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Docket No. 50-366 HL-6049

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Edwin I. Hatch Nuclear Plant – Unit 2
Request to Revise Technical Specifications:
Limited Change to Intermediate Range Monitor
Limiting Condition for Operation

Ladies and Gentlemen:

In accordance with the provisions of 10 CFR 50.90, as required by 10 CFR 50.59(c)(1), Southern Nuclear Operating Company (SNC) is proposing a limited (temporary) change to the Plant Hatch Unit 2 Technical Specifications, Appendix A to Operating License NPF-5. The proposed change allows Mode 2 (startup) operation with two required intermediate range monitor (IRM) channels per trip system and will be in effect from May 1, 2001 (the proposed date of NRC issuance of this amendment request) until the Fall 2001 Unit 2 refueling outage.

Since the IRMs are not required for power operation, they were retracted from the Unit 2 reactor core following start up from the Spring 2000 refueling outage. Unfortunately, two IRMs began experiencing erratic operation during the run cycle, and troubleshooting indicated the problem is inside the inerted Unit 2 drywell. Shutting down the unit prior to the fall outage to repair the IRMs requires an unnecessary power maneuver that involves proceeding to cold shutdown and de-inerting the drywell. In addition, a detailed review of the safety function of the IRMs has concluded that this limited change can be implemented with no adverse impact on plant safety.

Enclosure 1 provides a detailed description of the proposed change and the circumstances necessitating the change request. Enclosure 2 describes SNC's determination that the proposed change does not involve a significant hazardous consideration. Enclosure 3 provides page change instructions for incorporating the proposed change, the revised Technical Specifications page, and the associated markup of the Technical Specifications current page.

In accordance with the requirements of 10 CFR 50.91, a copy of this letter and all applicable enclosures will be sent to the designated State official of the Environmental Protection Division of the Georgia Department of Natural Resources.

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Mr. H. L. Sumner, Jr. states he is Vice President of Southern Nuclear Operating Company and is authorized to execute this oath on behalf of Southern Nuclear Operating Company, and to the best of his knowledge and belief, the facts set forth in this letter are true.

Respectfully submitted,

H. L. Sumner, Jr.

Sworn to and subscribed before me this 9th *day of* March *2001.*

Notary Public

Commission Expiration Date: 5-25-2003

GKM/eb

Enclosures:
1. Description, Background, and Justification for Proposed Change
2. 10 CFR 50.92 No Significant Hazards Evaluation and Environmental Assessment
3. Page Change Instructions, Revised Technical Specifications Page, and Associated Marked-up Page.

cc: Southern Nuclear Operating Company
Mr. P. H. Wells, Nuclear Plant General Manager
SNC Document Management (R-Type A02.001)

U.S. Nuclear Regulatory Commission, Washington, D.C.
Mr. L. N. Olshan, Project Manager - Hatch

U.S. Nuclear Regulatory Commission, Region II
Mr. L. A. Reyes, Regional Administrator
Mr. J. T. Munday, Senior Resident Inspector - Hatch

State of Georgia
Mr. L. C. Barrett, Commissioner - Department of Natural Resources

Enclosure 1

Edwin I. Hatch Nuclear Plant
Request to Revise Technical Specifications:
Limited Change to Intermediate Range Monitor
Limiting Condition for Operation

Description, Background, and Justification for Proposed Change

Description of Proposed Change

Southern Nuclear Operating Company (SNC) proposes to temporarily revise the Plant Hatch Unit 2 Technical Specifications Limiting Condition for Operation (LCO) for the intermediate range monitors (IRMs). Specifically, the number of required operable IRM channels per trip system specified in Technical Specifications Table 3.3.1.1-1 will be reduced from 3 to 2 for the period of time between issuance of the amendment and shut down of the reactor for the Unit 2 Fall refueling outage currently scheduled to commence on September 15, 2001. A footnote will be added to Table 3.3.1.1-1 to require one IRM to be operable in each quadrant of the core and require a second operator to verify rod moves whenever the rod worth minimizer (RWM) has to be operable.

Background

Since the IRMs are not required for power operation, they were retracted from the Unit 2 reactor core following start up from the Spring 2000 refueling outage. Detector 2C51-K601C (IRM C) began functioning improperly on August 2, 2000, and detector 2C51-K601E (IRM E) began functioning improperly on January 15, 2001. Troubleshooting verified the problem is inside the inerted Unit 2 drywell. Since the unit has been in power operation since startup, there has been no opportunity to replace the faulty detectors.

The most likely failure mode of the IRMs is a phenomenon known as copper migration, which shortens detector life and causes erratic operation. Prior to last summer, SNC's experience demonstrated that acceptable IRM performance could be expected by changing out the IRMs approximately every 10 years. It should be noted that detector life, even with copper migration, usually exceeds one fuel cycle. For example, IRMs C and E were installed in 1995.

The IRM vendor, Reuter-Stokes, traced the origin of the copper migration to a manufacturing problem, which they reportedly corrected. According to Reuter-Stokes, the problem occurred with detectors manufactured between 1992 and 1997. Detectors manufactured before 1992 rarely exhibited spiking, while detectors manufactured after 1997 have exhibited improved performance after at least 30 months of operation. Although Reuter-Stokes reported in an industry meeting that the IRM manufacturing process had been enhanced, they did not notify SNC of the problem with the "suspect" IRM detectors until June 2000. This notification was prompted by SNC inquiries to Reuter-Stokes after three Unit 1 IRM detectors failed during an operating cycle. Unit 1 operated at full power until a turbine trip occurred on July 10, 2000 (reference LER 50-321/2000-004). The cause of the turbine trip and subsequent reactor scram was the failure of a single instrument on the main turbine that senses vibration. The cause of the

turbine trip was promptly diagnosed and corrected. However, because the IRMs were inoperable, it was necessary to maneuver the unit to cold shutdown, de-inert the drywell, and install new detectors. Unit 1 was shut down again in the fall for its scheduled refueling outage. During the refueling outage, additional IRMs were replaced.

In addressing the Unit 2 short-term IRM problem, SNC has considered the following three options:

1. **Shut down the reactor this spring and replace the two inoperable IRM detectors.**

   This option involves going to cold shutdown and de-inerting the drywell. The outage duration will be approximately 5 days. During the Fall 2001 refueling outage, an additional five or six detectors will have to replaced.

2. **Continue power operation through the summer and replace all old or "suspect" detectors during the Fall 2001 outage.**

   The problem with this option is that there is always the possibility of a forced shutdown or scram occurring during peak summer months. If the unit is taken out of the RUN mode (Mode 1), it will be necessary to:

   a. enter the drywell to replace the detectors, or

   b. place IRM trip system A in the tripped condition and start up with a one-half scram signal, or

   c. request a Notice of Enforcement Discretion (NOED) from the NRC.

   Last summer, SNC elected to enter the drywell and replace the detectors rather than either start up with a one-half scram signal or request an NOED.

3. **Request relief from the NRC until the next Unit 2 outage.**

   SNC prefers this option. As discussed below in the "Justification for Proposed Change," the request has no adverse impact on plant safety.

Justification of Proposed Change

The function of the IRMs is discussed in Unit 2 Technical Specification Bases B 3.3.1.1 and the Final Safety Analysis Report (FSAR). The IRMs monitor neutron flux levels from the upper range of the source range monitors (SRMs) to the lower range of the average power range monitors (APRMs). The IRM system consists of eight incore detectors and their associated drive mechanisms and electronic circuitry (or channels). The eight IRM channels are divided into two groups of four, such that four IRM channels provide input into each reactor protection system (RPS) trip system. The Technical Specifications permit bypassing one IRM channel in each trip system, as delineated in Technical Specifications Table 3.3.1.1-1, Functions 1.a. and 1.b.

The IRM is credited with mitigating the consequences of a continuous control rod withdrawal transient for an out-of-sequence rod in the startup range (reference FSAR Chapter 15, paragraph 15.2.6.1.3). The event scenario assumes the following:

1. The operator mistakenly selects and continuously withdraws an out-of-sequence rod.

2. The RWM fails to give a "select error" and provide a rod block.

However, if the RWM is operable, any attempt to move a control rod out of sequence will not be possible because the RWM prevents the withdrawal of the error rod by initiating a control rod block. Thus, the postulated event cannot occur with the RWM operable. In addition, an in-sequence rod withdrawal error cannot occur, because the RWM will provide rod blocks at the banked-position withdrawal sequence (BPWS) specified withdrawal limits. The Technical Specifications currently require the RWM to be operable below 10% of rated reactor power, except during one startup per calendar year. In that case, the startup is currently allowed by utilizing a second operator to verify compliance with the BPWS compliant control rod pull sequence. **As a compensatory action associated with the proposed change, SNC will ensure both the RWM and a second licensed operator verify compliance with the withdrawal sequence.**

NEDO-23842, "Continuous Control Rod Withdrawal Transient in the Startup Range,"[(1)] which is referenced by the Technical Specification Bases, presents the results of General Electric's (GE's) generic analysis addressing the withdrawal error in the startup range. A detailed review of NEDO-23842[(1)] identified the following key points:

1. The analysis assumed one IRM in each RPS trip system was bypassed.

2. The resultant peak fuel enthalpies due to the continuous withdrawal of an out-of-sequence rod in the startup range were well below the licensing basis criterion of 170 cal/gm.

3. Although the scram terminates the event, Doppler feedback limits the prompt power increase that occurs when the control rod is withdrawn at its maximum rate.

4. Doppler feedback and the APRM flux scram (not the IRMs) actually terminate the limiting case that occurs at approximately 1% power. The peak fuel enthalpy reported for the limiting case was < 60 cal/gm.

NEDO-23842[(1)] further adds that at lower power, non-limiting cases are terminated by a combination of Doppler feedback and a scram signal from the IRMs. However, the analysis was conservative in assigning detector locations for the analysis. The detector closest to the control rod being withdrawn was bypassed on trip systems A and B. The power level registered by the second closest detector of both trip systems had to reach the scram trip level before the scram was initiated in the analysis. The further away the scram-initiating detector is from the rod being withdrawn, the longer the duration of the transient before scram initiation. By assuming the furthest possible distance between the control rod and a scram-initiating IRM detector, the analysis was very conservative. The peak fuel enthalpy reported for this scenario was < 20 cal/gm.

SNC requested the NSSS vendor, GE, to review the NEDO-23842[1] analysis and consider more recent analyses performed for the BWR industry to qualitatively determine if the licensing basis criterion for this event is satisfied with only two IRMs operable per channel. The conclusion of GE's evaluation[2] is that operation with two operable IRMs per channel is technically justified based upon the following:

1. The prompt power burst in the event of an RWE transient in the startup range is mainly terminated by the Doppler feedback rather than a scram which has a slower response due to the instrument delay time.

2. The APRM system can initiate the reactor scram; therefore, there is no need for the IRMs as long as the 15% APRM trip is available.

3. In the past, GE performed a sensitivity analysis for a rod withdrawal transient in the startup range to determine whether the licensing basis criterion for fuel failure would be exceeded if the IRM system failed to provide a scram initiation signal. This analysis assumed that the transient was terminated by a scram based upon an APRM scram setpoint of either 15% or 120% of rated power. Evaluation results showed that the peak fuel enthalpies that result from the continuous withdrawal of an out-of-sequence control rod in the startup range (assuming failure of the IRM scram function with an APRM scram at 15% of rated power) are less than the licensing basis criterion of 170 cal/gm for control rod worths up to 2.5% Δk. Rod worths for out-of-sequence rods are not expected to be > 2.5% Δk during startup.

4. In the event the 15% APRM scram fails, two operable IRMs per trip channel are still available and are expected to initiate a delayed scram independent of the specific detector location in the core. This scram would occur well before the 170 cal/gm limit is reached for the postulated low power (non-limiting) event scenario.

At this point, presenting a summary of the defense-in-depth protection against this event is appropriate. First, either the RWM or a second operator should prevent the continuous rod withdrawal event from occurring if the operator moving the rod makes an error. SNC has committed to have the RWM and the second operator available to verify compliance with the withdrawal sequence. Assuming both these barriers fail, there is inherent protection limiting the prompt power burst from Doppler reactivity feedback. Also, the limiting event is terminated by a combination of Doppler reactivity and the APRM neutron flux scram that is "set down" in the startup range (Technical Specification Table 3.3.1.1-1, Function 2.a.) For the non-limiting event starting from low powers, the IRM scram terminates the continuous rod withdrawal event. However, a qualitative review[2] of the NEDO-23842[1] analysis concluded that the licensing basis criterion would not be exceeded even if the IRM scram signal was initiated by detectors located on the opposite side of the core from the rod being withdrawn. The evaluation also concluded that the IRM scram signal is not necessary to mitigate the event. The 170 cal/gm criterion could be met with the 15% APRM flux scram.

The Technical Specifications Bases also state that IRMs can mitigate cold water injection events during startup, although no credit is specifically assumed. Vessel subcooling or cold water

injection transients cause core-wide changes in reactor power. Two IRM channels per trip system will continue to effectively mitigate this type of event.

The control rod drop accident was analyzed using the methodology described in NEDE-24011-P-A-14,[3] which credits the APRM system for initiating a reactor scram.

The IRMs are an important monitoring system during startup. It is necessary to verify adequate overlap exists between source range, intermediate range, and power range monitors and that the operators are always provided with the necessary feedback on core reactivity. This function can be adequately accomplished by having one IRM per quadrant operable during plant startup. (Currently, having only one operable IRM per quadrant is allowed implicitly since one is allowed to be bypassed in each RPS trip system.) Therefore, the proposed Technical Specifications amendment request allows two IRM channels per trip system to be operable while specifying that at least one IRM per quadrant must be available.

The detailed discussion above supports the defense-in-depth design of reactivity controls and demonstrates that SNC's proposed change does not adversely affect plant safety.

References:

1. "Continuous Control Rod Withdrawal Transient in the Startup Range," GE Report NEDO-23842, April 1978.

2. GE Letter Report NSA 01-107, Nadar Sadeghi to Daryl Bouchie, "Plant Hatch IRM Technical Specification," dated March 8, 2001.

3. "General Electric Standard Application for Reactor Fuels," NEDE-24011-P-A-14 (and US Supplement), June 2000, as amended.

Enclosure 2

Edwin I. Hatch Nuclear Plant
Request to Revise Technical Specifications:
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10 CFR 50.92 No Significant Hazards Evaluation
and
Environmental Assessment

In 10 CFR 50.92(c), the Nuclear Regulatory Commission (NRC) provides the following standards to be used in determining the existence of a significant hazards consideration:

> ...a proposed amendment to an operating license for a facility licensed under §50.21(b) or §50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not: (1) Involve a significant increase in the probability or consequences of an 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.

Basis for no significant hazards consideration determination:

Southern Nuclear Operating Company (SNC) reviewed the proposed license amendment request and determined its adoption does not involve a significant hazards consideration based upon the following discussion:

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

   The intermediate range monitors (IRMs) monitor neutron flux levels in the reactor core during startup. The IRM detectors are capable of generating a trip signal during a continuous rod withdrawal error in the startup range. However, the IRMs perform no function related to the probability of occurrence of a previously evaluated accident. Also, the IRM trip signal is not necessary to mitigate the limiting control rod withdrawal error. The limiting case assumes the trip signal is generated from the safety-related average power range monitor (APRM). Therefore, the consequences of this previously evaluated abnormal operating transient are not increased.

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

   The proposed change reduces the number of required operable IRM channels per trip system from three to two. However, the manner in which the actuation logic functions and the systems respond are unaffected by the proposed change. Furthermore, the IRMs will continue to perform their design function of core monitoring during startup and mitigating non-limiting transient events that are postulated to occur during startup. Therefore, the

proposed change cannot create the possibility of a new or different kind of accident from any previously evaluated.

3. *Does the proposed change involve a significant reduction in a margin of safety?*

   The Bases for Unit 2 Technical Specifications Table 3.3.1.1-1 state that the "IRMs are capable of generating trip signals that can be used to prevent fuel damage resulting from abnormal operating transients in the intermediate power [startup] range." The proposed change ensures the IRMs will still effectively mitigate these events. The most significant source of reactivity change is due to a control rod withdrawal error. With the proposed change, the IRMs will continue to provide protection against rod withdrawal errors, and peak fuel energy depositions will remain below the 170 cal/gm threshold criterion defined in the Technical Specifications Bases. Therefore, the proposed change does not reduce a margin of safety.

Environmental Assessment

10 CFR 51.22(c)(9) provides criteria for identification of licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed license amendment would not:

1. Involve a significant hazards consideration;

2. Result in a significant change in the types of significant increase in the amounts of any effluents that may be released offsite;

3. Result in a significant increase in individual or cumulative occupational radiation exposure.

SNC has determined the proposed Technical Specifications change described in Enclosure 1 meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.229(c)(9). Accordingly, pursuant to 10 CFR 51.22, no environmental impact statement needs to be prepared in connection with the issuance of the license amendment for the proposed change. The basis for this determination using the above criteria follows:

1. As demonstrated in this enclosure, the proposed change does not involve a significant hazards consideration.

2. The proposed change does not result in a significant change to the types of effluents or in the amounts of effluents released offsite. The proposed change involves the required number of operable channels associated with instruments that monitor neutron flux during reactor startup. It does not involve changes to the radioactive waste processing systems or to radioactive waste effluent monitors. Accordingly, the proposed change does not require the radioactive waste processing systems to perform a function different from the function they are designed to perform, nor does the proposed change alter the operation or testing of any such system.

3. The proposed change does not result in a significant increase in occupational radiation exposure. Monitoring reactor neutron flux is performed by licensed reactor operators from the main control room. The number of operable channels is not related to occupational radiation exposure.

Enclosure 3

Edwin I. Hatch Nuclear Plant
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Page Change Instructions

Unit 2

Page	Replace
3.3-7	3.3-7

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitor					
a. Neutron Flux – High	2	3(d)	G	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 120/125 divisions of full scale
	5(a)	3(d)	H	SR 3.3.1.1.1 SR 3.3.1.1.5 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 120/125 divisions of full scale
b. Inop	2	3(d)	G	SR 3.3.1.1.4 SR 3.3.1.1.15	NA
	5(a)	3(d)	H	SR 3.3.1.1.5 SR 3.3.1.1.15	NA
2. Average Power Range Monitor					
a. Neutron Flux – High (Setdown)	2	3(c)	G	SR 3.3.1.1.1 SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.13	≤ 20% RTP
b. Simulated Thermal Power – High	1	3(c)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.13	≤ 0.58 W + 58% RTP and ≤ 115.5% RTP(b)
c. Neutron Flux – High	1	3(c)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.13	≤ 120% RTP
d. Inop	1,2	3(c)	G	SR 3.3.1.1.10	NA

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) 0.58 W + 58% - 0.58 ΔW RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

(c) Each APRM channel provides inputs to both trip systems.

(d) Only two channels required per trip system until the Fall 2001 refueling outage, provided one channel is operable in each quadrant of the core and both the RWM and a second licensed operator verify compliance with the withdrawal sequence.

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitor					
a. Neutron Flux – High	2	3(d)	G	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 120/125 divisions of full scale
	5(a)	3(d)	H	SR 3.3.1.1.1 SR 3.3.1.1.5 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 120/125 divisions of full scale
b. Inop	2	3(d)	G	SR 3.3.1.1.4 SR 3.3.1.1.15	NA
	5(a)	3(d)	H	SR 3.3.1.1.5 SR 3.3.1.1.15	NA
2. Average Power Range Monitor					
a. Neutron Flux – High (Setdown)	2	3(c)	G	SR 3.3.1.1.1 SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.13	≤ 20% RTP
b. Simulated Thermal Power – High	1	3(c)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.13	≤ 0.58 W + 58% RTP and ≤ 115.5% RTP(b)
c. Neutron Flux – High	1	3(c)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.13	≤ 120% RTP
d. Inop	1,2	3(c)	G	SR 3.3.1.1.10	NA

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) 0.58 W + 58% - 0.58 ΔW RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

(c) Each APRM channel provides inputs to both trip systems.

(d) Only 2 channels required per trip system until the fall 2001 refueling outage provided one channel is operable in each quadrant of the core and both the RWM and a second licensed operator verify compliance with the withdrawal sequence.