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L47-91(09-30)LP
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JSP-0609-91
September 20, 1991
10CFR50.90

Docket No. 50-461

Document Control Desk
Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Clinton Power Station
Proposed Amendment of Facility
Operating License No. NPF-62

Dear Sir:

Pursuant to 10CFR50.90, Illinois Power Company (IP) hereby applies for amendment of Facility Operating License No. NPF-62, Appendix A - Technical Specifications, for Clinton Power Station (CPS). This request consists of proposed changes to the CPS Technical Specifications to incorporate reliability-based improvements to instrumentation Action Statements and surveillance test intervals based on Topical Reports which have previously been submitted to the NRC by the Boiling Water Reactor Owners' Group (BWROG). It should be noted that several of the proposed changes are based on one Topical Report (GENE-770-06-1) which has not yet been approved by the NRC. However, as described in Attachment 2, these proposed changes are bounded by the analyses provided in the Topical Reports which have already been approved by the NRC. IP is requesting these additional changes at this time to provide a complete request with respect to these reliability-based improvements. If the portion of this request based on GENE-770-06-1 is not approved, some of the remaining proposed changes (which are based on NRC-approved Topical Reports) would not be able to be implemented. This is because these instruments perform multiple functions which are addressed by separate Technical Specifications and hence, are addressed by separate Topical Reports.

For each of the above-noted proposed Technical Specification changes, a description and the associated justification (including a Basis For No Significant Hazards Consideration) are provided in Attachment 2. Marked-up copies of pages from the current CPS Technical Specifications are provided in Attachment 3. In addition, an affidavit supporting the facts set forth in this letter and its attachments is provided in Attachment 1.

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IP has reviewed the proposed changes against the criteria of 10CFR51.22 for categorical exclusion from environmental impact considerations. The proposed changes do not involve a significant hazards consideration, or significantly increase the amounts or change the types of effluents that may be released offsite, nor do they significantly increase individual or cumulative occupational radiation exposures. Based on the foregoing, IP concludes the proposed changes meet the criteria given in 10CFR51.22(c)(9) for a categorical exclusion from the requirement for an Environmental Impact Statement.

Sincerely yours,

J. S. Perry
Vice President

DAS/alh

Attachments

cc: NRC Clinton Licensing Project Manager
NRC Resident Inspector, V-690
NRC Region III, Regional Administrator
Illinois Department of Nuclear Safety

Attachment 1
to U-601671

STATE OF ILLINOIS

COUNTY OF DEWITT

J. Stephen Perry, being first duly sworn, deposes and says: That he is Vice President of Illinois Power Company; that the application for amendment of Facility Operating License NPF-62 has been prepared under his supervision and direction; that he knows the contents thereof; and that to the best of his knowledge and belief said application and the facts contained therein are true and correct.

DATE: This 20 day of September 1991.

Signed: J Stephen Perry
J. Stephen Perry

"OFFICIAL SEAL"
DIANE L. MILES
NOTARY PUBLIC, STATE OF ILLINOIS
MY COMMISSION EXPIRES 10/28/93

Subscribed and sworn to before me this 20 day of September 1991.

Diane L. Miles
Notary Public

Background

During late 1983, the BWR Owners' Group (BWROG) formed a Technical Specification Improvement (TSI) Committee, of which Illinois Power (IP) is a member. This committee subsequently established a program for the development of reliability analyses to justify improvements to surveillance test intervals (STIs) and allowable out-of-service times (AOTs) for instrumentation specified in the BWR Standard Technical Specifications. The primary objective of this program was to minimize, for applicable instrumentation, unnecessary testing and excessively restrictive AOTs that could potentially degrade overall plant safety and availability. Examples of some of the problems experienced with the current Technical Specification requirements are: inadvertent scrams or engineered safety feature actuations due to frequent testing; AOTs which are not long enough to perform repairs on a reasonable basis; excessive actuation of equipment for testing contributing to wear-out; and unnecessary radiation exposure to personnel performing Technical Specification required testing. A reduction in the number of Technical Specification required surveillance tests will allow plant personnel to perform other activities to increase the overall safety of the plant.

Within the same time frame, the NRC Staff issued NUREG-1024, "Technical Specifications - Enhancing the Safety Impact," which recommended that surveillance test requirements and Technical Specification Action Statements be reviewed to assure that they have an adequate technical basis and do indeed minimize plant risk. Use of reliability analyses to support engineering judgement was recognized as a primary basis for improving the Technical Specification requirements. NUREG-1024 thus reinforced the BWROG's program objectives and implementation methodology.

To this end, the BWROG submitted a series of Licensing Topical Reports addressing the Technical Specification instrumentation requirements for the Reactor Protection System (NEDC-30851P), Emergency Core Cooling Systems (NEDC-30936P), the Control Rod Block System (NEDC-30851P, Supplement 1), and the Containment and Reactor Vessel Isolation Control System (NEDC-30851P, Supplement 2 and NEDC-31677P). Each of these Licensing Topical Reports has been reviewed and approved by the NRC. In addition, the BWROG has submitted a Licensing Topical Report (GENE-770-06-1) which addresses Technical Specification requirements for other instruments which are similar to those addressed in the Licensing Topical Reports previously reviewed and approved by the NRC. However, GENE-770-06-1 has not yet been approved by the NRC.

As a member of the BWROG Technical Specifications Committee, IP is requesting that the results of the BWROG Licensing Topical Reports on Technical Specification improvements be applied to Clinton Power Station (CPS). For convenience in reviewing this request, this submittal (i.e., this attachment) has been divided into five separate parts addressing the functional areas and associated BWROG Licensing Topical Report(s). Each part contains its own description of proposed changes, justification, and Basis for No Significant Hazards Consideration. Included in Attachment 3 are marked-up copies of pages from the current CPS Technical Specifications indicating the combined effect of the changes requested in each part of this attachment.

Part I - Reactor Protection System (RPS)

Description of Proposed Changes

In accordance with 10CFR50.90, the following changes to Technical Specification 3/4.3.1, "Reactor Protection System Instrumentation," are proposed:

1. The repair allowable out-of-service time (AOT) of Action a.2 is being increased from one hour to six hours.

2. The surveillance AOT of footnote "*" is being increased from two hours to six hours.

3. The CHANNEL FUNCTIONAL TEST interval specified on Technical Specification Table 4.3.1.1-1, "Reactor Protection System Instrumentation Surveillance Requirements," is being increased from weekly (W) or monthly (M), as applicable, to quarterly (Q) for the following Functional Units:

   a. item 2.b, Average Power Range Monitor (APRM) Flow-Biased Simulated Thermal Power - High,
   b. item 2.c, APRM Neutron Flux - High,
   c. item 2.d, APRM Inoperative,
   d. item 3, Reactor Vessel Steam Dome Pressure - High,
   e. item 4, Reactor Vessel Water Level - Low, Level 3,
   f. item 5, Reactor Vessel Water Level - High, Level 8,
   g. item 6, Main Steam Line Isolation Valve - Closure,
   h. item 7, Main Steam Line Radiation - High,
   i. item 8, Drywell Pressure - High,
   j. item 9.a, Scram Discharge Volume Water Level - High, Level Transmitter,
   k. item 10, Turbine Stop Valve - Closure,
   l. item 11, Turbine Control Valve Fast Closure Valve Trip System Oil Pressure - Low, and
   m. item 13, Manual Scram.

4. The analog trip module calibration interval specified by footnote (g) to Technical Specification Table 4.3.1.1-1 is being increased from 31 days to 92 days.

5. An editorial change is being proposed to delete footnote "**" associated with Surveillance Requirement 4.3.1.2 since this footnote was only applicable until the first refueling outage.

Justification for Proposed Changes

On May 31, 1985 the BWROG submitted Licensing Topical Report NEDC-30851P, "Technical Specification Improvement Analyses for BWR Reactor Protection System," for NRC review. (This report provides justification for the proposed changes identified as 1 through 4 above.) The analyses documented in NEDC-30851P utilized fault tree modeling (based upon the CPS design) to estimate the impact of the proposed changes on the average Reactor Protection System (RPS) failure frequency.

The average RPS failure frequency is a function of the frequency of scram demands and the probability that the RPS is unavailable when demanded. The initiating events which require successful operation of the RPS for ensuring safe reactor shutdown were identified and their annual occurrence frequencies were estimated. The initiating events were divided into three groups based on the number of diverse sensors that initiate the scram for that event.

For each initiating event, a top-level failure event was identified using the success criteria described below. For each top failure event, a fault tree was developed which modeled all of the components needed for generation and processing of the RPS signals including the sensors, analog trip modules, logic cards, load drivers and scram solenoids. The common cause failure of these components was also modeled. A fault tree analysis was then performed using the WAM series computer code, WAMCUT, to obtain the major failure cut sets that contribute to the top failure event probability. The failure cut sets obtained were then analyzed using the FRANTIC III computer code to determine the average RPS system unavailability upon demand.

The average RPS unavailability was calculated for each initiating event group based on inputs which included component failure rates (time and demand related), common cause failure rates, human error rates, testing intervals, and test and repair times. Sensitivity studies were conducted by changing the input parameters by factors of 2, 5 and 10 (and 30, where appropriate) to determine the resultant impact on the average RPS unavailability and the total RPS failure frequency. The surveillance test intervals (STIs) and AOTs were then varied to determine the resulting effect on the average RPS failure frequency.

The scram success criteria used for this analysis is defined below for two specific failure modes:

a. Failure Mode A: One or more RPS electrical control rod groups fail to insert into the core. The success criteria for this failure mode was that two of the total four rod groups must fully insert.

b. Failure Mode B: One or more control rods in a random pattern fail to insert. The success criteria for this failure mode was that, if the control rods are inserted in a random manner, 69% of all the rods must fully insert to achieve success.

The acceptance guideline used by the BWROG for the proposed changes is based on a net change in risk. The net change in risk is the difference between the increase in risk that would result from the proposed changes and the decrease in risk that would result from the reduced likelihood of inadvertent scrams. If the net change in risk is determined to be insignificant, the BWROG considered the proposed changes to be acceptable.

The BWROG concluded that the overall effect of the proposed RPS Technical Specification changes provides a net increase in safety and improves plant operation. The improvement is achieved by reducing the potential for: (a) unnecessary plant scrams (reduced challenges to plant shutdown systems and improved plant availability); (b) excessive test cycles on equipment (reduced wear-out potential); and (c) diversion of plant personnel and resources on unnecessary testing (potential safety and operational improvement). The BWROG report concluded that for CPS the calculated average RPS failure frequency increases from $2.0x10^{-6}$/year to $2.3x10^{-6}$/year and a 0.1% increase in plant capacity factor can be achieved with incorporation of the proposed RPS Technical Specification changes.

By letter from Ashok C. Thadani (NRC) to Robert F. Janacek (BWROG) dated January 24, 1988, the NRC provided their Safety Evaluation Report on NEDC-30851P. The NRC concluded in their Safety Evaluation Report that NEDC-30851P applies directly to CPS and that the proposed changes would have a negligible impact on plant risk. On this basis, the NRC determined that these proposed changes are acceptable. However, the Staff identified that NEDC-30851P does not confirm that the allowable calibration period for instrumentation used in the RPS (for example, the solid-state analog trip units) can be extended from monthly to quarterly without creating excessive drift. Therefore, the Safety Evaluation Report states that the licensee must demonstrate, by use of current drift information provided by the equipment vendor or by use of plant-specific data, that a change of the functional test interval from monthly to quarterly can be supported.

With respect to the Staff's concern about instrument drift, the instrument setpoint calculations for the RPS instruments at CPS include the effects of instrument drift over 18 months for all instrument loop components except for the analog trip modules. To address drift of the analog trip modules, IP reviewed the results of monthly calibration checks performed over a one-year period on the affected RPS analog trip modules. Review of these calibration checks showed that the quarterly drift is within the present calibration tolerances. As a result, IP has concluded that lengthening the CHANNEL FUNCTIONAL TEST interval and analog trip module calibration interval, as applicable, for the RPS instruments from weekly or monthly to quarterly will not result in excessive instrument drift relative to the current, established setpoints. In addition, a CHANNEL CHECK is required at least once per 12 hours for those instruments with redundant channels. These routine CHANNEL CHECKS will help to identify excessive drift of the RPS instrumentation.

With respect to NRC approval of plant-specific changes to the RPS Technical Specifications based upon NEDC-30851P, IP understands that the NRC has expressed concern that the changes proposed in NEDC-30851P would allow continued plant operation for up to 12 hours with a combination of failures which could prevent a reactor scram as assumed for a particular plant transient. This could occur for a relay-type plant (with one-out-of-two-twice logic) if, for example, both channels of high reactor pressure and both channels of APRM neutron flux-high were inoperable in one trip system. These are the two RPS scram functions that are assumed to mitigate the pressure regulator failure - increasing transient (see

Table F-1 of NEDC-30851P). The proposed resolution to this issue is to require the affected trip system to be placed in the tripped condition within one hour (rather than 12 hours) when such a loss-of-function condition exists.

Because the solid-state RPS design of CPS is arranged in a two-out-of-four type logic scheme, this issue is not applicable to CPS. The proposed changes do not allow continued operation when any parameter is unable to provide a reactor scram. Therefore, resolution of the "loss-of-function" issue for NEDC-30851P does not impact approval of this request for CPS.

Basis For No Significant Hazards Consideration

In accordance with 10CFR50.92, a proposed change to the operating license (Technical Specifications) involves no significant hazards considerations if operation of the facility in accordance with the proposed change would not: (1) involve a significant increase in the probability or consequences of any accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety. The proposed RPS Technical Specification changes are evaluated against each of these criteria below.

(1) These proposed changes do not involve a change to the plant design or operation, only to the allowable out-of-service times (AOTs) and frequency at which testing of the RPS instrumentation is performed. As a result, these proposed changes cannot increase the probability of any design basis accident previously evaluated. As identified in NEDC-30851P, these proposed changes increase the average RPS failure frequency from $2.0x10^{-6}$/year to $2.3x10^{-6}$/year. This increase ($3x10^{-7}$/year) is considered to be insignificant.

As identified in the NRC Staff's Safety Evaluation Report of NEDC-30851P, this increase in average RPS failure frequency would contribute to a very small increase in core-melt frequency. The small increase in average RPS failure frequency is offset by safety benefits such as a reduction in the number of inadvertent test-induced scrams, a reduction in wear due to excessive equipment test cycling, and better optimization of plant personnel resources. Hence, the net change in risk resulting from these proposed changes would be insignificant. Therefore, these proposed changes do not result in a significant increase in the probability or the consequences of any accident previously evaluated.

(2) These proposed changes do not result in any change to the plant design or operation, only to the AOT and frequency at which testing of the RPS instrumentation is performed. Since failure of the RPS instrumentation itself cannot create an accident, these proposed changes can at most affect only accidents which have been previously evaluated. Therefore, these proposed changes cannot create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) As identified above, these proposed changes increase the average RPS failure frequency from 2.0×10^{-6}/year to 2.3×10^{-6}/year. The NRC Staff's Safety Evaluation Report of NEDC-30851P concluded that this small average RPS failure frequency increase would contribute to a very small increase in core-melt frequency. This small increase in average RPS failure frequency would be offset by safety benefits such as a reduction in the number of inadvertent test-induced scrams, a reduction in wear due to excessive equipment test cycling, and better optimization of plant personnel resources. Hence, the net change in risk resulting from these proposed changes would be insignificant. In addition, IP has confirmed that the proposed changes to the functional test intervals will not result in excessive instrument drift relative to the current, established setpoints. Therefore, these proposed changes do not result in a significant reduction in a margin of safety.

Based upon the foregoing, IP concludes that these proposed changes do not involve a significant hazards consideration.

Part II - Emergency Core Cooling System (ECCS)

Description of Proposed Changes

In accordance with 10CFR50.90, the following changes to Technical Specification 3/4.3.3, "Emergency Core Cooling System Actuation Instrumentation," are proposed:

1. The repair allowable out-of-service times (AOTs) of Technical Specification Table 3.3.3-1, "Emergency Core Cooling System Actuation Instrumentation," Actions 30, 33, 36, 37, and 40 are being increased from one hour to 24 hours; Action 35 is being increased from eight hours to 24 hours; and Action 32 is being identified as 24 hours.

2. The surveillance AOT of footnote (a) to Technical Specification Table 3.3.3-1 is being increased from two hours to six hours.

3. The CHANNEL FUNCTIONAL TEST interval specified on Technical Specification Table 4.3.3.1-1, "Emergency Core Cooling System Actuation Instrumentation Surveillance Requirements," is being increased from monthly (M) to quarterly (Q) for the following Trip Functions:

    a. item A.1.a, Division I Trip System, RHR-A (LPCI Mode) and LPCS System, Reactor Vessel Water Level - Low Low Low, Level 1,
    b. item A.1.b, Drywell Pressure - High,
    c. item A.1.c, Reactor Vessel Pressure - Low (LPCI and LPCS Injection Valve Permissive),
    d. item A.1.d, LPCI Pump A Start Time Delay Logic Card,
    e. item A.1.e, LPCS Pump Discharge Flow - Low,
    f. item A.1.f, LPCI Pump (A) Discharge Flow - Low,
    g. item A.2.a, Automatic Depressurization System Trip System "1", ADS Logic "A" and "E", Reactor Vessel Water Level - Low Low Low, Level 1,
    h. item A.2.b, Drywell Pressure - High,
    i. item A.2.c, ADS Timer,
    j. item A.2.d, Reactor Vessel Water Level - Low, Level 3,
    k. item A.2.e, LPCS Pump Discharge Pressure - High,
    l. item A.2.f, LPCI Pump A Discharge Pressure - High,
    m. item A.2.g, ADS Drywell Pressure Bypass Timer,
    n. item A.2.h, Manual Inhibit ADS Switch,
    o. item B.1.a, Division II Trip System, RHR B and C (LPCI Mode), Reactor Vessel Water Level - Low Low Low, Level 1,
    p. item B.1.b, Drywell Pressure - High,
    q. item B.1.c, Reactor Vessel Pressure - Low (LPCI Injection Valve Permissive),
    r. item B.1.d, LPCI Pump B Start Time Delay Logic Card,
    s. item B.1.e, LPCI Pump (B) Discharge Flow - Low,
    t. item B.1.f, LPCI Pump (C) Discharge Flow - Low,
    u. item B.2.a, Automatic Depressurization System Trip System "2", ADS Logic "B" and "F", Reactor Vessel Water Level - Low Low Low, Level 1,
    v. item B.2.b, Drywell Pressure - High,

w. item B.2.c, ADS Timer,
x. item B.2.d, Reactor Vessel Water Level - Low, Level 3,
y. item B.2.e, LPCI Pump (B and C) Discharge Pressure - High,
z. item B.2.f, ADS Drywell Pressure Bypass Timer,
aa. item B.2.g, Manual Inhibit ADS Switch,
bb. item C.1.a, Division III Trip System, HPCS System, Reactor Vessel Water Level - Low Low, Level 2,
cc. item C.1.b, Drywell Pressure - High,
dd. item C.1.c, Reactor Vessel Water Level - High, Level 8,
ee. item C.1.d, RCIC Storage Tank Level - Low,
ff. item C.1.e, Suppression Pool Water Level - High,
gg. item C.1.f, HPCS Pump Discharge Pressure - High, and
hh. item C.1.g, HPCS System Flow Rate - Low.

4. The analog trip module calibration interval specified by footnote (a) to Technical Specification Table 4.3.3.1-1 is being increased from 31 days to 92 days.

5. An editorial change is being proposed to delete footnote "*" associated with Surveillance Requirement 4.3.3.2 since this footnote was only applicable until the first refueling outage.

Justification for Proposed Changes

On July 23, 1987 the BWROG submitted Licensing Topical Report NEDC-30936P, "BWR Owners' Group Technical Specification Improvement Methodology (with Demonstration for BWR ECCS Actuation Instrumentation) Part 2," for NRC review. (This report provides justification for the proposed changes identified as 1 through 4 above.) Similar to the RPS report discussed in Part I of this submittal, the analyses documented in NEDC-30936P (Part 2) utilized fault tree modeling (based upon the CPS design) to estimate the impact of the proposed changes on the average water injection function failure frequency.

The calculation of average water injection failure frequency depends on two sets of parameters. The first set consists of initiating events which eventually call for water injection. The second set consists of the probability that the water injection function is unavailable given a demand for injection. Depending on each initiating event, the number of components that are needed for successful completion of the water injection function varies. Therefore, the water injection unavailability for a given initiating event may differ from that of another initiating event.

A function fault tree was developed for each initiating event in order to quantify the water injection unavailability per demand. The function fault tree modeled the logical relationship of the faults that contribute to the water injection unavailability. The function fault tree was used to estimate the water injection unavailability based upon the current Technical Specification requirements and the effect of proposed changes. The results were considered acceptable by the BWROG if the proposed changes resulted in less than a 4% increase in the average water injection failure frequency or if the average water injection failure frequency was calculated to be less than 1.0×10^{-6}/ year.

The only initiating events studied in this analysis were loss of offsite power (LOSP) initiating events. The LOSP event was chosen for consideration because, based on prior Probabilistic Risk Assessment calculations, LOSP events contribute from 40% to 90% of the calculated core damage frequency for most BWRs. Also, the LOSP analysis is a more severe test of ECCS actuation instrumentation than other accident sequences such as turbine trip, loss of feedwater, and recirculation pump failure. Therefore, the effect of the proposed changes on water injection unavailability and failure frequency for the LOSP initiating event will dominate contributions from all initiating events.

By letter from Charles E. Rossi (NRC) to Donald N. Grace (BWROG) dated December 9, 1988, the NRC provided their Safety Evaluation Report of NEDC-30936P (Part 2). The NRC concluded in their Safety Evaluation Report that the methods and acceptance criteria provided in NEDC-30936P (Part 2) are acceptable for implementation on a plant-specific basis. However, the NRC's Safety Evaluation Report states that in order for a licensee to use the generic analyses provided in NEDC-30936P (Part 2), the licensee must confirm the applicability of the generic analyses to the plant and confirm that any increase in instrument drift due to the extended surveillance intervals is properly accounted for in the setpoint calculation methodology.

The CPS ECCS configuration was specifically modeled (identified as BWR-6 solid-state) in NEDC-30936P (Part 2). Therefore, the generic analyses are directly applicable to CPS. As identified on Table 3-1 of NEDC-30936P (Part 2), these proposed changes increase the calculated average water injection failure frequency for CPS from 7.69×10^{-6}/year to 7.82×10^{-6}/year. This represents an increase of 1.3×10^{-7}/year (1.7%), which is well within the acceptance criteria of NEDC-30936P (Part 2) and the NRC's Safety Evaluation Report.

With respect to the NRC's concern about instrument drift, the instrument setpoint calculations for the ECCS actuation instrumentation at CPS include the effects of drift over 18 months for all instrument loop components except for the analog trip modules. To address drift of the analog trip modules, IP reviewed the results of monthly calibration checks performed over a one-year period on the affected ECCS actuation analog trip modules. Review of these calibration checks showed that the quarterly drift is within the present calibration tolerances. As a result, IP has concluded that lengthening the CHANNEL FUNCTIONAL TEST interval and analog trip module calibration interval, as applicable, for the ECCS actuation instruments from monthly to quarterly will not result in excessive drift relative to the current, established setpoints. In addition, a CHANNEL CHECK is required at least once per 12 hours for those instruments with redundant channels. These routine CHANNEL CHECKS will help to identify excessive drift of the ECCS actuation instrumentation.

Basis For No Significant Hazards Consideration

In accordance with 10CFR50.92, a proposed change to the operating license (Technical Specifications) involves no significant hazards considerations if operation of the facility in accordance with the proposed change would not: (1) involve a significant increase in the probability or consequences of any accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety. The proposed ECCS actuation instrumentation Technical Specification changes are evaluated against each of these criteria below.

(1) These proposed changes do not involve a change to the plant design or operation, only to the allowable out-of-service time (AOT) and frequency at which testing of the ECCS instrumentation is performed. Failure of the ECCS actuation instrumentation itself cannot create an accident. As a result, these proposed changes cannot increase the probability of any accident previously evaluated.

As identified in NEDC-30936P (Part 2), these proposed changes increase the calculated average water injection failure frequency from 7.69×10^{-6}/year to 7.82×10^{-6}/year. This represents an increase of 1.3×10^{-7}/year (1.7%), which is well within the acceptance criteria (4% or 1.0×10^{-6}/year) provided in NEDC-30936P (Part 2) and the NRC's Safety Evaluation Report. This small increase in average water injection failure frequency is offset by benefits such as a reduction in the number of inadvertent test-induced scrams and engineered safety feature actuations, a reduction in wear due to excessive test cycling, and better optimization of plant personnel resources. Therefore, these proposed changes do not result in a significant increase in the consequences of any accident previously evaluated.

(2) These proposed changes do not result in any change to the plant design or operation, only to the AOT and frequency at which testing of the ECCS instrumentation is performed. Since failure of the ECCS actuation instrumentation itself cannot create an accident, these proposed changes can at most affect only accidents which have been previously evaluated. Therefore, these proposed changes cannot create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) As identified above, these proposed changes increase the calculated average water injection failure frequency from 7.69×10^{-6}/year to 7.82×10^{-6}/year. This increase is well within the acceptance criteria found acceptable in the NRC Staff's Safety Evaluation Report for NEDC-30936P (Part 2). Further, this small increase in average water injection failure frequency would be offset by safety benefits such as a reduction in the number of inadvertent test-induced scrams and engineered safety feature actuations, a reduction in wear due to excessive test cycling, and better optimization of plant personnel resources. In addition, IP has confirmed that the proposed changes to the functional test

intervals will not result in excessive instrument drift relative to the current, established setpoints. Therefore, these proposed changes do not result in a significant reduction in a margin of safety.

Based on the foregoing, IP concludes that these proposed changes do not involve a significant hazards consideration.

Part III - Control Rod Block

Description of Proposed Changes

In accordance with 10CFR50.90, the following changes to Technical Specification 3/4.3.6, "Control Rod Block Instrumentation," are proposed*:

1. The surveillance allowable out-of-service time (AOT) of footnote (e) to Technical Specification Table 3.3.6-1, "Control Rod Block Instrumentation," is being increased from two hours to six hours.

2. The CHANNEL FUNCTIONAL TEST interval specified on Technical Specification Table 4.3.6-1, "Control Rod Block Instrumentation Surveillance Requirements," is being increased from monthly (M) to quarterly (Q) for the following Trip Functions:

   a. item 1.a, Rod Pattern Control System, Low Power Setpoint**,
   b. item 1.b, Rod Pattern Control System, RWL High Power Setpoint**,
   c. item 2.a, APRM Flow Biased Neutron Flux - Upscale,
   d. item 2.b, APRM Inoperative,
   e. item 2.c, APRM Downscale,
   f. item 2.d, APRM Neutron Flux - Upscale, Startup,
   g. item 5.a, Scram Discharge Volume, Water Level - High, and
   h. item 6.a, Reactor Coolant System Recirculation Flow, Upscale.

3. The analog trip module calibration interval specified by footnote (f) to Technical Specification Table 4.3.6-1 is being increased from 31 days to 92 days.

* Additional changes to Technical Specification 3/4.3.6 are proposed in Part V of this submittal.

** Currently, as indicated on the attached marked-up page from the CPS Technical Specifications (page 3/4 3-68), two footnotes [footnotes (d) and (e)] are attached to the CHANNEL FUNCTIONAL TEST requirement for the RPCS LPSP and RWL HPSP on Table 4.3.6-1. As the required CHANNEL FUNCTIONAL TEST frequency will be changed from monthly (M) to quarterly (Q) per this request, it appears that footnote (d) will no longer be consistent since the purpose of the footnote was to modify the current monthly (M) test frequency requirement by attaching the words, "at least once per 31 days while operation continues within a given power range above the RPCS low power setpoint." However, by IP letter dated October 30, 1987 (reference U-601048), IP requested a change to delete footnote (d). With the approval of the October 30, 1987 request, no inconsistency should result from changing the CHANNEL FUNCTIONAL TEST frequency from monthly (M) to quarterly (Q) per this request.

Justification for Proposed Changes

On June 23, 1986 the BWROG submitted Licensing Topical Report NEDC-30851P, Supplement 1, "Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation," for NRC review. This report provides justification for each of the proposed changes identified above.

Unlike the analyses discussed in Parts I and II of this submittal, no specific fault trees were developed for the control rod block instrumentation. Instead, the impact on the average control rod block failure rate was estimated based upon the results of the analyses presented in Part I of this submittal. This approach was taken because the Reactor Protection System (RPS) and control rod block functions share common instrument inputs. The BWROG report determined that the average control rod block failure rate would increase less than 10^{-4}/year (0.06%) from the current failure rate of 0.16/year (based on industry experience). NEDC-30851P, Supplement 1 states that the benefits associated with the proposed changes to the RPS instrumentation offset any potential negative impact of extending the control rod block instrumentation test intervals.

By letter from Charles E. Rossi (NRC) to Donald N. Grace (BWROG) dated September 22, 1988, the NRC provided their Safety Evaluation Report of NEDC-30851P, Supplement 1. The NRC concluded in their Safety Evaluation Report that NEDC-30851P, Supplement 1 provides an acceptable basis for implementing the above proposed control rod block instrumentation changes. However, the NRC's Safety Evaluation Report states that in order for a licensee to use the generic analyses provided in NEDC-30851, Supplement 1, the licensee must confirm the applicability of the generic analyses to the plant and confirm that any increase in instrument drift due to the extended intervals is properly accounted for in the setpoint calculation methodology.

IP has confirmed that the control rod block instrumentation configuration (described in NEDC-30851P and Supplement 1 as the Rod Control and Information System) is identical to that at CPS. As a result, the analyses presented in NEDC-30851P, Supplement 1 are directly applicable to CPS.

With respect to the NRC's concern about instrument drift, the instrument setpoint calculations for the control rod block instrumentation at CPS include the effects of instrument drift over 18 months for all instrument loop components except for the analog trip modules. To address drift of the analog trip modules, IP reviewed the results of monthly calibration checks performed over a one-year period on the affected control rod block analog trip modules. Review of these calibration checks showed that the quarterly drift is within the present calibration tolerances. As a result, IP has concluded that lengthening the CHANNEL FUNCTIONAL TEST interval and analog trip module calibration interval, as applicable, for the control rod block instrumentation from monthly to quarterly will not result in excessive drift relative to the current, established setpoints. In addition, a CHANNEL CHECK is required at least once per 12 hours for those instruments with redundant

channels. These routine CHANNEL CHECKS will help to identify excessive drift of the control rod block instrumentation.

Basis For No Significant Hazards Consideration

In accordance with 10CFR50.92, a proposed change to the operating license (Technical Specifications) involves no significant hazards considerations if operation of the facility in accordance with the proposed change would not: (1) involve a significant increase in the probability or consequences of any accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety. The proposed control rod block instrumentation Technical Specification changes are evaluated against each of these criteria below.

(1) These proposed changes do not involve a change to the plant design or operation, only to the allowable out-of-service time (AOT) and frequency at which testing of the control rod block instrumentation is performed. Failure of the control rod block instrumentation itself cannot create an accident. As a result, these proposed changes cannot increase the probability of any accident previously evaluated.

As identified in NEDC-30851P, Supplement 1, these proposed changes increase the average control rod block failure frequency less than 0.06%. As provided in the NRC Staff's Safety Evaluation Report of NEDC-30851P, Supplement 1, this increase is very slight and is offset by the safety benefits associated with the proposed changes to the RPS instrumentation. As a result, the combined effect of the changes proposed for the RPS and control rod block instrumentation requirements should result in an overall improvement in plant safety. Therefore, these proposed changes do not result in a significant increase in the consequences of any accident previously evaluated.

(2) These proposed changes do not result in any change to the plant design or operation, only to the AOT and frequency at which testing of the control rod block instrumentation is performed. Since failure of the control rod block instrumentation itself cannot create an accident, these proposed changes can at most affect only accidents which have been previously evaluated. Therefore, these proposed changes cannot create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) As identified above, these proposed changes increase the average control rod block failure frequency less than 0.06%. This increase is very slight and is offset by the safety benefits associated with the proposed changes to the RPS instrumentation. As a result, the combined effect of the changes proposed for the RPS and control rod block instrumentation requirements should result in an overall improvement in plant safety. In addition, IP has confirmed that the proposed changes to the functional test intervals will not result in excessive instrument drift relative

to the current, established setpoints. Therefore, these proposed changes do not result in a significant reduction in a margin of safety.

Based on the foregoing, IP concludes that these proposed changes do not involve a significant hazards consideration.

Part IV - Containment and Reactor Vessel Isolation Control System (CRVICS)

Description of Proposed Changes

In accordance with 10CFR50.90, the following changes to Technical Specification 3/4.3.2, "Containment and Reactor Vessel Isolation Control System," are proposed:

1. The repair allowable out-of-service time (AOT) of Action b.2 is being increased from one hour to six hours.

2. The repair AOT of Action c.1 is being increased from one hour to 24 hours. The repair AOT of footnote "**" associated with this Action is also being increased, from two hours to six hours.

3. The Surveillance AOTs identified in footnote "*" to the Limiting Condition for Operation and footnotes (a) and (k) to Technical Specification Table 3.3.2-1, "CRVICS Instrumentation," are being increased from two hours to six hours.

4. The CHANNEL FUNCTIONAL TEST interval specified on Technical Specification Table 4.3.2.1-1, "CRVICS Instrumentation Surveillance Requirements," is being increased from monthly (M) to quarterly (Q) for the following Trip Functions:

   a. item 1.a, Primary and Secondary Containment Isolation, Reactor Vessel Water Level - Low Low, Level 2,
   b. item 1.b, Reactor Vessel Water Level - Low Low, Level 2 (ECCS Div. I and II),
   c. item 1.c, Reactor Vessel Water Level - Low Low, Level 2 (ECCS Div. III),
   d. item 1.d, Drywell Pressure - High,
   e. item 1.e, Drywell Pressure - High (ECCS Div. I and II),
   f. item 1.f, Drywell Pressure - High (ECCS Div. III),
   g. item 1.g, Containment Building Fuel Transfer Pool Ventilation Plenum Radiation - High,
   h. item 1.h, Containment Building Exhaust Radiation - High,
   i. item 1.i, Containment Building Continuous Containment Purge (CCP) Exhaust Radiation - High,
   j. item 1.j, Reactor Vessel Water Level - Low Low Low, Level 1,
   k. item 1.k, Containment Pressure - High,
   l. item 1.l, Main Steam Line Radiation - High,
   m. item 1.m, Fuel Building Exhaust Radiation - High,
   n. item 2.a, Main Steam Line Isolation, Reactor Vessel Water Level - Low Low Low, Level 1,
   o. item 2.b, Main Steam Line Radiation - High,
   p. item 2.c, Main Steam Line Pressure - Low,
   q. item 2.d, Main Steam Line Flow - High,
   r. item 2.e, Condenser Vacuum - Low,
   s. item 2.f, Main Steam Line Tunnel Temp. - High,
   t. item 2.g, Main Steam Line Tunnel Delta Temp. - High,
   u. item 2.h, Main Steam Line Turbine Bldg. Temp. - High,

v. item 3.a, Reactor Water Cleanup System Isolation, Delta Flow - High,
w. item 3.b, Delta Flow Timer,
x. item 3.c.1, Equipment Area Temp. - High, Pump Rooms - A,B,C,
y. item 3.c.2, Equipment Area Temp. - High, Heat Exchanger Rooms - East, West,
z. item 3.d.1, Equipment Area Delta Temp. - High, Pump Rooms - A,B,C,
aa. item 3.d.2, Equipment Area Delta Temp. - High, Heat Exchanger Rooms - East, West,
bb. item 3.e, Reactor Vessel Water Level - Low Low, Level 2,
cc. item 3.f, Main Steam Line Tunnel Ambient Temp. - High,
dd. item 3.g, Main Steam Line Tunnel Delta Temp. - High,
ee. item 3.h, SLCS Initiation,
ff. item 4.a, Reactor Core Isolation Cooling System Isolation, RCIC Steam Line Flow - High,
gg. item 4.b, RCIC Steam Line Flow - High Timer,
hh. item 4.c, RCIC Steam Supply Pressure - Low,
ii. item 4.d, RCIC Turbine Exhaust Diaphragm Pressure - High,
jj. item 4.e, RCIC Equipment Room Ambient Temperature - High,
kk. item 4.f, RCIC Equipment Room Delta Temp. - High,
ll. item 4.g, Main Steam Line Tunnel Ambient Temp. - High,
mm. item 4.h, Main Steam Line Tunnel Delta Temp. - High,
nn. item 4.i, Main Steam Line Tunnel Temp. Timer,
oo. item 4.j, Drywell Pressure - High,
pp. item 4.l, RHR/RCIC Steam Line Flow - High,
qq. item 4.m, RHR Heat Exchanger A,B Ambient Temperature - High,
rr. item 4.n, RHR Heat Exchanger A,B Delta Temp. - High,
ss. item 5.a, RHR System Isolation, RHR Heat Exchanger Rooms A,B Ambient Temp. - High,
tt. item 5.b, RHR Heat Exchanger Rooms A,B Delta Temp. - High,
uu. item 5.c, Reactor Vessel Water Level - Low, Level 3,
vv. item 5.d, Reactor Vessel Water Level - Low Low Low, Level 1,
ww. item 5.e, Reactor Vessel (RHR Cut-in Permissive) Pressure - High,
xx. item 5.f.1, Drywell Pressure - High, RHR Test Line, and
yy. item 5.f.2, Drywell Pressure - High, Fuel Pool Cooling.

5. The staggered test interval specified by footnote (a) to Technical Specification Table 4.3.2.1-1 is being increased from 31 days to 92 days.

6. The analog trip module calibration interval specified by footnote (b) to Technical Specification Table 4.3.2.1-1 is being increased from 31 days to 92 days.

7. An editorial change is being proposed to delete footnote "**" associated with Surveillance Requirement 4.3.2.2 since this footnote was only applicable until the first refueling outage.

Justification for Proposed Changes

On August 29, 1986 the BWROG submitted Licensing Topical Report NEDC-30851P, Supplement 2, "Technical Specification Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," for NRC review. On June 27, 1989 the BWROG submitted Licensing Topical Report NEDC-31677P, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," for NRC review. The combination of the results from these two reports provides justification for the proposed changes identified as 1 through 6 above.

As stated in NEDC-30851P, Supplement 2, Technical Specification requirements for isolation instrumentation were originally established largely on the basis of RPS and ECCS requirements. That is, the surveillance test intervals and allowable out-of-service times generally do not need to be more stringent for isolation than for RPS or ECCS. Even though isolation is a safety function, failure to isolate would not of itself result in an accident. The overall containment and reactor vessel isolation function is made up of several subfunctions, each of which must operate upon demand during an accident. Failure of an isolation subfunction during an accident could potentially increase the offsite release risks.

The analysis presented in NEDC-30851P, Supplement 2 applies only to those CRVICS instruments which are common to the RPS or ECCS actuation instruments. Similar to the analyses discussed in Parts I and II of this submittal, fault trees were developed for each of the common isolation Trip Functions. These fault trees were then evaluated probabilistically to determine the impact of the proposed changes on isolation unavailability. As provided in NEDC-30851P, Supplement 2, the impact on the average isolation unavailability for the affected isolation subfunctions due to the proposed changes was determined to be negligible (an increase of less than 1%) when combined with the individual valve failure probabilities. The analyses demonstrate that the individual valve failure probabilities dominate the overall isolation failure probability.

By letter from Charles E. Rossi (NRC) to Donald N. Grace (BWROG) dated January 6, 1989, the NRC provided their Safety Evaluation Report of NEDC-30851P, Supplement 2. The NRC concluded in their Safety Evaluation Report that the methods and results provided in NEDC-30851P, Supplement 2 are acceptable for implementation on a plant-specific basis. However, the NRC's Safety Evaluation Report states that in order for a licensee to use the generic analyses provided in NEDC-30851P, Supplement 2, the licensee must confirm the applicability of the generic analyses to the plant and confirm that any increase in instrument drift due to the extended surveillance intervals is properly accounted for in the setpoint calculation methodology.

With respect to the NRC Staff's concern about confirming the plant-specific applicability of NEDC-30851P, Supplement 2, the CPS CRVICS configuration was specifically modeled (identified as BWR-6 solid-state) in NEDC-30851P, Supplement 2. Therefore, the generic results are directly applicable to CPS.

With respect to the NRC's concern about instrument drift, the instrument setpoint calculations for the CRVICS instrumentation at CPS include the effects of instrument drift over 18 months for all instrument loop components except for the analog trip modules. To address drift of the analog trip modules, IP reviewed the results of monthly calibration checks performed over a one-year period on the affected CRVICS analog trip modules which are common to RPS or ECCS. Review of these calibration checks showed that the quarterly drift is within the present calibration tolerances. As a result, IP has concluded that lengthening the CHANNEL FUNCTIONAL TEST interval and analog trip module calibration interval, as applicable, for the CRVICS instrumentation common to RPS or ECCS from monthly to quarterly will not result in excessive drift relative to the current, established setpoints. In addition, a CHANNEL CHECK is required at least once per 12 hours for those instruments with redundant channels. These routine CHANNEL CHECKS will help to identify excessive drift of the CRVICS instrumentation.

The analysis presented in NEDC-31677P applies to the remaining CRVICS Trip Functions (i.e., those CRVICS instruments which are <u>not</u> common to RPS or ECCS actuation instrumentation). Similar to previous analyses discussed above, the analyses presented in NEDC-31677P are based upon fault trees (based upon the CPS design) which were evaluated to determine the impact of the proposed changes on the average isolation failure frequency. In this case, the average isolation failure frequency is defined as the product of the accident initiating event frequency (such as a pipe break or high radiation event) and the probability of failure of the isolation function given a demand. The proposed changes were considered acceptable by the BWROG if the proposed changes resulted in less than a 10% increase in the average isolation failure frequency or if the average failure frequency was calculated to be less than 1.0×10^{-7}/year.

The results for the BWR-6 solid-state plant (which are directly applicable to CPS) demonstrate that these proposed changes only slightly increase the overall average isolation failure frequency for these instruments. As identified on Table 5-3 of NEDC-31677P, the calculated average isolation failure frequency increased 4.4×10^{-8}/year from 1.10×10^{-7}/year to 1.54×10^{-7}/year.

By letter from Charles E. Rossi (NRC) to S. D. Floyd (BWROG) dated June 18, 1990, the NRC provided their Safety Evaluation Report of NEDC-31677P. The NRC concluded in their Safety Evaluation Report that the methodology and acceptance criteria provided in NEDC-31677P are acceptable for implementation on a plant-specific basis. However, the NRC's Safety Evaluation Report states that in order for a licensee to use the generic analyses presented in NEDC-31677P, the licensee must confirm the applicability of the generic analyses to the plant and confirm that any increase in instrument drift due to the extended surveillance intervals is properly accounted for in the setpoint calculation methodology.

As identified above, the CPS CRVICS configuration was specifically modeled (identified as BWR-6 solid-state) in NEDC-31677P. Therefore, the generic results of NEDC-31677P are directly applicable to CPS.

With respect to the NRC's concern about instrument drift, the instrument setpoint calculations for the CRVICS instrumentation at CPS include the effects of instrument drift over 18 months for all instrument loop components except for the analog trip modules. To address drift of the analog trip modules, IP reviewed the results of monthly calibration checks performed over a one-year period on the affected CRVICS analog trip modules which are not common to RPS or ECCS. Review of these calibration checks showed that the quarterly drift is within the present calibration tolerances. As a result, IP has concluded that lengthening the CHANNEL FUNCTIONAL TEST interval and analog trip module calibration interval, as applicable, for the CRVICS instrumentation not common to RPS or ECCS from monthly to quarterly will not result in excessive drift relative to the current, established setpoints. In addition, a CHANNEL CHECK is required at least once per 12 hours for those instruments with redundant channels. These routine CHANNEL CHECKS will help to identify excessive drift of the CRVICS instrumentation.

It should be noted that the format of the Action Statements for the CRVICS instrumentation proposed in this request differs from that provided in NEDC-30851P, Supplement 2 and NEDC-31677P and from that provided in the NRC Safety Evaluation Reports for these reports. The above documents recommended replacing Action b (equivalent to CPS Action c.1) with an Action Statement which incorporated footnote "**" and recommended separate requirements for those instruments that are common to RPS. This format was recommended because the BWR-6 relay-type plants' requirements should be based on whether the channel can be placed in the tripped condition without resulting in an actuation while recognizing the shorter AOT of 12 hours for the relay plants' RPS instrumentation. Therefore, when placing the inoperable channel in the tripped condition would cause an isolation, the above documents recommended a repair AOT of six hours (equivalent to the surveillance AOT). When placing the inoperable channel in the tripped condition would not cause an isolation, the above documents recommended that the inoperable channel be placed in the tripped condition within 12 hours for instruments common to RPS and 24 hours for instruments common to ECCS.

As previously identified, the CPS RPS logic design is solid-state. This solid-state RPS logic is combined in a two-out-of-four arrangement such that a trip of any two of the four RPS channels will result in a reactor scram. Because of this redundancy, one RPS channel is currently allowed to be inoperable for 48 hours. As proposed in Part 1 of this submittal, two RPS channels may be inoperable for up to six hours.

Because one RPS channel may be inoperable for up to 48 hours (vs. 12 hours for BWR-6 relay-type plants), there is no need to specify a shorter AOT for CRVICS instruments common to RPS. Hence, an Action Statement that provides a separate AOT for CRVICS instruments common to RPS is not required for CPS. The proposed Action c.1 will permit, for all CRVICS instruments (including those common to RPS), a maximum AOT of 24 hours (vs. 48 hours per Technical Specification 3.3.1, "Reactor Protection System Instrumentation."). Additionally, footnote "**" has been retained, but the repair AOT has been increased from two hours to six hours (consistent with the proposed surveillance AOT). As a result, proposed Action c.1 is consistent with the model Technical

Specifications provided in the BWROG reports as applied to the BWR-6 solid-state design.

One additional area that was not specifically identified in the model Technical Specifications provided in the BWROG reports relates to CPS Action b for the CRVICS Main Steam Line Isolation Trip Functions. These Trip Functions utilize logic which is identical to the RPS. As such, these requirements should be identical to the Action Statements of Technical Specification 3.3.1. Accordingly, Action b.2 has been revised to match the changes proposed in Part I of this submittal.

Basis For No Significant Hazards Consideration

In accordance with 10CFR50.92, a proposed change to the operating license (Technical Specifications) involves no significant hazards considerations if operation of the facility in accordance with the proposed change would not: (1) involve a significant increase in the probability or consequences of any accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety. The proposed CRVICS instrumentation Technical Specification changes are evaluated against each of these criteria below.

(1) These proposed changes do not involve a change to the plant design or operation, only to the allowable out-of-service time (AOT) and frequency at which testing of the CRVICS instrumentation is performed. Failure of the CRVICS instrumentation itself cannot create an accident. As a result, these proposed changes cannot increase the probability of any accident previously evaluated.

As identified in NEDC-30851P, Supplement 2, the proposed changes to the requirements for the CRVICS instruments which are common to RPS and ECCS have a negligible (less than 1%) impact on the average isolation unavailability when combined with the individual valve failure probability. The analyses demonstrate that the individual valve failure probabilities dominate the overall isolation failure probability. As identified in NRC Staff's Safety Evaluation Report of NEDC-30851P, Supplement 2, these proposed changes would have a very small impact on plant risk. As a result, overall plant safety is not reduced by these proposed changes.

As identified in NEDC-31677P, the proposed changes to the requirements for CRVICS instrumentation not common to RPS or ECCS result in a small increase of 4.40×10^{-8}/year in the average isolation failure frequency. As identified in the NRC Staff's Safety Evaluation Report of NEDC-31677P, the NRC agreed that these proposed changes are acceptable because the failure frequency impact is minimal. As a result, overall plant safety is not reduced by these proposed changes.

The small increase in the average failure frequency due to the proposed changes to the CRVICS instrumentation requirements is offset by safety benefits such as a reduction in the number of

inadvertent test-induced scrams and engineered safety feature actuations, a reduction in wear due to excessive test cycling, and better optimization of plant personnel resources. Therefore, the proposed changes do not represent a significant increase in the consequences of any accident previously evaluated.

(2) These proposed changes do not result in any change to the plant design or operation, only to the AOT and frequency at which testing of the CRVICS instrumentation is performed. Since failure of the CRVICS instrumentation itself cannot create an accident, these proposed changes can at most affect only accidents which have been previously evaluated. Therefore, these proposed changes cannot create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) As identified above, these proposed changes to the requirements for CRVICS instruments common to RPS or ECCS have a negligible impact on the average isolation unavailability when combined with the individual valve failure probability. The analyses demonstrate that the individual valve failure probabilities dominate the overall isolation failure probability. The proposed changes to the requirements for CRVICS instruments not common to RPS or ECCS increase their average isolation failure frequency approximately 4.4×10^{-8}/year (from 1.10×10^{-7}/year to 1.54×10^{-7}/ year). This increase is negligible.

The small increases in average CRVICS instrumentation failure frequency are offset by safety benefits such as a reduction in the number of inadvertent test-induced scrams and engineered safety feature actuations, a reduction in wear due to excessive equipment test cycling, and better optimization of plant personnel resources. As a result, the NRC Staff's Safety Evaluation Reports for these BWROG reports concluded that these proposed changes would have a very small impact on plant risk. In addition, IP has confirmed that the proposed changes to the functional test intervals will not result in excessive instrument drift relative to the current, established setpoints. Therefore, these proposed changes do not result in a significant reduction in a margin of safety.

Based upon the foregoing, IP concludes that these proposed changes do not involve a significant hazards consideration.

Part V - Other Technical Specification Instrumentation

Description of Proposed Changes

In accordance with 10CFR50.90, the following changes are proposed:

1. Technical Specification 3/4.3.4.1, "ATWS Recirculation Pump Trip System Instrumentation"

   a. The surveillance allowable out-of-service time (AOT) of footnote (a) to Technical Specification Table 3.3.4.1-1, "ATWS Recirculation Pump Trip System Instrumentation," is being increased from two hours to six hours.

   b. The CHANNEL FUNCTIONAL TEST interval specified on Technical Specification Table 4.3.4.1-1, "ATWS Recirculation Pump Trip Actuation Instrumentation Surveillance Requirements," is being increased from monthly (M) to quarterly (Q) for the following Trip Functions:

      (i) item 1, Reactor Vessel Water Level - Low Low, Level 2, and
      (ii) item 2, Reactor Vessel Pressure - High.

   c. The trip unit calibration interval specified by footnote "*" to Technical Specification Table 4.3.4.1-1 is being increased from 31 days to 92 days.

2. Technical Specification 3/4.3.4.2, "End-of-Cycle Recirculation Pump Trip System Instrumentation"

   a. The repair AOT of Action c is being increased from one hour to six hours.

   b. The surveillance AOT of footnote (a) to Technical Specification Table 3.3.4.2-1, "End-of-Cycle Recirculation Pump Trip System Instrumentation," is being increased from two hours to six hours.

   c. The CHANNEL FUNCTIONAL TEST interval specified on Technical Specification Table 4.3.4.2-1, "End-of-Cycle Recirculation Pump Trip System Surveillance Requirements," is being increased from monthly (M) to quarterly (Q) for the following Trip Functions:

      (i) item 1, Turbine Stop Valve - Closure, and
      (ii) item 2, Turbine Control Valve - Fast Closure.

   d. An editorial change is being proposed to delete the "Total Number of Channels" and the "Channels to Trip" columns of Technical Specification Table 3.3.4.2-1 and revise the Minimum OPERABLE Channels per Trip Function requirements from "3" to "4". Since this logic is arranged in a two-out-of-four scheme and the Action Statements address channel inoperability based on four channels, this proposed change

is being made to make Table 3.3.4.2-1 match the current Action Statements. This proposed change does not reduce the number of channels required to be OPERABLE.

c. An editorial change is being proposed to delete footnote "*" associated with Surveillance Requirement 4.3.4.2.2 since this footnote was only applicable until the first refueling outage.

3. Technical Specification 3/4.3.5, "Reactor Core Isolation Cooling System Actuation Instrumentation"

a. The repair AOTs of Technical Specification Table 3.3.5-1, "Reactor Core Isolation Cooling System Actuation Instrumentation," Actions 50.a and 52 are being increased from one hour to 24 hours; Action 53 is being increased from eight hours to 24 hours; and Action 51 is being identified as 24 hours.

b. The surveillance AOT of footnote (a) to Technical Specification Table 3.3.5-1 is being increased from two hours to six hours.

c. The CHANNEL FUNCTIONAL TEST interval specified on Technical Specification Table 4.3.5.1-1, "Reactor Core Isolation Cooling System Actuation Instrumentation Surveillance Requirements," is being increased from monthly (M) to quarterly (Q) for the following Functional Units:

(i) item a, Reactor Vessel Water Level - Low Low, Level 2,
(ii) item b, Reactor Vessel Water Level - High, Level 8,
(iii) item c, RCIC Storage Tank Level - Low, and
(iv) item d, Suppression Pool Water Level - High.

d. The analog trip module calibration interval specified by footnote (a) to Technical Specification Table 4.3.5.1-1 is being increased from 31 days to 92 days.

e. An editorial change is being proposed to delete footnote "*" associated with Surveillance Requirement 4.3.5.2 since this footnote was only applicable until the first refueling outage.

4. Technical Specification 3/4.3.6, "Control Rod Block Instrumentation"

a. Action 64 is being added to Technical Specification Table 3.3.6-1, "Control Rod Block Instrumentation," to allow control rod block instrumentation channels to be inoperable for up to 24 hours rather than one hour per existing Action 62. The reference to Action 62 on Table 3.3.6-1 is being replaced with a reference to Action 64 for the following Trip Functions:

(i) item 5.a, Scram Discharge Volume, Water Level - High, and
(ii) item 6.a, Reactor Coolant System Recirculation Flow, Upscale.

5. <u>Technical Specification 3/4.3.7.1, "Radiation Monitoring Instrumentation"</u>

a. The repair AOT of Technical Specification Table 3.3.7.1-1, "Radiation Monitoring Instrumentation," Action 70.a is being increased from one hour to 24 hours.

b. The CHANNEL FUNCTIONAL TEST interval specified on Technical Specification Table 4.3.7.1-1, "Radiation Monitoring Instrumentation Surveillance Requirements," is being increased from monthly (M) to quarterly (Q) for item 1, Main Control Room Air Intake Radiation Monitor.

6. <u>Technical Specification 3/4.3.9, "Plant Systems Actuation Instrumentation"</u>

a. The repair AOT of Action b.1 and Technical Specification Table 3.3.9-1, "Plant Systems Actuation Instrumentation," Action 50.a is being increased from one hour to 24 hours, and a repair AOT of 24 hours is being added to Technical Specification Table 3.3.9-1 Action 51.

b. The surveillance AOT of footnote "*" to Technical Specification Table 3.3.9-1 is being increased from two hours to six hours.

c. The CHANNEL FUNCTIONAL TEST interval specified on Technical Specification Table 4.3.9.1-1, "Plant Systems Actuation Instrumentation Surveillance Requirements," is being increased from monthly (M) to quarterly (Q) for the following Trip Functions:

(i) item 1.a, Containment Spray System, Drywell Pressure - High,
(ii) item 1.b, Containment Pressure - High,
(iii) item 1.c, Reactor Vessel Water Level - Low Low Low, Level 1,
(iv) item 1.d, Timers,
(v) item 2.a, Feedwater System/Main Turbine Trip System, Reactor Vessel Water Level - High, Level 8,
(vi) item 3.a, Suppression Pool Makeup, Drywell Pressure - High,
(vii) item 3.b, Reactor Vessel Water Level - Low Low Low, Level 1,
(viii) item 3.c, Suppression Pool Water Level - Low Low, and
(ix) item 3.d, Suppression Pool Makeup Timer.

d. The analog trip module calibration interval specified by footnote (a) to Technical Specification Table 4.3.9.1-1 is being increased from 31 days to 92 days.

e. The analog comparator unit calibration interval specified by footnote (b) to Technical Specification Table 4.3.9.1-1 is being increased from 31 days to 92 days.

f. An editorial change is being proposed to correct the spelling of the word "comparator" in footnote (b) to Technical Specification Table 4.3.9.1-1.

g. An editorial change is being proposed to Action a to make it consistent with the wording utilized in similar Action Statements of other Technical Specifications.

h. An editorial change is also being proposed to delete footnote "*" associated with Surveillance Requirement 4.3.9.2 since this footnote was only applicable until the first refueling outage.

7. Technical Specification 3/4.4.2.1, "Safety/Relief Valves"

a. The surveillance AOT of footnote "*" is being increased from two hours to six hours.

b. The CHANNEL FUNCTIONAL TEST interval of Surveillance Requirement 4.4.2.1.2.a is being increased from 31 days to 92 days.

c. An editorial change is being proposed to delete footnote "**" associated with Surveillance Requirement 4.4.2.1.2.b since this footnote was only applicable until the first refueling outage.

8. Technical Specification 3/4.4.2.2, "Safety/Relief Valves Low-Low Set Function"

a. The surveillance AOT of footnote "*" is being increased from two hours to six hours.

b. The CHANNEL FUNCTIONAL TEST interval of Surveillance Requirement 4.4.2.2.a is being increased from 31 days to 92 days.

c. An editorial change is being proposed to delete footnote "+" associated with Surveillance Requirement 4.4.2.2.b since this footnote was only applicable until the first refueling outage.

Justification for Proposed Changes

On February 19, 1991 the BWROG submitted Licensing Topical Report GENE-770-06-1, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," for NRC review. This report provides the justification for the proposed changes identified above (except those identified as editorial changes). Although GENE-770-06-1 has not yet been approved by

the NRC Staff, IP is requesting the Technical Specification changes identified in that report (as described above) at this time in order to provide a complete request with respect to the instrumentation reliability-based improvements. If the proposed changes in Part V of this request are not approved, some of the improvements in Parts I through IV of this request would not be able to be implemented. This is because these instruments perform multiple functions which are addressed by separate Technical Specifications and hence, are addressed by separate Topical Reports.

As noted in GENE-770-06-1, the primary purpose for requesting these changes is to ensure consistency with the changes proposed for the RPS, ECCS actuation instrumentation and CRVICS actuation instrumentation. The instrumentation affected by the proposed changes in Part V of this request affect either the same or similar instrumentation addressed in Parts I through IV. The primary difference is the safety function performed by the instrumentation.

As also noted in GENE-770-06-1, a detailed analysis of those proposed changes that are associated with instrumentation that is common to previously analyzed instrumentation was not performed since the analyses discussed in Parts I through IV bound them. The remaining proposed changes involve instruments which are of a similar type to the instruments included in the analyses discussed in Parts I through IV. Existing redundancy of this instrumentation is either more extensive or comparable to the redundancy of the instruments discussed in Parts I through IV. Further, analyses have generally shown that the most significant contributor to safety function failure probability is associated with the actuated device (such as valves) rather than associated with the actuation instrumentation. Therefore, the analyses discussed in Parts I through IV of this request can be used to justify the proposed changes identified in this part.

As discussed in Parts I through IV of this request, any expected increase in the probability of function failure as a result of these proposed changes will be offset by safety benefits such as a reduction in the number of inadvertent test-induced scrams and engineered safety feature actuations, a reduction in wear due to excessive equipment test cycling, and better optimization of plant personnel resources. As a result, these proposed changes do not result in a degradation to overall plant safety.

The basis for IP's determination that each of these proposed changes are bounded by the analyses discussed in Parts I through IV of this request is discussed below for each of the affected systems.

1. <u>Technical Specification 3/4.3.4.1, "ATWS Recirculation Pump Trip Systems Instrumentation"</u>

   The ATWS-RPT instrumentation is part of the mitigation system that initiates in the unlikely event of a scram failure. The trip function is initiated by either high reactor pressure or low low reactor water level (Level 2). The ATWS-RPT logic for CPS is two-out-of-two channels per trip system for each Trip Function. Each of the two trip systems initiates a trip of both recirculation

pumps. The effect of the proposed changes to the ATWS-RPT instrumentation requirements on the reactivity shutdown failure frequency is negligible based on the low average RPS failure frequency ($2.3x10^{-6}$/year from NEDC-30851P, page 6-11) and the small change in overall ATWS-RPT function unavailability due to the proposed changes (less than $1x10^{-2}$/demand calculated from failure rates of similar instruments as given in Appendix B and C of NEDC-30851P).

2. Technical Specification 3/4.3.4.2, "End-of-Cycle Recirculation Pump Trip System Instrumentation"

The EOC-RPT is initiated by signals and instrumentation common to the RPS (turbine stop valve closure and turbine control valve low hydraulic pressure). The proposed changes for this instrumentation were evaluated in NEDC-30851P for the RPS function. Although the EOC-RPT trip functions were not explicitly identified in NEDC-30851P, these proposed changes can be considered bounded by that analysis. The basis for this conclusion is similar to the basis established in NEDC-30851P, Supplement 2 for the control rod block instrumentation common to the RPS. That is, although failure of the EOC-RPT trip function could potentially lead to exceeding the Minimum Critical Power Ratio (MCPR) limit (similar to the consequences of an unmitigated rod withdrawal error), the slight increase in risk of an MCPR violation due to the proposed EOC-RPT changes is offset by the safety benefits associated with the proposed changes for the RPS instrumentation.

3. Technical Specification 3/4.3.5, "Reactor Core Isolation Cooling System Actuation Instrumentation"

The proposed changes to the RCIC system actuation instrumentation were evaluated in the BWROG analysis of ECCS actuation instrumentation (NEDC-30936P (Part 2)). The RCIC fault tree models and input data were developed for the CPS design (BWR-6 solid-state). In NEDC-30936P (Part 2), the water injection function failure frequency was analyzed as a function of the STIs and AOTs for the ECCS (including RCIC) actuation instrumentation. The RCIC actuation instrumentation surveillance test interval (STI) was changed from 1 to 3 months and the associated AOT was changed from 1 to 24 hours for repair and from 2 to 6 hours for test. The analysis results are summarized in NEDC-30936P (Part 2); however, model Technical Specification changes for the RCIC actuation instrumentation were not specifically included in NEDC-30936P (Part 2).*

* Model Technical Specification changes for the RCIC actuation instrumentation were thus later provided in GENE-770-06-1.

An analysis was conducted to demonstrate the specific effect of individual changes to the RCIC actuation instrumentation STIs on the overall average water injection function unavailability. As noted above, the analysis was performed using the models and input data developed and documented in NEDC-30936P (Parts 1 and 2). In order to determine the specific effect of the STI change on the RCIC actuation instrumentation, the RCIC actuation instrumentation STI was held constant (i.e., STI = one month) while the STI for other ECCS actuation instrumentation was changed to three months. This calculation demonstrated that there is a very small change in the calculated average water injection function unavailability (less than 1%) for this case when compared with the results of NEDC-30936P (Part 2). The NEDC-30936P (Part 2) analysis results indicated that the effect of AOT changes is significantly less than STI changes. On this basis, a similar negligible change in average water injection function unavailability can be expected when the RCIC actuation instrumentation AOTs (one hour repair and two hours test) are held constant. Therefore, it can be concluded that the STI and AOT changes to the RCIC actuation instrumentation are justified based on the small effect on the calculated average water injection function unavailability and consistency with comparable changes to the actuation instrumentation for the ECCS subsystems.

4. <u>Technical Specification 3/4.3.6, "Control Rod Block Instrumentation"</u>

NEDC-30851P, Supplement 2 provided the bases for changing the STIs for the control rod block instrumentation from one month to three months. Although the above changes to the repair and test AOTs were not explicitly identified in NEDC-30851P, Supplement 2, the same bases used for changing the STIs applies to the AOT changes. The reason for this is because analyses indicate that the effect of AOT changes is significantly less than the effect of STI changes. The proposed changes to the AOTs for the control rod block instrumentation are therefore supported by the basis provided in NEDC-30851P, Supplement 2.

5. <u>Technical Specification 3/4.3.7.1, "Radiation Monitoring Instrumentation"</u>

The main control room ventilation system is provided with radiation monitors to monitor radiation levels at the two outside minimum air intakes. Upon detecting a high radiation signal, the main control room ventilation system is automatically placed into the filtration mode. A Division I and a Division II radiation monitor is provided at each air intake. The radiation monitor outputs are combined in a one-out-of-two-twice logic to actuate the automatic filtration mode. This instrumentation arrangement is similar to the CRVICS radiation monitoring instrumentation. Therefore, the analysis of the isolation actuation instrumentation provided in NEDC-31677P supports similar STI and AOT changes to the Technical Specifications for the main control room ventilation intake radiation monitors.

6. Technical Specification 3/4.3.9, "Plant Systems Actuation Instrumentation"

This Technical Specification addresses the requirements for those instruments that provide automatic actuation of the containment spray system, feedwater/main turbine trip system, and the suppression pool makeup system. Each of these systems are discussed separately below.

a. Cor nment Spray System

The containment spray system actuation instrumentation contains instrumentation common to the ECCS actuation instrumentation. In addition, the actuation function performed (i.e., closing and opening selected valves) is similar to the function performed by the isolation and ECCS actuation instrumentation. The dominant contributor to the unavailability for this type of function is valve unavailability. Therefore, the analyses of isolation actuation instrumentation provided in NEDC-30851P, Supplement 2 and NEDC-31677P support similar STI and AOT changes to the containment spray system instrumentation.

b. Feedwater System/Main Turbine Trip System

The BWR-6 plant design incorporates a direct scram from high reactor vessel water level (Level 8) trip instrumentation (included in the RPS instrumentation). The bases for changes to the STIs and AOTs for the reactor vessel water Level 8 trip instrumentation associated with the feedwater system/main turbine trip system are therefore bounded by the changes to the RPS reactor vessel water Level 8 trip instrumentation provided in NEDC-30851P.

c. Suppression Pool Makeup

The same bases given for the containment spray system instrumentation applies for the suppression pool makeup system instrumentation. The suppression pool makeup system instrumentation contains instrumentation which is common to the ECCS actuation instrumentation. In addition, the actuation function performed (i.e., opening selected valves) is similar to the function performed by the isolation and ECCS actuation instrumentation. The dominant contributor to the unavailability for this type of function is valve unavailability. Therefore, the analyses of isolation actuation instrumentation provided in NEDC-30851P, Supplement 2 and NEDC-31677P support similar STI and AOT changes to the suppression pool makeup system instrumentation.

7. Technical Specification 3/4.2.4.2.1, "Safety/Relief Valves"

For CPS, five of the 16 safety/relief valves (SRVs) are required to open in the relief mode (actuated by a pressure transmitter)

and six are required to open in the safety mode (actuating against spring pressure) to prevent reactor vessel overpressurization. SRV safety mode actuation is diverse from the relief mode actuation. The relief function of the SRVs is performed by three separate sets of logic. Each logic set is actuated by one of two two-out-of-two reactor steam dome pressure logic combinations. The first logic group controls the relief function for one valve, the second logic group controls eight valves, and the third logic group controls seven valves. If a relief function logic group should fail (which requires at least two channel failures), overpressure protection can be provided by the remaining relief logic groups in combination with SRV actuations in the safety mode.

Based on the level of redundancy, unavailability of the relief valve pressure actuation function is a small contributor to the overall SRV function unavailability. Changes to the STI and AOT for the SRV pressure actuation instrumentation will therefore have an insignificant effect on the probability of failure to prevent reactor overpressurization. These STI and AOT changes will also be consistent with STI and AOT changes to similar instrumentation in the ECCS and isolation actuation systems.

8. <u>Technical Specification 3/4.4.2.2, "Safety/Relief Valves Low-Low Set Function"</u>

The Low-Low Set (LLS) logic for CPS consists of three individual LLS circuit groups which control five LLS SRVs. This logic is designed so that no more than one SRV reopens following a reactor vessel isolation event, ensuring that the containment design basis is met. After an LLS SRV initially opens in the relief mode, the associated LLS logic is activated and the SRV's closing setpoint is lowered such that the SRV stays open longer than without LLS. Two of the LLS circuit groups each control an individual SRV. These two logic circuit groups also lower the SRV's reopening setpoint such that the SRV will open prior to activating additional SRVs in the relief mode. The third logic circuit group controls a group of three LLS SRVs and only lowers their closure setpoints. The LLS function can normally be performed by either of the first two LLS logic groups.

Because energization of either SRV solenoid pilot valve results in opening the SRV, both solenoid pilot valves must be de-energized for the SRV to close. Opening of the first two LLS logic groups is accomplished by actuation of one of the two two-out-of-two SRV relief mode logic trains. Subsequent closure and reopening of these two LLS logic groups is accomplished by actuation of a one-out-of-one logic for each solenoid pilot valve. The third LLS logic group opens upon actuation of one of two two-out-of-two logic trains and recloses upon deactivation of both two-out-of-two logic trains.

Although the LLS logic has an important safety function, its function is not as critical to overall plant safety as the water injection or isolation functions. Therefore, changes to the STIs

and AOTs for the LLS pressure actuation instrumentation will have less risk impact on the overall plant safety than ECCS and isolation actuation STI and AOT changes. Existing analyses of ECCS and isolation actuation instrumentation STI and AOT changes can be applied to the LLS logic based on the use of the same or similar type of components (i.e., relays, transmitters, trip units, etc.), designed redundancy, and safety significance of the LLS logic. The extensive redundancy in the LLS circuit logic is comparable with the logic redundancy in the ECCS and isolation actuation instrumentation. Based on this redundancy, similarity of components, and safety function significance of the LLS logic, it can be concluded that the effect of changes for the LLS logic STIs and AOTs is bounded by the basis established for similar STI and AOT changes for the ECCS and isolation actuation instrumentation.

With respect to instrument drift for all of the instrumentation addressed in this part, the instrument setpoint calculations for these instruments include the effects of instrument drift over 18 months for all instrument loop components except for the trip units. To address drift of the trip units, IP reviewed the results of monthly calibration checks performed over a one-year period on the affected trip units. Review of these calibration checks showed that the quarterly drift is within the present calibration tolerances. As a result, IP has concluded that lengthening the CHANNEL FUNCTIONAL TEST interval and trip unit calibration interval, as applicable, for the affected instrumentation from monthly to quarterly will not result in excessive drift relative to the current, established setpoints. In addition, a CHANNEL CHECK is required at least once per 12 hours for those instruments with redundant channels. These routine CHANNEL CHECKS will help to identify excessive drift of the instrumentation affected by these proposed changes.

Basis for No Significant Hazards Consideration

In accordance with 10CFR50.92, a proposed change to the operating license (Technical Specifications) involves no significant hazards considerations if operation of the facility in accordance with the proposed change would not: (1) involve a significant increase in the probability or consequences of any accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety. The proposed Technical Specification changes are evaluated against each of these criteria below.

(1) These proposed changes do not involve a change to the plant design or operation, only to the allowable out-of-service time (AOT) and frequency at which testing of the associated instrumentation is performed. These instruments are designed to mitigate the consequences of previously analyzed accidents. Failure of these instruments cannot increase, and is independent of, the probability of occurrence of such accidents. As a result, these proposed changes cannot increase the probability of any accident previously evaluated. As identified in GENE-770-06-1, although not specifically analyzed, these proposed changes are bounded by

the results of the analyses discussed in Parts I through IV of this request. Such analyses have shown that the safety function failure frequency is not significantly impacted by similar proposed changes. In addition, any increase in the probability of failure of these instruments to perform their required functions would be offset by safety benefits such as a reduction in the number of inadvertent test-induced scrams and engineered safety feature actuations, a reduction in wear due to excessive equipment test cycling, and better optimization of plant personnel resources. As a result, these proposed changes should reduce overall plant risk. Therefore, these proposed changes do not result in a significant increase in the probability or the consequences of any accident previously evaluated.

(2) These proposed changes do not result in any change to the plant design or operation, only to the AOT and frequency at which testing of the associated instrumentation is performed. As a result, these proposed changes can at most affect only accidents which have been previously evaluated. Therefore, these proposed changes cannot create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) As identified in GENE-770-06-1, although not specifically analyzed, these proposed changes are bounded by the results of the analyses discussed in Parts I through IV of this request. Such analyses have shown that the safety function failure frequency is not significantly impacted by similar proposed changes. In addition, any increase in the probability of failure of these instruments to perform their required functions would be offset by safety benefits such as a reduction in the number of inadvertent test-induced scrams and engineered safety feature actuations, a reduction in wear due to excessive equipment test cycling, and better optimization of plant personnel resources. As a result, these proposed changes will reduce overall plant risk. In addition, IP has confirmed that the proposed changes to the functional test intervals will not result in excessive instrument drift relative to the current, established setpoints. Therefore, these proposed changes do not involve a significant reduction in a margin of safety.

Based upon the foregoing, IP has concluded that these proposed changes do not involve a significant hazards consideration.