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United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

Byron Station, Units 1 and 2
Facility Operating License Nos. NPF-37 and NPF-66
NRC Docket Nos. STN 50-454 and STN 50-455

Subject: Regulatory Commitment Change Summary Report

Please find enclosed the "Regulatory Commitment Change Summary Report" for Byron Station. This report contains summary information from January 1, 2001, through December 31, 2001. Revisions to docketed regulatory commitments were processed using Nuclear Energy Institute's document NEI 99-04, "Guidelines for Managing Nuclear Regulatory Commission (NRC) Commitment Changes," Revision 0.

If you have any questions concerning this letter, please contact William Grundmann, Regulatory Assurance Manager, at (815) 406-2800.

Respectfully,

Richard P. Lopriore
Site Vice President
Byron Nuclear Generating Station

RPL/GS/dpk

Attachment

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Byron Station
NRC Project Manager – NRR – Byron Station
Office of Nuclear Facility Safety – Illinois Department of Nuclear Safety

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ATTACHMENT

BYRON STATION

REGULATORY COMMITMENT CHANGE SUMMARY REPORT

BYRON STATION

REGULATORY COMMITMENT CHANGE SUMMARY REPORT

Original Document: Commitment 454-251-88-25100 (Response to NRC Notice of Violation 454/87041-01 and 455/87038-03)

Subject of Change:

This commitment was made in response to a NRC Violation issued for failure to develop a training matrix and training standards for technical activities performed by technical staff engineers. The commitment was to revise the Byron Station administrative procedure for engineering personnel qualifications to include reference to a surveillance and test equipment matrix and add a section on requirements for special qualification or test equipment qualification and add training requirements for the same. Also referenced in the response to the violation was a testing manual to be used as a text for annual training and requalification for performance of surveillances and tests. This commitment was deleted.

Basis:

This commitment was made in 1988. At the time of the NRC Violation, a programmatic deficiency in the Byron Station engineering training program had been identified. This deficiency was corrected by incorporation of a training qualification matrix and training standards for technical activities performed by technical staff engineers. The training qualification matrix and training standards have been programmatically implemented for an extended period of time and continue to be implemented. Engineering training programs now require training on test equipment as part of initial training and certification guides have been developed which identify surveillances performed on an assigned system. The training program periodically reviews conduct of testing for changes in how testing is conducted or for changes in procedures to determine if training is appropriate. These processes have replaced the use of a testing manual. The intent of the original commitment continues to be met.

Status:

This commitment was deleted through Commitment Change Identification Number 01-002.

Original Document: Commitments 454-104-97-00600, 455-441-97-00500, and 455-441-97-00500-01 (NRC Generic Letter 97-06 Guidance)

Subject of Change:

These commitments were made in response to NRC Generic Letter (GL) 97-06, "Degradation of Steam Generator Internals," as stated in a ComEd to NRC transmittal dated March 20, 1998. In that letter, ComEd (now Exelon) committed to perform the following two inspections each refueling outage in the Westinghouse Model D-5 steam generators (SGs) for Byron Unit 2.

1. Inspection of Tube Support Plates (TSPs)

    a. At present, 100% of the tube-to-tube support plate intersections are inspected each refueling outage with eddy current.

b. 100% of the tube support plate intersections will be inspected each refueling outage with the conventional bobbin coil probe for identifying distorted tube support plate signals, anomalies and to verify proper location of the TSPs.

2. Inspection of Pre-heater Water Box Tubes

a. Eddy current inspections of peripheral and T-slot tubes within the pre-heater will be undertaken at each scheduled refueling outage to detect loose parts and any tubes with significant tube wall degradation.

The intent of the above commitments was to perform inspections in all four SGs each refueling outage. These commitments were revised to limit the scope of the inspections to the Byron Unit 2 SGs selected for inspection and now read as follows:

1. Inspection of Tube Support Plates (TSPs)

a. 100% of the tube-to-tube support plate intersections will be inspected with eddy current in accordance with the EPRI PWR Steam Generator Examination Guidelines sampling plan requirements, at a minimum.

b. 100% of the tube support plate intersections in those steam generators selected for inspection will be inspected with the conventional bobbin coil probe for identifying distorted tube support plate signals, anomalies and to verify proper location of the TSPs.

2. Inspection of Pre-heater Water Box Tubes

a. Eddy current inspections of peripheral and T-slot tubes within the pre-heater in those steam generators selected for inspection will be undertaken to detect loose parts and any tubes with significant tube wall degradation.

Basis:

These commitments were made in 1998. The ComEd response to Generic Letter 97-06, dated March 20, 1998, stated that ComEd's SG inspection programs are in accordance with NEI 97-06 Steam Generator Program Guidelines. The guidelines allow up to two fuel cycles between inspections, provided that steam generator integrity is shown to be maintained throughout the next operating period until these inspections are performed. Industry guidelines and Exelon procedures require these types of evaluations to be conducted to ensure the appropriate inspections are performed to maintain SG integrity over the current and future operating periods. Our steam generator inspection programs continue to meet NEI 97-06 Steam Generator Program Guidelines.

Westinghouse WCAP-15093, "Evaluation of EDF Steam Generator Internals Degradation – Impact of Causal Factors on the Westinghouse Models F, 44F, D, and E2 Steam Generators," concludes that the Byron/Braidwood Unit 2 steam generators are not susceptible to the type of degradation described in the GL, which includes patch plate weld degradation, TSP ligament cracking, wrapper drop, or TSP flow hole erosion-corrosion. Westinghouse Technical Bulletin NSD-TB-97-05, "Water Box Erosion," provides potential degradation growth data that can be used in determining the appropriate water box inspection interval using NEI 97-06 methods. The data supports more than one cycle between inspections while maintaining SG integrity within structural limits. The inspections described in the ComEd response to GL 97-06 were performed in Byron Unit 2 refueling outages B2R07 and B2R08 and no degradation was found.

The Westinghouse Model D-5 design and the Exelon SG Management Program, which implements industry guidance, ensures SG integrity is maintained throughout the entire operating period between inspections. Based upon the implementation of the industry guidance and Exelon procedures to evaluate all degradation mechanisms prior to each inspection to ensure SG integrity is met throughout the appropriate inspection interval, these commitment changes are justified. These commitment changes are in accordance with inspection and evaluation methods described in NEI 97-06, EPRI PWR Steam Generator Examination Guidelines, EPRI Steam Generator Integrity Assessment Guidelines, and Byron Technical Specification 5.5.9, "Steam Generator Tube Surveillance Program."

Status:

These commitments were revised through Commitment Change Identification Number 01-011.

Original Document: Commitments 454-251-88-275 and 454-251-88-276 (NRC Generic Letter 87-12 Guidance)

Subject of Change:

These commitments were made in response to Item 2 of NRC Generic Letter 87-12, "Loss of Residual Heat Removal (RHR) While the Reactor Coolant System (RCS) is Partially Filled." Item 2 requested a description of instrumentation and alarms provided to operators for controlling the thermal and hydraulic aspects of the Nuclear Steam Supply System (NSSS) during operation with the RCS partially filled. A description of temporary connections, piping, and instrumentation was requested. The subject commitments encompassed the portion of our response to Item 2, which addresses installation of tygon tube for RCS level indication and also addressed subsequent removal from service of both the tygon tube and level instrument LI-RY046. These commitments were deleted.

Basis:

These commitments were made in 1987 and were incorporated into Station operating procedures for installation and removal from service of RCS level instrumentation. Station operating procedures continue to provide direction for installation and removal of RCS level instrumentation. Required valves to isolate level instrumentation will continue to be closed under direction of Station procedures, but these valves will no longer be taken out-of-service. The intent of the original commitments continues to be met.

Status:

These commitments were deleted through Commitment Change Identification Number 01-012.

Original Document: Commitment 454-251-88-26700 (NRC Generic Letter 87-12 Guidance)

Subject of Change:

This commitment was made in response to Item 5 of NRC Generic Letter 87-12, "Loss of Residual Heat Removal (RHR) While the Reactor Coolant System (RCS) is Partially Filled." Item 5 requested information providing reference to and a summary description of procedures in the control room describing operation while the RCS is partially filled. The subject commitment encompassed the portion of our response to Item 5, which addressed operating procedures for normal plant operation while the RCS is partially filled, specifically for Mode 5 operation. This portion of our response provided a general description of procedures used for plant shutdown and cooldown to take the plant from Startup Conditions (Mode 2) to Cold Shutdown (Mode 5). Included in the discussion were placement of RHR on line, RCS cooldown, stopping reactor coolant pumps, RCS depressurization, and draining of the entire RCS to a pre-determined level or draining a single RCS loop following isolation. This commitment was deleted.

Basis:

This commitment was made in 1987 and the intent of the commitment continues to be met. Direction has been provided in our procedures for an extended period of time and continues to be provided to address Mode 5 operations while the RCS is partially filled. The commitment deletion does not change actual practice.

Status:

This commitment was deleted through Commitment Change Identification Number 01-048.

Original Document: Commitments 454-251-91-11000 and 455-225-90-30200 (Response to NRC Notice of Violation 455/90023-01)

Subject of Change:

These commitments stated that modification work packages which have Pre Out-of-Service/Limiting Conditions for Operation Action Requirements (LCOAR) work will be scheduled and statused on the routine or outage work schedule (as applicable) on a sub-package level. These commitments were deleted.

Basis:

These commitments were made in 1990 and the intent of the commitments continues to be met. The separation of work tasks is currently performed in accordance with a standard Exelon procedure governing preparation of maintenance work packages. Work is subdivided non-outage, pre-outage, or outage work. Work request tasks are scheduled to the task (sub-package) level. These are standard work processes that have been in place for an extended period of time.

Status:

These commitments were deleted through Commitment Change Identification Number 01-013.

Original Document: Commitment 454-251-87-00800 (Unit 2 LER 87-002-01, "Reactor Trips and Feedwater Isolations Due to Operator Difficulty in Controlling Steam Generator Level Transients at Low Power")

Subject of Change:

This original commitment addressed main steam line warm-up by controlling steam line pressurization using Main Steam Isolation bypass valves for equalization of pressure around the Main Steam Isolation Valves (MSIVs). For Unit 2, performance of pressure equalization around the MSIVs was limited to Operational Modes 3 and 4 only. This commitment was revised to state that Unit 2 MSIVs will only be opened if the unit is in Modes 3 or 4; or opening of Unit 2 MSIVs in Mode 2 is allowed only if the corresponding MSIV bypass valve is opened and the steamline is warmed to normal operating temperature.

Basis:

The original commitment was made in 1988 during revision of a 1987 Licensee Event Report (LER). Subsequent plant changes and operating procedures have provided increased capability to adequately control Steam Generator level during various plant conditions and startup to preclude plant transients. The revised commitment clarifies the intent of the LER corrective action(s) to prevent an inadvertent steam draw during opening of an MSIV in Mode 2. The revised commitment preserves the intent of the original commitment.

Status:

This commitment was revised through Commitment Change Identification Number 01-014.

Original Document: Commitments 454-180-97-SCAQ00014-01, 02, 03 (Unit 1 LER 97-014, "Testing of P-11 Permissive Missed Due to Inadequate Procedure")

Subject of Change:

These commitments were to revise Instrument Maintenance Department 92-day surveillance functional test procedures (now referred to as channel operational test procedures) to properly test input relays to the Solid State Protection System for pressurizer pressure channels to confirm P-11 permissives operability. These commitments have been revised to include test of the P-11 permissives operability in 18-month channel calibration surveillance procedures as well.

Basis:

These original commitments were made in 1997 and were applied to the 92-day channel operational test procedures for testing pressurizer pressure inputs to confirm P-11 permissives operability and P-11 permissives operability was verified on a quarterly frequency. Since that time, 18-month channel calibration surveillance procedures have been revised to also satisfy the 92-day channel operational test requirement(s) to confirm P-11 operability. The intent of the original commitments to properly test the P-11 permissives on a 92-day frequency continues to be met.

Status:

These commitments were revised through Commitment Change Identification Number 01-015.

Original Document: Commitment 454-180-97-0003-02 (Unit 1 LER 97-003, "Equipment Hatch Gallery Not Properly Attached to the Containment Structure")

Subject of Change:

This original commitment was to revise the Byron Station maintenance procedure writer's guide to provide enhanced guidance for maintenance procedure writers and work analysts to greater assure equipment design requirements are considered when writing or revising procedures and work instructions. This commitment was deleted.

Basis:

This commitment was made in 1997. A standard Exelon writer's guide for procedures has since been developed that continues to provide direction and guidance for incorporation of equipment design requirements when writing or revising procedures or written instructions. The intent of the original commitment continues to be met.

Status:

This commitment was deleted through Commitment Change Identification Number 01-016.

Original Document: Commitment 454-251-89-21900

Subject of Change:

This original commitment was to implement a Byron Station administrative procedure that provided department guidance for establishing Quality Control (QC) hold points. The procedure provided guidelines for consistently and adequately identifying hold points. This commitment was deleted.

Basis:

This commitment has been in place since 1989. The Byron Station procedure that provided department guidance for QC hold points is being superceded by a standard corporate procedure that contains the Exelon QC inspection plan which prescribes the minimum required hold points. This standard procedure will continue to prescribe necessary guidelines for consistently and adequately identifying hold points. The intent of the original commitment continues to be met.

Status:

This commitment was deleted through Commitment Change Identification Number 01-034.

Original Document: Commitment 454-100-94-01005-01 (Response to NRC Notice of Violation 454(455)94010-05)

Subject of Change:

This original commitment required revision of the Station administrative procedure for Quality Control Field Inspections and implementation of a Station policy addressing Quality Control Field Inspection Involvement in Safety/Regulatory Related Minor and Exempt Changes, to require 100% dimensional verification inspections on piping and component support installations installed under the minor/exempt change process. The commitment was revised such that installed piping modifications require 100% dimensional verification inspection by certified Quality Control inspector(s) utilizing standard Nuclear Station Work Procedures for Fabrication and Installation of Piping and Tubing; and for Pipe Support Installation and Inspection.

Basis:

The commitment was made in 1994. Requirements for dimensional verification have now been incorporated directly into the procedures utilized for installation. The terms minor modification and exempt change are no longer utilized as part of the facility change process, the standard term modification is used. The intent of the original commitment continues to be met.

Status:

This commitment was revised through Commitment Change Identification Number 01-035.

Original Document: Commitments 454-251-88-61100 and 454-251-88-63300 (Unit 1 LER 88-007, "Loss of Shutdown Cooling During Reactor Cavity Level Lowering Evolution Due to Unexpected Flow Phenomenon," and related Response to NRC Notice of Violation 454/88019-01)

Subject of Change:

In the LER response in 1988, these original commitments stated that operating procedure revisions had been initiated to provide licensed operators with better control of reactor cavity drain rate and to provide more guidance and cautions. It was further stated, as part of a related NRC violation response in 1989, that the operating procedure governing pump down of the reactor cavity to the refueling water storage tank had been revised to require two functional methods of level indication to be used for any draining below the 403' elevation. In addition, as part of the violation response, it was mentioned that this same procedure stated visual indication of the reactor vessel level at or below the "Top Hat" area with the upper internals installed was not reliable. The violation response also stated that the reactor coolant system drain operating procedure required use of the Chemical and Volume Control System when draining below the reactor vessel flange, resulting in a lower drain down rate.

These commitments were deleted.

Basis:

These original commitments were made in 1988 and 1989 and the intent of the commitments continues to be met. In 1991, a portion of the commitments was revised to state that two independent level indications must be available below 402'6" reactor cavity elevation. This same item was revised in year 2000 to state that two independent level indications must be available below 402'. These changes were necessary since level instrumentation does not come on scale until 402'6" and this allowed a channel check to be performed from 402'6" to 402'.

The original commitments identified in the LER response and in the related NRC violation response have been proceduralized for a long period of time. The existing operating procedure governing pump down of the refueling cavity to the refueling water storage tank and the existing operating procedure governing reactor coolant system drain continue to provide direction previously stated in these commitments. Deletion of these commitments does not change actual practice.

Status:

These commitments were deleted through Commitment Change Identification Numbers 01-049 and 01-050.

Original Document: Commitment 454-251-85-04600 (Response to NRC Notice of Violation 454/85002-02)

Subject of Change:

A NRC Violation was issued for having isolated both safety injection (SI) pumps from the RCS cold leg injection header in Mode 3 while raising SI accumulator level. This had rendered both trains of safety injection inoperable for approximately 15 minutes. A portion of the response to the violation committed to revise the operating procedure used for raising SI accumulator level to allow the use of either SI pump to fill the accumulators, depending on plant conditions. The original procedure had been very restrictive allowing only the A train SI pump to be used for raising accumulator level. This commitment has been deleted.

Basis:

This commitment was made in 1985 and the intent of the commitment continues to be met. Direction has been provided in our operating procedure(s) for a long period of time and continues to be provided to address the use of either SI pump for filling SI accumulators and under what conditions each pump can and/or must be used to prevent a LCO violation. This is a well-established standard practice.

Status:

This commitment was deleted through Commitment Change Identification Number 01-055.

Original Document: Commitment 455-180-95-0004-02 (Unit 2 LER 95-004, "Inadequate Diesel Generator Post Maintenance Testing Due to Management Deficiency")

Subject of Change:

This original commitment was to implement a corrective action recommendation as a permanent test procedure, which verifies the operability of the emergency diesel generator following voltage regulator and/or governor adjustments, repairs, or replacements and will be applicable for all unit operating modes. The commitment was revised to state that the recommendation will be implemented in either a permanent test procedure or a special plant procedure (SPP) which verifies the operability of the emergency diesel generator following voltage regulator and/or governor adjustments, repairs, or replacements and will be applicable for all unit operating modes consistent with Technical Specifications.

Basis:

This commitment was made in 1995. A permanent site procedure and SPP have equivalent administrative and technical controls. Either procedure process provides acceptable controls for testing methodology. Additionally, a clarifying comment was added to state that the applicable testing mode must be consistent with Technical Specification requirements. The intent of the original commitment continues to be met.

Status:

This commitment was revised through Commitment Change Identification Number 01-057.

Original Document: Commitments 454-251-90-07600 and 454-251-90-08500 (NRC Generic Letter 89-13 Guidance)

Subject of Change:

These commitments were originally made pertaining to infrequently used cooling piping for the Auxiliary Feedwater system (AF) diesel engines and their auxiliary equipment, in response to NRC Generic Letter (GL) 89-13, "Service Water System Problems Affecting Safety-Related Equipment." Item I of GL 89-13 required implementation and maintenance of an ongoing program of surveillance and control techniques to significantly reduce the incidence of flow blockage problems as a result of biofouling. Under item I, Byron Station had committed to Control Technique C, which addressed flushing and flow testing for redundant and infrequently used cooling loops. The Essential Service Water system (SX) supply and return piping to the AF diesel engine auxiliary coolers and room coolers had been normally stagnant except when the diesel engine was running. By virtue of operating the diesel driven pump for surveillance testing, a flowpath was established through the diesel engine and pump auxiliary coolers and room coolers. This surveillance testing was conducted on a monthly and quarterly basis for approximately 30 minutes and two hours, respectively. These commitments to periodically flush the AF piping have been deleted.

Basis:

These commitments were made in 1990. Since that time the SX cooling flowpath to both the Unit 1 and Unit 2 diesel driven AF pumps and auxiliary equipment has been modified to provide continuous flow. This provides continuous flushing and biocide injection, such that the original commitment to periodically flush AF piping is no longer necessary.

Status:

These commitments were deleted through Commitment Change Identification Number 01-059.

Original Document: Commitment 454-251-88-93700 (October 29, 1986 letter from K. A. Ainger (ComEd) to H. R. Denton (NRR), "Byron Station Units 1 and 2 Application for Amendment to Facility Operating License NPF-37 Appendix A, Technical Specifications")

Subject of Change:

This original commitment was to not intentionally take one SX pump out of service (OOS) from each unit at the same time for maintenance. The commitment was revised to state one SX pump from each unit will not intentionally be taken out of service at the same time for maintenance unless:

a) One or both units are shutdown (i.e., MODES 5 or 6); or

b) If both units are at power, one pump from each unit may be taken out of service, provided compensatory measures are put in place to restore one of the out of service SX pumps to service within 4 hours if a loss of service water event occurs on one of the units.

The commitment revision was required to facilitate maintenance of SX pump suction isolation valves.

Basis:

The original commitment was made in 1986 in response to NRC concerns with the probability and consequences of a loss of all essential service water on one unit and was credited in the NRC Safety Evaluation for Amendment 24 to the Technical Specifications for Byron Units 1 and 2. Subsequently, the commitment was relaxed to allow annual inspections of the SX Cooling Tower basins with one SX pump on each unit OOS (as documented in August 20, 1991 letter from T. K. Schuster (ComEd) to Dr. T. E. Murley (NRR), "Byron Station Units 1 and 2 Annual Inspection of the Essential Service Water Cooling Tower Basin"). The relaxation was allowed based on the ability to quickly and easily restore the OOS SX pumps to service if a loss of all SX event occurred.

A loss of all SX event is not specifically evaluated as part of the design basis for Byron. Generic evaluations were performed for multi-plant sites by the NRC (NUREG/CR-5526, "Analysis of Risk Reduction Measures Applied to Shared Essential Service Water Systems at Multi-Unit Sites, June 1991;" and NRC Generic Issue 130, "Essential Service Water Pump Failures at Multiplant Sites," Rev. 2). The generic evaluations resulted in the issuance of NRC Generic Letter (GL) 91-13, "Request for Information Related to the Resolution of Generic Issue 130, Essential Service Water System Failures at Multi-Unit Sites," September 19, 1991. In part, GL 91-13 recommended that licensees of multi-unit sites implement Technical Specification changes to 1) require the SX unit cross-tie to be capable of being opened from the control room as a flow path between the two units, and 2) when one unit was at power and the opposite unit was in Modes 5 or 6, at least one pump on the opposite unit is operable and available to provide service water to the operating unit. The GL did not impose the Byron commitment (to not intentionally take one SX pump OOS from each unit at the same time) on any of the other multi-unit sites. The commitment was not discussed in the Byron response to GL 91-13 (March 16, 1992 letter from D. J. Chrzanowski (ComEd) to Dr. T. E. Murley (NRR), "ComEd Response to Generic Letter 91-13 for Byron and Braidwood Stations").

A recently conducted 10 CFR 50.92 Significant Hazards Consideration evaluation of the revised commitment concluded that a significant hazards consideration does not exist. In MODES 5 and 6, as described in Byron Technical Specification Bases, the operability requirements of the unit-specific SX system are determined by the systems it supports and there are no opposite-unit SX system requirements. When both units are at power, the consequences of a loss of service water event are not significantly increased because existing procedures will be used to crosstie the units and provide cooling water to vital equipment and appropriate compensatory measures will be in place to quickly restore one of the OOS SX pumps to service within 4 hours (prior to the need for Residual Heat Removal shutdown cooling). These actions will prevent reactor coolant pump seal failure or core melt accidents. The revised commitment description meets the intent of the original commitment and subsequent relaxation in 1991.

Status:

This commitment was revised through Commitment Change Identification Number 01-060.