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JUL 17 1984

Mr. A. Schwencer, Chief
Licensing Branch No. 2
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Docket Nos.: 50-352
50-353

Subject: Limerick Generating Station, Units 1 and 2 Deferral of Certain Pre-Operational Tests and Construction Completion Items Until After Fuel Load

Reference: Meeting between NRC Staff and Philadelphia Electric on June 6, 1984 to review project status

File: GOVT 1-1 (NRC)

Dear Mr. Schwencer:

At the reference meeting Philadelphia Electric discussed with the NRC staff the potential deferral of certain items beyond the scheduled fuel load date of September 15, 1984. The attachments to this letter identify those items which we propose to defer beyond fuel load, together with a justification for each proposed deferral.

Attachment 1 provides a listing of proposed pre-operational test deferrals, and corresponding individual test descriptions, each with justification for deferral. The safety consequences for each deferred test have been evaluated and found to represent neither increased risk to the public nor non-compliance with NRC regulations. All deferred pre-operational tests will be completed prior to the intended use of the system being tested, as described in Attachment 1. Similar test deferrals have been requested by recent operating license applicants and approved by the NRC.

Attachment 2 provides a listing of six construction completion items proposed for deferral, and corresponding individual descriptions, each with justification for deferral until a specified time after fuel load. The safety consequences for each deferred construction completion item have been evaluated and found to represent neither increased risk to the public, nor non-compliance with NRC regulations. Similar construction completion deferrals have been requested by recent operating license applicants and approved by the NRC.

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In order for us to schedule with confidence the balance of construction and startup activities, expedited review and approval of our proposals by the NRC is requested.

We would be pleased to discuss these matters in greater detail should you require additional information.

Sincerely,

John S Kemper

HDH/gra/07058402

cc: See Attached Service List

cc: Judge Lawrence Brenner (w/enclosure)
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Martha W. Bush, Esq. (w/enclosure)
Mr. James Wiggins (w/enclosure)
Mr. Timothy R. S. Campbell (w/enclosure)
Ms. Phyllis Zitzer (w/enclosure)
Judge Peter A. Morris (w/enclosure)

ATTACHMENT #1

Preoperational Test Deferral List

	Test Identification	Defer Until
1P13.5	Fire Protection Halon System	2
1P16.1	Residual Heat Removal Service Water System	1
1P31.1,A,B,C,D	Process Computer System	7
1P33.1	Turbine Enclosure HVAC System	4
1P34.1	Reactor Enclosure HVAC System	2
1P43.1	Condenser and Air Removal System	4
1P45.1	Feedwater System	2
1P58.2	Redundant Reactivity Control System	7
1P68.1	Solid Radwaste	2
1P68.1B	Radwaste Crane	2
1P70.1	Standby Gas Treatment System	2
1P72.1	Gaseous Radwaste Recombiners and Filters	4
1P73.1	Containment Atmospheric Control System	2
1P76.2	Post-Accident Sampling System	6
1P79.2A	Digital Process Radiation Monitoring System	2
1P79.2B	Offgas Pretreatment and Radiation Monitoring	4
1P79.2C	Main Steam Line Radiation Monitoring	4
1P79.2F	Gaseous Effluent Radiation Monitoring	5
1P83.1	Main Steam System	3
1P83.3	Steam Leak Detection System	3
1P93.2	Main Turbine Control System	4

1 = Prior to CRD functional testing (See STP-5, Control Rod Drive System Testing performed after fuel load).
2 = Prior to initial criticality.
3 = Prior to installation of the RPV head.
4 = Prior to opening of the MSIV's.
4a = Prior to opening the MSIV's except the RFPT performance test must be completed before or in conjunction with preoperational test 1P45.1.
5 = Prior to pressurization of piping in primary containment.
6 = Prior to 5% power.
7 = Prior to exceeding 5% power.

Description and Justification for Deferred Preoperational Tests

Description and justification for each preoperational test which will be deferred until after fuel load is provided in the following attachments.

1P13.5 Fire Protection Halon System

Description:

The underfloor sections of the auxiliary equipment room and remote shutdown room are provided with an automatic halon suppression system to control/suppress a cable fire in the floor sections.

Justification:

The Halon system preoperational testing will be completed prior to initial criticality. Delays experienced during the floor sections sealing program have set back the pre-operational testing schedule which includes testing for operabilitv of some 170 smoke and heat detectors. A roving fire watch will be established until system preoperational testing is complete.

1P16.1 Residual Heat Removal Service Water (RHRSW) System

Description:

The RHRSW System serves to cool the RHR System which in turn cools the reactor and/or the suppression pool.

Justification:

The RHRSW System is not required prior to initial criticality because there is no nuclear decay heat to remove. However, the RHRSW system preoperational testing will be completed prior to initial control rod drive functional testing.

LGS Process Computer System

SLA (Scan-Log-Alarm) Functional Testing of the Operator's and Engineer's Console Functions. (Test Nos. 1P31.1A, 1P31.1C, 1P31.1A1, and 1P31.1C1)

Description

These tests are designed to verify and document that the Process Computer Software associated with the various consoles (Operator's and Engineer's) is installed properly and is responding properly to operator requests from the consoles as per the functional specifications for the Process Computer.

Justification

Test 1P31.1A is complete. PECO proposes to perform tests 1P31.1A1, 1P31.1C and 1P31.1C1 after fuel load but prior to exceeding 5% power, due to a delay in powering up of the Engineer's Console and Backup Computer.

Test 1P31.1A, which is complete, ensures that the operator can perform basic console functions and hence can monitor computer operability.

Tests 1P31.1A1 and 1P31.1C1 involve the Backup Computer System. Test 1P31.1C is a rigorous functional testing of the functional capabilities of the Operator's and Engineer's Consoles.

None of the Core Performance Software operates below 5% power. No console actions by the operator are required for the operation of the Rod Block Circuitry.

For the above reasons, the proposed schedule for operation of the Scan-Log-Alarm Functions of the Operator's and Engineer's Consoles will provide for safe operation of the plant in accordance with NRC regulatory requirements.

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LGS Process Computer System

Input Test (Analog and Digital) (Test No. 1P-31.1B)

Description

This test is designed to verify and document that every plant input to the Process Computer produces the correct printout and display (message and value or status) when the input is printed out on the computer typers and is displayed on the computer CRTs.

Justification

The Core Performance Software, which includes three-dimensional power density distribution calculations, is not operated either automatically or on operator-demand below 5% power.

Therefore, only digital inputs used for the Rod Block Circuitry (Rod Worth Minimizer) are required for fuel load. These points have been tested.

PECO proposes to perform this test prior to exceeding 5% power.

This schedule for performance of this test will provide for safe operation of the plant in accordance with NRC regulatory requirements.

LGS Process Computer System

Supervisory System Tests (Test No.'s 1P31.1D and 1P31.1D1)

Description

These tests are designed to verify and document the correct operation of the electrical substation (220 kV and 500 kV), and the pumping stations (Perkiomen and Bradshaw). The tests also verify and document that all the inputs to the computer from the supervisory system are correctly printed out and displayed.

Justification

The Supervisory System of the PCS is not required for fuel load and low power testing.

None of the substations or pumping stations involve safety-related equipment or systems.

PECO proposes to perform these tests prior to exceeding 5% power.

This schedule will provide for safe operation of the plant in accordance with NRC regulatory requirements.

1P33.1 Turbine Enclosure HVAC System

Description:

The Turbine Enclosure HVAC systems are designed to provide ventilation, heating and cooling of the Turbine Enclosure. They ensure that the process systems will operate within proper ambient limits during normal operation, and they provide proper ALARA air flows and filtration, and maintain a negative pressure with respect to atmosphere during normal operation. This system is not safety related.

Justification:

The Turbine Enclosure HVAC system preoperational testing need not be completed prior to initial heatup. Main steam will not be present in the turbine enclosure until opening of the MSIV's. Therefore, the requirement for cooling of the equipment and enclosure is not required until that time. Heating will be accomplished by unit heaters as necessary.

Radioactive products from main steam will not be present until opening of the MSIV's. Therefore the need for maintenance of negative enclosure pressure, proper ALARA air flow directions and filtered exhaust is not present until that time.

Hence, the Turbine Enclosure HVAC systems preoperational testing will be completed prior to opening of the MSIV's.

1P34.1 Reactor Enclosure HVAC

Description:

The Reactor Enclosure HVAC systems are designed to provide ventilation heating and cooling of the Reactor Enclosure so that the process systems will operate within proper ambient limits during normal operation. They also provide proper ALARA air flows and filtration, and maintain a negative pressure with respect to atmosphere during normal operation. In the event of an accident the system design provides for isolation of the Reactor Enclosure from the environment with the exception of a single controlled filtered exhaust which maintains negative pressure in the enclosure.

Justification:

Reactor Enclosure HVAC systems preoperational testing need not be completed prior to fuel load for the following reasons:

1. Large heat loads from the operating equipment in the Reactor Enclosure will not be present prior to criticality. Ventilation cooling will not be required. Heating will be accomplished by unit heaters as necessary.

2. Radioactive fission and activation products will not be present prior to criticality so the requirement to maintain negative enclosure pressure, proper ALARA air flow directions and filtered exhaust will not be present.

3. Prior to criticality the possibility of an accident releasing fission products to the Reactor Enclosure is not present, so there is no need to isolate the Reactor Enclosure HVAC system and initiate recirculation flow mixing and filtration.

Consequently, the Reactor Enclosure HVAC systems preoperational testing will be completed prior to initial criticality.

1P43.1 Condenser and Air Removal System

Description:

The main condenser system is designed to condense and deaerate the exhaust steam from the main turbine and provide a heat sink for the turbine bypass system. The main condenser system is not safety related.

The main condenser evacuation system establishes a vacuum in the condenser during startup and removes noncondensible gases during normal operation. The system is not safety related.

Justification:

Prior to opening of the MSIV's and introduction of steam to the condenser, the Condenser and Air Removal System Preoperational testing need not be complete because no radioactive steam is present. The mechanical vacuum pumps will be available.

The condenser and air removal system preoperational testing will be completed prior to opening of the MSIV's.

1P45.1 Feedwater System

Description:

The Feedwater System takes cleaned up condensate, preheats it and transports it to the reactor to be converted into steam.

Justification:

The Feedwater system is not needed prior to fuel load because the reactor vessel can be manually flooded to support fuel load through many sources such as core spray or residual heat removal system. No automatic control functions would be required.

The feedwater and feedwater control system are required to maintain the reactor water level in order to achieve initial criticality. Consequently, the feedwater and feedwater control system preoperational testing will be completed prior to initial criticality.

1P46.1 Extraction Steam and Feedwater Heater System

Description:

The Extraction Steam System is designed to transport exahust steam from intermediate stages of the turbine to the feedwater heaters to increase the thermal efficiency of the plant. The system is not safety related.

Justification:

The turbine bypass valves can be used up to 25% reactor load to avoid having extraction steam. Extraction steam will not be present prior to turbine roll. Extraction steam system preoperational testing is not required prior to turbine roll. The extraction steam and feedwater heater system preoperational testing will be completed prior to turbine roll. The portion of the preoperational test that deals with the steam seal system will be completed prior to opening of the MSIV's.

1P58.2 Redundant Reactivity Control System

Description:

The Redundant Reactivity Control System (RRCS) is designed to provide a redundant and diverse method of shutting down the reactor for the unlikely occurrence of an Anticipated Transient Without Scram (ATWS) event. The system senses reactor pressure and level to determine if an ATWS event is underway, and if appropriate, the RRCS logic will initiate alternate rod insertion, recirculation pump trip, feedwater runback, and standby liquid control injection. Actuation of the RRCS logic will shut down the reactor, independent of the control rod drive system.

Justification:

The completion of the RRCS preoperational test will be deferred prior to reactor power exceeding 5%. This is necessary because of equipment delays which have postponed the completion of the system installation. Additionally, this will preclude the inadvertent initiation of automatic boron injection to the reactor during this low power test period.

Prior to initial criticality the control rods will be inserted and the reactor shutdown. After initial criticality it is very unlikely that an ATWS would occur during this low power period, and if uch an event did occur, the decay heat removal systems would be capable of removing the generated heat. This would provide the plant operators with significantly more time than during a full power ATWS event to shutdown the reactor. Consequently, the Redundant Reactivity Control System preoperational testing will be completed prior to the reactor exceeding 5% power.

1P68.1 Solid Radwaste System (Packaging)

Description:

The solid radwaste system is designed to collect and process radioactive spent resins, filter sludges and dry wastes generated as a result of plant operation. It also packages these wastes in suitable containers for offsite shipment and disposal. The solid radwaste system is not required to mitigate any accident situation.

Justification:

Since significant amounts of radioactivity are not expected prior to the initial heatup phase of power ascension testing, the solid radwaste system (packaging) need not be preoperationally tested until prior to initial criticality.

Hence, the packaging portion of the solid radwaste system preoperational testing will be completed prior to initial criticality.

1P68.1B Radwaste Crane

Description:

The radwaste crane is utilized primarily to move empty high integrity containers (HICs) to the filling areas and full HICs from the filling areas to the decontamination, storage or truck loading areas. The radwaste crane is not safety related and is not required to mitigate any accident.

Justification:

Since significant amounts of radioactivity are not expected prior to initial criticality, the radwaste crane need not be preoperationally tested until prior to initial criticality.

Hence, radwaste crane preoperational testing will be completed prior to initial criticality.

1P70.1 Standby Gas Treatment System

Description:

The Standby Gas Treatment System is designed to provide a filtered single exhaust point from either the reactor enclosure or refueling area, and to maintain a specific negative pressure of the enclosure in the event of an accident producing radiation products. It is also used in conjunction with purging primary containment in order to de-inert containment.

Justification:

The Standby Gas Treatment preoperational testing need not be completed prior to fuel load for the following reasons:

1. Prior to criticality, radioactive fission and activation products will not be present, so there is no need to maintain negative enclosure pressure and filtered exhaust.
2. Prior to criticality, drywell or suppression pool purge for de-inerting will not be required. See 1P73.1 for a discussion of inerting requirements.

Consequently, the Standby Gas Treatment System preoperational testing will be completed prior to initial criticality.

1P72.1 Gaseous Radwaste Recombiners and Filters

Description:

The function of the offgas system is to collect and delay release of non-condensible radioactive gases removed from the main condenser by the steam jet air ejectors during normal plant operation. The system is not safety related.

Justification:

Non condensible radioactive gases will not collect in the main condenser when the MSIV's are closed. Completion of preoperational testing of the offgas system is therefore not required until prior to opening of the MSIV's.

Hence, the offgas system preoperational testing will be completed prior to the MSIV's opening.

1P73.1 Containment Atmospheric Control System

Description:

The Containment Atmospheric Control (CAC) system incorporates features for accomplishing a number of functions, including inerting of the primary containment with nitrogen, purging of the primary containment, limiting the differential pressure between drywell and wetwell, monitoring of hydrogen and oxygen concentrations in the primary containment, and controlling combustible gas concentrations in the primary containment after a LOCA. Portions of the system are safety related.

Justification:

The Containment Atmospheric Control system preoperational testing need not be completed prior to fuel load for the following reasons:

The generation of hydrogen and other gases which could form a combustible mixture can potentially occur primarily as a result of a metal water reaction involving the fuel cladding and coolant at high temperatures, radiolytic decomposition of the coolant water or corrosion of certain materials in the containment. During the period before initial criticality there is no heat generation from the core, coolant temperatures are low, and no radioactive fission products or activation products have been produced. There are no conditions for which any of the CAC system safety functions would be required prior to criticality.

Technical Specification LCO 3.6.6.3 for Drywell and Suppression Chamber Oxygen concentration only requires operability of the CAC nitrogen inerting system during operational condition #1 at 15% power or greater. Additionally, LCO 3.10.5 suspends the inerting requirement "... until 6 months after initial criticality". Therefore, no technical specification limitations would be exceeded if the CAC inerting function is inoperable prior to criticality.

Consequently, the Containment Atmospheric Control System preoperational testing will be completed prior to initial criticality.

1P76.2 Post-Accident Sampling System

Description:

The post-accident sampling systems (PASS) are designed to obtain representative liquid and gas grab samples from the primary coolant system and from within the primary and secondary containments for radiological and chemical analysis under accident conditions. The grab samples are subsequently transported to the radwaste enclosure chemistry laboratory and counting facility for chemical and radioisotopic analyses, or shipped offsite for analysis. This system does not perform a safety function.

Justification:

The Post-Accident Sampling System need not be implemented until prior to 5% power operation because even if an accident occurred prior to 5% power operation, there would not be sufficient fission products released into the reactor coolant to necessitate the use of PASS. PASS would not be required to mitigate the accident or shutdown the plant. The safety of the public would not be affected by not having PASS available until 5% power operation.

Hence, the Post-Accident Sampling System preoperational testing will be completed prior to 5% power.

1P79.2A Digital Process Radiation Monitoring System

Description:

This system consists of the North Stack Effluent Radiation Monitors, the South Stack Effluent Radiation Monitors, the Drywell Post LOCA High Range Radiation Monitors, the Control Room Normal Fresh Air Supply Radiation Monitors, the Control Room Emergency Fresh Air Supply Radiation Monitors, the Wide-Range Accident Monitor, and the Hot Maintenance Shop Ventillation Exhaust Radiation Monitor. With the exception of the Drywell Post LOCA High Range Radiation Monitors, this system monitors the concentration of radioactive fission and activation products in the applicable process flow paths. The Drywell Post LOCA High Range Radiation Monitors detect the dose rate due to fission and activation products within the drywell. Portions of this system are safety related.

Justification:

The Digital Process Radiation Monitoring System Preoperational Testing need not be completed prior to fuel load for the following reason:

1. Prior to criticality, radioactive fission and activation products will not be present, so there is no need to monitor the drywell and the applicable process flow paths for radioactivity.

Consequently, the Digital Process Radiation Monitoring System Preoperational Testing will be completed prior to initial criticality.

1P79.2B Offgas Pretreatment Radiation Monitoring

Description:

This system monitors radioactivity in the main condenser offgas prior to the offgas steam entering the charcoal delay system. The system is not safety related.

Justification:

This system is required only when the offgas system is operating. Completion of preoperational testing of the offgas pretreatment monitoring system is therefore not required until prior to opening of the MSIV's.

Hence, the offgas pretreatment radiation monitoring preoperational testing will be completed prior to opening of the MSIV's.

1P79.2C Main Steam Line Radiation Monitoring

Description:

This system monitors radioactivity in the main steam lines downstream of the MSIV's.

Justification:

The main steam line radiation monitoring is required only when the MSIV's are open. Completion of preoperational testing of this system is therfore not required until prior to opening of the MSIV's.

Hence, main steam line radiation monitoring preoperational testing will be completed prior to opening of the MSIV's.

1P79.2F Gaseous Effluent Radiation Monitoring

Description:

The charcoal offgas trains exhaust the main condenser process gases via HEPA filters to the north stack. The majority of the gaseous effluent monitors are used in the exhaust lines to detect malfunctions in the corresponding offgas trains. One of the gaseous effluent radiation monitors, however, is used as the containment leak detector. This monitor samples the containment atmosphere and gives a qualitative indication of reactor coolant pressure boundary leakage based on gross noble gas radioactivity. The system is not safety related.

Justification:

The majority of the system is required only when radioactive steam is directed to the main condenser and the offgas system is operating. Completion of preoperational testing of the gaseous effluent radiation monitors is therefore not required before initial criticality.

Because of the containment leak detector, the gaseous effluent radiation monitoring system preoperational testing will be completed prior to pressurization of piping in primary containment.

1P83.1 Main Steam System

Description:

The main steam supply system transports steam from the nuclear steam supply system to the power conversion system and to auxiliary equipment.

Justification:

The Main Steam System is not required prior to initial criticality because there is no steam generated. However, the Main Steam System preoperational testing will be completed prior to installation of the reactor pressure vessel head.

1P83.3 Steam Leak Detection

Description:

The steam leak detection system is used to identify any steam leaks outside of primary containment.

Justification:

The steam leak detection system is not required prior to initial criticality because there is no steam generated. However, the steam leak detection system preoperational testing will be completed prior to installation of the reactor pressure vessel head.

1P93.2 Main Turbine Control (EHC) System

Description:

The turbine-generator control system is a GE electrohydraulic control (EHC) system. It is designed to maintain constant reactor pressure during normal operation and to operate the steam bypass system up to 25% of full load to maintain constant reactor pressure during plant startup, transients, and shutdown. The EHC system is not safety related.

Justification:

Prior to opening of the MSIV's, the EHC system, which controls the Main Stop Valves, the Control Valves, the Bypass Valves, and the Combined Intermediate Valves, is not required for operation.

Hence, the EHC system preoperational testing will be completed prior to opening of the MSIV's.

Attachment 2

Construction Completion Deferrals

Item		Defer Until
1.	Wrapping of Cables for Raceway Separation Criteria Compliance	First Refueling Outage
2.	Operability of Redundant Reactivity Control System	Prior to Exceeding 5% Power
3.	Control Panel Human Factors Enhancements	Prior to Exceeding 5% Power
4.	Post Accident Sampling System Modifications	Prior to Exceeding 5% Power
5.	Safety Parameter Display System and Emergency Response Facility Data System Operability	April 1, 1985
6.	Process Computer System	Prior to Exceeding 5% Power

HDH/gra/07058403

Wrapping of Cables For Raceway Separation Criteria Compliance

Description:

"Drop outs" occur where cables drop from raceways to enter equipment or another raceway. Where these drop outs come into proximity with cables or raceways of redundant electrical divisions, they are wrapped with a flame retardant material to provide separation. It is this wrapping that PECO proposes to defer until the first refueling outage. All other raceway separation barriers required by the Limerick separation criteria will be installed prior to fuel load.

Justification

The safe operation of Limerick will not be compromised by the deferral of this work. The need to wrap these cables was identified by a Limerick-unique test program conducted by Wyle Laboratories, and documented in Wyle Test Report #46960-3 which was transmitted to the NRC by letter from J. S. Kemper to A. Schwencer dated May 18, 1984. The Limerick cable separation criteria are conservatively based on the assumed failure of a primary overcurrent protection device to clear a high impedance fault with the resulting overcurrent magnitude being just below the trip setpoint of the next higher level overcurrent protection device. In addition, this overcurrent is assumed to maintain its magnitude even when the resistance of the cable increases due to the heating effects. Finally, this current is assumed to be present on a cable whose size will allow the longest duration of the fault prior to cable failure, thus generating the maximum amount of heat over time.

These conservative assumptions assure that the resulting criteria are more than adequate to mitigate any electrical failures. In reality, the occurrence of these types of failures is extremely unlikely. For example, testing shows that, as the cable resistance increases due to heatup, the current magnitude will decrease.

The unlikelihood of occurrence of this design basis failure supports the conclusion that deferral of wrapping the drop out cables until the first refueling outage will not adversely affect the safe operation of Limerick.

HDH/gra/07058404

Operability of Redundant Reactivity Control System

Description:

The redundant reactivity control system (RRCS) provides a redundant, diverse method of shutting down the reactor in the unlikely occurrence of an anticipated transient without scram (ATWS). The system senses reactor pressure and level to determine if an ATWS event is underway, and if appropriate, automatically initiates Alternate Rod Insertion, Recirculation Pump Trip, Feedwater Runback, and Standby Liquid Control System boron injection. Actuation of the RRCS logic shuts down the reactor independent of the control rod drive system. Operability of RRCS is proposed for deferral until prior to exceeding 5% power by blocking the RRCS logic.

Justification:

This deferral will preclude inadvertant initiation of SLCS boron injection during the low power test period. Prior to initial criticality, all control rods are inserted and the reactor is in the shutdown mode. From initial criticality to 5% power, in the highly unlikely event that an ATWS did occur, the small amount of heat being generated provides plant operators with significantly more time than would exist at full power to manually initiate the actions which the RRCS would automatically initiate.

For the above reasons, it is concluded the deferral of RRCS operability until prior to exceeding 5% power does not adversely affect the safe operation of Limerick.

HDH/gra/07058405

Control Panel Human Factors Enhancements

Description

The Limerick Control Room Design Review Final Report was transmitted to the NRC via letter from J. S. Kemper to A. Schwencer dated June 25, 1984. That report called for enhancements (paint, tape, and label) to the control room panels, re-scaling some instruments using acceptable human factors methods, and changes to some standard control switch shapes and colors to be made before fuel load. PECO is proposing that the completion of these enhancements, re-scalings and switch changes be deferred from fuel load until prior to exceeding 5% power.

Justification

This deferral will have no impact on the safe operation of Limerick. First, operator training has been conducted on the Limerick simulator which does not as yet incorporate the above human factors enhancements; thus appropriate and timely operator response to an accident would be unaffected by deferral of these enhancements. Second, because of the low level of decay heat present at 5% power, significantly more time is available to the operators to consider and initiate mitigative actions than would be available at full power.

For these reasons, it is concluded that deferral of the above human factors enhancements until prior to exceeding 5% power does not adversely affect the safe operation of Limerick.

Post Accident Sampling System Modifications

Description

The post accident sampling system (PASS) is designed to obtain representative liquid and gas grab samples from the primary coolant system and from within the primary and secondary containments for radiological and chemical analysis under accident conditions. The grab samples are subsequently transported to the radwaste enclosure chemistry laboratory and counting facility for chemical and radioisotopic analyses, or shipped offsite for analysis. This system does not perform a safety function.

The dissolved gas portion of PASS requires modifications to provide the accuracy, range, and sensitivity of measurement required by the NRC. PECO proposes to complete these modifications prior to exceeding 5% power.

Justification

These modifications need not be implemented until prior to exceeding 5% power operation because even if an accident occurred during 5% power operation, there would not be sufficient fission products released into the reactor coolant to necessitate the use of PASS. PASS is not required to mitigate the accident or shutdown the plant. The time required for design changes, delivery of parts, and implementation of required modifications at all plants employing a GE PASS was discussed at the May 2, 1984 meeting on between GE, the BWR Owners Group, and the NRC staff. The NRC staff indicated acceptance of the required delay in all cases and stated that the 5% power limitation imposed on WNP-2 was appropriate for all other plants in startup.

Safety Parameter Display System (SPDS) and Emergency Response Facility Data System (ERFD) Operability

Description

PECO stated in its October 25, 1983 letter from V. S. Boyer to A. Schwencer that the SPDS would be operational by February 29, 1984 and that operator training would be completed by May 1, 1984. FSAR Section 1.13 states that the system is expected to be functional prior to fuel load. The SPDS is a "subsystem" of the ERFDS, a non-safety related computerized data handling and display system.

All system hardware is presently installed and powered up. The system is undergoing calibration and de-bugging. The display formats for the SPDS are functional and can be called up in the Control Room, TSC, and EOF. Operator training on use of the system is complete. Additional time will be required to complete the debugging process; until completed, the ERFDS and SPDS cannot be considered functional. PECO proposes the following schedule for operation of the ERFDS and SPDS:

- Hardware Installed and Powered	Complete
- SPDS Display Formats Loaded into ERFDS	Complete
- Operator Training	Complete
- SPDS Displays Functional	March 1, 1985
- Reg. Guide 1.97 Displays Functional	April 1, 1985

Justification

All parameters which are part of the SPDS displays are also displayed on hardwired indicators in the Control Room in accordance with the Limerick commitments to Regulatory Guide 1.97; SPDS provides no additional information to the operators. Further, NUREG-0737 Supplement 1, Section 4.3 allows for the licensee to propose an implementation schedule. There is no requirement that all provisions of Supplement 1 be implemented prior to fuel load. There are no regulatory requirements governing ERFDS operability.

For the above reasons, the proposed schedule for SPDS and ERFDS operation will provide for safe operation of the plant in accordance with NRC regulatory requirements.

Process Computer System

Description

The Process Computer System (PCS) consists of a Primary and Backup Computer, analog/digital I/O (Input/Output) hardware, three (3) consoles, and four (4) Remote Terminals for communication with Bradshaw Pumping Station, Perkiomen Pumping Station, the 500kV and 220kV Substations.

All hardware is presently installed with the exception of the Remote Terminals at Bradshaw and Perkiomen. All of the installed hardware is powered up and undergoing functional testing with the exception of the Backup Computer and the Engineer's Console.

PECO proposes to have the entire PCS operational before exceeding 5% power.

Justification

The PCS is designed to periodically determine the three-dimensional power density distribution for the reactor core and near-continuous monitoring of core power loads. It also provides automatic and operator-initiated isotopic composition data for each fuel bundle in the core (FSAR Section 7.1.2.1.13.1).

The PCS provides interlock inputs to the rod block circuitry to supplement and aid in the enforcement of procedural restrictions on control rod manipulations.

Digital input/output hardware and software required for the rod block circuitry will be operational prior to fuel load.

No other monitoring or calculational aspects of the PCS are required below 10% power. Furthermore, there are neither safety design bases nor specific regulatory requirements for this system.

For the above reasons, the proposed implementation schedule for the PCS operation will provide for safe operation of the plant in accordance with NRC regulatory requirements.