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ComEd

December 27, 1999

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Quad Cities Nuclear Power Station, Units 1 and 2
Facility Operating License Nos. DPR-29 and DPR-30
NRC Docket Nos. 50-254 and 50-265

Subject: Proposed Technical Specifications Change
Surveillance Test Intervals and Allowable Outage Times for Protective Instrumentation

Reference: Letter from D. C. Trimble (USNRC) to R. A. Anderson (CP&L), dated March 30, 1995, "Issuance of Amendment No. 175 to Facility Operating License No DPR-71 and Amendment No. 206 to Facility Operating License No. DPR-62 Regarding Increases in Surveillance Test Intervals and Allowable Out-Of-Service Times for Selected Instrumentation - Brunswick Stream Electric Plant, Units 1 and 2"

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," Commonwealth Edison (ComEd) Company proposes to amend Appendix A, Technical Specifications (TS), of Facility Operating License Nos. DPR-29 and DPR-30. The proposed changes will increase allowable out-of-service times (AOTs) and surveillance test intervals (STIs) for selected TS actuation instrumentation.

The proposed changes implement recommendations resulting from generic evaluations (i.e., AOT/STI licensing topical reports) performed by General Electric and the Boiling Water Reactor Owners' Group and subsequently approved by the NRC. These topical reports assessed the reliability of TS actuation instrumentation and concluded that extending AOTs and STIs for test and repair activities enhances operational safety because: 1) the potential for inadvertent plant scrams is reduced, 2) the number of test cycles on equipment is minimized and 3) the use of plant personnel can be optimized. The proposed changes are consistent with the instrument STIs and AOTs found in the Improved Standard Technical Specification, ISTS (i.e., NUREG-1433, Revision 1, "Standard Technical Specifications, General Electric Plants, BWR/4"). A similar change was approved by the NRC for the Brunswick Steam Electric Plant, Units 1 and 2 (see referenced letter).

PDR ADOCK 05000254

Change: NRC PDR ltr encl 1 1w/ogup A001

The proposed amendment request is subdivided as follows:

1. Attachment A gives a description and safety analysis of the proposed changes.

2. Attachment B includes the marked-up TS pages with the requested changes indicated.

3. Attachment C provides information supporting a finding of no significant hazards in accordance with 10 CFR 50.92(c).

4. Attachment D provides information supporting an environmental assessment.

5. Attachment E provides the site-specific evaluations performed by GE supporting the proposed changes. Attachment E contains information considered by GE to be proprietary and an affidavit to that effect has been included with Attachment E. Accordingly, pursuant to 10 CFR 2.790, ComEd requests that the proprietary information in Attachment E be withheld from public disclosure.

In order to support our conversion to ISTS, ComEd requests NRC approval of this amendment request by August 15, 2000, to be effective no later than 120 days following approval. This implementation period will permit the appropriate procedural/program revisions and training necessary to implement the proposed changes.

Note that ComEd is not proposing the surveillance interval relaxation for the Reactor Pressure Vessel (RPV) steam dome high-pressure and the RPV low-low level instruments. Design changes are planned to improve the reliability of these instruments. The RPV steam dome high-pressure instruments will be upgraded during the upcoming Unit 2 and Unit 1 refueling outages (i.e., currently scheduled to start in January 2000 and October 2000 respectively). Upgrades for the RPV low-low level instruments are planned for future Unit 2 and Unit 1 refueling outages (i.e., currently scheduled to start in February 2002 and October 2002 respectively). As part of our corrective action plan, ComEd has conservatively increased the calibration frequency for these instruments under station administrative controls until the instrument upgrades are completed.

This proposed TS change has been reviewed and approved by the Plant Operations Review Committee and the Nuclear Safety Review Board in accordance with the Quality Assurance Program.

ComEd is notifying the State of Illinois of this application for amendment by transmitting a copy of this letter and its attachments to the designated State Official.

If there are any questions or comments concerning this letter, please refer them to Mr. C. C. Peterson, Regulatory Assurance Manager, at (309) 654-2241, extension 3609.

Sincerely,

R. M. Krich
Vice President - Regulatory Services

Attachments: Affidavit
A. Description and Safety Analysis for Proposed Changes
B. Marked-Up Technical Specifications Pages
C. Information Supporting a Finding of No Significant Hazards
D. Information Supporting an Environmental Assessment
E. General Electric Site Specific Evaluations

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Quad Cities Nuclear Power Station
Office of Nuclear Facility Safety – Illinois Department of Nuclear Safety

STATE OF ILLINOIS	**)**	
COUNTY OF DUPAGE	**)**	
IN THE MATTER OF	**)**	
COMMONWEALTH EDISON (COMED) COMPANY	**)**	**Docket Numbers**
QUAD CITIES NUCLEAR POWER STATION UNITS 1 AND 2	**)**	**STN 50-254 and STN 50-265**

SUBJECT: **Proposed Technical Specifications Change Surveillance Test Intervals and Allowable Outage Times for Protective Instrumentation**

AFFIDAVIT

I affirm that the content of this transmittal is true and correct to the best of my knowledge, information and belief.

R. M. Krich
Vice President - Regulatory Services

Subscribed and sworn to before me, a Notary Public in and for the State above named, this 27th day of December, 1999.

Notary Public

" OFFICIAL SEAL "
Joseph V. Sipek
Notary Public, State of Illinois
My Commission Expires 11/24/2001

Attachment A
DESCRIPTION AND SAFETY ANALYSIS FOR PROPOSED CHANGES

A. SUMMARY OF PROPOSED CHANGES

In accordance with 10 CFR 50.90, Commonwealth Edison (ComEd) Company is proposing changes to the Technical Specifications (TS) for Quad Cities Nuclear Power Station, Units 1 and 2. The proposed TS changes represent revisions to TS Sections 3/4.1, "Reactor Protection System," and 3/4.2, "Instrumentation," by increasing allowable out-of-service times (AOTs) and surveillance test intervals (STIs) for specified actuation instrumentation. The proposed changes will permit channel functional tests to be conducted quarterly rather than weekly or monthly. In addition, the AOTs for repairs will be increased from 1 hour to 12 or 24 hours, and the AOTs for required surveillance tests will be increased from 2 hours to 6 hours. All proposed STI and AOT changes are consistent with General Electric (GE) Company licensing topical reports (LTRs) which have been reviewed and approved by the NRC. The development of these LTRs is described below. These topical reports assessed the reliability of TS actuation instrumentation and concluded that extending AOTs and STIs for test and repair activities enhances operational safety because: 1) the potential for inadvertent plant scrams is reduced, 2) the number of test cycles on equipment is minimized and 3) the use of plant personnel can be optimized.

<u>**Background**</u>

In 1983 the Boiling Water Reactor Owners' Group (BWROG) formed a Technical Specifications Improvement (TSI) Committee. This committee established a program to identify improvements to allowable out-of-service times and surveillance test intervals (i.e., AOT/STI) specified in BWR Standard TS. The primary objective was to minimize unnecessary testing and restrictive out-of-service times that could potentially degrade the overall plant safety and availability. Examples of some of the problems experienced with current TS are inadvertent scrams due to frequent testing, AOTs that are not of sufficient duration to perform repairs on a reasonable basis, excessive actuation of equipment contributing to component wear-out, and unwarranted radiation exposure to personnel performing surveillance testing.

During April 1984, the TSI Committee met with the NRC and outlined the Boiling Water Reactor (BWR) TS Improvement Program. The NRC expressed agreement with the overall approach. Subsequently, the BWROG developed a series of LTRs which provided the basis for extending the AOTs and STIs for key actuation instrumentation including the reactor protection system (RPS), emergency core cooling system (ECCS), containment isolations, control rod block functions, and other miscellaneous functions. These GE LTRs were reviewed and approved by the NRC. The NRC required that plant specific applications confirm the applicability of the generic analyses to the specific plant, and confirm that setpoint drift, which could be expected under the extended test intervals, is within the existing allowances in the respective instrument setpoint calculations.

As described in more detail below, ComEd has confirmed the applicability of the generic analyses to the Quad Cities Nuclear Power Station, Units 1 and 2. Attachment E provides the site-specific evaluations performed by GE supporting the proposed changes. In addition, we have confirmed that expected setpoint drift is within the existing allowances in the respective instrument setpoint calculations.

Note that ComEd is not proposing the surveillance interval relaxation for the reactor pressure vessel (RPV) steam dome high-pressure and RPV low-low level instruments. Design changes are planned to improve the reliability of these instruments. The RPV steam dome high-pressure instruments will be upgraded during the upcoming Unit 2 and Unit 1 refueling outages (i.e., currently scheduled to start in January 2000 and October 2000, respectively). Upgrades for the RPV low-low level instruments are planned for future Unit 2 and Unit 1 refueling outages (i.e., currently scheduled to start in February 2002 and October 2002, respectively). As part of our corrective action plan, ComEd has conservatively increased the calibration frequency for these instruments under station administrative controls until the instrument changeouts are completed.

In order to support our efforts to covert to the Improved Standard Technical Specification, ISTS, NUREG-1433, Revision 1, "Standard Technical Specifications, General Electric Plants, BWR/4," other administrative changes are also proposed.

B. DESCRIPTION OF THE CURRENT REQUIREMENTS

The TS require instrumentation important to safety be tested at a specified interval to ensure a high degree of safety system reliability. A CHANNEL FUNCTIONAL TEST is defined in the TS to be the injection of a simulated signal into the CHANNEL as close to the sensor as practicable to verify OPERABILITY including required alarm and/or trip functions and CHANNEL failure trips. The scope of the proposed changes are instruments being functionally tested on a monthly, or in some cases weekly, frequency at Quad Cities Nuclear Power Station, Units 1 and 2.

The current TS also provide AOTs for instrument testing and repairs. The current TS, in general, allow a one-hour AOT for repairs and a two-hour AOT for testing prior to exercising TS Action requirements. The changes proposed in this request extend the AOTs for test and repair activities based on an extensive effort among the BWROG, GE, and the NRC to determine proper STIs and AOTs for TS instruments.

C. BASES FOR THE CURRENT REQUIRMENTS

TS Section 3/4.1.A, Reactor Protection System (RPS)
The protection and monitoring functions of the RPS have been designed to ensure safe operation of the reactor. The RPS initiates a reactor scram when one or more monitored parameters exceed their specified limits, to preserve the integrity of the fuel cladding and the Reactor Coolant System (RCS) and minimize the energy that must be absorbed following a loss of coolant accident (LOCA). A reactor scram can be accomplished either automatically or manually. This protection is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the RPS. TS Section 3/4.1.A provides limiting conditions for operation (LCO), including associated AOTs and Action Statements, to ensure a high performance level of RPS for safe

operation. Surveillance requirements are also provided to ensure the quality of RPS and associated components is maintained.

In addition to the RPS instrumentation, protective instrumentation has been provided which initiate actions to mitigate the consequences of transients and accidents which are beyond the operator's ability to control. Each of the functional areas within the scope of the AOT/STI improvements are discussed below.

TS Section 3/4.2.A Instrumentation - Isolation Actuation
The isolation actuation instrumentation automatically initiates closure of appropriate isolation valves and/or dampers which are necessary to prevent or limit the release of fission products from the reactor coolant system, the primary containment and the secondary containment in the event of a loss-of-coolant accident or other reactor coolant pressure boundary leak. The isolation instrumentation includes the sensors, relays, and switches that are necessary to cause initiation of primary and secondary containment and reactor coolant pressure boundary system isolation.

TS Section 3/4.2.B Instrumentation - Emergency Core Cooling System (ECCS) Actuation
The ECCS instrumentation generates signals to automatically actuate safety systems which provide adequate core cooling in the event of a design basis transient or accident.

TS Section 3/4.2.C Instrumentation - Anticipated Transients Without Scram (ATWS) - Recirculation Pump Trip (RPT)
The ATWS RPT provides a means to limit the consequences of the unlikely occurrence of a failure to scram event.

TS Section 3/4.2.D Instrumentation - Reactor Core Isolation Cooling (RCIC) Actuation Instrumentation
The RCIC system actuation instrumentation is provided to initiate actions to assure adequate core cooling in the event of a reactor isolation from its primary heat sink and the loss of feedwater flow.

TS Section 3/4.2.E Instrumentation - Control Rod Blocks Actuation
The purpose of the control rod block instrumentation is to mitigate control rod withdrawal errors and prevent withdrawal of control rods when reactor core/plant conditions do not warrant control rod motion. Excessive reactivity insertion via control rod withdrawal at high power levels could challenge fuel integrity limits. The actual rod block signals originate in different systems such as the neutron monitoring system, but are enforced by the reactor manual control system.

TS Section 3/4.2.I Instrumentation - Suppression Chamber and Drywell Spray Actuation
Instrumentation is provided to monitor the parameters that are necessary to permit initiation of the suppression chamber and drywell spray mode of the low pressure coolant injection/containment cooling. The spray mode results in quicker depressurization following completion of the blowdown following a loss of coolant accident.

D. NEED FOR REVISION OF THE REQUIREMENTS

The primary objective of the BWROG AOT/STI TS Improvement Program is to minimize unnecessary testing and excessively restrictive AOTs that could potentially degrade overall plant safety and availability. The following are examples of some of the generic problems experienced with the present TS requirements:

- Inadvertent scrams or engineered safety feature actuations caused during the performance of frequent surveillance tests;
- AOTs which are not long enough to permit completion of surveillance tests, repairs, or maintenance activities on a reasonable basis;
- Excessive actuation of equipment which contributes to component wear-out, shortening equipment lifetimes and increased failure rates;
- Unnecessary radiation exposure to personnel performing required surveillance tests; and,
- The allocation of plant resources to perform excessive surveillance testing prevents plant personnel from performing other activities which may have a more significant contribution to plant safety.

In NUREG-1024, "Technical Specifications - Enhancing the Safety Impact," the NRC suggested that TS action statements be reviewed to assure that they have an adequate technical basis and do indeed minimize plant risk. The use of reliability analyses to support engineering judgment was recognized as a primary basis for improving TS requirements. Consistent with this approach, the BWROG generated a series of LTRs justifying STI and AOT extensions in the TS for the RPS, Isolation System, ECCS, Control Rod Block System Instrumentation, and additional reports that were also generated to support similar extensions to other functions such as ATWS-RPT. The NRC evaluated and subsequently approved each of these licensing topical reports.

E. DESCRIPTION OF THE PROPOSED CHANGES

The following TS change descriptions have been grouped by TS section. The reference(s) for each change represent the key supporting documentation including LTRS, BWROG correspondence and/or site specific evaluation reports as appropriate. Attachment B provides the TS marked-up pages.

TS Section 1.0, Definitions

Change No.	Description of Change	Reference(s)
1)	**Page 1-8:** Surveillance Frequency Notation BIM, equal to 60 days, has been added to Table 1-1 of TS Section 1.0, Definitions. The new frequency definition will be used on certain instruments as detailed below.	None

TS Section 3/4.1.A, Reactor Protection System (RPS)

Change No.	Description of Change	Reference(s)
2)	**Page 3/4.1-1:** Revise Actions 1 and 2 and Footnotes a and b to incorporate a 1 hour check for trip capability (loss-of-function) and to provide a 12 hour AOT for maintenance activities. Note that the manual scram functions (Functional Units 13 and 14) were not changed to allow a 12 hour AOT because their configuration is not consistent with the generic model evaluated in Reference 3.	2,3
3)	**Page 3/4.1-6:** Revise Table 3.1.A-1, Note (a), to increase the AOT for required surveillance testing from 2 hours to 6 hours and to incorporate a check for trip capability consistent with ISTS.	1,3
4)	**Page 3/4.1-7:** Add Note (r) to Table 4.1.A-1 to provide for weekly testing of automatic scram contactors. As discussed in Reference 3, sensor channel tests can be increased to quarterly provided the automatic scram contactors are tested on a weekly basis.	3
5)	**Pages 3/4.1-7, 3/4.1-8 and 3/4.1-9:** Revise Table 4.1.A-1 to extend the CHANNEL FUNCTIONAL TEST frequency to quarterly for the following Functional Units: 2.b, 2.c, 2.d, 3, 4, 5, 6, 7, 9, 10, 11 and 12. Note (q) to TS Table 4.1.A-1 has been added to specifically address planned upgrades to Functional Unit 3. Upgrades are planned for the upcoming Unit 2 and Unit 1 refueling outages for Functional Unit 3 that will support a CHANNEL FUNCTIONAL TEST frequency of Q and CHANNEL CALIBRATION frequency of E. A shiftly (S) CHANNEL check has also been provided consistent with ISTS. Table 4.1.A-1, Note (h), has been changed to reflect a 92-day calibration of associated trip units.	3
6)	**Page B 3/4.1-1:** Bases changes indicating that specified surveillance intervals and surveillance and maintenance allowable outage times have been determined in accordance with the NRC approved methodologies.	3

TS Section 3/4.2.A – Isolation Actuation

Change No.	Description of Change	Reference(s)
7)	**Pages 3/4.2-1 and 3/4.2-2:** Revise Action 2 to incorporate a 1 hour check for loss-of-function and to provide AOTs of 12 hours to repair Functional Units that are common to RPS and 24 hours to repair Functional Units that are not common to RPS. Due to loss-of-function check provided in Action 2, Action 3 has been deleted. Due to the deletion of Action 3, Footnotes (b) and (c) have been deleted. Revise Footnote (a) to reflect change in AOT requirement.	4,5,13,11
8)	**Page 3/4.2-7:** Revise Table 3.2.A-1, Note (a), to increase the AOT for required surveillance testing from 2 hours to 6 hours and to incorporate a check for trip capability consistent with ISTS.	1,4,5,11
9)	**Pages 3/4.2-8, 3/4.2-9 and 3/4.2-10:** Revise Table 4.2.A-1 to increase the Channel Functional Test intervals from M to Q for the following Functional Units: 1.a, 1.b, 1.c, 2.a, 2.b, 2.c, 2.d, 3.a, 3.b, 3.c, 3.d, 4.b, 5.a, 5.b, 6.a, 6.b, 7.a and 7.b. Table 4.2.A-1, Note (a), has been modified to reflect the increase in the Channel Calibration interval for the corresponding Trip Units from 31 to 92 days.	4,5,11
10)	**Page B 3/4.2-1:** Bases change indicating that specified surveillance intervals and surveillance and maintenance allowable outage times have been determined in accordance with the NRC approved methodologies.	4,5

TS Section 3/4.2.B – Emergency Core Cooling System (ECCS) Actuation

Change No.	Description of Change	Reference(s)
11)	**Pages 3/4.2-11 and 3/4.2-12:** Action 3 has been deleted. The Action 3 requirements have been incorporated into a new Action 38 for ADS initiation instrumentation. This change is consistent with ISTS.	1
12)	**Pages 3/4.2-14 and 3/4.2-15:** For ADS permissive functions 4e, 4f, 5e and 5f, the Minimum Channels per Trip Function requirement has been increased from 1/pump to 2/pump. This more restrictive change ensures each ADS trip system has a sufficient number of operable channels to initiate during a design basis event. This change is consistent with the ISTS.	1
13)	**Page 3/4.2-14:** A new Action 37 is proposed for HPCI Initiation Functional Units 3a and 3b. This conservative change ensures HPCI injection capability and is consistent with the ISTS.	1
14)	**Pages 3/4.2-14 and 3/4.2-15:** A new Action 38 has been incorporated for ADS initiation functions 4a, 4b, 5a, and 5b. This change is consistent with ISTS.	1

Attachment A
DESCRIPTION AND SAFETY ANALYSIS FOR PROPOSED CHANGES

Change No.	Description of Change	Reference(s)
15)	**Page 3/4.2-16:** Revise Action 30 to incorporate a 1 hour loss-of-function check and a 24 hour AOT for maintenance activities. Action 30a is only applicable in Modes 1, 2 and 3 because in Modes 4 and 5 the specific initiation time of low pressure ECCS is not assumed and the probability of a LOCA is lower. This change is consistent with the ISTS.	1,6,14,15
16)	**Page 3/4.2-16:** Revise Action 31 to incorporate a 1 hour loss-of-function check and a 24 hour AOT for maintenance activities. For CS, LPCI and HPCI, Action 31a is only applicable in Modes 1, 2 and 3 because in Modes 4 and 5 the specific initiation time of ECCS is not assumed and the probability of a LOCA is lower. This change is consistent with the ISTS.	1,6,14,15
17)	**Page 3/4.2-16:** Revise Action 32 to extend the AOT for maintenance activities to 24 hours.	6,14,15
18)	**Page 3/4.2-16:** Revise Action 33 to incorporate a 1 hour loss-of-function check. For Functional Units 1.d and 2.d, Action 33a is only applicable in Modes 1, 2 and 3 because in Modes 4 and 5 the specific initiation time of low pressure ECCS is not assumed and the probability of a LOCA is lower. This change is consistent with the ISTS.	1
19)	**Page 3/4.2-16:** Revise Action 34 to extend the AOT for maintenance activities to 24 hours.	6,14,15
20)	**Page 3/4.2-16:** Revise Action 35 to incorporate a 1 hour loss-of-function check and a 24 hour AOT for maintenance activities. This change is consistent with the ISTS.	1,6,14,15
21)	**Page 3/4.2-16:** A New Action 37 is proposed for HPCI initiation functions 3a and 3b. This change incorporates a 1 hour loss-of-function check and a revised AOT for maintenance activities. This change is consistent with ISTS.	1,6,14,15
22)	**Page 3/4.2-16:** A new Action 38 is proposed for ADS initiation functions 4a, 4b, 5a, and 5b. This new action incorporates a 1 hour loss-of-function check and a revised AOT for maintenance activities. If the action requirements can not be met, the ADS relief valves are declared inoperable. The actions for inoperable ADS relief valves are provided in TS 3.6.F, which provides ACTION requirements consistent with 3.2.B, Action 3 (which has been deleted by this proposed change).	1,6,14,15
23)	**Page 3/4.2-17:** Revise Table 3.2.B-1, Note (a), to increase the AOT for required surveillance testing from 2 hours to 6 hours and to incorporate a check for trip capability. These changes are consistent with the ISTS and the reliability analysis performed in Reference 6.	1,6,15

Change No.	Description of Change	Reference(s)
24)	**Pages 3/4.2-18, 3/4.2-19, and 3/4.1-20:** Revise Table 4.2.B-1 to increase the Channel Functional Test intervals from M to Q for the following Functional Units: 1.b, 1.c, 1.d, 2.b, 2.c, 2.d, 3.b, 3.c, 3.d, 3.f, 4.b, 4.e and 4.f. The Channel Calibration intervals for Functional Units 1.a, 2.a, 3.a, 3.e, and 4.a have been changed to BIM to reflect the current station practices. Table 4.2.B-1, Note (e), has been changed to reflect a 92-day calibration of the associated trip units.	6,15
25)	**Page B 3/4.2-2:** Bases changes indicating that specified surveillance intervals and surveillance and maintenance allowable outage times have been determined in accordance with the NRC approved methodologies.	6

TS Section 3/4.2.C – Anticipated Transient Without Scram (ATWS) - Recirculation Pump Trip (RPT)

Change No.	Description of Change	Reference(s)
26)	**Page 3/4.2-23:** Revise Table 3.2.C-1, Note (a), to allow a 6-hour AOT for testing and to incorporate a check for trip capability consistent with ISTS.	1,7
27)	**Page 3/4.2-24:** Revise Table 4.2.C-1 to increase the Channel Functional Test intervals from M to Q for the following Functional Units: 1 and 2.	7
28)	**Page B 3/4.2-2:** Bases changes indicating that specified surveillance intervals and surveillance and maintenance allowable outage times have been determined in accordance with the NRC approved methodologies.	7

TS Section 3/4.2.D – Reactor Core Isolation Cooling (RCIC) Actuation Instrumentation

Change No.	Description of Change	Reference(s)
29)	**Page 3/4.2-26:** Revise Table 3.2.D-1, Note (a), to allow a 6-hour AOT for testing and to incorporate a check for trip capability. These changes are consistent with the ISTS and the reliability analysis performed in Reference 8.	1,8
30)	**Page 3/4.2-27:** Revise Action 40 to incorporate a 1 hour loss-of-function check and a 24 hour AOT for maintenance activities. This change is consistent with the Improved Standard Technical Specifications.	1,8
31)	**Page 3/4.2-27:** Change Action 41 to allow 24 AOT for maintenance activities.	8

Change No.	Description of Change	Reference(s)
32)	**Page 3/4.2-27:** Revise Action 42 to incorporate a 1 hour loss-of-function check and a 24 hour AOT for maintenance activities. This change is consistent with ISTS.	1,8
33)	**Page 3/4.2-27:** Change Action 43 to allow 24 AOT for maintenance activities.	8
34)	**Page 3/4.2-28:** Revise Table 4.2.D-1 to increase the Channel Functional Test intervals from M to Q for the following Functional Units: 3 and 4. The Channel Calibration intervals for Functional Units 1 and 2 have been changed to BIM to reflect the current station practices.	8
35)	**Page B 3/4.2-2:** Bases changes indicating that specified surveillance intervals and surveillance and maintenance allowable outage times have been determined in accordance with the NRC approved methodologies.	8

TS Section 3/4.2.E – Control Rod Block Actuation

Change No.	Description of Change	Reference(s)
36)	**Page 3/4.2-33:** Revise Table 3.2.E-1, Action 52, to increase the AOT for maintenance activities from one hour to 12 hours.	7
37)	**Page 3/4.2-34:** Revise Table 3.2.E- 1, Note (i), to increase the AOT for required surveillance testing from 2 hours to 6 hours and to incorporate a check for trip capability consistent with ISTS.	1,7
38)	**Page 3/4.2-35:** Revise Table 4.2.E-1 to increase the Channel Functional Test intervals from M to Q for the following Functional Units: 1.a, 1.b, 1.c, 2.a.1, 2.a.2, 2.b, 2.c and 2.d.	9
39)	**Page B 3/4.2-4:** Bases changes indicating that specified surveillance intervals and surveillance and maintenance allowable outage times have been determined in accordance with the NRC approved methodologies.	7,9

TS Section 3/4.2.I – Suppression Chamber and Drywell Spray Actuation

Change No.	Description of Change	Reference(s)
40)	**Page 3/4.2-49:** Revise Table 3.2.I-1, Action 80.a, to increase the AOT for maintenance activities from one hour to 24 hours.	7
41)	**Page 3/4.2-49:** Revise Table 3.2.I-1, Note (c), to increase the AOT for required surveillance testing from 2 hours to 6 hours and to incorporate a check for trip capability consistent with ISTS.	1,7
42)	**Page 3/4.2-50:** Revise Table 4.2.I-1 to increase the Channel Functional Test interval from M to Q for the following Functional Units: 1 and 2. In addition, Table 4.2.I-1, Note (a), has been changed to reflect a 92-day calibration of the associated trip units.	7

Change No.	Description of Change	Reference(s)
43)	**Page B 3/4.2-5:** Bases changes indicating that specified surveillance intervals and surveillance and maintenance allowable outage times have been determined in accordance with the NRC approved methodologies.	7

F. SAFETY ANALYSIS OF THE PROPOSED CHANGES

ComEd has reviewed the referenced AOT/STI LTRs and completed the necessary plant-specific evaluations to confirm that the generic results and conclusions apply to Quad Cities Nuclear Power Station, Units 1 and 2. As stated within the NRC Safety Evaluations (SEs) for the AOT/STI LTRs, three conditions for NEDC-30851P-A (Reference 3) and two conditions for the remaining LTRs (References 4 through 9) must be addressed to justify application of the generic analyses to a plant-specific application. The following discussion provides the information requested by the NRC staff in plant-specific submittals.

NRC CONDITION NO. 1

Confirm the applicability of the generic analyses to the plant.

RESPONSE TO NRC CONDITION NO. 1

1. Licensing Topical Report NEDC-30851P-A (Reference 3) provides the justification for TS improvements for RPS actuation functions. A NRC Safety Evaluation for Reference 3 was transmitted in letter A.Thadani (USNRC) to T. Pickens (BWROG) dated July 15, 1987. Appendix L of Reference 3 identifies ComEd (including Quad Cities Nuclear Power Station, Units 1 and 2) as a participating utility in the development of the RPS TS Improvement Analysis. Section 7.4 of the Reference 3 states the following:

   "The evaluation found various differences between the RPS configuration of various plants and the generic plant. These differences include HFA relays, four scram contactors for BWR/2, sensor differences, scram parameter differences, and SDV sensor diversity differences. The assessment of these differences shows that while the HFA relays and the four scram contactors for BWR/2 would result in a higher overall RPS failure frequency, the improved technical specification intervals and allowable out-of-service times based on the generic plant would result in a net improvement to plant safety for plants with such differences. The effect of other differences on the RPS failure frequency is insignificant. Therefore, the generic results can be applied to plants in the BWROG Technical Specification Improvement Program."

   Included in Attachment E are the Quad Cities Nuclear Power Station plant-specific evaluations provided by GE (Reference 10) for the RPS actuation functions. The evaluations confirm the applicability of the Reference 3 generic analysis to Quad Cities Nuclear Power Station. In order to comply with the results of the NEDO-30851P-A analysis for RPS, Quad Cities Nuclear Power Station has included in the proposed TS change the requirement to test, on a weekly basis, the automatic scram contactors. This testing will be accomplished using the RPS system subchannel keylock switches, which de-energize the associated automatic scram contactor relays or an equivalent method.

2. Licensing Topical Report NEDC-30851P-A, Supplement 2 (Reference 4), provides the justification for TS improvements for BWR Isolation Functions common to RPS and ECCS. An NRC Safety Evaluation for Reference 4 was transmitted in letter C. Rossi (USNRC) to D. Grace (BWROG) dated January 6, 1989. Appendix A of Reference 4 identifies ComEd, including Quad Cities Nuclear Power Station as a participating utility in the development of the BWR Isolation Instrumentation Common to the RPS and ECCS Technical Specification Improvement Analysis. Section 3.3 of Reference 4 specifically addresses BWR 3/4 plants and includes common isolation functions consistent with those at Quad Cities Nuclear Power Station. Furthermore, included in Attachment E is a plant specific evaluation performed by General Electric for isolation actuation functions (Reference 11). The evaluation confirms the applicability of the Reference 4 generic analysis to Quad Cities Nuclear Power Station.

3. Licensing Topical Report NEDC-31677P-A (Reference 5) provides the justification for TS improvements for BWR Isolation Functions not common to the RPS or ECCS. A NRC Safety Evaluation for Reference 5 was transmitted in letter C. Rossi (USNRC) to S. Ployd (BWROG), dated June 18, 1990. Appendix E of Reference 5 identifies ComEd (including Quad Cities Nuclear Power Station) as a participating utility in the development of the Reference 5 analysis. The results for the BWR/3 product line are presented in Sections 5.1. Furthermore, included in Attachment E is a plant specific evaluation performed by General Electric for isolation actuation functions (Reference 11). The evaluation confirms the applicability of the Reference 5 generic analysis to Quad Cities Nuclear Power Station. Note that the secondary containment isolation functions also isolate the control room ventilation system (CREVS). The CREVS isolation function was generically evaluated in Reference 7. The NRC SE for Reference 7 approved AOT/STI extensions because the CREVS initiation logic and diversity was similar to previously analyzed systems, thus the change was acceptable. ComEd has reviewed the Reference 7 analysis with respect to the CREVS isolation logic, including the logic scheme and level of diversity, and has concluded the generic analysis is applicable to Quad Cities Nuclear Power Station.

4. Licensing Topical Report NEDC-30936P-A, Parts 1 and 2 (Reference 6) provides the justification for TS improvements for BWR ECCS Functions. NRC Safety Evaluations for the Reference 6 evaluation were provided in letters from A. Thadani (USNRC) and C. Rossi (USNRC) to D. Grace (BWROG) dated December 9, 1988. Appendix N of NEDC-30936P-A, Part 1, and Appendix B of NEDC-30936P-A, Part 2, identifies ComEd (including Quad Cities Nuclear Power Station) as a participating utility in the development of the BWROG Technical Specification Improvement Methodology for ECCS Actuation Instrumentation. Section 5.4 of NEDC-

30936P-A, Part 2, describes the generic analyses performed for BWR 3/4 plants. In order to support the full BWR product line, GE identified major differences from the population of BWRs that participated in the AOT/STI program. The differences were evaluated by changing the generic fault trees and re-evaluating the impact of proposed AOT/STI changes. These are documented as Case Studies in the BWROG Topical Report NEDC-30936P-A. Using the results of the Case Studies, individual plant–specific reports were produced to support each individual Utility's submittal to the NRC.

Note that GE generically evaluated a plant with loop selection logic similar to Quad Cities Nuclear Power Station, Units 1 and 2. This analysis is documented in the NRC approved Topical Report NEDC-30936P-A for ECCS Actuation Instrumentation (Case 4B). For the Quad Cities Nuclear Power Station site-specific analysis, GE compared the Quad Cities Nuclear Power Station design configuration with the configuration of the Case 4B plant. Only a single difference required additional analysis - the effect of the 18 month surveillance interval for the ADS Drywell Pressure Bypass Timers and ADS Initiation Timers. Case 4B assumed a one month surveillance interval. GE determined the effect of this change to be insignificant. Included in Attachment E is the plant specific evaluation provided by General Electric for ECCS actuation functions (Reference 15). The evaluation confirms the applicability of the Reference 6 generic analysis to Quad Cities Nuclear Power Station.

5. Licensing Topical Report GENE-770-06-1-A (Reference 7) provides the justification for TS improvements for selected instrumentation functions. A NRC Safety Evaluation for the Reference 7 evaluation was provided in letter C. Rossi (USNRC) to R. Binz (BWROG) dated July 21, 1992. The Reference 7 report identifies changes to surveillance test intervals and allowable out-of-service times for selected instrumentation for BWR plants. The report concluded that extending functional test frequencies and AOT for certain instruments was appropriate. The site-specific actuation functions encompassed by the Reference 7 evaluation are as follows: ATWS-RPT and Suppression Chamber and Drywell Spray Actuation. ComEd has reviewed the Reference 7 analysis and has verified applicability to the Quad Cities Nuclear Power Station as summarized below:

   ATWS-RPT: Instrumentation for the ATWS-RPT actuation consists of two low-low RPV level instruments and two high RPV pressure instruments. The signals are combined in a two-out-of two taken once logic scheme. The Reference 7 analysis (Section 3.3) evaluated the following logic schemes: one-out-of-two taken twice (BWR 4) and two-out-of-two taken once (BWR 6). These logic schemes are consistent with the design at Quad Cities Nuclear Power Station. The Reference 7 evaluation concluded that the proposed changes to ATWS instrumentation AOTs and STIs have a negligible effect on the reactivity shutdown failure frequency. For these reasons, the proposed changes are acceptable.

Suppression Chamber and Drywell Spray Actuation: As stated in Reference 7, certain BWRs employ manually initiated suppression pool and drywell spray functions. This design approach is consistent with the Quad Cities Nuclear Power Station containment spray functions. The Reference 7 evaluation (Section 3.1) concluded that extending the AOTs and STIs for these functions is acceptable because there are no automatic functions and, therefore, no corresponding automatic initiation functions. Similar extensions in AOTs and STIs for automatic actuation functions (e.g., ECCS and Isolation functions) were found to have a negligible impact on plant safety.

6. Licensing Topical Report NEDC-30851P-A, Supplement 1 (Reference 9), provides the TS improvement analysis for Control Rod Block actuation functions. A Safety Evaluation approving the Reference 9 evaluation was transmitted in letter C. Rossi (USNRC) to D. Grace (BWROG) dated September 22, 1988. Appendix B of Reference 9 identifies ComEd (including Quad Cities Nuclear Power Station) as a participating utility in the development of the TS improvement analysis for BWR Control Rod Block instrumentation. Extending the AOTs and STIs for the Control Rod Block actuation functions was found acceptable due to the benefits associated with reduced testing. Note that Reference 7 (Section 3.10) provides the basis for extending specific Control Rod Block AOTs for testing and maintenance activities. Similar to the RPS evaluation, extending the AOTs and STIs for the Control Rod Block functions has no significant impact on the availability of the Control Rod Block functions. Therefore, the proposed changes have a negligible impact on reactor safety. ComEd has reviewed the Reference 9 generic analysis and has confirmed the applicability to Quad Cities Nuclear Power Station, Units 1 and 2.

7. Licensing Topical Report GENE-770-06-02-A (Reference 8) provided the bases for TS improvements for the Reactor Core Isolation Cooling (RCIC) system. A NRC Safety Evaluation for Reference 8 was transmitted in letter C. Rossi (USNRC) to G. Beck (BWROG) dated July 30, 1992. As indicated in Reference 8 (Section 3.1) the BWR 3/4 product line was specifically evaluated, and was based primarily on the Reference 6 evaluations performed for the ECCS actuation instrumentation. The proposed AOT and STI changes for RCIC actuation instrumentation are justified based on their small affect on the water injection function unavailability and are consistent with the proposed changes for ECCS instrumentation. The RCIC actuation logic is comparable to other ECCS actuation functions at Quad Cities Nuclear Power Station, and is consistent with the actuation logic in the Reference 8 generic evaluation for BWR 3/4 plants. For these reasons, the Reference 8 generic analysis is applicable to the Quad Cities Nuclear Power Station, Units 1 and 2.

<u>NRC CONDITION NO. 2</u>

Demonstrate, by use of current drift information provided by the equipment vendor or plant specific data, that the drift characteristics for instrumentation used in the channels in the plant are bounded by the assumptions used in the generic analyses when the functional test interval is extended from monthly (or weekly) to quarterly.

RESPONSE TO NRC CONDITION NO. 2

The AOT/STI LTRs do not contain specific instrument drift assumptions. For this reason, the requirements for plant specific applications were clarified in letter C.E. Rossi (USNRC) to R.F. Janecek (BWROG) dated April 27, 1988 (Reference 12). As indicated in Reference 12:

> *"...licensees need only confirm that the setpoint drift which could be expected under the extended STIs has been studied and either (1) has been shown to remain within the existing allowance in the RPS and ESFAS instrument setpoint calculation or (2) that the allowance and setpoint have been adjusted to account for the additional expected drift."*

The setpoint methodology at Quad Cities Nuclear Power Station uses the instrument *calibration* frequency to account for potential instrument drift. For the instruments within the scope of AOT/STI, the calibration intervals are not being modified; therefore, the proposed changes are within the existing allowances in the instrument setpoint calculations. The only exceptions are instrument loops that contain Rosemount analog trip units. These devices are calibrated every 31 days in accordance with the current TS. ComEd has explicitly evaluated the setpoint drift associated with the Rosemount analog trip units. The results confirm that extending the current calibration frequency from 31 days to 92 days is acceptable and within existing setpoint allowances. This is consistent with the evaluation of analog trip units provided in Reference 3, Section 5.7.3, which states: "Current vendor drift information on analog trip units indicate that the calibration interval could be extended to 6 months." Therefore, increasing the Channel Functional Test intervals in accordance with the AOT/STI LTRs and, where applicable, the Rosemount analog trip unit calibration intervals from 31 days to 92 days is consistent with the current setpoint methodology at Quad Cities Nuclear Power Station.

Note that ComEd is not proposing the surveillance interval relaxation for the steam dome high-pressure and the reactor low-low level instruments. Design changes are planned to improve the reliability of these instruments. As part of our corrective action plan, ComEd has conservatively increased the calibration frequency for these instruments under station administrative controls.

NRC CONDITION NO. 3

Confirm that the differences between the parts of the RPS that perform the trip functions in the plant and those of the base case plant were included in the analysis for its plant done using the procedures of Appendix K of NEDC-30851P (This issue applies only to NEDC-30851P-A, Reference 3).

RESPONSE TO NRC CONDITION NO. 3

Attachment E includes the site-specific RPS evaluation report performed by GE (Reference 10). The analysis utilized the methodology outlined in Appendix K of Reference 3 to identify and evaluate the differences between the parts of the RPS that perform the trip functions at Quad Cities Nuclear Power Station and those analyzed in the generic study. The results of the site-specific analysis indicate that although the

RPS configurations for Quad Cities Nuclear Power Station differ in some respects from the configuration of the base case plant, the differences do not have a significant impact on the results and conclusions of the generic evaluation. Therefore, the Reference 3 generic analysis is applicable to Quad Cities Nuclear Power Station. Furthermore, as required by Reference 3, the proposed TS changes include a weekly surveillance of the automatic scram contactors.

RPS INSTRUMENT UPGRADE

The proposed change modifies the Surveillance Requirements for Reactor Protection System (RPS) Functional Unit 3, "Reactor Vessel Steam Dome Pressure-High." This change supports a planned upgrade to the Reactor Vessel Steam Dome Pressure-High instrumentation from pressure switches (Barksdale) to analog trip units (Rosemount). Analog trip units are a proven technology that are more reliable than the existing pressure switches which are sensitive to vibration and difficult to calibrate. Analog trip units are used in various applications at Quad Cities Nuclear Power Station, including the RPS low water level trip function. The proposed change extends the CHANNEL FUNCTIONAL TEST from M to Q, extends the CHANNEL CALIBRATION from Q to E and provides a CHANNEL CHECK of S consistent with the Improved Technical Specifications (Reference 1). ComEd has performed the appropriate setpoint calculations that supports these proposed intervals. In addition, these proposed intervals are consistent with RPS Functional Unit 4, "Reactor Vessel Water Level - Low," which also employ a Rosemount Transmitter/Trip Unit arrangement.

COMED RISK ASSESSMENT

The proposed AOT/STI changes implement recommendations resulting from generic reliability evaluations performed by General Electric and the Boiling Water Reactor Owners' Group. These topical reports assessed the reliability of TS actuation instrumentation and concluded that extending STIs and AOTs for test and repair activities does not adversely impact risk and may enhance operational safety by minimizing challenges to the plant. In accordance with the NRC SE conditions, plant specific applications must ensure plant design features are consistent with those of the corresponding reference plant.

Since the issuance of the AOT/STI SEs, the NRC has developed much more detailed acceptance criteria for Technical Specifications changes that involve a "risk informed" approach. These more detailed criteria are provided in Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" and Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications."

Regulatory Guide 1.177 states that licensee-initiated TS changes that are consistent with currently approved staff positions [e.g., regulatory guides, standard review plans, branch technical positions, or the Standard Technical Specifications (STS)] are normally evaluated by the staff using traditional engineering analyses. A licensee would not be expected to submit risk information in support of the proposed change. The proposed AOT/STI changes have been generically accepted by the NRC and incorporated by reference into the Improved Standard Technical Specifications (Reference 1). For this reason, a rigorous PRA (probabilistic risk assessment) evaluation - consistent with the current Regulatory Guides - is not required. However, ComEd has completed a risk assessment of the AOT/STI changes at Quad Cities Nuclear Power Station.

The purpose of this risk assessment was to ensure the acceptability of the original PRA evaluations performed to support the generic BWROG submittals. This assessment included the individual sensitivity cases performed to extend the applicability of the generic studies to Quad Cities Nuclear Power Station and a review of the recently upgraded Quad Cities Nuclear Power Station PRA and its insights to ensure the proposed changes are acceptable.

In general, because the proposed changes are limited to instrumentation STIs (functional testing only) and AOTs, the risk change is very small. This is due to the high degree of redundancy and in many cases diversity of instrumentation and signals to provide automatic actuation. Industry PRA analyses, including Quad Cities Nuclear Power Station, have generally confirmed that instrumentation for actuation of RPS, isolation, and ECCS are not dominant contributors to risk (relative to system related reliability). In addition, the proposed changes provide tangible benefits including:

- Reduction in inadvertent scrams (estimated by GE to be a 0.3% decrease in CDF) or engineered safety feature actuations caused during the performance of frequent surveillance tests.
- Reduction in personnel radiation exposure.
- Reduction in number of plant shutdowns. This has a small positive impact on reducing plant risk. AOTs which are not long enough to permit completion of surveillance tests, repairs, or maintenance activities on a reasonable basis are avoided.
- Reduction in failures due to equipment wear-out. Excessive actuation of equipment which contributes to component wear-out, shortening equipment lifetimes and increased failure rates.

In addition, Quad Cities Nuclear Power Station has significant plant design features (not included in the generic AOT/STI evaluations) that reduce the impact of certain instrumentation failures - therefore reducing the already low risk associated with STI and AOT extensions:

- Station Blackout Diesels afford an alternate power source to substantially reduce the Loss of Offsite Power (LOOP) induced Core Damage Frequency (CDF) (i.e., the major contributor assessed in the ECCS actuation logic report)
- The availability of external injection to the RPV via the feedwater system (i.e., use of Standby Coolant System) affords an injection path not dependent on the low pressure permissive.

- The Safe Shutdown Makeup Pump provides an alternate safe shutdown injection method (redundant motor driven high-pressure injection pump) affords yet another RPV makeup source.

- Manual initiation of equipment (ECCS), isolation, or RPS is a backup to the automatic initiation logic. The manual initiation logic is important in the integrated PRA model and is not generally included in the generic AOT/STI evaluations.

The risk assessment was aimed at addressing the following items:

- The appropriateness of any differences between Quad Cities and the reference plant in terms of safety significance.

- The appropriateness of the risk conclusions in light of the different risk spectrum currently calculated for Quad Cities compared with the reference plant.

- The appropriateness of the generic methodology which has received NRC approval.

- The feasibility of assuring that the more rigorous requirements of Regulatory Guides 1.174 and 1.177 would be met if such a quantitative analysis were performed.

The following conclusions were reached by the Quad Cities specific risk assessment review:

- The BWROG AOT/STI methodology is applied in a manner consistent with the approved SERs and the results are consistent with the Improved Standard Technical Specifications (Reference 1).

- The results from the application of the NRC approved BWROG methodology and the associated criteria are that the AOT/STI changes are acceptable.

- The BWROG AOT/STI methodology and evaluation process is a reasonable approach that captures many of the risk significant issues associated with changes in risk associated with changing AOTs or STIs for instrumentation.

- GE implemented the BWROG methodology for Quad Cities consistent with the plant specific implementation in previous plants and consistent with NRC SEs.

- The generic evaluations were tested against operating experience to determine whether the change was appropriate. For example, the Yarway channel functional test frequency has not been relaxed.

- The Base case identifies that the increase in frequency of scram contactor testing creates a low failure probability of scram failure. This is a significant safety improvement relative to the current Quad Cities Nuclear Power Station operation. This change to increase the frequency of testing is a major safety improvement that is judged to be larger than any very small decrement in risk associated with extending STIs or AOTs on other equipment.

In summary, Regulatory Guides 1.174 and 1.177 recognize that there may be substantial, yet not easily quantifiable, safety benefits associated with a change. The net change can, in fact, be considered to be a positive safety benefit based solely on these non-quantifiable changes. It is judged that the Quad Cities Nuclear Power Station AOT/STI changes may result in a net positive increase in safety based upon the conclusions stated above and the increase in scram contactor testing relative to the current Quad Cities practices.

G. IMPACT ON PREVIOUS SUBMITTALS

ComEd has submitted two previous license amendments that involving actuation instrumentation:

1) J. Dimmette letter to USNRC, SVP 99-182, dated October 12, 1999, "Request for Technical Specifications Change, Reactor Protection System"

   This proposed change eliminates the reactor protection system (RPS) electro-hydraulic control (EHC) low oil pressure trip. This change is currently under review by the USNRC.

2) J. Dimmette letter to USNRC, SVP 99-205, dated November 16, 1999, "Request for Technical Specifications Change, Reactor Protection System Instrumentation, reactor Vessel Steam Dome Pressure - High."

   This proposed change provided a CHANNEL CHECK and Monthly Trip Unit calibration requirement for the upgrade to RPS Functional Unit 3, "Reactor Vessel Steam Dome Pressure - High." This change was submitted to allow installation of the instrument upgrade during Q2R15. This change is currently under review by the USNRC.

H. SCHEDULE REQUIREMENTS

In order to ensure consistency with our ISTS conversion effort, ComEd requests NRC approval of this request by August 15, 2000, to be effective no later than 120 days following approval. This implementation period will permit the appropriate procedural/program revisions and training necessary to implement the proposed changes.

I. REFERENCES

1) NUREG-1433, Revision 1, "Standard Technical Specifications, General Electric Plants, BWR/4," April 1995
2) Letter from C. L. Tully (BWROG) to B. Grimes (USNRC), BWROG-92102, dated November 4, 1992, "BWR Owners' Group (BWROG) Topical Reports on Technical Specification Improvement Analysis for BWR Reactor Protection Systems"
3) General Electric Licensing Topical Report NEDC-30851P-A, "Technical Specification Improvement Analyses for BWR Reactor Protection System," March 1988
4) General Electric Licensing Topical Report, NEDC-30851P-A, Supplement 2, "Technical Specification Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," March 1989
5) General Electric Licensing Topical Report, NEDC 31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," July 1990
6) General Electric Licensing Topical Report, NEDC-30936P-A, Part 1 and Part 2, "Technical Specification Improvement Methodology With Demonstration for BWR ECCS Actuation Instrumentation," December 1988
7) General Electric Licensing Topical Report, GENE-770-06-1-A, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," December 1992
8) General Electric Licensing Topical Report, GENE 770-06-2-A, "Addendum to Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," December 1992
9) General Electric Licensing Topical Report, NEDC-30851P-A, Supplement 1, "Technical Specification Improvement Analysis For BWR Control Rod Block Instrumentation," October 1988
10) Technical Specification Improvement Analysis for the Reactor Protection System for Quad Cities Station, Units 1 and 2, GE Report Nos. GE-NE E11-00105-00-01-01 and GE-NE E11-00105-00-02-01, dated December 1999
11) Technical Specification Improvement Analysis For Isolation Actuation Instrumentation For Quad Cities Nuclear Power Station, Units 1 & 2, GE Report No. GE-NE E11-00105-00-04-01, dated December 1999
12) Letter from C. Rossi (USNRC) to R. Janecek (BWROG) dated April 27, 1988, "Staff Guidance for Licensee Determination That Drift Characteristics for Instrumentation Used in RPS Channels are Bounded By NEDC-30851P Assumptions When the Functional Test Interval is Extended from Monthly to Quarterly"
13) BWROG letter to M. Wohl (USNRC) dated June 25, 1990, "Implementation Enhancements to Technical Specification Changes Given in Isolation Actuation Instrumentation Analysis"

14) W. Sullivan (GE) letter to M. Wohl (USNRC) dated March 22, 1990, "Clarification of Technical Specification Changes Given in ECCS Actuation Instrumentation Analysis"

15) Technical Specification Improvement Analysis for the Emergency Core Cooling System Actuation Instrumentation for Quad Cities Nuclear Power Station, Units 1 and 2, GE Report No. GE-NE E11-00105-00-03-01, dated December 1999.

Attachment B
MARKED-UP TECHNICAL SPECIFICATIONS PAGES

TABLE 1-1

SURVEILLANCE FREQUENCY NOTATION

	NOTATION	FREQUENCY
1. Shift	S	At least once per 12 hours
2. Day	D	At least once per 24 hours
3. Week	W	At least once per 7 days
4. Month	M	At least once per 31 days
6. ~~5.~~ Quarter	Q	At least once per 92 days
7. ~~6.~~ Semiannual	SA	At least once per 184 days
8. ~~7.~~ Annual	A	At least once per 366 days
9. ~~8.~~ Sesquiannual	E	At least once per 18 months (550 days)
10. ~~9.~~ Startup	S/U	Prior to each reactor startup
11. ~~10.~~ Not Applicable	NA	Not applicable
5. Bimonthly	BIM	At least once per 60 days

3.1 - LIMITING CONDITIONS FOR OPERATION

A. Reactor Protection System (RPS)

The reactor protection system (RPS) instrumentation CHANNEL(s) shown in Table 3.1.A-1 shall be OPERABLE.

APPLICABILITY:

As shown in Table 3.1.A-1.

ACTION:

1. ~~With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per TRIP SYSTEM requirement for one TRIP SYSTEM, place the inoperable CHANNEL(s) and/or that TRIP SYSTEM in the tripped condition[a] within 1 hour.~~

2. ~~With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per TRIP SYSTEM requirement for both TRIP SYSTEM(s), place at least one TRIP SYSTEM in the tripped condition[b] within 1 hour and take the ACTION required by Table 3.1.A-1.~~

INSERT 1

4.1 - SURVEILLANCE REQUIREMENTS

A. Reactor Protection System

1. Each reactor protection system instrumentation CHANNEL shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL MODE(s) and at the frequencies shown in Table 4.1.A-1.

2. LOGIC SYSTEM FUNCTIONAL TEST(s) of all CHANNEL(s) shall be performed at least once per 18 months.

3. The response time of each reactor trip functional unit shown in Table 3.1.A-1 shall be demonstrated at least once per 18 months. Each test shall include at least one CHANNEL per TRIP SYSTEM such that all CHANNEL(s) are tested at least once every N times 18 months where N is the total number of redundant CHANNEL(s) in a specific reactor TRIP SYSTEM.

INSERT 2

a ~~An inoperable CHANNEL need not be placed in the tripped condition when this would cause the trip function to occur. In these cases, the inoperable CHANNEL shall be restored to OPERABLE status within 2 hours or the ACTION required by Table 3.1.A-1 for that trip function shall be taken.~~

b ~~The TRIP SYSTEM need not be placed in the tripped condition if this would cause the trip function to occur. When a TRIP SYSTEM can be placed in the tripped condition without causing the trip function to occur, place the TRIP SYSTEM with the most inoperable CHANNEL(s) in the tripped condition; if both systems have the same number of inoperable CHANNEL(s), place either TRIP SYSTEM in the tripped condition.~~

Insert 1, Page 3/4.1-1

1. With one CHANNEL required by Table 3.1.A-1 inoperable for Functional Units 1 through 12 in one or more Functional Units, place the inoperable CHANNEL and/or that TRIP SYSTEM in the tripped condition[(a)] within 12 hours.

2. With two or more CHANNELS required by Table 3.1.A-1 inoperable for Functional Units 1 through 12 in one or more Functional Units:

   a. Within one hour, verify sufficient CHANNELS remain OPERABLE or tripped[(a)] to maintain trip capability in the Functional Unit, and

   b. Within 6 hours, place the inoperable CHANNEL(s) in one TRIP SYSTEM and/or that TRIP SYSTEM[(b)] in the tripped condition[(a)], and

   c. Within 12 hours, restore the inoperable CHANNELS in the other TRIP SYSTEM to an OPERABLE status or tripped[(a)].

   Otherwise, take the ACTION required by Table 3.1.A-1 for the Functional Unit.

3. With one or more CHANNEL(s) required by Table 3.1.A-1 inoperable for Functional Units 13 or 14, within one hour place the inoperable CHANNEL(s) in the tripped condition[(a)].

   Otherwise, take the ACTION required by Table 3.1.A-1 for the Functional Unit.

Insert 2, Page 3/4.1-1

a. An inoperable CHANNEL or TRIP SYSTEM need not be placed in the tripped condition where this would cause the trip function to occur. In these cases, if the inoperable CHANNEL is not restored to OPERABLE status within the required time, the ACTION required by Table 3.1.A-1 for the Functional Unit shall be taken.

b. This ACTION applies to that TRIP SYSTEM with the most inoperable CHANNELS; if both TRIP SYSTEMS have the same number of inoperable CHANNELS, the ACTION can be applied to either TRIP SYSTEM.

TABLE 3.1.A-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

INSERT 3

TABLE NOTATION

(a) ~~A CHANNEL may be placed in an inoperable status for up to 2 hours for required surveillance without placing the TRIP SYSTEM in the tripped condition provided at least one OPERABLE CHANNEL in the same TRIP SYSTEM is monitoring that parameter.~~

(b) This function may be bypassed, provided a control rod block is actuated, for reactor protection system logic reset in Refuel and Shutdown positions of the reactor mode switch.

(c) Deleted.

(d) With THERMAL POWER greater than or equal to 45% of RATED THERMAL POWER.

(e) An APRM CHANNEL is inoperable if there are fewer than 2 LPRM inputs per level or there are less than 50% of the normal complement of LPRM inputs to an APRM CHANNEL.

(f) This function is not required to be OPERABLE when the reactor pressure vessel head is unbolted or removed per Specification 3.12.A.

(g) Required to be OPERABLE only prior to and during required SHUTDOWN MARGIN demonstrations performed per Specification 3.12.B.

(h) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(i) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.10.I or 3.10.J.

Insert 3, Page 3/4.1-6

(a) When a CHANNEL is placed in an inoperable status solely for performance of required surveillances, entry into associated Limiting Conditions for Operation and required ACTIONs may be delayed for up to 6 hours provided the Functional Unit maintains RPS trip capability.

TABLE 4.1.A-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	Applicable OPERATIONAL MODES	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST (r)	CHANNEL[a] CALIBRATION
1. Intermediate Range Monitor:				
a. Neutron Flux - High	2 3, 4, 5	S[b] S	S/U[c], W[o] W[o]	E[o] E[o]
b. Inoperative	2, 3, 4, 5	NA	W[o]	NA
2. Average Power Range Monitor[f]:				
a. Setdown Neutron Flux - High	2 3, 5[m]	S[b] S	S/U[c], W[o] W[o]	SA[o] SA[o]
b. Flow Biased Neutron Flux - High	1	S, D	~~W~~→Q	W[d,e], SA
c. Fixed Neutron Flux - High	1	S	~~W~~→Q	W[d], SA
d. Inoperative	1, 2, 3, 5[m]	NA	~~W~~→Q	NA
3. Reactor Vessel Steam Dome Pressure - High	1, 2[i]	~~NA~~ S[q]	~~M~~→Q[q]	~~Q~~→E[q][h]
4. Reactor Vessel Water Level - Low	1, 2	D	~~M~~→Q	E[h]
5. Main Steam Line Isolation Valve - Closure	1	NA	~~M~~→Q	E
6. Main Steam Line Radiation - High	1, 2[i]	S	~~M~~→Q	E[p]
7. Drywell Pressure - High	1, 2[n]	NA	~~M~~→Q	Q

TABLE 4.1.A-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	Applicable OPERATIONAL MODES	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST (r)	CHANNEL[a] CALIBRATION
8. Scram Discharge Volume Water Level - High				
a. ΔP Switch, and	1, 2, 5[l,k]	NA	Q	E
b. Thermal Switch	1, 2, 5[l,k]	NA	Q	NA
9. Turbine Stop Valve - Closure	1[l]	NA	~~M~~ → Q	E
10. Turbine EHC Control Oil Pressure - Low	1[l]	NA	~~M~~ → Q	Q
11. Turbine Control Valve Fast Closure	1[l]	NA	~~M~~ → Q	E
12. Turbine Condenser Vacuum - Low	1	NA	~~M~~ → Q	Q
13. Reactor Mode Switch Shutdown Position	1, 2, 3, 4, 5	NA	E	NA
14. Manual Scram	1, 2, 3, 4, 5	NA	M	NA

TABLE 4.1.A-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATION

(a) Neutron detectors may be excluded from the CHANNEL CALIBRATION.

(b) The IRM and SRM channels shall be determined to overlap for at least (½) decades during each startup after entering OPERATIONAL MODE 2 and the IRM and APRM channels shall be determined to overlap for at least (½) decades during each controlled shutdown, if not performed within the previous 7 days.

(c) Within 24 hours prior to startup, if not performed within the previous 7 days. The weekly CHANNEL FUNCTIONAL TEST may be used to fulfill this requirement.

(d) This calibration shall consist of the adjustment of the APRM CHANNEL to conform, within 2% of RATED THERMAL POWER, to the power values calculated by a heat balance during OPERATIONAL MODE 1 when THERMAL POWER is ≥25% of RATED THERMAL POWER. This adjustment must be accomplished: a) within 2 hours if the APRM CHANNEL is indicating lower power values than the heat balance, or b) within 12 hours if the APRM CHANNEL is indicating higher power values than the heat balance. Until any required APRM adjustment has been accomplished, notification shall be posted on the reactor control panel.

Any APRM CHANNEL gain adjustment made in compliance with Specification 3.11.B shall not be included in determining the above difference. This calibration is not required when THERMAL POWER is <25% of RATED THERMAL POWER. The provisions of Specification 4.0.D are not applicable.

(e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.

(f) The LPRMs shall be calibrated at least once per 2000 effective full power hours (EFPH).

(g) Deleted.

(h) Trip units are calibrated at least once per ~~31~~ 92 days and transmitters are calibrated at the frequency identified in the table.

(i) This function is not required to be OPERABLE when the reactor pressure vessel head is unbolted or removed per Specification 3.12.A.

(j) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.10.I or 3.10.J.

(k) This function may be bypassed, provided a control rod block is actuated, for reactor protection system reset in Refuel and Shutdown positions of the reactor mode switch.

TABLE 4.1.A-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

(l) With THERMAL POWER greater than or equal to 45% of RATED THERMAL POWER.

(m) Required to be OPERABLE only prior to and during required SHUTDOWN MARGIN demonstrations performed per Specification 3.12.B.

(n) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(o) The provisions of Specification 4.0.D are not applicable to the CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION surveillances for a period of 24 hours after entering OPERATIONAL MODE 2 or 3 when shutting down from OPERATIONAL MODE 1.

(p) A current source provides an instrument channel alignment every 3 months.

(q) The CHANNEL CHECK frequency will remain NA, the CHANNEL FUNCTIONAL TEST frequency will remain M, and the CHANNEL CALIBRATION frequency will remain Q for Functional Unit 3 until instrument upgrades are completed (Design Change Package Nos. 9900090 for Unit 1 and 9900091 for Unit 2).

(r) A Functional Test of each Automatic Scram contactor will be performed on a surveillance frequency of W.

BASES

3/4.1.A REACTOR PROTECTION SYSTEM INSTRUMENTATION

The reactor protection system (RPS) automatically initiates a reactor scram to:

a. preserve the integrity of the fuel cladding,

b. preserve the integrity of the primary system, and

c. minimize the energy which must be absorbed and prevent criticality following a loss-of-coolant accident.

This specification provides the Limiting Conditions for Operation necessary to preserve the ability of the system to perform its intended function, even during periods when instrument CHANNEL(s) may be out-of-service because of maintenance. When necessary, one CHANNEL may be made inoperable for brief intervals to conduct required surveillance.

The reactor protection system is made up of two independent TRIP SYSTEM(s), each having a minimum of two CHANNEL(s) of tripping devices. Each CHANNEL has an input from at least one instrument CHANNEL which monitors a critical parameter. The outputs of the CHANNEL(s) are combined in a one-out- of-two-logic, i.e., an input signal on either one or both of the CHANNEL(s) will cause a TRIP SYSTEM trip. The outputs of the TRIP SYSTEM(s) are arranged so that a trip on both systems is required to produce a reactor scram. This system meets the intent of the proposed IEEE 279, "Standard for Nuclear Power Plant Protection Systems" issued September 13, 1966. The system has a reliability greater than that of a two-out-of-three system and somewhat less than that of a one-out-of-two system (reference APED 5179). The bases for the trip settings of the RPS are discussed in the Bases for Specification 2.2.A.

The primary reactivity control functions during refueling are the refueling interlocks and the SHUTDOWN MARGIN calculations, which together provide assurance that adequate SHUTDOWN MARGIN is available. The IRMs also provide backup protection for any significant reactivity excursions.

INSERT 4

Insert 4, Page B 3/4.1-1

Specified surveillance intervals and surveillance and maintenance allowable outage times have been determined in accordance with the NRC approved methodologies contained in:

> General Electric Licensing Topical Report NEDC-30851P-A, "Technical Specification Improvement Analyses for BWR Reactor Protection System," March 1988.

In order to maintain consistency with the reliability analysis performed in NEDC-30851P-A, the automatic scram contractors will be exercised on a weekly basis. The NEDC-30851P-A analysis concluded that extending surveillance intervals and allowed outage times for RPS instrumentation was acceptable provided the scram contactors were functionally tested on a weekly interval.

3.2 - LIMITING CONDITIONS FOR OPERATION

A. Isolation Actuation

The isolation actuation instrumentation CHANNEL(s) shown in Table 3.2.A-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column.

APPLICABILITY:

As shown in Table 3.2.A-1.

ACTION:

1. With an isolation actuation instrumentation CHANNEL trip setpoint less conservative than the value shown in the Trip Setpoint column of Table 3.2.A-1, declare the CHANNEL inoperable until the CHANNEL is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.
2. ~~With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per TRIP SYSTEM requirement for one TRIP SYSTEM, place the inoperable CHANNEL(s) and/or TRIP SYSTEM in the tripped condition[a] within one hour.~~

INSERT 5

4.2 - SURVEILLANCE REQUIREMENTS

A. Isolation Actuation

1. Each isolation actuation instrumentation CHANNEL shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL MODE(s) and at the frequencies shown in Table 4.2.A-1.
2. LOGIC SYSTEM FUNCTIONAL TEST(s) of all CHANNEL(s) shall be performed at least once per 18 months.

~~a An inoperable CHANNEL need not be placed in the tripped condition where this would cause the trip function to occur. In these cases, the inoperable CHANNEL shall be restored to OPERABLE status within 2 hours or the ACTION required by Table 3.2.A-1 for that trip function shall be taken.~~

3.2 - LIMITING CONDITIONS FOR OPERATION

4.2 - SURVEILLANCE REQUIREMENTS

~~3. With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per TRIP SYSTEM requirement for both TRIP SYSTEMS, place at least one TRIP SYSTEM[b] in the tripped condition[c] within one hour and take the ACTION required by Table 3.2.A-1.~~

INSERT 5

DELETE

~~b. If more CHANNEL(s) are inoperable in one TRIP SYSTEM than in the other, select the TRIP SYSTEM with the greater number of inoperable CHANNEL(s) to place in the tripped condition except when this would cause the trip function to occur; if both TRIP SYSTEM(s) have the same number of inoperable CHANNEL(s), place either TRIP SYSTEM in the tripped condition.~~

~~c. An inoperable CHANNEL need not be placed in the tripped condition where this would cause the trip function to occur. In these cases, the inoperable CHANNEL shall be restored to OPERABLE status within one hour or the ACTION required by Table 3.2.A-1 for that trip function shall be taken.~~

DELETE

Insert 5, Pages 3/4.2-1, 3/4.2-2

2. With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per TRIP SYSTEM requirement:

   a) Within 1 hour, verify sufficient CHANNELS remain OPERABLE or in the tripped condition to ensure automatic isolation capability.
   b) Within 12 hours, place the inoperable CHANNEL(s) and/or TRIP SYSTEM in the tripped condition[a] for Table 3.2.A-1 Functional Units common to RPS: 1a, 1b, 2a, 2b, 3a, 3b, 4b, and 7a, and
   c) Within 24 hours, place the inoperable CHANNEL(s) and/or TRIP SYSTEM in the tripped condition[a] for Table 3.2.A-1 Functional Units not common to RPS.

   OR

   Take the ACTION required by Table 3.2.A-1.

a An inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, if the inoperable channel is not restored to OPERABLE status within the required time, the ACTION required by Table 3.2.A-1 for the Functional Unit shall be taken.

TABLE 3.2.A-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TABLE NOTATION

* During CORE ALTERATIONS or operations with a potential for draining the reactor vessel.

** When handling irradiated fuel in the secondary containment.

INSERT 6

~~(a) A CHANNEL may be placed in an inoperable status for up to 2 hours for required surveillance without placing the CHANNEL in the tripped condition provided the Functional Unit maintains isolation actuation capability.~~

(b) Also trips the mechancal vacuum pump and isolates the steam jet air ejectors.

(c) Isolates the reactor building ventilation system and actuates the standby gas treatment system.

(d) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(e) Only one TRIP SYSTEM.

(f) Closes only reactor water cleanup system isolation valves.

(g) Only one trip system requied in OPERATIONAL MODE(s) 4 and 5 with RHR Shutdown Cooling System integrity maintained. System integrity is maintained provided the piping is intact and no maintenance is being performed that has the potential for draining the reactor vessel through the system.

(h) Normal background is as measured during full power operation without hydrogen being injected.

(i) Includes a time delay of $3 \leq t \leq 9$ seconds.

(j) Reactor vessel water level settings are expressed in inches above the top of active fuel (which is 360 inches above vessel zero).

(k) Also isolates the control room ventilation system.

Insert 6, Page 3/4.2-7

(a) When a CHANNEL is placed in an inoperable status solely for performance of required surveillances, entry into associated Limiting Conditions for Operation and required ACTIONs may be delayed for up to 6 hours provided the Functional Unit maintains isolation actuation capability.

TABLE 4.2.A-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	Applicable OPERATIONAL MODE(s)
4. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Standby Liquid Control System Initiation	NA	E	NA	1, 2, 3
b. Reactor Vessel Water Level - Low	S	~~M~~ → Q	E[a]	1, 2, 3
5. REACTOR CORE ISOLATION COOLING ISOLATION				
a. Steam Flow - High	NA	~~M~~ → Q	Q	1, 2, 3
b. Reactor Vessel Pressure - Low	NA	~~M~~ → Q	Q	1, 2, 3
c. Area Temperature - High	NA	E	E	1, 2, 3
6. HIGH PRESSURE COOLANT INJECTION ISOLATION				
a. Steam Flow - High	NA	~~M~~ → Q	E[a]	1, 2, 3
b. Reactor Vessel Pressure - Low	NA	~~M~~ → Q	E[a]	1, 2, 3
c. Area Temperature - High	NA	E	E	1, 2, 3
7. RHR SHUTDOWN COOLING MODE ISOLATION				
a. Reactor Vessel Water Level - Low	S	~~M~~ → Q	E[a]	3, 4, 5
b. Reactor Vessel Pressure - High (Cut-in Permissive)	NA	~~M~~ → Q	Q	1, 2, 3

TABLE 4.2.A-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

TABLE NOTATION

* During CORE ALTERATIONS or operations with a potential for draining the reactor vessel.

** When handling irradiated fuel in the secondary containment.

(a) Trip units are calibrated at least once per ~~31~~ 92 days and transmitters are calibrated at the frequency identified in the table.

(b) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(c) Isolates the reactor building ventilation system and actuates the standby gas treatment system.

(d) Also isolates the control room ventilation system.

(e) These instrument channels will be calibrated using simulated electrical signals once every three months. In addition, calibration including the sensors will be performed every 18 months.

3.2 - LIMITING CONDITIONS FOR OPERATION

B. Emergency Core Cooling Systems (ECCS) Actuation

The ECCS actuation instrumentation CHANNEL(s) shown in Table 3.2.B-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column.

APPLICABILITY:

As shown in Table 3.2.B-1.

ACTION:

1. With an ECCS actuation instrumentation CHANNEL trip setpoint less conservative than the value shown in the Trip Setpoint column of Table 3.2.B-1, declare the CHANNEL inoperable until the CHANNEL is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.
2. With one or more ECCS actuation instrumentation CHANNEL(s) inoperable, take the ACTION required by Table 3.2.B-1.
3. ~~With either ADS TRIP SYSTEM inoperable, restore the inoperable TRIP SYSTEM to OPERABLE status within:~~
   a. ~~7 days provided that both the HPCI and RCIC systems are OPERABLE, or~~
   b. ~~72 hours.~~

   ~~With the above provisions of this ACTION not met, be in at least HOT~~

DELETE

4.2 - SURVEILLANCE REQUIREMENTS

B. ECCS Actuation

1. Each ECCS actuation instrumentation CHANNEL shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL MODE(s) and at the frequencies shown in Table 4.2.B-1.
2. LOGIC SYSTEM FUNCTIONAL TEST(s) of all CHANNEL(s) shall be performed at least once per 18 months.

3.2 - LIMITING CONDITIONS FOR OPERATION

4.2 - SURVEILLANCE REQUIREMENTS

~~SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to ≤150 psig within the following 24 hours.~~

DELETE

TABLE 3.2.B-1 (Continued)

ECCS ACTUATION INSTRUMENTATION

Functional Unit	Trip Setpoint[h]	Minimum CHANNEL(s) per Trip Function[a]	Applicable OPERATIONAL MODE(s)	ACTION
3. HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM[d]				
a. Reactor Vessel Water Level - Low Low	≥84 inches	4	1, 2, 3	~~35~~ 37
b. Drywell Pressure - High[f]	≤2.5 psig	4	1, 2, 3	~~35~~ 37
c. Condensate Storage Tank Level - Low[i]	≥10,000 gal	2	1, 2, 3	35
d. Suppression Chamber Water Level - High[i]	≤14'8" above bottom of chamber	2	1, 2, 3	35
e. Reactor Vessel Water Level - High (Trip)	≤201 inches	2	1, 2, 3	31
f. HPCI Pump Discharge Flow - Low (Bypass)	≥600 gpm	1	1, 2, 3	33
g. Manual Initiation	NA	1/system	1, 2, 3	34
4. AUTOMATIC DEPRESSURIZATION SYSTEM - TRIP SYSTEM 'A' [d]				
a. Reactor Vessel Water Level - Low Low	≥84 inches	2	1, 2, 3	~~30~~ 38
b. Drywell Pressure - High[f]	≤2.5 psig	2	1, 2, 3	~~30~~ 38
c. Initiation Timer	≤120 sec	1	1, 2, 3	31
d. Low Low Level Timer	≤9.0 min	1	1, 2, 3	31
e. CS Pump Discharge Pressure - High (Permissive)	≥100 psig & ≤150 psig	2 ~~1~~/pump	1, 2, 3	31
f. LPCI Pump Discharge Pressure - High (Permissive)	≥100 psig & ≤150 psig	2 ~~1~~/pump	1, 2, 3	31

TABLE 3.2.B-1 (Continued)

ECCS ACTUATION INSTRUMENTATION

Functional Unit	Trip Setpoint[h]	Minimum CHANNEL(s) per Trip Function[a]	Applicable OPERATIONAL MODE(s)	ACTION
5. AUTOMATIC DEPRESSURIZATION SYSTEM - TRIP SYSTEM 'B' [d]				
a. Reactor Vessel Water Level - Low Low	≥84 inches	2	1, 2, 3	~~30~~ 38
b. Drywell Pressure - High[f]	<2.5 psig	2	1, 2, 3	~~30~~ 38
c. Initiation Timer	≤120 sec	1	1, 2, 3	31
d. Low Low Level Timer	≤9.0 min	1	1, 2, 3	31
e. CS Pump Discharge Pressure - High (Permissive)	≥100 psig & ≤150 psig	2/pump	1, 2, 3	31
f. LPCI Pump Discharge Pressure - High (Permissive)	≥100 psig & ≤150 psig	2/pump	1, 2, 3	31

Functional Unit	Trip Setpoint	Minimum CHANNEL(s) per Trip Function	Applicable OPERATIONAL MODE(s)	ACTION
6. LOSS OF POWER				
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	3045 ± 152 volts decreasing voltage	2/bus	1, 2, 3, 4[e], 5[e]	36
b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	≥3845 volts (Unit 1)[g][j] ≥3845 volts (Unit 2)[g][j]	2/bus	1, 2, 3, 4[e], 5[e]	36

TABLE 4.2.A-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	Applicable OPERATIONAL MODE(s)
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low	S	~~M~~ → Q	E[a]	1, 2, 3
b. Drywell Pressure - High[b]	NA	~~M~~ → Q	Q	1, 2, 3
c. Drywell Radiation - High	S	~~M~~ → Q	E	1, 2, 3
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low[c,d]	S	~~M~~ → Q	E[a]	1, 2, 3 & *
b. Drywell Pressure - High[b,c,d]	NA	~~M~~ → Q	Q	1, 2, 3
c. Reactor Building Ventilation Exhaust Radiation - High[c,d]	S	~~M~~ → Q	Q	1, 2, 3 & **
d. Refueling Floor Radiation - High[c,d]	S	~~M~~ → Q	Q	1, 2, 3 & **
3. MAIN STEAM LINE (MSL) ISOLATION				
a. Reactor Vessel Water Level - Low Low	S	~~M~~ → Q	E[a]	1, 2, 3
b. MSL Tunnel Radiation - High	S	~~M~~ → Q	E[a]	1, 2, 3
c. MSL Pressure - Low	NA	~~M~~ → Q	Q	1
d. MSL Flow - High[d]	S	~~M~~ → Q	E	1, 2, 3
e. MSL Tunnel Temperature - High	NA	E	E	1, 2, 3

TABLE 3.2.B-1 (Continued)

ECCS ACTUATION INSTRUMENTATION

ACTION

INSERT 8

ACTION 30 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per Trip Function requirement:

a. With one CHANNEL inoperable, place the inoperable CHANNEL in the tripped condition within one hour or declare the associated ECCS system(s) inoperable.

b. With more than one CHANNEL inoperable, declare the associated ECCS system(s) inoperable.

ACTION 31 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per Trip Function requirement:

a. For ADS, declare the associated ADS TRIP SYSTEM inoperable.

b. For CS, LPCI or HPCI, declare the associated ECCS system(s) inoperable.

INSERT 9

ACTION 32 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per Trip Function requirement, place the inoperable CHANNEL in the tripped condition within ~~one hour~~ 24 hours.

INSERT 10

ACTION 33 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per Trip Function requirement, place the inoperable CHANNEL in the tripped condition within one hour; restore the inoperable CHANNEL to OPERABLE status within 7 days or declare the associated ECCS system(s) inoperable.

ACTION 34 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per Trip Function requirement, restore the inoperable CHANNEL to OPERABLE status within ~~8 hours~~ 24 hours or declare the associated ECCS system(s) inoperable.

INSERT 11

ACTION 35 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per Trip Function requirement, place at least one inoperable CHANNEL in the tripped condition within one hour or declare the HPCI system inoperable.

ACTION 36 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per Trip Function requirement, place the inoperable CHANNEL in the tripped condition within one hour, or declare the associated emergency diesel generator inoperable and take the ACTION required by Specification 3.9.A or 3.9.B, as appropriate.

ACTION 37 (INSERT 12)
ACTION 38 (INSERT 13)

Insert 8, Page 3/4.2-16

ACTION 30 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per Trip Function requirement (Action 30a only applies in OPERATIONAL MODES 1, 2 and 3):

a. Within one hour from discovery of loss of initiation capability declare the associated ECCS systems inoperable, AND

b. Place the inoperable CHANNEL(s) in the tripped condition within 24 hours or declare the associated ECCS system inoperable.

Insert 9, Page 3/4.2-16

ACTION 31 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per Trip Function requirement:

For CS, LPCI and HPCI (For Functional Units 1.c and 2.c, Action 31a applies only in OPERATIONAL MODES 1, 2, and 3):

a. Within one hour from discovery of loss of initiation capability declare the associated ECCS systems inoperable, AND
b. Restore the inoperable CHANNEL(s) to OPERABLE status within 24 hours or declare the associated ECCS system(s) inoperable.

For ADS:

a. Within one hour from discovery of loss of initiation capability in both ADS trip systems, declare the ADS relief valves inoperable, AND
b. With RCIC or HPCI inoperable, restore the inoperable CHANNEL(s) to OPERABLE status within 96 hours or declare the ADS relief valves inoperable, AND
c. With RCIC and HPCI OPERABLE, restore the inoperable CHANNEL(s) to OPERABLE status within 8 days or declare the ADS relief valves inoperable.

Insert 10, Page 3/4.2-16

ACTION 33 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per Trip Function requirement (For Functional Units 1.d and 2.d, Action 33a only applies in OPERATIONAL MODES 1, 2 and 3):

a. Within one hour from discovery of loss of initiation capability declare the associated ECCS system(s) inoperable, AND
b. Restore the CHANNEL(s) to OPERABLE status within 7 days or declare the associated ECCS system(s) inoperable.

Insert 11, Page 3/4.2-16

ACTION 35 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per Trip Function requirement:

a. Within one hour from discovery of loss of initiation capability, declare HPCI inoperable, AND

b. Place the inoperable CHANNEL(s) in the tripped condition within 24 hours or declare the HPCI system inoperable.

Insert 12, Page 3/4.2-16

ACTION 37 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per Trip Function requirement:

a. Within one hour from discovery of loss of initiation capability declare HPCI inoperable, AND

b. Place the inoperable CHANNEL(s) in the tripped condition within 24 hours or declare HPCI inoperable.

INSERT 13, Page 3/4.2-16

ACTION 38 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per Trip Function requirement:

a. Within one hour from discovery of loss of initiation capability in both ADS trip systems, declare the ADS relief valves inoperable, AND

b. With RCIC or HPCI inoperable, place the inoperable CHANNEL(s) in the tripped condition within 96 hours or declare the ADS relief valves inoperable, AND

c. Place the inoperable CHANNEL(s) in the tripped condition within 8 days or, declare the ADS relief valves inoperable.

TABLE 3.2.B-1 (Continued)

ECCS ACTUATION INSTRUMENTATION

TABLE NOTATION

INSERT 14

~~(a) A CHANNEL may be placed in an inoperable status for up to 2 hours for required surveillance without placing the CHANNEL in the tripped condition provided the associated Functional Unit maintains ECCS initiation capability.~~

(b) Also actuates the associated emergency diesel generator.

(c) When the system is required to be OPERABLE per Specification 3.5.B.

(d) Not required to be OPERABLE when reactor steam dome pressure is ≤150 psig.

(e) Required when the associated diesel generator is required to be OPERABLE per Specification 3.9.B.

(f) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(g) With no LOCA signal present, there is an additional time delay of 5 ± 0.25 minutes.

(h) Reactor water level settings are expressed in inches above the top of active fuel (which is 360 inches above vessel zero).

(i) Provides signal to pump suction valves only.

(j) There is an inherent time delay of 7 ± 1.4 seconds on degraded voltage.

Insert 14, Page 3/4.2-17

(a) When a CHANNEL is placed in an inoperable status solely for performance of required surveillances, entry into associated Limiting Conditions for Operation and required ACTIONs may be delayed as follows:

1) For up to six hours for Functional Units 3.e, 3.f, and 3.g; and
2) For up to six hours for Functional Units other than 3.e, 3.f, and 3.g provided the functional unit maintains actuation capability.

TABLE 4.2.B-1

ECCS ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	Applicable OPERATIONAL MODE(s)
1. CORE SPRAY (CS) SYSTEM				
a. Reactor Vessel Water Level - Low Low	S	M	~~E~~ → BIM	1, 2, 3, 4[b], 5[b]
b. Drywell Pressure - High[d]	NA	~~M~~ → Q	Q	1, 2, 3
c. Reactor Vessel Pressure - Low (Permissive)	NA	~~M~~ → Q	Q	1, 2, 3, 4[b], 5[b]
d. CS Pump Discharge Flow - Low (Bypass)	NA	~~M~~ → Q	E[e]	1, 2, 3, 4[b], 5[b]
2. LOW PRESSURE COOLANT INJECTION (LPCI) SUBSYSTEM				
a. Reactor Vessel Water Level - Low Low	S	M	~~E~~ → BIM	1, 2, 3, 4[b], 5[b]
b. Drywell Pressure - High[d]	NA	~~M~~ → Q	Q	1, 2, 3
c. Reactor Vessel Pressure - Low (Permissive)	NA	~~M~~ → Q	Q	1, 2, 3, 4[b], 5[b]
d. LPCI Pump Discharge Flow - Low (Bypass)	NA	~~M~~ → Q	E[e]	1, 2, 3, 4[b], 5[b]
3. HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM[a]				
a. Reactor Vessel Water Level - Low Low	S	M	~~E~~ → BIM	1, 2, 3
b. Drywell Pressure - High[d]	NA	~~M~~ → Q	Q	1, 2, 3
c. Condensate Storage Tank Level - Low	NA	~~M~~ → Q	NA	1, 2, 3
d. Suppression Chamber Water Level - High	NA	~~M~~ → Q	NA	1, 2, 3
e. Reactor Vessel Water Level - High (Trip)	NA	M	~~E~~ → BIM	1, 2, 3
f. HPCI Pump Discharge Flow - Low (Bypass)	NA	~~M~~ → Q	E	1, 2, 3
g. Manual Initiation	NA	E	NA	1, 2, 3

TABLE _.B-1 (Continued)

ECCS ACTUATION INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	Applicable OPERATIONAL MODE(s)
4. AUTOMATIC DEPRESSURIZATION SYSTEM[a]				
a. Reactor Vessel Water Level - Low Low	S	M	~~Q~~ → BIM	1, 2, 3
b. Drywell Pressure - High[d]	NA	~~M~~ → Q	Q	1, 2, 3
c. Initiation Timer	NA	E	E	1, 2, 3
d. Low Low Level Timer	NA	E	E	1, 2, 3
e. CS Pump Discharge Pressure - High (Permissive)	NA	~~M~~ → Q	Q	1, 2, 3
f. LPCI Pump Discharge Pressure - High (Permissive)	NA	~~M~~ → Q	Q	1, 2, 3
5. LOSS OF POWER				
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	NA	E	E	1, 2, 3, 4[c], 5[c]
b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	NA	E	E	1, 2, 3, 4[c], 5[c]

TABLE 4.2.B-1 (Continued)

ECCS ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATION

(a) Not required to be OPERABLE when reactor steam dome pressure is ≤150 psig.

(b) When the system is required to be OPERABLE per Specification 3.5.B.

(c) Required when the associated diesel generator is required to be OPERABLE per Specification 3.9.B.

(d) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(e) Trip units are calibrated at least once per ~~31~~ 92 days and transmitters are calibrated at the frequency identified in the table.

TABLE 3.2.C-1

ATWS - RPT INSTRUMENTATION

Functional Unit	Trip Setpoint[c]	Minimum OPERABLE CHANNEL(s) per TRIP SYSTEM[a]
1. Reactor Vessel Water Level - Low Low	≥84 inches[b]	2
2. Reactor Vessel Pressure - High	≤1250 psig	2

INSERT 16

a ~~A CHANNEL may be placed in an inoperable status for up to 2 hours for required surveillance without placing the TRIP SYSTEM in the tripped condition provided at least one OPERABLE CHANNEL in the same TRIP SYSTEM is monitoring that parameter.~~

b Includes a time delay of $8 \leq t \leq 10$ seconds.

c Reactor vessel water level settings are expressed in inches above the top of active fuel (which is 360 inches above vessel zero).

Insert 16, Page 3/4.2-23

a When a CHANNEL is placed in an inoperable status solely for performance of required surveillances, entry into associated Limiting Conditions for Operation and required ACTIONs may be delayed for up to 6 hours provided the Functional Unit maintains ATWS actuation capability

TABLE 4.2.C-1

ATWS - RPT INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Reactor Water Level - Low Low	S	~~M~~ → Q	E[a]
2. Reactor Vessel Pressure - High	S	~~M~~ → Q	E[a]

a Trip units are calibrated at least once per 31 days and transmitters are calibrated at the frequency identified in the table.

TABLE 3.2.D-1

REACTOR CORE ISOLATION COOLING ACTUATION INSTRUMENTATION

Functional Unit	Trip Setpoint[c]	Minimum CHANNEL(s) per Trip Function[a]	ACTION
1. Reactor Vessel Water Level - Low Low	≥84 inches	4	40
2. Reactor Vessel Level - High (Trip)	≤201 inches	2	41
3. Condensate Storage Tank Level - Low	≥598' El.	2[b]	42
4. Suppression Chamber Water Level - High	≤14'8" above bottom of chamber	2[b]	42
5. Manual Initiation	NA	1/system	43

INSERT 18

~~a A CHANNEL may be placed in an inoperable status for up to 2 hours for required surveillance without placing the CHANNEL in the tripped condition provided the Functional Unit maintains RCIC actuation capability.~~

b Provides signal to pump suction valves only.

c Reactor vessel water level settings are expressed in inches above the top of active fuel (which is 360 inches above vessel zero).

Insert 18, Page 3/4.2-26

a When a CHANNEL is placed in an inoperable status solely for performance of required surveillances, entry into associated Limiting Conditions for Operation and required ACTIONs may be delayed as follows:

1) For up to six hours for Functional Units 2 and 5; and
2) For up to six hours for Functional Units other than 2 and 5 provided the functional unit maintains actuation capability.

TABLE 3.2.D-1 (Continued)

REACTOR CORE ISOLATION COOLING ACTUATION INSTRUMENTATION

ACTION

INSERT 19

~~ACTION 40 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per TRIP SYSTEM requirement:~~

~~a. With one CHANNEL inoperable, place the inoperable CHANNEL in the tripped condition within one hour or declare the RCIC system inoperable.~~

~~b. With more than one CHANNEL inoperable, declare the RCIC system inoperable.~~

ACTION 41 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per TRIP SYSTEM requirement, declare the RCIC system inoperable within 24 hours.

~~ACTION 42 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per TRIP SYSTEM requirement, place at least one inoperable CHANNEL in the tripped condition within one hour or declare the RCIC system inoperable.~~

INSERT 20

ACTION 43 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum OPERABLE CHANNEL(s) per TRIP SYSTEM requirement, restore the inoperable CHANNEL to OPERABLE status within ~~8 hours~~ 24 hours or declare the RCIC system inoperable.

Insert 19, Page 3/4.2-27

ACTION 40- With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per Trip Function requirement:

a. Within one hour from discovery of loss of initiation capability declare RCIC inoperable, AND

b. Place the inoperable CHANNEL(s) in the tripped condition within 24 hours or declare RCIC inoperable.

Insert 20, Page 3/4.2-27

ACTION 42 – With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per Trip Function requirement:

a. Within one hour from discovery of loss of initiation capability declare RCIC inoperable, AND

b. Place the inoperable CHANNEL(s) in the tripped condition within 24 hours or declare RCIC inoperable.

TABLE 4.2.D-1

REACTOR CORE ISOLATION COOLING ACTUATION INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Reactor Vessel Water Level - Low Low	S	M	~~E~~ → BIM
2. Reactor Vessel Water Level - High (Trip)	S	M	~~E~~ → BIM
3. Condensate Storage Tank Level - Low	NA	~~M~~ → Q	NA
4. Suppression Chamber Water Level - High	NA	~~M~~ → Q	NA
5. Manual Initiation	NA	E	NA

TABLE 3.2.E-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION

ACTION

ACTION 50 - Declare the rod block monitor inoperable and take the ACTION required by Specification 3.3.M.

ACTION 51- With the number of OPERABLE CHANNEL(s):

a. One less than required by the Minimum CHANNEL(s) per Trip Function requirement, restore the inoperable CHANNEL to OPERABLE status within 7 days or place the inoperable CHANNEL in the tripped condition within the next hour.

b. Two or more less than required by the Minimum CHANNEL(s) per Trip Function requirement, place at least one inoperable CHANNEL in the tripped condition within one hour.

ACTION 52 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per Trip Function requirement, place the inoperable CHANNEL in the tripped condition within ~~one hour~~ 12 hours.

TABLE 3.2.E-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION

TABLE NOTATION

(a) The RBM shall be automatically bypassed when a peripheral control rod is selected.

(b) This function shall be automatically bypassed if the IRM channels are on range 3 or higher.

(c) This function shall be automatically bypassed when the associated IRM channels are on range 8 or higher.

(d) This function shall be automatically bypassed when the IRM channels are on range 1.

(e) With THERMAL POWER ≥30% of RATED THERMAL POWER.

(f) With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.10.I or 3.10.J.

(g) The Average Power Range Monitor rod block function is varied as a function of recirculation loop flow (W). The trip setting of this function must be maintained in accordance with Specification 3.11.B. W is equal to the percentage of the drive flow required to produce a rated core flow of 98 x 10^6 lbs/hr.

(h) Required to be OPERABLE only during SHUTDOWN MARGIN demonstrations performed per Specification 3.12.B.

INSERT 22

~~(i) A CHANNEL may be placed in an inoperable status for up to 2 hours for required surveillance without placing the CHANNEL in the tripped condition provided the Functional Unit maintains control rod block capability.~~

(j) With detector count rate less than or equal to 100 cps.

Insert 22, Page 3/4.2-34

(i) When a CHANNEL is placed in an inoperable status solely for performance of required surveillances, entry into associated Limiting Conditions for Operation and required ACTIONs may be delayed for up to 6 hours provided the Functional Unit maintains rod block actuation capability.

TABLE 4.2.E-1

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	Applicable OPERATIONAL MODE(s)
1. ROD BLOCK MONITORS				
a. Upscale	NA	S/U[b,c], ~~M[c]~~ → Q[c]	Q	1[d]
b. Inoperative	NA	S/U[b,c], ~~M[c]~~ → Q[c]	NA	1[d]
c. Downscale	NA	S/U[b,c], ~~M[c]~~ → Q[c]	Q	1[d]
2. AVERAGE POWER RANGE MONITORS				
a. Flow Biased Neutron Flux - High				
1. Dual Recirculation Loop Operation	NA	S/U[b], ~~M~~ → Q	SA	1
2. Single Recirculation Loop Operation	NA	S/U[b], ~~M~~ → Q	SA	1
b. Inoperative	NA	S/U[b], ~~M~~ → Q	NA	1, 2, 5[e]
c. Downscale	NA	S/U[b], ~~M~~ → Q	SA	1
d. Startup Neutron Flux - High	NA	S/U[b], ~~M~~ → Q	SA	2, 5[e]
3. SOURCE RANGE MONITORS				
a. Detector not full in[f]	NA	S/U[b], W	E	2[i][h], 5[h]
b. Upscale[g]	NA	S/U[b], W	E	2[i], 5
c. Inoperative[g]	NA	S/U[b], W	NA	2[i], 5

TABLE 3.2.I-1

SUPPRESSION CHAMBER AND DRYWELL SPRAY ACTUATION INSTRUMENTATION

Functional Unit	Trip Setpoint[a]	Minimum CHANNEL(s) per TRIP SYSTEM[c]	ACTION
1. Drywell Pressure - High (Permissive)	$0.5 \leq \rho \leq 1.5$ psig	2	80
2. Reactor Vessel Water Level - Low (Permissive)	$\geq$ -48 inches	1	80

ACTION

ACTION 80 -

a. With the number of OPERABLE CHANNEL(s) less than required by the Minimum OPERABLE CHANNEL(s) per TRIP SYSTEM requirement for one TRIP SYSTEM, place at least one inoperable CHANNEL in the tripped condition[b] within ~~one hour~~ 24 hours or declare the Suppression Chamber and Drywell Spray Actuation mode of the Residual Heat Removal system inoperable.

b. With the number of OPERABLE CHANNEL(s) less than required by the Minimum OPERABLE CHANNEL(s) per TRIP SYSTEM requirement for both TRIP SYSTEM(s), declare the Suppression Chamber and Drywell Spray Actuation mode of the Residual Heat Removal system inoperable.

---

a Reactor vessel water level settings are expressed in inches above the top of active fuel (which is 360 inches above vessel zero).

b If an instrument is inoperable, it shall be placed (or simulated) in a tripped condition so that it will not prevent a containment spray.

~~c A CHANNEL may be placed in an inoperable status for up to 2 hours for required surveillance without placing the CHANNEL in the tripped condition provided the Functional Unit maintains Suppression Chamber and Drywell Spray Actuation capability.~~

INSERT 24

Insert 24, Page 3/4.2-49

c When a CHANNEL is placed in an inoperable status solely for performance of required surveillances, entry into associated Limiting Conditions for Operation and required ACTIONs may be delayed for up to 6 hours provided the Functional Unit maintains Suppression Chamber and Drywell Spray actuation capability.

TABLE 4.2.I-1

SUPPRESSION CHAMBER AND DRYWELL SPRAY ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Drywell Pressure - (Permissive)	NA	~~M~~ → Q	Q
2. Reactor Vessel Water Level - Low (Permissive)	D	~~M~~ → Q	E[a]

a Trip units are calibrated at least once per ~~31~~ 92 days and transmitters are calibrated at the frequency indicated in the table.

BASES

3/4.2 INSTRUMENTATION

In addition to reactor protection instrumentation which initiates a reactor scram (Sections 2.2 and 3/4.1), protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operator's ability to control, or which terminates operator errors before they result in serious consequences. The objectives of these specifications are to assure the effectiveness of the protective instrumentation when required and to prescribe the trip settings required to assure adequate performance. As indicated, one CHANNEL may be required to be made inoperable for brief intervals to conduct required surveillance. Some of the settings have tolerances explicitly stated where the high and low values are both critical and may have a substantial effect on safety. It should be noted that the setpoints of other instrumentation, where only the high or low end of the setting has a direct bearing on safety, are chosen at a level away from the normal operating range to prevent inadvertent actuation of the safety system involved and exposure to abnormal situations. Surveillance requirements for the instrumentation are selected in order to demonstrate proper function and OPERABILITY. Additional instrumentation for REFUELING operations is identified in Sections 3/4.10.B.

Current fuel designs incorporate slight variations in the length of the active fuel and, thus, the actual top of active fuel, when compared with the original fuel designs. Safety Limits, instrument water level setpoints, and associated LCOs refer to the top of active fuel. In these cases, the top of active fuel is defined as 360 inches above vessel zero. Licensing analyses, both accident and transient, utilize this definition for the automatic initiation and manual intervention associated with these events.

3/4.2.A Isolation Actuation Instrumentation

The isolation actuation instrumentation automatically initiates closure of appropriate isolation valves and/or dampers, which are necessary to prevent or limit the release of fission products from the reactor coolant system, the primary containment and the secondary containment in the event of a loss-of-coolant accident or other reactor coolant pressure boundary (RCPB) leak. The parameters which result in isolation of the secondary containment also actuate the standby gas treatment system. The isolation instrumentation includes the sensors, relays, and switches that are necessary to cause initiation of primary and secondary containment and RCPB system isolation. Functional diversity is provided by monitoring a wide range of dependent and independent parameters. Redundant sensor input signals for each parameter are provided for initiation of isolation (one exception is standby liquid control system initiation).

The reactor low level instrumentation is set to trip at greater than or equal to 144 inches above the top of active fuel (which is defined to be 360 inches above vessel zero). This trip initiates closure of Group 2 and 3 primary containment isolation valves but does not trip the recirculation pumps. For this trip setting and a 60-second valve closure time, the valves will be closed before perforation of the cladding occurs, even for the maximum break.

INSERT 7

Insert 7, Page B 3/4.2-1

Specified surveillance intervals and surveillance and maintenance allowable outage times have been determined in accordance with the NRC approved methodologies contained in:

> General Electric Licensing Topical Report, NEDC-30851P-A, Supplement 2, "Technical Specification Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," March 1989.
>
> General Electric Licensing Topical Report, NEDC 31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," July 1990.

BASES

3/4.2.B Emergency Core Cooling System Actuation Instrumentation

The emergency core cooling system (ECCS) instrumentation generates signals to automatically actuate those safety systems which provide adequate core cooling in the event of a design basis transient or accident. The instrumentation which actuates the ECCS is generally arranged in a one-out-of-two taken twice logic circuit. The logic circuit is composed of four CHANNEL(s) and each CHANNEL contains the logic from the functional unit sensor up to and including all relays which actuate upon a signal from that sensor. For core spray and low pressure coolant injection, the divisionally powered actuation logic is duplicated and the redundant components are powered from the other division's power supply. The single-failure criterion is met through provisions for redundant core cooling functions, e.g., sprays and automatic blowdown and high pressure coolant injection. Although the instruments are listed by system, in some cases the same instrument is used to send the actuation signal to more than one system at the same time.

For effective emergency core cooling during small pipe breaks, the high pressure coolant injection (HPCI) system must function since reactor pressure does not decrease rapidly enough to allow either core spray or the low pressure coolant injection (LPCI) system to operate in time. The automatic pressure relief function is provided as a backup to HPCI, in the event HPCI does not operate. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the specification are adequate to assure the above criteria are met. The specification preserves the effectiveness of the system during periods of maintenance, testing or calibration and also minimizes the risk of inadvertent operation, i.e., only one instrument CHANNEL out-of-service.

INSERT 15

3/4.2.C ATWS - RPT Instrumentation

The anticipated transient without scram (ATWS) recirculation pump trip (RPT) provides a means of limiting the consequences of the unlikely occurrence of a failure to scram concurrent with the associated anticipated transient. The response of this plant to this postulated event falls within the bounds of study events in General Electric Company Topical Report NEDO-10349, dated March 1971 and NEDO24222, dated December 1979. Tripping the recirculation pumps adds negative reactivity by increasing steam voiding in the core area as core flow decreases.

INSERT 17

3/4.2.D Reactor Core Isolation Cooling Actuation Instrumentation

The reactor core isolation cooling system provides makeup water to the core in the event of a postulated isolation of the reactor from the main condenser with a loss of feedwater. The system automatically initiates upon receipt of a reactor vessel low-low water level signal utilizing level indicating switches in a one-out-of-two taken twice logic scheme. The system may also be manually started.

INSERT 21

Insert 15, Page B 3/4.2-2

Specified surveillance intervals and surveillance and maintenance allowable outage times have been determined in accordance with the NRC approved methodologies contained in:

> General Electric Licensing Topical Report, NEDC-30936P-A, Part 1 and Part 2, "Technical Specification Improvement Methodology With Demonstration for BWR ECCS Actuation Instrumentation," December 1988.

Insert 17, Page B 3/4.2-2

Specified surveillance intervals and surveillance and maintenance allowable outage times have been determined in accordance with the NRC approved methodologies contained in:

> General Electric Licensing Topical Report, GENE-770-06-1-A, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," December 1992.

Insert 21, Page B 3/4.2-2

Specified surveillance intervals and surveillance and maintenance allowable outage times have been determined in accordance with the NRC approved methodologies contained in:

> General Electric Licensing Topical Report, GENE 770-06-2-A, "Addendum to Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," December 1992.

BASES

discharge volume, high water level rod block provides annunciation for operator action. The alarm setpoint has been selected to provide adequate time to allow for the determination of the cause for the level increase and corrective action prior to automatic scram initiation. INSERT 23

3/4.2.F Accident Monitoring Instrumentation

Instrumentation is provided to monitor sufficient accident conditions to adequately assess important variables and provide operators with necessary information to complete the appropriate mitigation actions. OPERABILITY of the instrumentation listed provides adequate monitoring of the containment following a loss-of-coolant accident. Information from this instrumentation will provide the operator with a detailed knowledge of the conditions resulting from the accident; based on this information, the operator can make logical decisions regarding post accident recovery. Allowable outage times are based on diverse instrumentation availability for guiding the operator should an accident occur, and on the low probability of an instrument being out-of-service concurrent with an accident. As noted in the surveillance requirements, the instrumentation CHANNEL(s) associated with Torus Pressure provides a dual function and is shared in common with the Drywell Pressure (Narrow and Wide Ranges) instrumentation. This instrumentation is identified in response to Generic Letter 82-33 and the associated NRC Safety Evaluation Report, and some instrumentation is included in accordance with the response to Generic Letter 83-36.

3/4.2.G Source Range Monitoring Instrumentation

The source range monitors (SRM) provide the operator with the status of the neutron flux in the core at very low power levels during startup and shutdown. The consequences of reactivity accidents are functions of the initial neutron flux. Therefore, the requirements for a minimum count rate assures that any transient, should it occur, begins at or above the initial value used in the analyses of transients from cold conditions. Two OPERABLE SRM CHANNEL(s) are adequate to monitor the approach to criticality using homogeneous patterns of scattered control rod withdrawal. Three OPERABLE SRMs provide an added conservatism. When the intermediate range monitors are on scale, adequate information is available without the SRMs and they can be retracted.

3/4.2.H Explosive Gas Monitoring Instrumentation

Instrumentation is provided to monitor the concentrations of potentially explosive mixtures in the off-gas holdup system to prevent a possible uncontrolled release via this pathway. This instrumentation is included in accordance with Generic Letter 89-01.

Insert 23, Page B 3/4.2-4

Specified surveillance intervals and surveillance and maintenance allowable outage times have been determined in accordance with the NRC approved methodologies contained in:

> General Electric Licensing Topical Report, GENE-770-06-1-A, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," December 1992.
>
> General Electric Licensing Topical Report, NEDC-30851P-A, Supplement 1, "Technical Specification Improvement Analysis For BWR Control Rod Block Instrumentation," October 1988.

BASES

3/4.2.I Suppression Chamber and Drywell Spray Actuation Instrumentation

Instrumentation is provided to monitor the parameters which are necessary to permit initiation of the containment cooling mode of the residual heat removal system to condense steam in the containment atmosphere. The spray mode does not significantly affect the rise of drywell pressure following a loss of coolant accident, but does result in quicker depressurization following completion of the blowdown.

INSERT 25

3/4.2.J Feedwater Trip System Actuation

The feedwater trip system actuation instrumentation is designed to detect a potential failure of the feedwater control system which causes excessive feedwater flow. If undetected, this would lead to reactor vessel water carryover into the main steam lines and to the main turbine. This instrumentation is included in response to Generic Letter 89-19.

3/4.2.K Toxic Gas Monitoring

Toxic gas monitoring instrumentation is provided in or near the control room ventilation system intakes to allow prompt detection and the necessary protective actions to be initiated. Isolation from high toxic chemical concentration has been added to the station design as a result of the "Control Room Habitability Study" submitted to the NRC in December 1981 in response to NUREG-0737 Item III D.3.4. As explained in Section 3 of this study, ammonia, chlorine, and sulphur dioxide detection capability has been provided. In a report generated by Sargent and Lundy in April 1991, justification was provided to delete the chlorine and sulphur dioxide detectors from the plant. The setpoints chosen for the control room ventilation isolation are based on early detection in the outside air supply at the odor threshold, so that the toxic chemical will not achieve toxicity limit concentrations in the Control Room.

Insert 25, Page B 3/4.2-5

Specified surveillance intervals and surveillance and maintenance allowable outage times have been determined in accordance with the NRC approved methodologies contained in:

> General Electric Licensing Topical Report, GENE-770-06-1-A, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," December 1992.

Attachment C
INFORMATION SUPPORTING A FINDING OF NO SIGNIFICANT HAZARDS

Commonwealth Edison (ComEd) Company has evaluated the proposed Technical Specifications (TS) changes for Quad Cities Nuclear Power Station, Units 1 and 2, and has determined that it involves no significant hazards consideration. According to 10 CFR 50.92(c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

Involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated;

Create the possibility of a new or different kind of accident from any previously analyzed; or

Involve a significant reduction in a margin of safety.

In accordance with 10 CFR 50.90, ComEd proposes to revise Appendix A, Technical Specifications (TS), of Facility Operating License Nos. DPR-29 and DPR-30. The proposed changes increase surveillance test intervals (STIs) and allowable out-of-service times (AOTs) for selected TS actuation instrumentation. The proposed changes implement recommendations resulting from generic evaluations performed by General Electric and the Boiling Water Reactor Owners' Group (i.e., AOT/STI licensing topical reports) and subsequently approved by the NRC.

These topical reports assessed the reliability of TS actuation instrumentation and concluded that extending STIs and AOTs for test and repair activities is desirable because: 1) the potential for inadvertent plant scrams is reduced, 2) the number of test cycles on equipment is minimized and 3) the use of plant personnel can be better optimized. The proposed changes are consistent with STIs and AOTs found in the Improved Standard Technical Specification, ISTS (i.e., NUREG-1433, Revision 1, "Standard Technical Specifications, General Electric Plants, BWR/4").

The determination that the criteria set forth in 10 CFR 50.92 are met for this amendment request is provided below:

Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed TS changes increases the Allowable Outage Times and Surveillance Test Intervals (AOT/STI) for actuation instrumentation based on analyses developed and approved by the Nuclear Regulatory Commission (NRC). TS requirements that govern operability or routine testing of plant instruments are not assumed to be initiators of any analyzed event because these instruments are intended to prevent, detect, or mitigate accidents. Therefore, these changes will not involve an increase in the probability of occurrence of an accident previously evaluated. Additionally, these changes will not increase the consequences of an accident previously evaluated because the proposed changes do not involve any physical changes to plant systems, structures or components (SSCs), or the manner in which these SSCs are operated. These changes will not alter the operation of equipment assumed to be available for the mitigation of accidents or transients by the plant safety analysis or licensing basis. As justified and approved in the AOT/STI licensing topical reports, the proposed changes establish or maintain adequate assurance that components are operable when necessary for the prevention or mitigation of accidents or transients and that plant variables are maintained within limits necessary to satisfy the assumptions for initial conditions in the safety analyses. The proposed changes establish or modify time limits allowable for operation with inoperable instrument channels based on analyses which have been approved by the NRC. Furthermore, there will be no change in the types or significant increase in the amounts of any effluents released offsite. For these reasons, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed changes do not involve any physical changes to SSCs, or the manner in which these SSCs function. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated. The changes in methods governing normal plant operation are consistent with the current safety analysis assumptions. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Does the change involve a significant reduction in a margin of safety?

The proposed changes increase the STIs and AOTs for actuation instrumentation based on generic analyses completed by the Boiling Water Reactor Owners' Group (BWROG). The NRC has reviewed and approved the generic studies and has concurred with the BWROG that the proposed changes do not significantly affect the probability of failure or availability of the affected instrumentation systems. The analysis determined that there is no significant change in the availability and/or reliability of instrumentation as a result of the proposed changes in STIs and AOTs. Furthermore, the change to increase the frequency of the reactor protection system scram contactor testing has been shown to improve plant safety. ComEd has determined these studies are applicable to Quad Cities Nuclear Power Station, Units 1 and 2. The proposed changes to AOTs provide realistic times to complete required testing and maintenance actions without increasing the overall instrument failure frequency. Likewise, the extended STIs do not result in significant changes in the probability of instrument failure. Furthermore, the proposed changes will reduce the probability of test-induced plant transients and equipment failures. Therefore, it is concluded that the proposed changes will not result in a reduction in the margin of safety.

Therefore, based upon the above evaluation, ComEd has concluded that these changes involve no significant hazards considerations.

Attachment D
INFORMATION SUPPORTING AN ENVIRONMENTAL ASSESSMENT

ComEd has evaluated this proposed operating license amendment request against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. ComEd has determined that this proposed license amendment request meets the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9) and as such, has determined that no irreversible consequences exist in accordance with 10 CFR 50.92(b). This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR 50 that changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or that changes an inspection or a surveillance requirement, and the amendment meets the following specific criteria:

(i) the amendment involves no significant hazards consideration.

As demonstrated in Attachment C, this proposed amendment does not involve any significant hazards consideration.

(ii) there is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite.

As documented in Attachment C, there will be no change in the types or significant increase in the amounts of any effluents released offsite.

(iii) there is no significant increase in individual or cumulative occupational radiation exposure.

The proposed changes will not result in changes in the operation or configuration of the facility. There will be no change in the level of controls or methodology used for processing of radioactive effluents or handling of solid radioactive waste, nor will the proposal result in any change in the normal radiation levels within the plant. Therefore, there will be no significant increase in individual or cumulative occupational radiation exposure resulting from this change.

Attachment E
GENERAL ELECTRIC SITE SPECIFIC EVALUATIONS

NOTE

ATTACHMENT E CONTAINS INFORMATION CONSIDERED BY GE TO BE PROPRIETARY AND AN AFFIDAVIT TO THAT EFFECT HAS BEEN INCLUDED WITH THIS ATTACHMENT.

ACCORDINGLY, PURSUANT TO 10 CFR 2.790, COMED REQUESTS THAT THE INFORMATION IN THIS ATTACHMENT BE WITHHELD FROM PUBLIC DISCLOSURE.

GE Nuclear Energy

General Electric Company
175 Curtner Avenue, San Jose, CA 95125

December 3, 1999

Mr. Mark Wagner
Quad Cities Nuclear Power Station
Commonwealth Edison
22710 206 Avenue North
Cordova, IL 61242

Dear Mark,

Attached are the Revision 1 Technical Specification Improvement Analysis reports for the Reactor Protection System (RPS), Emergency Core Cooling System (ECCS), and Isolation Actuation Instrumentation for Quad Cities 1 and 2. These reports incorporate the ComEd review comments dated November 16, 1999.

This transmittal contains GE-NE proprietary information which is provided under the ComEd/GE-NE proprietary information agreement. GE-NE customarily maintains this information in confidence and withholds it from public disclosure.

The attached affidavit identifies that the designated information has been handled and classified as proprietary to GE-NE. Along with the affidavit this information is suitable for review by the NRC. GE-NE hereby requests that the designated information be withheld from public disclosure in accordance with the provisions of 10 CFR 2.790. In addition, all copies previously transmitted to ComEd should be destroyed.

Please call me if you have any questions.

Kelly Fletcher
Manager, Regulatory Services
M/C 182
408-925-6535
bcc:

General Electric Company

AFFIDAVIT

I, George B. Stramback, being duly sworn, depose and state as follows:

(1) I am Project Manager, Regulatory Services, General Electric Company ("GE") and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in the GE proprietary report GE-NE E11-00105-00-01-01, *Technical Specification Improvement Analysis for the Reactor Protection System for Quad Cities Station, Unit 1,* Class III (General Electric Company Proprietary Information), dated December 1999, GE-NE E11-00105-00-02-01, *Technical Specification Improvement Analysis for the Reactor Protection System for Quad Cities Station, Unit 2,* Class III (General Electric Company Proprietary Information), dated December 1999, and GE-NE E11-00105-00-03-01, *Technical Specification Improvement Analysis for the Emergency Core Cooling System Actuation Instrumentation for Quad Cities Nuclear Power Station, Units 1 & 2,* Class III (General Electric Company Proprietary Information), dated December 1999. This information is delineated by bars or brackets marked in the margin adjacent to the specific material.

(3) In making this application for withholding of proprietary information of which it is the owner, GE relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), 2.790(a)(4), and 2.790(d)(1) for "trade secrets and commercial or financial information obtained from a person and privileged or confidential" (Exemption 4). The material for which exemption from disclosure is here sought is all "confidential commercial information", and some portions also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, <u>Critical Mass Energy Project v. Nuclear Regulatory Commission,</u> 975F2d871 (DC Cir. 1992), and <u>Public Citizen Health Research Group v. FDA</u>, 704F2d1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by General Electric's competitors without license from General Electric constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

c. Information which reveals cost or price information, production capacities, budget levels, or commercial strategies of General Electric, its customers, or its suppliers;

d. Information which reveals aspects of past, present, or future General Electric customer-funded development plans and programs, of potential commercial value to General Electric;

e. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in both paragraphs (4)b. and (4)d., above.

(5) The information sought to be withheld is being submitted to NRC in confidence. The information is of a sort customarily held in confidence by GE, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GE, no public disclosure has been made, and it is not available in public sources. All disclosures to third parties including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within GE is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his delegate), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GE are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers,

and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information identified in paragraph (2), above, is classified as proprietary because it would provide other parties, including competitors, with a valuable interpretive information regarding the application of reliability based methodology to BWR instrumentation. A substantial effort has been expended by General Electric to develop this information in support of the BWR Owners' Group Technical Specifications Improvement Program.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GE's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GE's comprehensive BWR technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods.

The research, development, engineering, and analytical costs comprise a substantial investment of time and money by GE.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

GE's competitive advantage will be lost if its competitors are able to use the results of the GE experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to GE would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive GE of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing these very valuable analytical tools and expertise to determine and apply the appropriate evaluation process.

STATE OF CALIFORNIA)
) ss:
COUNTY OF SANTA CLARA)

George B. Stramback, being duly sworn, deposes and says:

That he has read the foregoing affidavit and the matters stated therein are true and correct to the best of his knowledge, information, and belief.

Executed at San Jose, California, this 2nd day of December 1999.

George B. Stramback

George B. Stramback
General Electric Company

Subscribed and sworn before me this 2nd day of December 1999.

VICKY D. SCHROER
Commission # 1224251
Notary Public - California
Santa Clara County
My Comm. Expires Jun 12, 2003

Vicky D. Schroer

Notary Public, State of California