UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 21, 2010

Mr. Thomas Joyce
President and Chief Nuclear Officer
PSEG Nuclear
P.O. Box 236, N09
Hancocks Bridge, NJ 08038

SUBJECT: HOPE CREEK GENERATING STATION - ISSUANCE OF AMENDMENT RE: CONTROL ROD NOTCH TESTING FREQUENCY (TAC NO. ME2209)

Dear Mr. Joyce:

The Commission has issued the enclosed Amendment No. 182 to Facility Operating License No. NPF-57 for the Hope Creek Generating Station. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated September 9, 2009.

The amendment changes the frequency of control rod notch testing, as specified in TS surveillance requirement 4.1.3.1.2.a, from at least once per 7 days to at least once per 31 days. The amendment also adds the word "fully" to the Action for TS Limiting Condition for Operation 3.9.2 to clarify the requirement to fully insert all insertable control rods when the required source range monitor (SRM) instrumentation is inoperable. The proposed amendment is based on TS Task Force (TSTF) change, TSTF-475, Revision 1, "Control Rod Notch Testing Frequency and SRM Insert Control Rod Action."

A copy of our safety evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Richard B. Ennis, Senior Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-354

Enclosures:
1. Amendment No. 182 to
   License No. NPF-57
2. Safety Evaluation

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

PSEG NUCLEAR LLC

DOCKET NO. 50-354

HOPE CREEK GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 182
License No. NPF-57

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment filed by PSEG Nuclear LLC dated September 9, 2009, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-57 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 182, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into the license. PSEG Nuclear LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Harold K. Chernoff, Chief
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the License
and Technical Specifications

Date of Issuance: July 21, 2010

ATTACHMENT TO LICENSE AMENDMENT NO. 182

FACILITY OPERATING LICENSE NO. NPF-57

DOCKET NO. 50-354

Replace the following page of the Facility Operating License with the revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

Remove	Insert
3	3

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
3/4 1-5	3/4 1-5
3/4 5-3	3/4 5-3

(4) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(5) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(6) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

PSEG Nuclear LLC is authorized to operate the facility at reactor core power levels not in excess of 3840 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 182, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into the license. PSEG Nuclear LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Inservice Testing of Pumps and Valves (Section 3.9.6, SSER No. 4)*

This License Condition was satisfied as documented in the letter from W. R. Butler (NRC) to C. A. McNeill, Jr. (PSE&G) dated December 7, 1987. Accordingly, this condition has been deleted.

*The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

electrically or hydraulically shall be demonstrated OPERABLE by moving each control rod at least one notch:

a. At least once per 31 days, and

b. Within 24 hours when any control rod is immovable as a result of excessive friction or mechanical interference.

4.1.3.1.3 All control rods shall be demonstrated OPERABLE by performance of Surveillance Requirements 4.1.3.2, 4.1.3.4, 4.1.3.5, 4.1.3.6 and 4.1.3.7.

4.1.3.1.4 The scram discharge volume shall be determined OPERABLE by demonstrating:

a. The scram discharge volume drain and vent valves OPERABLE at least once per 18 months, by verifying that the drain and vent valves:

    1. Close within 30 seconds after receipt of a signal for control rods to scram, and

    2. Open when the scram signal is reset.

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 At least 2 source range monitor* (SRM) channels shall be OPERABLE and inserted to the normal operating level with:##

a. Annunciation and continuous visual indication in the control room,

b. One of the required SRM detectors located in the quadrant where CORE ALTERATIONS are being performed and the other required SRM detector located in an adjacent quadrant, and

c. Unless adequate shutdown margin has been demonstrated per Specification 3.1.1, the "shorting links" removed from the RPS circuitry prior to and during the time any control rod is withdrawn.#

d. During a SPIRAL UNLOAD, the count rate may drop below 3 cps when the number of assemblies remaining in the core drops to sixteen or less.

e. During a SPIRAL RELOAD, up to four fuel assemblies may be loaded in the four bundle locations immediately surrounding each of the four SRMs prior to obtaining 3 cps. Until these assemblies have been loaded, the 3 cps count rate is not required.

APPLICABILITY: OPERATIONAL CONDITION 5.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS and fully insert all insertable control rods.

SURVEILLANCE REQUIREMENTS

4.9.2 Each of the above required SRM channels shall be demonstrated OPERABLE by:

a. At least once per 12 hours:

1. Performance of a CHANNEL CHECK,

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* The use of special movable detectors during CORE ALTERATIONS in place of the normal SRM nuclear detectors is permissible as long as these special detectors are connected to the normal SRM circuits.

\# Not required for control rods removed per Specification 3.9.10.1 and 3.9.10.2.

\## Three SRM channels shall be OPERABLE for critical shutdown margin demonstrations. An SRM detector may be retracted provided a channel indication of at least 100 cps is maintained.

UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 182 TO FACILITY OPERATING LICENSE NO. NPF-57

PSEG NUCLEAR LLC

HOPE CREEK GENERATING STATION

DOCKET NO. 50-354

1.0 INTRODUCTION

By letter dated September 9, 2009 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML092600469), PSEG Nuclear, LLC (PSEG or the licensee) requested changes to the Technical Specifications (TSs) for the Hope Creek Generating Station (HCGS). The proposed amendment would change the frequency of control rod notch testing, as specified in TS surveillance requirement (SR) 4.1.3.1.2.a, from at least once per 7 days to at least once per 31 days. The purpose of this SR is to confirm control rod insertion capability which is demonstrated by inserting each partially or fully withdrawn control rod at least one notch and observing that the control rod moves. This ensures that the control rod is not stuck and is free to insert on a scram signal. The proposed amendment would also add the word "fully" to the Action for TS Limiting Condition for Operation (LCO) 3.9.2 to clarify the requirement to fully insert all insertable control rods when the required source range monitor (SRM) instrumentation is inoperable.

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (CFR), Part 50, Appendix A, General Design Criterion (GDC) 26 - "Reactivity control system redundancy and capability," states that:

> Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

Enclosure

GDC 29, "Protection against anticipated occurrences," states that:

> The protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in an event of anticipated operational occurrences.

Section 3.1 of the HCGS Updated Final Safety Analysis Report (UFSAR) discusses conformance of the HCGS design with the GDC in Appendix A of 10 CFR Part 50. With respect to GDC 26, UFSAR Section 3.1.2.3.7.1 states, in part, that the design of the control rod system includes appropriate margin for malfunctions, such as stuck rods, in the highly unlikely event they do occur. This section of the UFSAR also states that the unlikely occurrence of a limited number of rods stuck during a scram would not adversely affect the capability to maintain the core within fuel design limits. With respect to GDC 29, UFSAR Section 3.1.2.3.10.1 states that an extremely high reliability of timely response to anticipated operational occurrences is maintained by a thorough program of inservice testing and surveillance.

Section 50.36(c)(3) of 10 CFR Part 50 requires that TSs include SRs "relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met."

3.0 TECHNICAL EVALUATION

3.1 Background

Control rods are components of the control rod drive (CRD) system, which is the primary reactivity control system for the reactor. The CRD system, in conjunction with the Reactor Protection System, provides the means for reliable control of reactivity changes to ensure that under conditions of normal operation, including anticipated operational occurrences, specified fuel design limits are not exceeded. In addition, the control rods provide the capability to maintain the reactor core subcritical under all conditions and to limit the potential amount and rate of reactivity increase if there were to be a malfunction in the CRD system.

The CRD system consists of a CRD mechanism (CRDM) by which the control rods are moved, and a hydraulic control unit (HCU) for each control rod. The CRDM is a mechanical-hydraulic latching cylinder that positions the control blades. The CRDM is a highly reliable mechanism for inserting a control rod to the full-in position. The collet piston mechanism design feature ensures that the control rod will not be inadvertently withdrawn. This is accomplished by engaging the collet fingers, mounted on the collet piston, in notches located on the index tube. Due to the tapered design of the index tube notches, the collet piston mechanism will not impede rod insertion under normal insertion or scram conditions.

The collet retainer tube (CRT) is a short tube, welded to the upper end of the CRD, which houses the collet mechanism. The collet mechanism (which consists of the locking collet, collet piston, collet return spring and unlocking cam), provides the locking and unlocking functions that allow the insertion and withdrawal of the control rod. The CRT has three primary functions: (a) to carry the hydraulic unlocking pressure to the collet piston; (b) to provide an outer cylinder, with a suitable wear surface for the metal collet piston rings; and (c) to provide mechanical support for the guide cap, a component which incorporates the cam surface for holding the collet fingers open and also provides the upper rod guide or bushing.

3.2 TSTF-475, Revision 1

As discussed in the licensee's application dated September 9, 2009, the proposed amendment is based on Nuclear Regulatory Commission (NRC or the Commission) approved Technical Specification Task Force (TSTF) Standard Technical Specifications (STS) change traveler, TSTF-475, Revision 1, "Control Rod Notch Testing Frequency and SRM Insert Control Rod Action." A notice of availability for this TS improvement was published in the *Federal Register* on November 13, 2007 (72 FR 63935). The notice included a model safety evaluation (SE) that may be referenced by licensees in plant-specific applications to adopt the TSTF-475 changes.

TSTF-475 contains the following changes to the STS: (1) revise the TS control rod notch surveillance frequency for fully withdrawn control rods from 7 days to 31 days; (2) clarify the TS requirements for inserting control rods with one or more inoperable SRMs; and (3) clarify the applicability of the 1.25 surveillance test interval extension provision in STS SR 3.0.2. Due to differences between the STS and the HCGS TSs, the licensee stated in its application dated September 9, 2009, that the third change in TSTF-475 (i.e., clarify the applicability of the 1.25 surveillance test interval extension) is not applicable to HCGS. In addition, the licensee's application provided the following discussion to justify a variation from the first change contained in TSTF-475 (i.e., revise the TS control rod notch surveillance frequency):

> The revised notch testing frequency addressed in TSTF-475, Revision 1, is specific to fully withdrawn control rods since partially withdrawn control rods already have a 31-day test frequency in Standard Technical Specifications. Currently, HCGS TS requires partially and fully withdrawn control rods to be exercised at least once per seven days. The proposed amendment addresses changes to include "withdrawn control rods" to be inclusive of fully and partially withdrawn. The change is based upon STS and TSTF-475, Revision 1.
>
> The purpose of the surveillance to test partially withdrawn rods is the same as for fully withdrawn rods. It is performed in order to confirm control rod insertion capability. As discussed in the safety evaluation for TSTF-475, Revision 1, a stuck control rod is an extremely rare event. The proposed change for the surveillance frequency has been determined to be acceptable based upon the demonstrated historical reliability of the Control Rod Drive System. As discussed in the safety evaluation, monthly surveillances would still provide a large number of tests in order to provide confidence that any problems with the system would be identified. The industry operating experience is inclusive of notch testing for both partially and fully withdrawn control rods.

The licensee also stated in its application dated September 9, 2009, that it has reviewed the model SE for TSTF-475 and concluded that the justifications presented in the TSTF proposal and SE prepared by the NRC staff are applicable to HCGS and justify the proposed amendment.

As discussed in the model SE, the NRC staff approved the TSTF-475, Revision 1, proposal to revise the TS control rod notch surveillance frequency in the STS from 7 days to monthly based on a number of considerations including: (1) slow crack growth rate of the CRT; (2) improved CRT design; (3) a higher reliable method (scram time testing) to monitor CRD scram system functionality; and (4) no known CRD failures being detected during notch testing. The NRC staff concluded that this change would reduce the number of control rod manipulations, thereby reducing the opportunity for potential reactivity events while having a very minimal impact on the high reliability of the CRD system. Specifically, the NRC staff's model SE for TSTF-475, Revision 1 stated, in part, that:

> According to the BWROG [Boiling Water Reactor Owners Group], at the time of the first CRT crack discovery in 1975 each partially or fully withdrawn operable control rod was required to be exercised one notch at least once each week. It was recognized that notch testing provided a method to demonstrate the integrity of the CRT. Control rod insertion capability was demonstrated by inserting each partially or fully withdrawn control rod at least one notch and observing that the control rod moves. The control rod may then be returned to its original position. This ensures the control rod is not stuck and is free to insert on a scram signal.
>
> It was determined that during scrams, the CRT temperature distribution changes substantially at reactor operating conditions. Relatively cold water moves upward through the inside of the CRT and exits via the flow holes into the annulus on the outside. At the same time hot water from the reactor vessel flows downward on the outside surface of the CRT. There is very little mixing of the cold water flowing from the three flow holes into the annulus and the hot water flowing downward. Thus, there are substantial through wall and circumferential temperature gradients during scrams which contribute to the observed CRT cracking.
>
> Subsequently, many BWRs [boiling-water reactors] have reduced the frequency of notch testing for partially withdrawn control rods from weekly to monthly. The notch test frequency for fully withdrawn control rods are still performed weekly. The change, for partially withdrawn control rods, was made because of the potential power reduction required to allow control rod movement for partially withdrawn control rods, the desire to coordinate scheduling with other plant activities, and the fact that a large sample of control rods are still notch tested on the weekly basis. The operating experience related to the changes in CRD performance also provided additional justification to reduce the notch test frequency for the partially withdrawn control rods.
>
> In response to the NRC staff RAIs [requests for additional information] and to support their position to reduce the CRD notch testing frequency, the BWROG provided plant data and GE [General Electric] Nuclear Energy report, CRD Notching Surveillance Testing for Limerick Generating Station (CRDNST). The GE report provided a description of the cracks noted on the original design CRT surfaces. These cracks, which were later determined to be intergranular, were generally circumferential, and appeared with greatest frequency below and between the cooling water ports, in the area of the change in wall thickness.

Subsequently, cracks associated with residual stresses were also observed in the vicinity of the attachment weld. Continued circumferential cracking could lead to 360 degree severance of the CRT that would render the CRD inoperable which would prevent insertion, withdrawal or scram. Such failure would be detectable in any fully or partially withdrawn control rod during the surveillance notch testing required by the Technical Specifications. To a lesser degree, cracks have also been noted at the welded joint of the interim design CRT but no cracks have been observed in the final improved CRT design.

To date, operating experience data shows no reports of a severed CRT at any BWR. No collet housing failures have been noted since 1975. On a numerical basis for instance, based on [the] BWROG assumption that there are 137 control rods for a typical BWR/4 and 193 control rods for a typical BWR/6, the yearly performance would be 6590 rod notch tests for a BWR/4 plant and 9284 for a BWR/6 plant. For example, if all BWRs operating in the US are taken into consideration, the yearly performances of rod notch data would translate into approximately 240,000 rod notch tests without detecting a failure.

Also, the BWR scram system has extremely high reliability. In addition to notch testing, scram time testing can identify failure of individual CRD operation resulting from IGSCC [intergranular stress-corrosion cracking]-initiated cracks and mechanical binding. Unlike the CRD notch tests, these single rod scram tests cover the other mechanical components such as scram pilot solenoid operated valves, the scram inlet and outlet air operated valves, and the scram accumulator, as well as operation of the control rods. Thus, the primary assurance of scram system reliability is provided by the scram time testing since it monitors the system scram operation and the complete travel of the control rod.

Also, the HCUs, CRD drives, and control rods are also tested during refueling outages, approximately every 18-24 months. Based on the data collected during the preceding cycle of operation, selected control rod drives, are inspected and, as required, their internal components are replaced. Therefore, increasing the CRD notch testing frequency to monthly would have very minimal impact on the reliability of the scram system.

3.3 <u>Proposed Change to HCGS SR 4.1.3.1.2.a</u>

TSTF-475, Revision 1, is only applicable in decreasing the notch test surveillance frequency for fully withdrawn control rods, as many licensees have already decreased the surveillance frequency for partially withdrawn control rods (i.e., prior to TSTF-475). The proposed change for HCGS would apply to both partially withdrawn and fully withdrawn control rods. As discussed above in SE Section 3.2, the licensee has reviewed the model SE for TSTF-475 and concluded that it justifies the proposed amendment. The NRC staff has also reviewed the model SE and determined that the justifications for reducing control rod notch surveillance frequency are also applicable to partially withdrawn control rods.

Consistent with the discussion above in SE Section 3.2, the NRC staff concludes that the proposed change to HCGS SR 4.1.3.1.2.a will reduce the number of control rod manipulations, thereby reducing the opportunity for potential reactivity events while having a very minimal impact on the high reliability of the CRD system. Therefore, the NRC staff concludes that changing the notch test surveillance frequency from 7 days to 31 days for both partially withdrawn and fully withdrawn control rods is acceptable.

3.4 Proposed Change to HCGS LCO 3.9.2

HCGS LCO 3.9.2 provides the operability requirements for the SRMs. The LCO Action currently states that:

> With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS and insert all insertable control rods.

The proposed amendment would add the word "fully" before the word "insert." Consistent with the discussion in the model SE for TSTF-475, the current HCGS TS requirement to "insert all insertable control rods" is meant to require that the control rods be fully inserted. As such, adding the word "fully" does not change but clarifies the intent of the action. Therefore, the NRC staff concludes that this change is acceptable.

3.5 Technical Evaluation Conclusion

Based on the discussion in SEs Sections 3.1 through 3.4, the NRC staff concludes that the proposed amendment is acceptable.

PSEG's application dated September 9, 2009, provided proposed changes to the TS Bases to be implemented with the associated TS changes. These pages were provided for information only and will be revised in accordance with the HCGS TS Bases Control Program.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New Jersey State Official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes SRs. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (74 FR 62836). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: R. Grover
R. Ennis

Date: July 21, 2010

July 21, 2010

Mr. Thomas Joyce
President and Chief Nuclear Officer
PSEG Nuclear
P.O. Box 236, N09
Hancocks Bridge, NJ 08038

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A copy of our safety evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/ra/

Richard B. Ennis, Senior Project Manager
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Office of Nuclear Reactor Regulation

Docket No. 50-354

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2. Safety Evaluation

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