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New York Power Authority

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July 30, 1993
JPN-93-053

U. S. Nuclear Regulatory Commission
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Washington, D. C. 20555

Subject: James A. FitzPatrick Nuclear Power Plant
Docket No. 50 - 333
Response to Bulletin 93-03
BWR Reactor Vessel Water Level Instrumentation Issues

References: 1. NRC Bulletin 93-03 "Resolution of Issues Related to Reactor Vessel Water Level Instrumentation in BWRs," dated May 28, 1993.

2. NRC Letter B. C. McCabe to R. E. Beedle, dated April 5, 1993 "Response to Generic Letter 92-04," (TAC No. M84280).

Dear Sir:

This letter provides the Authority's response to NRC Bulletin 93-03 (Reference 1). The short term actions requested by Bulletin 93-03 were completed on June 12, 1993, and are described in Attachment I. The hardware modifications to the level instrumentation system and their implementation schedule, requested in References 1 and 2, are described below and detailed in Attachment II.

A "back-fill" modification will be installed on the five reactor vessel instrumentation reference legs prior to start-up from the maintenance outage scheduled to begin on September 11, 1993. These modifications will not be made operable immediately, but will be phased in to reduce the probability of unanticipated transients.

The Authority's implementation schedule is based upon timely resolution of technical and safety issues. The back-fill modification on the refuel zone reference leg, which is the only reference leg which can not initiate a plant transient during plant operation, will be made operational prior to start-up from the September, 1993 maintenance outage. Observation and on-line testing of this modified reference leg will provide data to determine the modification's effects on indicated reactor water level. The results of this test program will ensure that the other reference leg modifications will not generate false reactor water level indications which can result in plant transients.

Contingent upon acceptable operation of the refuel zone reference leg, the back-fill modification for one narrow range and one wide range reactor level instrument will be made operable prior to start-up from the scheduled April, 1994 maintenance outage. The remaining two back-fill modifications will be made operable prior to startup from the

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Refuel 11/Cycle 12 refueling outage which is scheduled for the first quarter of 1995. This schedule may be accelerated, if warranted by satisfactory on-line performance. In addition, the back-fills would be available at any time following the September, 1993 outage to remedy an instrument anomaly should it occur.

The back-fill modification can not be made operable before October 7, 1993. This date reflects the early June, 1993 start date for modification design, expedited delivery schedules for materials, and most significantly, the engineering analyses required to ensure that all technical and safety issues are resolved. Detailed justification is provided in Attachment II. If the FitzPatrick plant enters a cold shutdown condition after July 30, 1993, but before the scheduled September, 1993 maintenance outage, the Authority may elect to restart and operate the FitzPatrick plant until the September maintenance outage without first installing the modification.

This schedule is justified because the FitzPatrick condensing chamber temperature monitoring program continues to indicate functioning condensing chambers. Increasingly detailed examination of level instrumentation during shutdowns has found no evidence of level indication anomalies, such as notching, characteristic of gas saturated reference columns. The reactor level instrumentation enhanced monitoring program combined with operator training and new plant process computer displays provide plant operators with early indication of potential instrumentation problems and the guidance for responding to postulated reactor level instrumentation anomalies.

If you have any questions, please contact J. A. Gray, Jr.

Very truly yours,

Ralph E. Beedle

STATE OF NEW YORK
COUNTY OF WESTCHESTER

Subscribed and sworn to before me
this thirtieth day of July 1993.

Geraldine Strand
Notary Public

GERALDINE STRAND
Notary Public, State of New York
No. 4991272
Qualified in Westchester County
Commission Expires Jan. 27, 1994

Attachments:

I - Description of Short Term Compensatory Actions Taken
II - Description of Planned Hardware Modification to Back-fill Reactor Vessel Water Instrumentation Reference Legs - Modification F1-93-075

cc: See next page

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Attachment I to JPN-93-053

RESPONSE TO BULLETIN 93-03

RESOLUTION OF BWR REACTOR VESSEL WATER LEVEL INSTRUMENTATION ISSUES

DESCRIPTION OF SHORT TERM COMPENSATORY ACTIONS TAKEN

July 30, 1993

New York Power Authority

James A. FitzPatrick Nuclear Power Plant

Docket 50-333

DESCRIPTION OF SHORT TERM COMPENSATORY ACTIONS TAKEN

(Section numbers in this attachment correspond to Bulletin 93-03 requested action numbers.)

1. (a) (1) Establish enhanced monitoring of all RPV level instruments to provide early detection of level anomalies associated with de-gassing from the reference legs.

New displays were created in the Emergency and Plant Information Computer (EPIC) to be used by the "notch watch" operators established by the procedure changes identified in section 1.(a)(2) below. These new displays will provide the operators with the ability to monitor level from the various reference legs during cool-downs.

In addition, a modification completed on June 11, 1993 created two new EPIC data points for the wide range level instrumentation. The new data points provide expanded ranges of indication for each of the reference columns associated with condensing chambers 2A and 2B. The wide range instrumentation uses different density compensation calibration factors than the narrow range instrumentation. As a result, the wide range indicated level increases to an off-scale high condition during normal shutdown. Expanded monitoring ranges on the EPIC will provide the operator with the ability to read (for information purposes only) "uncalibrated" information beyond the normal calibrated range of the instruments. These EPIC points were added to the two new EPIC displays to provide continued monitoring of the associated level instrumentation. The two new EPIC CRT displays show level instruments and their current indications grouped by condensing chambers as well as the temperature data (individual and differential) from the chambers.

1. (a) (2) Develop enhanced procedures and additional restrictions and controls for valve alignments and maintenance that have a potential to drain the RPV during Mode 3.

Pump down or drain down events due to shutdown cooling (SDC) valve position errors were already precluded by hardware and procedural controls which existed prior to Bulletin 93-03. These controls are effective regardless of the potential for errors in water level indication discussed in NRC Bulletin 93-03. Inadvertent reactor vessel drain down would require the failure of operators to follow procedures, ignoring warnings in the procedures, and a failure of valve position interlock switches on two valves located in separate areas of the plant.

<u>Controls existing prior to Bulletin 93-03:</u>

a. Motor operated valve position limit switch interlocks were installed during the 1988 refueling outage in response to INPO SOER 87-2 which identified a concern with draining the vessel to the suppression pool. Even before these interlocks were installed, key lock switches, which were part of the original plant design, required control room supervisory review to operate valves associated with the potential for drain down or pump down events.

The interlocks between residual heat removal (RHR) system SDC suction valves (10MOV-15) from the vessel recirculation system suction line, and the key locked RHR torus suction valves (10MOV-13), prevent a drain down event through the common suction line connecting the reactor vessel and torus to the RHR pump suction. The interlocks between the same RHR SDC suction valves (10MOV-15), and the RHR system discharge line key locked motor operated valve (10MOV-39) to the suppression chamber spray and torus return lines, prevent opening both valves at the same time. These interlocks therefore prevent pumping or draining the vessel inventory to the torus.

b. Operating Procedure 13, "Residual Heat Removal System," contained existing requirements for establishing SDC line-ups which recognized a potential for drain down event. The valves with reactor vessel drain or pump down potential are controlled by key-operated switches.

c. Outage risk assessments were performed in accordance with Plant Standing Order PSO-2, "JAF Outage Management Program," section 5.3, "Emergent Work Scheduling," and attachment 7.1, "Short Form Risk Assessment," which identify and evaluate shutdown tasks with a potential to drain the vessel.

d. The possibility of a drain down to the radioactive waste system was also considered. Although one RHR loop contains two motor operated discharge valves connecting the suction line and the discharge lines to a 4 inch diameter line to radwaste, operation of manual suction and discharge valves is required in both loops to create a path to the 4 inch line to radwaste. Normally closed pump suction and discharge manual valves are located in both loops. Thus, pumping or draining to radwaste is a deliberate event requiring a procedurally controlled opening of the manual valves. The operator would be aware of the need to closely monitor vessel water level, be alert for anomalies in indication, and be able to stop the event by closing either the manual valves or, in the "A" loop, closing of MOVs.

The opening of reactor vessel SDC suction containment isolation valve 10MOV-17, and 10MOV-18 are deliberate actions necessary to initiate shut down cooling. Valve 10MOV-18 is normally closed. The disconnect switch to the motor power supply for 10MOV-18 is located in a cable tunnel. The switch is locked open in the disconnect position. Procedures control the operation of the valve and use of the key to unlock and close the motor power switch. The valves are isolated above 75 psig in addition to low reactor water level (Level 3). The level 3 trip provides a 177 inch margin to top of active fuel. Therefore, a level indication error would have to exceed 14 feet to prevent isolation of these valves.

Short Term Responses to Bulletin 93-03

In addition to existing plant and procedural features designed to prevent a pump or drain down event, the Authority has responded to the Bulletin 93-03 by further enhancing the following procedures with additional cautions and steps. These enhancements specifically address the potential of draining the vessel due to an error in reactor vessel water level indication. Temporary changes to procedures identified in items "a." through "g." below were approved by the Plant Operations Review Committee (PORC) on June 8 and 9, 1993. Permanent revisions to incorporate these changes into items "a." through "d." were completed by July 9, 1993. Revisions to incorporate the temporary changes made to items "e." through "g." are in progress.

a. Work Activity Control Procedure (WACP) 10.1.1 was changed to prevent work with the potential to drain the vessel while in hot shutdown unless a safety evaluation has been performed and approved by PORC.

b. A new Administrative Procedure AP-10.01 "Problem Identification and Work Control," was approved by PORC on June 23, 1993 to replace WACP 10.1.1. The new procedure, includes in section 7 "Special Instructions," a step which states:

> *Maintenance which has potential for draining the reactor vessel shall not be performed with reactor coolant temperatures greater than 212 degrees F until a PORC approved safety evaluation addressing the maintenance item is completed. This safety evaluation shall include a review of NRC Bulletin 93-03, "Resolution of Issues Related to Reactor Vessel Water Level Instrumentation in BWRs.*

c. Operating Procedure 13, "Residual Heat Removal System" inserted cautions in the procedure for starting shutdown cooling, identified indications of non-condensible gases in instrument reference legs, and in the procedure for pumping coolant to the discharge header. These cautions identify the need to monitor multiple level indicators during the evolution.

d. Operating Procedure 28, "Reactor Water Clean-Up System" modified steps for blow down using the reactor water clean-up system identify indications of non-condensible gases in instrument reference legs and modified procedure steps to require monitoring of multiple level indicators.

e. Operating Procedure 65, "Start-Up and Shutdown Procedure" was revised to add precautions and identify requirements for a "notch watch" during depressurization or forced cool downs of the vessel. Steps were added in the procedure for depressurization to establish and secure the "notch watch" based on specific plant conditions.

f. Operations Department Standing Order OSDO-32, "Shutdown Operation" inserted a precaution and added steps to identify actions to be taken during reactor cool down. These steps include assigning a separate "notch watch" operator stationed in the control room during forced cool down of the reactor to monitor reactor vessel water level instrumentation. The operator is stationed whenever the reactor is shutdown and reactor coolant temperature is greater than 212 degrees F. If potential level indication inaccuracies are observed during shutdown operations, the forced cool down operation is stopped, and management is informed.

g. Operations Surveillance Test ST-26J, "Heat Up and Cool Down Temperature Checks," Revision 8 of May 27, 1993, inserting a precaution which identifies actions to be taken in the event of potential level indication inaccuracies during vessel cool down.

1. ***(a) (3) Alert operators to potentially confusing or misleading level indication that may occur during accidents or transients initiating from Mode 3. For example, a drain-down event could lead to automatic initiation of high-pressure emergency core cooling systems (ECCS) without automatic system isolation or low pressure ECCS actuation.***

The operating shifts, staff licenses and engineers on shift received the training specified in NRC Bulletin 93-03 prior to June 12, 1993. Two individuals who were not available prior to that date were subsequently trained prior to assuming on shift duties. The lesson plan was also used in the training of the current license class and the shift technical advisor class.

1. (b) By July 30, 1993, each licensee is requested to complete augmented operator tr ing on loss of RPV inventory scenarios during Mode 3, including RPV drain-down e nts and cracks or breaks in piping.

Five of the six shifts have completed training. The sixth shift and remaining licensed operators are scheduled to complete training on August 6, 1993. Personnel will not be permitted to resume operating shift responsibilities after July 30, 1993 without having completed this training.

The training simulator software was modified to incorporate the two new computer screens used to monitor water level during cool down and condensing chamber differential temperatures.

The simulator software modification was completed on June 14, 1993 to support simulator training on notching and non-conservative level effects. Three new malfunction scenarios were developed. Two provide the ability to induce level notching on one "A" and one "B" level column. These malfunctions are variable from 0 to 30 inches of level instrument range. The third is a Boolean logic 30 inch offset on all of the narrow range transmitters.

The malfunctions described above were incorporated into a plant shutdown and cool down simulator scenario. This scenario involves a cool down where notching is observed. Subsequently, after going into shutdown cooling, a 30 inch offset occurs. This is followed by a loss of inventory. With the level offset, automatic actions do not occur as expected.

Attachment II to JPN-93-053

RESPONSE TO BULLETIN 93-03

RESOLUTION OF BWR REACTOR VESSEL WATER LEVEL INSTRUMENTATION ISSUES

DESCRIPTION OF PLANNED HARDWARE MODIFICATION TO BACK-FILL REACTOR VESSEL WATER INSTRUMENTATION REFERENCE LEGS MODIFICATION F1-93-075

July 30, 1993

New York Power Authority

James A. FitzPatrick Nuclear Power Plant

Docket 50-333

(Section number in this attachment corresponds to Bulletin 93-03 reporting requirement number.)

(2) (b) " a description of the hardware modifications to be implemented at the next cold shutdown after July 30, 1993. If an addressee chooses not to take the requested actions specified in the Hardware Modifications section, the report shall contain a description of the proposed alternative course of action and the schedule for completing it, and a justification for any deviations from the requested actions."

A. DESCRIPTION OF THE MODIFICATION:

The Authority plans to install back-fill capability for each reactor vessel instrumentation reference leg. Back-fill will be provided from the control rod drive (CRD) system. One tap will be installed on the CRD system branching out to the five reference column back-fill modules.

The back-fill components will be assembled in interchangeable modular enclosures installed in the vicinity of the reactor vessel level instrumentation racks. These back-fill modules will contain filters, metering valves, flow monitoring equipment, flow indicator and associated electrical equipment. Manually operated metering valves will be used to adjust flow rate.

The outlet of each backfill module will be routed to instrument tubing connected to the reference column. Connection to the reference leg will be made as close as possible to, and outside of, the primary containment boundary excess flow check valves. This connection location is expected to minimize perturbations to the level instruments based on preliminary test results from the BWROG. Two isolation valves will be provided on both the inlet and the outlet of each back-fill module to facilitate removal and replacement of the enclosure for maintenance. Two check valves will be provided downstream of each enclosure. These check valves provide added assurance for system boundary isolation. Containment isolation is provided by the existing excess flow check valves and manual isolation valves on the reference leg supply tubing.

The back-fill modules will not be safety related. The back-fill injection lines connecting the modules to the reference legs will be safety related up to and including the isolation check valves.

Test taps will be provided to facilitate testing of new check valves. Two filters will be provided for each system, with one normally valved out of service. Bleed lines will be provided downstream (on the reactor side) of the filters and check valves to facilitate filter and enclosure replacement without introducing air into the system.

Based upon initial evaluations of condensing chamber operation, the anticipated range of back-fill flow to the reference legs will be between 2 and 6 pounds/hour. This range in flows will allow for manual operation of the system, eliminating the need for pressure control valves, thus maintaining a more simplified design.

Electrical power for the flow monitoring system will be provided from a 115 VAC instrument control power bus with emergency diesel generator backup. Individual fuses located at each enclosure will provide system electrical isolation. Local flow indication will be provided at the panels. Remote indication may be provided by a future connection to the Emergency and Plant Information Computer (EPIC) system.

B. SCHEDULE

Phased Implementation Schedule:

To reduce the probability of unplanned plant transients, a conservative schedule has been established for phasing the back-fill modules into operation on the five reference legs.

Contingent upon satisfactory resolution of technical and safety issues, the modification will be made operational in three stages. This schedule may be accelerated if warranted by satisfactory on-line operation of the modification during the earlier stages or evidence of "notching" during shutdown conditions.

Prior to Start-up from the September 1993 Maintenance Outage

> The Authority plans to install the modification on the five reactor vessel instrumentation reference columns. However, only the refuel zone reference column (which does not supply trip signals used during operation) back-fill module will be placed into operation at that time.
>
> Observation and on-line operational testing of the refuel zone reference will be conducted to ensure that plant trips will not be initiated by pressure fluctuations which could result when the modification is placed into operation on other reference legs. This testing and observation will be conducted during the following six months.

Prior to Start-up from the April 1994 Maintenance Outage

> Contingent upon acceptable operation of the refuel zone reference leg during the first phase, the back-fill module will be placed into operation on one narrow range and one wide range reference column.

Prior to Start-up from first quarter 1995 Refuel Outage

> The back-fill modules for the final two reference columns will be placed into operation. The back-fill module may be removed from the refuel zone reference column when the module has been made operable for the two wide range, and two narrow range, reference columns.

Earliest Possible Implementation Date October 7, 1993 (Scheduled Completion of the September Maintenance Outage)

> This is the earliest date by which the phase I modification can be placed into operation.

C. JUSTIFICATIONS:

1. Plant Restart After July 30, 1993

Start-up of the plant from a forced cold shutdown during the period from July 30, 1993 through the start of the planned September maintenance outage would not have an adverse impact on nuclear safety. The paragraphs below detail why operation during this period is justified.

a. Absence of Level Indication Anomalies

Anomalies in water level indication characteristic of those discussed in Generic Letter 92-04 and Bulletin 93-03 have not been observed at the FitzPatrick plant. Significantly upgraded instrumentation and computer aided analysis have been established to monitor reference column level indications during shutdown operations. Increasingly detailed examination of reference column level indications during shutdowns using the more sensitive instrumentation has found no evidence of the anomalies in level indication which are characteristic of gas saturated reference columns.

b. Condensing Chamber Temperature Monitoring

The Authority took the initiative, and an industry leading position, by installing temperature monitoring, data recording, and trending instrumentation on the condensing chambers for all five reactor vessel instrument columns. The data from these instruments is monitored and recorded. The differential and average chamber temperature data trends obtained from these instruments are stable and consistent with the characteristics indicative of functioning condensing chambers which have not reached gas saturation conditions. This is a continuing program which will provide early indication to the control room operator if conditions occur which are associated with potential anomalies in water level indication.

c. Shut Down Cooling Drain Down and Pump Down Events Precluded

As discussed in Attachment I to this letter, the valve interlocks, key locked switch controls, and enhanced procedures are currently in place to preclude the type of event discussed in Bulletin 93-03. Such an event would require both a failure of valve position switches and a violation of operator procedures.

d. Specialized Operator Simulator Training

As discussed in Attachment I to this letter, prior to resuming shift responsibilities, all licensed personnel will have completed simulator training with scenario specifically created to prepare them to handle these phenomena.

e. Financial Impact

The activities necessary to support installation of the required modification will not be complete before the start of the September outage. Therefore, the Authority would incur a significant financial impact based on the duration of a forced shutdown occurring after July 30 and before the scheduled start of the September outage.

2. Phased Implementation Program

Phased implementation of the back-fill modification is justified.

Paragraphs "a" through "d" in the preceding section provided an amplified discussion of the following items which provide assurance of continued safe operation. "Notching" and level anomalies would be recognized if they occur and would be properly responded to by our operators based on the additional training provided. Level indication anomalies of the type discussed in bulletin 93-03 have not been observed at this plant. Condensing chamber temperature data is available to provide an indication of continued satisfactory operation and to provide advance indication of conditions which have a potential to result in water level indication anomalies. The plant design and enhanced procedures preclude the type of SDC drain down and pump down events discussed in the bulletin.

Operation with the back-fill modification in service on the refuel zone reference column will provide bench mark information to bolster assurance of accurate level monitoring, without exposing the plant to the risks associated with reactor trips.

The phased implementation program is designed to address concerns with CRD system pressure effects on reactor instrumentation. The introduction of flow to the reactor vessel instrumentation reference legs has the potential for causing pressure fluctuations on the reference leg, predominately the result of perturbations caused by plant evolutions effecting the pressure and flow from the CRD. The perturbations "seen" by the pressure and level instruments are a function of the magnitude and duration of the pressure fluctuations and the sensitivity of the instruments. The current design inherently supports adequate surge suppression through existing controls and component selection. However, further evaluations are required. Additionally, the temperature effects of the introduction of CRD water to the reference legs on temperature compensation for instrument set points and accuracies requires evaluation.

The phased implementation schedule is essential to allow the effects of CRD system operation on the backfill modification and the reactor vessel reference leg instrumentation to be thoroughly evaluated without subjecting the plant to the unnecessary risks of trips resulting from water level indication errors which could result from the following conditions:

1. Shifting CRD Pumps
2. Failure of CRD flow control
3. Failure of new back fill components
4. Control Rod Movement
5. Scrams and scram resets
6. Containment Isolation
7. Ability of selected flow control devices to adequately regulate flow rates over the full 1000 pound range of reactor pressure.

3. Implementation Date

The Authority had originally intended to pursue one of the alternate modification methods to address the concerns of Generic Letter 92-04. Following the release of Bulletin 93-03 and the BWROG meeting with the NRC staff before the Commissioners on June 1, 1993, the Authority redirected these efforts toward the installation of a "back-fill" type modification with a completion target date of July 30, 1993. That target date could not be achieved because of the combined effects of the time required for design; material procurement, fabrication, and delivery; the need for extensive engineering safety analysis; and the continuing concerns being identified by the BWROG test program. The progress to date and schedule for installation during the scheduled September maintenance outage is discussed below.

a. Design

A contract with a outside engineering organization was put in place within one week of the June 1st meeting with the NRC Commissioners. The kickoff meeting was conducted the following week. The draft of the preliminary engineering package was distributed for review within two weeks. Additional design changes are being introduced as a result of information obtained at the July 14th BWROG meeting.

Final detailed design engineering is in progress with an anticipated completion during the first week of August. The review, and PORC approval process is expected to support start of installation during the last week of August.

b. Engineering Analysis and Safety Evaluation

Work in this area is proceeding in parallel with the design effort. This is the single longest lead time effort. Final outlines for performance of these analyses are scheduled for delivery from the BWROG by the end of the second week of August. Therefore, although some preliminary work can begin prior to that date, a large portion of the plant specific analysis can not start until the third week of August. The current and optimistic target date for completion of the safety evaluation and pre-operational test procedure is the end of the second week of September. Engineering review and PORC approval are expected to be complete by the fourth week of September. Among the analyses to be performed are:

Thermal Hydraulic and Thermal Stress Analyses

(Analyze effects of pumping cold back-fill water toward the vessel)
Condensing chambers
Condensing chamber steam supply pipes
Vessel instrumentation nozzles

Instrument End Point Calibration Bias Analysis

Flow Bias
Temperature Bias

System Impact Evaluations

CRD system transients
Operational conditions other than at full power
Single failure impact on safety trips

Low Frequency Standing Wave Phenomena

Interactions from pump pressure waves and reference column hydraulic coupling

Experimental investigation of a potential low frequency standing wave phenomena in reference columns is considered essential to the safety analysis and is not expected to begin on a generic basis until mid August. It is anticipated that plant specific analyses may have to be performed following completion of the generic testing.

c. Procurement

Procurement began upon issuance of the preliminary engineering package during the last week of June, although the bulk of the ordering awaited the preliminary bill of materials during the second week of July. Most of the material is expected to be delivered during the first week of August. This will permit the start of shop fabrication of the five modular units. The flow meters, essential for completion of the modular units, are currently not scheduled for delivery until the end of the third week in August.

d. Installation

Shop prefabrication of the five modular back-fill units will proceed in parallel with the PORC review process for the complete modification package. Shop fabrication is anticipated to begin during the first week of August and to be completed during the last week of August.

Installation in the plant must wait for approval of the modification package and the installation safety evaluation. Assuming these are completed during the last week in August, installation is planned to be performed during the third week in September.

e. Operating Procedures and Operator Training

Development of operating procedures and training modules is progressing in parallel with design efforts and is expected to be completed and incorporated into the next six week operator training cycle which will begin during the second week of August. The training cycle will be completed during the fourth week of September, prior to pre-operational testing.

f. Pre-operational Testing

Completion and PORC approval of the operational safety evaluation and the pre-operational test procedure will proceed in parallel with installation and are expected to coincide with completion of installation during the third week in September. Performance of the pre-operational test, and review and approval of results, are expected to require approximately two weeks with a planned completion date of October 7th at which time the initial back-fill module for the refuel zone reference column will be operational.