated isolation condensers are assumed to depressurize and cooldown the contained reactor coolant mass to a reactor pressure at which core spray flow makeup would start to raise reactor vessel water levels. The cooldown results in an increase in reactor coolant density, thereby causing an additional drop in reactor vessel water levels. The core water level analysis assumes no voids are present in the system at saturation conditions. Thus, the actual height of the two phase mixture in the core region is conservatively accounted for from a density viewpoint. Finally, the distribution of coolant inventory (between annulus and core) has been accounted for based on no forced recirculation flow (due to a reactor coolant pump trip on low-low water level) and a maximum of one unisolated recirculation loop. The above conditions will result in the most adverse distribution of coolant inventory within the reactor vessel.

In summary, the above combination of inventory loss and inventory distribution assumptions provides an adequately conservative basis upon which to calculate the minimum core water level attained during the limiting loss of coolant inventory transient.

II.B.5 Results

The results of the limiting loss of coolant inventory transient from initiation to 125 seconds, as calculated by the PTSBWR2 Code, are provided in Reference 3. The results show that vessel annulus water level drops rapidly reaching the low level reactor scram setpoint corresponding to 11'5" above the top of the active fuel within 4.5 seconds. At 15 seconds, the low-low level setpoint, corresponding to 7'2" above the top of the active fuel, is reached initiating MSIV closure and a trip of all reactor coolant pumps. Core spray pumps are signalled to start at this time although reactor pressure is sufficiently high to prevent any inventory addition. The voiding in the core caused by the tripped recirculation pumps causes level in the annulus to start increasing and recovering low-low water level after approximately 3.4 seconds. This result is not consistent with the 10 second sustained low-low water assumed for initiating opening of the isolation condenser drain valves described in Section II.B.4. However, the licensee has committed to propose Technical Specifications (see Section III) which will acceptably resolve this inconsistency. The proposed technical specifications will require a sustained low-low water level for three seconds or less to initiate opening of the isolation condenser drain valves. In view of the predicted margin to low-low-low water level for this limiting (see discussion below) event we consider giving credit for prompt manual initiation of the isolation condensor subsequent to reactor isolation acceptable until the proposed technical specification is implemented. The minimum annulus water level after MSIV closure and before cooldown depressurization begins is 5.36 ft. above the top of the active core and occurs at approximately 35 seconds. However, continued depressurization, cooldown, and shrinkage of the contained inventory occurs until core spray flow recovers the decreasing water levels in the reactor vessel. Based on the methods and assumptions (evaluated in Sections II.B.3 and II.B.4, respectively) used to extrapolate the water inventory distributions and levels, the minimum attained collapsed water level[6] is 6'7" above the top of the active fuel. This result includes the effect of recirculation loop discharge valve positions on steady-state water levels. That is, with only one recirculation loop assured unisolated, recirculation flow is sufficient to prevent boiloff from reducing core water level below 6'7" above the top of the active fuel.

II.B.6 Conclusions

The above result is acceptable in that the low-low-low water level fuel cladding integrity safety limit is not violated.

Our evaluation of the revisions to the plant Technical Specifications which are considered necessary and sufficient to complete the implementation of the important analysis assumptions and results appears in Section III.

III. TECHNICAL SPECIFICATIONS

The licensee has proposed several changes to the Oyster Creek Technical Specifications to clarify the appropriate limits for transient events which result in a loss of reactor coolant inventory and to provide assurance that the reactor coolant system configuration and mitigating equipment taken credit for in the bounding loss of coolant inventory analysis will be in accordance with the analysis assumptions. A discussion of these changes follows.

III.A Safety Limits

As discussed in Section II.B.1, the licensee has proposed[4] that the definition of the low-low-low water level fuel cladding integrity safety limit appearing in paragraph 2.1.D of the plant Technical Specifications, be clarified to specifically provide for applicability to all modes of reactor operation. Based on our evaluation in Section II.B.1, this is acceptable. The licensee has also proposed[4] to add a safety limit appearing as paragraph 2.1.F in the plant Technical Specifications which requires that during all modes of operation (except when the reactor head is off and the reactor is flooded to a level above the main steam nozzles) at least two (2) recirculation loop suction valves and their associated discharge valves will be in the full open position. Based on our evaluation appearing in Section II.B. herein, the acceptability of this requirement is conservative relative to the assumptions used in the bounding loss of coolant inventory transient analysis.

III.B Limiting Safety System Settings

The licensee has taken credit for the automatic protective operation of the isolation condensers for acceptably terminating the limiting loss of coolant inventory event. To assure the proper initiation and operation of the isolation condensers on low-low water level in the annulus in accordance with the bounding analysis assumptions, the licensee has proposed to add a limiting safety system setting requirement to Section 2.3 of the Oyster Creek plant Technical Specifications. The Specification will state that the limiting safety system setting is the low-low water level setpoint which was assumed in the bounding analysis, i.e., 7'2" above the top of the active fuel. The limiting safety system setting will incorporate a maximum three (3) second time delay to assure that the system will not fail to initiate because low-low water level momentarily clears as a result of the water level swell in the annulus caused by a simultaneous recirculation pump trip. Additionally, based on our review of

actual plant operating data of isolation condenser initiations and possible isolation, a time delay of three seconds or less will not cause the isolation condensors to reisolate on high flow conditions caused by recirculation pump coastdown effects. This time delay will also be adequate for recirculation flow coastdown effects applicable to four loop operation as well. The limiting conditions of operation and surveillance requirements for the isolation condenser will not be changed.

IV. OPERATING PROCEDURES

We have reviewed both the operating procedures (including standing orders) which were in effect at Oyster Creek at the time of the May 2, 1979 event, and the revisions of these operating procedures as a result of the event. The former procedures were reviewed to evaluate whether the operator actions during the event were wholly in conformance with the procedures then in effect. The revised procedures were reviewed to evaluate their consistency with the bounding analyses assumptions (discussed in Section II.B.4) and the resulting technical specification changes (described in Section III).

IV.A Operator Actions

The following is an evaluation of the correctness of the operator actions relative to the plant operating procedures which existed at the time of the May 2, 1979 event. Our evaluation is itemized by procedure.

1. Procedure 514, Rev 2 "Reactor Isolation Scram"

   This procedure is pertinent to the May 2, 1979 event since the operator manually closed all four main steam isolation valves 43 seconds after the reactor scrammed to minimize the loss of coolant inventory. The closure of the MSIVs, although not specifically required in the particular procedure, was the proper action to take and was taken promptly. This procedure requires the operator to verify that a reactor isolation was initiated if a reactor low-low water level or reactor high-pressure condition exists. The low-low water level is measured in the downcomer (annulus) but was never reached during the event. The high reactor pressure signal which occurred was spurious and was not sustained for the delay time needed to initiate isolation cooling. The subject procedure requires the operator to manually actuate systems that have not automatically actuated. Thus, he correctly actuated the isolation condenser in order to establish an alternate heat sink.

   All appropriate immediate and subsequent operator actions were completed by the operators as required by this procedure.

2. Procedure 511.1, Rev 1 "Feedwater Pump Failure"

This procedure states that a reactor low-low level condition may be experienced in the case of a trip of all three feedwater pumps. However, since the low-low water level condition was not attained for the subject event some of the automatic actions listed in the procedure did not occur. Among the significant immediate and subsequent operator actions required for this situation is to restart one or more feedwater pumps. Ten seconds after scram the operator did make an attempt to restart the only feedwater pump powered by a live bus. The control room operator was unsuccessful since a tripped overload condition existed on the motor drven auxiliary oil pump. No further attempt was made to restart this only available feedwater pump until 31 minutes and 54 seconds after the scram since the low water level alarm had cleared at 90 seconds and water level was normal. 31 minutes and 54 seconds, after the reactor scrammed the operator started the "C" recirculation pump which resulted in a level decrease in the downcomer. At this time, the operator made a second unsuccessful attempt to start the feedwater pump. However, operating personnel dispatched to the feedwater pump station locally started the auxiliary oil pump allowing feedwater pump A to be successfully restarted at 36 minutes and 48 seconds. No procedure violations occurred and all actions taken were in accordance with the procedure. The time delay to locally start the auxiliary oil pump appears to be somewhat long but is not considered unreasonable since other operator actions were being taken at the time. Additionally, since water level in the annulus during most of the event was normal, this delay is understandable.

One of the subsequent operator action steps required by the procedure is to place a recirculation pump back into service. This was done at 31 minutes, 54 seconds but the pump was immediately tripped manually. Another recirculation pump was started 36 minutes and 48 seconds after the scram.

3. Procedure 301, Rev 4 "Nuclear Steam Supply System"

This procedure addresses routine operation including startup and shut down of the main steam and recirculation systems. Section 7.0 of this procedure is entitled "Removing a Reactor Recirculation Pump from Service." The "Precautions and Limitations" subsection includes the

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following statements: "Never isolate all recirculation loops at the same time. The suction and discharge valves of at least one recirculation loop shall always remain open and, if possible, at least one pump should always be running to provide continuous circulation and indication of reactor vessel water level." This condition was violated since all five recirculation discharge valves were believed to be simultaneously closed 76 seconds after the scram and were definitely observed to be closed 186 seconds after scram. Furthermore, all five discharge valves remained closed until 31 minutes, 54 seconds after the scram when a recirculation pump was started and its associated discharge valve reopened. Although both the violation and precaution in the procedure are considered to be clear, the subject event caused some complications which may have contributed to the procedure violation. Standing Order "23 in effect at the time of the event requires the operator to close the A and E loop discharge valves to prevent the isolation condenser from isolating itself as a result of high flow conditions. The "D" loop discharge valve was closed prior to the event since the associated pump was out of service. The above precaution required the operator to have at least one pump running. To start one of the pumps, it is necessary (by procedure) to first close the associated discharge valve. The logical pump to start was pump C since it was powered by a live bus and was not in a loop connected to the isolation condenser. The pump was started at 31 minutes, 54 seconds. Therefore, the operator was required to close three discharge valves but made the error of closing four discharge values.

There would appear to be some basis for confusion since the term "isolated" as used in the procedures can be inferred to describe either closing the discharge valve or closing both the discharge valve and the discharge bypass valves simultaneously. During the entire event, all five discharge bypass valves were open, however.

4. Procedure 307, Rev 3 "Isolation Condenser System"

   Standing Order No. 23 Rev 0 (dated 11/15/77) -"Isolation Condensor Operation"

   The section of Procedure 307, applicable to this event requires the operator to control reactor pressure and limit the cooldown rate to less than 100°F/Hr. Procedure 307 does not mention the relationships between isolation condenser and "A" and "E" bypass and discharge valves.

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closing the discharge valves is intended to prevent automatic isolation of the isolation condenser system. Based on our review, we believe that there were no violations of any of the steps in either the procedure or standing order.

Procedure 502.3 Rev 1 "Loss of 4160V Bus 1A, B, C, D"

Procedure 510 Rev 2 "Turbine Trip"

Based on our review of these two procedures and the operator actions we believe that no procedure violations occurred during the May 2, 1979 event.

IV.B. Revised Plant Operating Procedures

The Office of Inspection and Enforcement has reviewed the revised Oyster Creek operating procedures to evaluate whether they adequately implement the revised Oyster Creek Technical Specification requirements.

V. START-UP SURVEILLANCE PROGRAM

As stated in Section IIA2, there is no indication from the primary coolant concentrations or the off-gas rates that any abnormal release of fission products from the fuel occurred during the transient. However, some fuel damage can be detected in either the primary coolant concentrations or the off-gas rates during startup and ascension to full power. The licensee has designed a surveillance program to identify signs of fuel damage occurring during restart. This program, which the licensee has committed to during this startup, is described below.

The off-gas rates from the air ejector and the stack will be continuously monitored. The primary coolant will be sampled and analyzed for gamma-emitting radionuclides on the following schedule:

1) Pre-startup
2) 250°F average reactor coolant temperature
3) 500 psig reactor system pressure
4) 20% thermal power
5) 10% thermal power increments up to full power
6) Daily for 14 days after reaching full power

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Air ejector off-gas samples will be taken and analyzed for isotopic content on the following schedule to ensure proper calibration of the continuous off-gas monitors:

1) 20% power
2) 40% power
3) 60% power
4) 80% power
5) full power
6) Weekly for 14 days after reaching full power.

Two additional air ejector samples will be taken each week for 14 days after reaching full power. These samples will be analyzed for gross gamma only. If the ratio of short and long-lived emitters changes, further sampling and analysis will be performed.

The surveillance program includes adequate frequency of sampling, analysis and monitoring of the primary coolant concentations and off-gas rates to ensure that signs of abnormal fission product release resulting from the transient will be identified quickly.

These are the criteria by which the licensee has committed to judge the information on primary coolant concentrations and off-gas rates from the surveillance program. Iodine-134 and Iodine-135 will be used as the indicator nuclides in the primary coolant analyses. These nuclides reach equilibrium concentrations for the various power levels quickly because of their short half-lives. The bases for the criteria will be the primary coolant concentrations and off-gas rates experienced at Oyster Creek at full power before the transient. All of these criteria will be applied to both the primary coolant concentrations and the off-gas release rates. For startup evaluations up to 50% power, no action will be taken if 100% of the base levels are not exceeded. If 100% of the base levels are exceeded, power level will be held and the samples and analyses repeated until the 100% criteria are met. If 200% of the base levels are exceeded, the licensee will promptly initiate a reactor shutdown until the problem is resolved.

For startup evaluations between 50% and full power, no action will be taken if 150% of the base levels are not exceeded. If 150% of the base levels are exceeded, power level will be held and the samples and analyses repeated until the 150% criteria are met. Again, if 200% of the base levels are exceeded, the licensee will shut down the reactor until the problem is resolved. For two weeks after reaching full power,

the licensee will continue evaluations of the off-gas rate. If 125% of the base level is exceeded, the sampling and analysis program will be augmented. If 150% of the base level is exceeded, the licensee will reduce power to stay within 150% of the base level. If 200% of the base level is exceeded, the licensee will shut down the reactor until the problem is resolved.

Before the incident, stack off-gas rates at full power were running at approximately 40,000 microcuries per second. A stack off-gas rate of 125%, 150% or 200% of this base level would still be less than one-third of the rates allowed by technical specifications. Therefore, the criteria are acceptable. We request that the licensee notify us within 24 hours if the reactor is shut down based on these criteria.

VI. OTHER CONSIDERATIONS

VI.A. Water Level Indication

The level instrumentation in the Oyster Creek reactor reads only "collapsed" water level. Such a collapsed level is an indicator of water inventory, but does not necessarily correspond to a liquid surface height. This distinction is especially true within the core area during operation, where liquid quality increases monotonically from the boiling boundary up to the steam separator, with no distinct liquid/vapor interface.

In the annulus, collapsed level roughly corresponds to the liquid level. Moreover, the water inventory within the annulus is generally greater than core inventory during operations and when annulus level is in the normal range, it is above the bottom of the steam separator skirts. Thus, the whole core area is submerged under these conditions. Also, water is normally being drawn from the annulus and forced into the core. Thus, when supply is interrupted or inventory is lost, it is the annulus level which will go down at first, while core inventory will not change greatly.

The annulus level is continously monitored by three electric level (GE/MAC) gauges (two narrow range and one wide range) and 8 Yarway level gauges. The Yarways read out in the control room and are used as inputs to the high, low, and low-low level setpoints. The GE/MAC gauges are recorded as well as read-out in the control room. Moreover, the GE/MAC signals are used for feedwater control. One of the narrow range GE/MAC signals is for a strip chart recorder in the control room.

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The water level in the core area is monitored by four Barton level gauges. The four Bartons tap into the core spray sparger for their lower tap, and share reference columns with the GE/MAC instruments described above. The Barton gauges read-out locally, but only send a signal which initiates the low-low-low level logic plus a control room alarm. Thus, core level is monitored only in the sense of an alarm signal; it is neither recorded nor observable by a control room operator.

At this point, it is essential to understand the purposes of the three low level setpoints. The low level setpoint is above the lower edge of the steam separator skirts. Actuation of this setpoint causes a reactor trip and a group II isolation. It is normal to reach the low setpoint following a reactor scram at power due to void collapse.

The low-low level causes a group I isolation (which includes the MSIVs), trips the recirculation pumps, initiates isolation cooling and starts the core spray pumps. (However, core spray flow will not start unless the reactor is depressurized.)

The low-low-low level starts the Automatic Depressurization System (ADS) timer provided there is coincident high drywell pressure and the core spray pumps are operating. This is the only use of the low-low-low level signal.

Limitations of Water Level Indication

Although annulus level is appropriate for feedwater control and water inventory monitoring during normal and most upset conditions, it has no intrinsic safety significance except through its relationship with core water level. The annulus, core area and recirculation lines form a large U-tube when the recirculation pumps are not running, and the two levels should be very nearly the same. When the recirculation pumps are running and the core is shut down, the level in the annulus should be lower than the (collapsed) core level, and therefore should be a conservative indicator. For the annulus level instrumentation to work properly, the annulus and the core area must be in good communication at the bottom. It is now apparent that the non-conservative situation (annulus level greater than core level) can exist if there is a restriction in the recirculation lines. (This is only possible in non-jet-pumps BWRs, since the more modern plants always have good communication between the two regions through the jet pumps.)

The core water level instrumentation provide meaningful results only when there is no liquid flow through the steam separators. When there is flow through the separators, the resulting differential pressure introduces a non-conservative error in the core water level reading. This is of no conse-

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quence during power operation, since the core is filled with a two-phase level-less mixture, and inventory is monitored in the annulus. The core area water level does become meaningful under low separator flow conditions. Thus, core area level indication will not work unless either the recirculation pumps are tripped or the collapsed core level drops so far that only steam enters the separators. This is the reason why the core level instrumentation is often called "accident" instrumentation: the instrumentation is not operative under normal and most upset conditions.

In addition to the limitation described above, collapsed core water level is not linearly related to core water inventory. The horizontal cross-sectional area of this water volume is large above the core, narrows rapidly through the dome, and is small through the standpipes. Therefore, a constant inventory loss (in gallons per minute) will cause collapsed level to drop very rapidly out of the standpipes, but much more slowly in the large cylindrical volume immediately above the core. The low-low-low setpoint is normally about half-way up the transitional (dome) area between the two volumes.

Finally, the lower taps of the core area level instrumentation are on the core spray spargers. The instrumentation cannot monitor water levels below these spargers.

Summary

The primary safety concern for level instrumentation is that the level setpoints must be assured to occur in proper sequence. This implies that the core and annulus water volumes must not be partially isolated from one another. Given this, all safety analyses remain valid and bounding.

In addition, it is recommended that a read-out of the core level instrumentation be provided in the control room. This read-out could inform plant operators during an accident situation. Such a read-out is provided in new plants.

The licensee plans [4] to add level instrumentation with a tap at a still lower elevation (e.g. the core differential pressure tap) be investigated in the longer term. Presently, this plant cannot monitor levels below the core spray sparger. Although such situations are not likely and also have been bounded by accident analyses, we consider additional level instrumentation prudent.

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VI.B Potential for Transients Due to Surveillance Testing

The event on May 2, 1979, at Oyster Creek occurred when a gauge valve was opened to verify that an excess flow check valve in the instrument line was still in the open position. The instrument line feeds pressure switches which actuate the reactor and recirculation pump trip systems. A simultaneous reactor trip and recirculation pump trip resulted from this surveillance testing of the isolation condenser pressure switches. Confirmation that the check valve is open is necessary after the surveillance test to assure that the pressure switches are hydraulically communicating with the reactor vessel and can, thereby, sense any changes in reactor coolant system pressure. As the valve was opened, fluid entered the gauge line and stopped abruptly when there was no more room for fluid motion. This caused a short term pressure transient of sufficient duration to actuate the pressure sensing switches for the reactor trip system.

A reactor trip resulting from the test to confirm that an excess flow check valve is in its proper position for normal operation has not been a recurring event at the Oyster Creek plant. Although a reactor trip is not an unexpected event and several may be expected over the lifetime of a plant, the possibility for unnecessary reactor trips of this nature should be minimized by proper procedure and design.

Unnecessary reactor trips can be eliminated procedurally by instructions which result in slower opening of valves or by closing the block valve to one set of sensors while the check of the excess flow check valve is made. It may also be possible by design to significantly reduce the spurious signals by utilizing either fine control test valves or self indicating excess flow check valves.

We conclude that it is desirable to reduce the likelihood of spurious scram signals but it is not necessary to assure the health and safety of the public.

The licensee has stated that modifications to surveillance testing procedures are in progress to reduce the likelihood of a similar occurrence and that the design will be reviewed to determine the need for equipment modifications for further improvement. We have requested that the licensee submit his findings prior to the startup after the next refueling outage.

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VII CONCLUSIONS

We have evaluated both the current condition of the core and the adequacy of changes being made to prevent a recurrence of the significant events that happened at Oyster Creek on May 2, 1979.

The condition of the core was evaluated both analytically and empirically. The analytical evaluation (Section II.A.1) demonstrates by conservative thermal-hydraulic calculations that the liquid level did not drop below the top of the core. The evaluation (Section II.A.2) of the radiochemical records supports a conclusion that the event caused no fuel failures. Therefore, we conclude that the Oyster Creek core is currently undamaged. Additionally, the licensee has established a surveillance program to monitor for signs of fuel damage occurring during the subsequent restart. We consider the surveillance an acceptably prudent measure and request notification if any of the criteria of Section V are exceeded.

We agree with the licensee's proposal to make the triple low alarm a safety limit for all reactor modes. This provides a measurable basis for concluding that the core is covered.

We conclude that the hydraulic communication between the annulus and the core is adequate whenever more than one recirculation loop discharge valve is open. This requirement will be assured by the proposed Technical Specification.

We conclude that the loss of reactor-vessel-inventory analysis provided adequately bounds transients of this nature. Two key assumptions of the analysis are now covered by Technical Specifications because of the importance of automatic actuation of the isolation condensor at double-low level. Both the double-low level and the maximum time delay before isolation condensor value opening shall be included as limiting safety system settings.

In addition, we have recommended improvements for Oyster Creek regarding level indication and surveillance testing. These were discussed in Section VI.

Based on the foregoing, we conclude that there is reasonable assurance that operation of the Oyster Creek facility can be resumed without undue risk to the health and safety of the public.

VIII REFERENCES

1. Letter dated May 12, 1979, from I. Finfrock (JCP&L) to NRC and an enclosed "Report on the May 2, 1979, Transient at the Oyster Creek Nuclear Generating Station."

2. Letter (undated), received by NRC on May 17, 1979, from I. Finfrock (JCP&L) transmitting ENC Report XN-NF-79-47, "Evaluation of the Oyster Creek Reactor Core Liquid Level Following the Inadvertent Reactor High Presure Scram on May 2, 1979," dated May 11, 1979 (Issue Date: May 15, 1979).

3. Letter dated May 19, 1979 from I. Finfrock (JCP&L) to NRC enclosing an analysis titled "Bounding Loss of Coolant Inventory Transient for the Oyster Creek Plant."

4. Letter dated May 19, 1979 from I. Finfrock (JCP&L) to NRC requesting changes to Technical Specification 2.1.D to extend the applicability of a safety limit, and a new Section 2.1.[illegible] to prevent the isolation of pump loops.

5. J. D. Kahn and M. S. Foster, PTSBWR2 - Plant Transient Simulation Code for Boiling Water Reactors," XN-74-6 Revision 3.

6. Telecopy, "Responses to Staff Questions on May 19, 1979 Submittal, Loss of Inventory Transient Analysis," received by the NRC on May 23, 1979.

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APPENDIX

DESCRIPTION OF TRANSIENT AND

SEQUENCE OF EVENTS RELATED TO SCRAM OF MAY 2, 1979, AT

OYSTER CREEK NUCLEAR GENERATING STATION

(EDITED FROM JERSEY CENTRAL POWER & LIGHT CO., LETTER/REPORT DATED 05/12/79)

INITIATING EVENT:

On May 2, 1979, at 1350 hours, an inadvertent reactor high pressure scram occurred during required surveillance testing on the isolation condenser high pressure initiation switches.

Two of the four reactor high pressure scram sensors share a common sensing line with the isolation condenser high pressure initiation switches being tested.

The technician performing the test was in the process of verifying that the sensing line excess flow check valve was open when the scram occurred.

The scram has been attributed to a momentary simultaneous operation of two of the reactor high pressure scram sensors due to a hydraulic disturbance associated with valve manipulations which was required by procedure to verify the position of the excess flow check valve. These sensors are also used in the automatic recirculation pump trip logic which tripped the four operating recirculation pumps. The hydraulic disturbance also caused a momentary trip of the isolation condenser initiation switches. These sensors were not closed long enough for automatic initiation of the isolation condensers since a time delay is involved in the initiation logic.

INITIAL CONDITIONS:

Plant Parameters at the Time of the Scram:

Reactor Power	1895 MWt
Reactor Water	79" Yarway (13'4" Above the top of the active fuel)
	6.4' GEMAC
Reactor Pressure	1020 psig
Feedwater Flow	7.1×10^6 lbs/hr
Recirculation Flow	14.8×10^4 gpm

Equipment Out of Service Relevent to Event Sequence

A. One (SB) of the two startup transformers was out of service as permitted by Techcnical Specifications to inspect associated 4160-Volt cabling. SB supplies offsite power to one half of the station electrical distribution system when power is not available through the station auxiliary transformer. The 4160 Volt buses which receive power from SB are 1B and 1D. Bus 1D supplies power to certain redundant safety systems. Bus 1D is designed to be powered from #2 Diesel Generator in the event power is not available from either the auxiliary transformer or startup transformer. Bus 1B supplies 4160-Volt power to non-safety related systems and hence, does not have a diesel backup power source.

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B. One (n) of the five recirculation loops was not in service due to a defective seal cooler cooling coil. The pump suction valve was open, the discharge valve was closed, and the discharge valve bypass valve was open. No other important systems or components were out of service.

EVENT SEQUENCE (Two = 1350 hours):

TIME OF EVENT (Sec)	EVENT DESCRIPTION
0	For the reason previously described a reactor scram occurred coupled with a simultaneous automatic trip of the four operating Recirculation Pumps. The Control Room operator verified that all control rods inserted and proceeded to drive-in the IRM and SRM Nuclear Instrumentation. At this time, 4160-Volt power was being supplied from the auxiliary transformer during the coast-down of the Turbine Generator System and the Feedwater System was in operation. Recirculation flow started decreasing due to pump coastdown. Steam flow started decreasing due to loss of heat production (scram) but feed flow rate remained at the rated level. Reactor vessel pressure decreased to the pressure regulator setpoint as steam flow decreased. Reactor water level began decreasing due to steam void collapse in the core.

TIME OF EVENT (cont)	EVENT DESCRIPTION (cont)
13	The turbine Generator tripped at the no load trip point which initiates an automatic transfer of power from auxiliary to the startup transformers. Power to Buses 1A and 1C successfully transferred from the auxiliary transformer to the SA startup transformer. Since SB was out of service at this time, power was lost to Buses 1B and 1D. As designed, Buses 1B and 1D separated through operation of breaker 1D and Diesel Generator No. 2 fast started to power emergency loads on Bus 1D. Loss of power to Bus 1B caused the loss of Feedwater pumps B and C and Condensate Pumps B and C. Although power was available to the A condensate and feedwater pump via Bus 1A, the A Feedwater Pump tripped on low suction pressure. Reactor water level and pressure decreased since water was leaving the Reactor Vessel through the Steam Bypass Valves to the Main Condensers and no high capacity source of high pressure makeup water was available.

TIME OF EVENT (cont)	EVENT DESCRIPTION (cont)
	In addition, the loss of power to Bus 1B caused the B Cleanup System Recirculation Pump to trip which, in turn, caused an isolation of the Cleanup System due to low flow through the cleanup filter. Furthermore, one condensate transfer pump and the operating fuel pool cooling pump tripped. An unsuccessful attempt was made to restart the A feedwater pump. (The reasons for the restart failure are described later.)
(Event Recorder) 13.6	Reactor water level decreased to the Low level scram setpoint which is 11'5" above the top of the active fuel region.
(Event Recorder) 16.8	The output breaker on the No. 2 Reactor Protection System M.G. Set tripped due to loss of power to the drive motor. The output voltage from the M.G. Set had been maintained by flywheel action since the time of the turbine trip. Power to the M.G. Set drive motor is fed indirectly through Bus 1D which was deenergized at this time.

TIME OF EVENT (cont)	EVENT DESCRIPTION (cont)
31	The No. 2 Diesel Generator Breaker closed and supplied power to the 1D Bus. A second control rod drive pump started.
33	Reactor water loss continued from steam flow to the main condenser. Reactor isolation was manually initiated to conserve water by closing the Main Steam Isolation Valves prior to an automatic isolation of the reactor on a Low-Low Reactor Water Level signal which occurs at 7'2" above the top of the active fuel region). This action was taken at an indicated water level of approximately 30" on the Yarway instrument which corresponds to 9'8" above the top of the active fuel region. Note, that the decrease in indicated water level and pressure was amplified by the effects of introducing cold feedwater into the vessel during the 13 second period prior to the Turbine Generator Trip. The cold feedwater reduced the steam voiding inside the vessel thereby shrinking the water level.

TIME OF EVENT (cont)	EVENT DESCRIPTION (cont)
49	The Main Steam Isolation Valves fully closed, thus stopping the loss of water from the vessel. The reactor steam pressure increased. Indicated reactor water level started to increase shortly after isolation when reactor decay heat reestablished a steam void distribution.
(Event Recorder)	
59.6	The operator transferred the mode switch from RUN to REFUEL.
76 (1 min. 16 sec)	The operator placed the B isolation condenser into service to establish a sink for the removal of decay heat from the reactor. At this time, the Control Room operator closed the A and E recirculation loop discharge valves (these valves take approximately two (2) minutes to close). It is postulated that at this time, the operator closed both B and C loop discharge valves. The conclusion that the five recirculation pump discharge valves were closed is based upon loop temperature response later in the event and is further supported by the Low-Low-Low level at 172 seconds. The D loop was isolated previously (see the equipment out of service section).

TIME OF EVENT (cont)	EVENT DESCRIPTION (cont)
(Event Recorder)	
90 (1 min. 30 sec.)	The reactor Low water level alarm cleared as water water was added from the isolation condenser to the Primary System.
96 (1 min. 36 sec.)	The B isolation condenser initiation valve fully opened after 20 seconds. The temperature of the E recirculation loop, which serves as the B isolation condenser water return path, decreased due to the effects of cold water from the isolation condenser. The D recirculation loop temperature did not change appreciably. A, B, and C recirculation loop temperatures increased slightly. The heat-up is attributed to natural circulation through the partially open discharge valves carrying hot water (536 F) warming the lines previously cooled by the effects of cold feedwater. The reduced flow area between the lower downcomer and lower plenum area, due to the slow closure of the discharge valves, started to cause a shift in water inventory from the core area to the upper and lower downcomer region. The shift was due to the isolation

TIME OF EVENT (cont)	EVENT DESCRIPTION (cont)
	condenser returning condensed steam from the core area to the downcomers. The water inventory shift continued as the discharge valves moved to the full closed position.
(Event Recorder)	
172 (2 min. 52 sec.) Last recorded point on	The reactor Low-Low-Low water level instrument trip point was reached.
186 sec. (3 min. 6 sec.)	All recirculation loop discharge valves fully closed. At this time, based upon closure initiation, the cooldown of the E recirculation loop stopped and a heat-up began. The indicated reactor water level increased due to the shift in water inventory. Recirculation loops A, B, and C continued to heat up. The mechanism of the heat up was due to heat transfer between the hot recirculation loop piping and the water in the piping. Reactor pressure continued to decrease as a result of isolation condenser operation.
250 (4 min. 10 sec.)	The operator removed B isolation condenser from service to reduce the rate of cooldown of the Primary System. The indicated annulus water level fell due to a return of water to the core region from the downcomer region through the five, two-inch bypass valves around the recirculation loop discharge valves. During this period, the water was stored in the recently secured isolation condenser. The recirculation loop discharge temperatures reached equilibrum and followed a slow cooldown trend.

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TIME OF EVENT (cont)	EVENT DESCRIPTION (cont)
270 (4 min. 30 sec.)	The reactor pressure increased due to the effects of removing B isolation condenser. The rate of decrease in water level shifted from a ramp of approximately 37 in/min to 2 in/min. The reason for this change is the isolation condenser tube assembly was completely filled. The flow through the five 2" bypass valves continued.
450 (7 min. 30 sec.)	Both isolation condensers were placed in service. This caused an increase in indicated water level and a decrease in pressure. The A recirculation loop temperature decreased because cold water from the A isolation condenser entered the A recirculation loop by design. A portion of the water passed through the loop via its 2" bypass line contributing to the cool-down.
528 (8 min. 48 sec.)	The operator removed the B isolation condenser from service to slow the rate of cooldown. The indicated annulus water level reached a maximum of approximately 14.4 feet above the top of the active fuel (88" on Yarway). This is considered to be above normal water level for full power operation. When the B isolation condenser

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TIME OF EVENT (cont)	EVENT DESCRIPTION (cont)
	was removed from service, indicated water level decreased to 13'8" above the top of the active fuel where it remained until approximately 1212 seconds when the A isolation condenser was removed from service. The reactor pressure continued to decrease and all recirculation loop temperatures continued to to trend downward. Indicated water level was stable at this time because the head of water in the downcomer region was sufficient to establish equilibrium between the water entering the core region via the 5 two inch bypass valves and condensed steam returning to the downcomer from the isolation condensers.
540 (approx) (9 min)	The four (4) Low-Low-Low water level indicators were verified locally to be below their alarm setpoint which is 10" above 4' 8", or 5' 6" above the core. The reading appeared to be at or below the instrument's lower level of detection.
810 (approx) (13 min 30 sec)	A recheck of the triple Low water level indicators showed that the pointers were active (moving) although they continued to read below their alarm point. The instrument was at or slightly above its lower level of detection.

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TIME OF EVENT (cont)	EVENT DESCRIPTION (cont)
1212 (20 min 12 sec)	The operator removed the A isolation condenser from service stopping the removal of water from the core region. Indicated water level decreased as the water in the downcomer region flowed into the core region. Reactor pressure started to increase due to the decay heat steam production.
1488 (24 min 48 sec)	The isolation condensers were used several more times to control the reactor cooldown with predictable increases in indicated water level and reduction in pressure. This mode of operation continued until 1914 seconds.
1914 (31 min 54 sec)	In order to more correctly determine the plant cooldown rate C recirculation pump was started and the discharge valve was opened. It was noted that the indicated water level dropped approximately 3 feet in less than 2 minutes. The operator shutdown the C recirculation pump and isolated it to investigate the reason for the drop in level. In response to the indicated water level drop, an additional attempt was made to start the A feedwater pump. The pump had failed to start earlier due to a tripped overload on the auxiliary oil pump that is interlocked in the pump starting sequence. The indicated water level started to increase due to the action of the operating isolation

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TIME OF EVENT (cont)	EVENT DESCRIPTION (cont)
	condenser transferring water to the downcomer region. When the C recirculation loop was started the loop temperature increased from approximately 400 F to 470 F. The other recirculation loop temperatures continued to trend down. At this time Low-Low-Low alarm may have cleared.
2208 (36 min 48 sec)	The A Feedwater pump was successfully started by locally starting the auxiliary oil pump which satisfied the required starting interlocks. Indicated water level increased to a level corresponding to 13'8" above the top of the active fuel region. Realization occurred that the indicated water level and core water level may not have been the same when it was recognized that the five recirculation loop discharge valves were closed.
2340 (39 min 0 sec)	The A recirculation pump was placed in service at a flow rate of approximately 1.9×10^4 gpm, thus removing the disparity between water level measuring systems. The Low Low Low water level alarms were known to be cleared at this time. Indicated water level dropped approximately three feet to 11'4" above the top of the active fuel. The A recirculation loop temperature rose from 375°F to 465°F when it was placed in service. Steps were initiated at this time to bring the plant to "cold shutdown condition".

298 320

TIME OF EVENT (cont)	EVENT DESCRIPTION (cont)
2700 (45 min. 0 sec.)	Reactor Protection System #2 restored and scram reset.
3600 (1 hr.)	The SB transformer was returned to service and Bus 1B was energized, and normal shutdown proceeded.
(8 hr. 40 min.)	Cold shutdown achieved.