Southern Nuclear

Regulatory Affairs Director

3535 Colonnade Parkway
Birmingham, AL 35243
205-992-5316

April 20, 2021

Docket No.: 50-366 NL-21-0413

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

Edwin I. Hatch Nuclear Plant Unit 2
Supplement to Emergency License Amendment Request for Technical Specification 3.5.1
Regarding One-Time Extension of Completion Time for 2D RHR Pump

Ladies and Gentlemen:

Pursuant to the provisions of Section 50.90 of Title 10 of the Code of Federal Regulations (CFR), on April 19, 2021, Southern Nuclear Operating Company (SNC) submitted a license amendment request (LAR) to the Technical Specifications (TS) for Hatch Nuclear Plant (HNP) Unit 2 renewed facility operating license NPF-5. The proposed amendment (ADAMS Accession Number ML21109A388) would revise Condition A of TS 3.5.1, ECCS Operability, by allowing a one-time increase in the Completion Time from 7 days to 15 days. The increased Completion Time would expire on May 1, 2021 at 0955 eastern daylight time (EDT).

Subsequent to submittal, SNC identified discrepancies affecting Attachment 4, Evaluation of Risk Impact and Compensatory Measures, of the LAR. The error has been entered into SNC's corrective action program. SNC hereby supplements the request with a revised Attachment 4.

SNC requests approval of the proposed license amendment as soon as possible and no later than April 22, 2021 based on emergent circumstances at HNP Unit 2 in accordance with the provisions of 10 CFR 50.91(a)(5). A discussion of the emergency situation is provided in SNC's original request. The amendment, if approved, will be implemented immediately upon issuance.

This supplement has no impact on the no significant hazards consideration or the environmental considerations of the original submittal.

This letter contains no NRC commitments. If you have any questions, please contact Jamie Coleman at 205.992.6611.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 20th day of April 2021.

Respectfully submitted,

Jamie M. Coleman

Jamie M. Coleman
Licensing Manager
Southern Nuclear Operating Company

JMC/tle

Revised Attachment 4: Evaluation of Risk Impact and Compensatory Measures

cc: NRC Regional Administrator, Region II
NRC NRR Project Manager – Hatch
NRC Senior Resident Inspector – Hatch
Director, Environmental Protection Division - State of Georgia
RType: CHA02.004

Edwin I. Hatch Nuclear Plant Unit 2

Supplement to Emergency License Amendment Request for Technical Specification 3.5.1 Regarding One-Time Extension of Completion Time for 2D RHR Pump

Revised Attachment 4

Evaluation of Risk Impact and Compensatory Measures

1.0 INTRODUCTION

1.1 PURPOSE

The purpose of this analysis is to assess the acceptability, from a risk perspective, of a change to extend the Hatch completion time (CT) for Tech Spec Condition 3.5.1.A from 7 days to 15 days for Unit 2 in order to allow for repair of the 2D RHR Pump. These proposed changes are requested to be effective only during a one-time extension.

1.2 BACKGROUND

1.2.1 Technical Specification Changes

Since the mid-1980s, the NRC has been reviewing and granting improvements to TS that are based, at least in part, on probabilistic risk assessment (PRA) insights. In its final policy statement on TS improvements of July 22, 1993, the NRC stated that it . . .

> . . . expects that licensees, in preparing their Technical Specification related submittals, will utilize any plant-specific PSA or risk survey and any available literature on risk insights and PSAs. . . Similarly, the NRC staff will also employ risk insights and PSAs in evaluating Technical Specifications related submittals. Further, as a part of the Commission's ongoing program of improving Technical Specifications, it will continue to consider methods to make better use of risk and reliability information for defining future generic Technical Specification requirements.

The NRC reiterated this point when it issued the revision to 10 CFR 50.36, "Technical Specifications," in July 1995. In August 1995, the NRC adopted a final policy statement on the use of PRA methods in nuclear regulatory activities that encouraged greater use of PRA to improve safety decision-making and regulatory efficiency. The PRA policy statement included the following points:

1. The use of PRA technology should be increased in all regulatory matters to the extent supported by the state of the art in PRA methods and data and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy.
2. PRA and associated analyses (e.g., sensitivity studies, uncertainty analyses, and importance measures) should be used in regulatory matters, where practical within the bounds of the state of the art, to reduce unnecessary conservatism associated with current regulatory requirements.
3. PRA evaluations in support of regulatory decisions should be as realistic as practicable and appropriate supporting data should be publicly available for review.
4. The Commission's safety goals and subsidiary numerical objectives are to be used with consideration of uncertainties in making regulatory judgments…

The movement of the NRC to more risk-informed regulation has led to the NRC identifying Regulatory Guides and associated processes by which licensees can submit changes to the plant design basis including Technical Specifications. These guides are discussed in the following section.

1.3 REGULATORY GUIDES

Three Regulatory Guides provide primary inputs to the evaluation of a Technical Specification change. Their relevance is discussed in this section.

1.3.1 Regulatory Guide 1.200, Revision 3

Regulatory Guide 1.200, Revision 3 [Ref. 1] describes an acceptable approach for determining whether the quality of the PRA, in total or the parts that are used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decision-making for light-water reactors. This guidance is intended to be consistent with the NRC's PRA Policy Statement and more detailed guidance in Regulatory Guide 1.174.

It is noted that RG 1.200, Revision 3 endorses Addendum A of the ASME/ANS PRA Standard [Ref. 4] as clarified in Appendix A of RG 1.200, Revision 3.

1.3.2 Regulatory Guide 1.174, Revision 3

Regulatory Guide 1.174 [Ref. 2] specifies an approach and acceptance guidelines for use of PRA in risk informed activities. RG 1.174 outlines PRA related acceptance guidelines for use of PRA metrics of Core Damage Frequency (CDF) and Large Early Release Frequency (LERF) for the evaluation of permanent TS changes. The guidelines given in RG 1.174 for determining what constitutes an acceptable permanent change specify that the ΔCDF and the ΔLERF associated with the change should be less than specified values, which are dependent on the baseline CDF and LERF, respectively.

RG 1.174 also specifies guidelines for consideration of external events. External events can be evaluated in either a qualitative or quantitative manner.

Since this LAR is for a one-time TS change, the ΔCDF and the ΔLERF of RG 1.174 do not specifically apply.

1.3.3 Regulatory Guide 1.177 Revision 2

Regulatory Guide 1.177 [Ref. 3] specifies a risk-informed approach and acceptance guidelines for the evaluation of plant technical specification changes. RG 1.177 identifies a three-tiered approach for the evaluation of the risk associated with a proposed TS change as identified below:

- Tier 1 is an evaluation of the plant-specific risk associated with the proposed TS change, as shown by the change in core damage frequency (CDF) and incremental conditional core damage probability (ICCDP). Where applicable, containment performance should be evaluated on the basis of an analysis of large early release frequency (LERF) and incremental conditional large early release probability (ICLERP). The acceptance guidelines given in RG 1.177 for determining an acceptable permanent TS change are that the

ICCDP and the ICLERP associated with the change should be less than 1E-06 and 1E-07, respectively. RG 1.177 also addresses risk metric requirements for one-time TS changes, as outlined in Section 1.3.4 (Acceptance Guidelines) of this risk assessment.

- Tier 2 identifies and evaluates, with respect to defense-in-depth, any potential risk-significant plant equipment outage configurations associated with the proposed change. The licensee should provide reasonable assurance that risk-significant plant equipment outage configurations will not occur when equipment associated with the proposed TS change is out-of-service.
- Tier 3 provides for the establishment of an overall configuration risk management program (CRMP) and confirmation that its insights are incorporated into the decision-making process before taking equipment out-of-service prior to or during the CT. Compared with Tier 2, Tier 3 provides additional coverage based on any additional risk significant configurations that may be encountered during maintenance scheduling over extended periods of plant operation. Tier 3 guidance can be satisfied by the Maintenance Rule (10 CFR 50.65(a)(4)), which requires a licensee to assess and manage the increase in risk that may result from activities such as surveillance, testing, and corrective and preventive maintenance.

This risk analysis supports the three tiers of RG 1.177, specifically the comparison of the results with the acceptance guidelines for ICCDP and ICLERP associated with changing a Technical Specification Completion Time, the assessment of risk-significant combinations, and the use of the Configuration Risk Management Program.

1.3.4 Acceptance Guidelines

Risk significance in a LAR is determined by comparison of changes in Core Damage Frequency (CDF) and Large Early Release Frequency (LERF) and values of Incremental Conditional Core Damage Probability (ICCDP) and Incremental Conditional Large Early Release Probability (ICLERP) produced by a permanent change to either the plant design basis or Technical Specifications to the guidelines given in Regulatory Guide 1.174 and Regulatory Guide 1.177. Reg. Guide 1.174 specifies the acceptable changes in CDF and LERF for permanent changes. Reg. Guide 1.177 specifies the acceptable ICCDP and ICLERP for permanent changes, usually associated with changing CT.

Reg. Guide 1.177 directly addresses the risk metric requirements for one-time TS changes, as reproduced below:

> *"For one-time only changes to TS CTs, the frequency of entry into the CT may be known, and the configuration of the plant SSCs may be established. Further, there is no permanent change to the plant CDF or LERF, and hence the risk guidelines of Regulatory Guide 1.174 cannot be applied directly. The following TS acceptance guidelines specific to one-time only CT changes are provided for evaluating the risk associated with the revised CT:*

1. *The licensee has demonstrated that implementation of the one-time only*

   TS CT change impact on plant risk from implementing the one-time only TS CT change is acceptable (Tier 1):

   - *An ICCDP of less than 1.0x10^{-6} and an ICLERP of less than 1.0x10^{-7}, or*
   - *An ICCDP of less than 1.0x10^{-5} and an ICLERP of less than 1.0x10^{-6} with effective compensatory measures implemented to reduce the sources of increased risk.*

2. *The licensee has demonstrated that there are appropriate restrictions on dominant risk-significant configurations associated with the change (Tier 2).*
3. *The licensee has implemented a risk-informed plant configuration control program. The licensee has implemented procedures to utilize, maintain, and control such a program (Tier 3)."*

Based on the available quantitative guidelines for other risk-informed applications, it is judged that the quantitative criteria shown in Table 1-1 represent a reasonable set of acceptance guidelines. For the purposes of this evaluation, these guidelines demonstrate that the risk impacts are acceptably low. This, combined with effective compensatory measures to maintain lower risk, will ensure that the TS change meets the intent of small risk increases consistent with the Commission's Safety Goal Policy Statement.

Table 1-1

PROPOSED RISK ACCEPTANCE GUIDELINES

RISK ACCEPTANCE GUIDELINE	BASIS
ICCDP < 1E-6, or ICCDP < 1E-5 with effective compensatory measures implemented to reduce the sources of increased risk	ICCDP is an appropriate metric for assessing risk impacts of out of service equipment per RG 1.177. This guideline is specified in Section 2.4 of RG 1.177.
ICLERP < 1E-7, or ICLERP < 1E-6 with effective compensatory measures implemented to reduce the sources of increased risk	ICLERP is an appropriate metric for assessing risk impacts of out of service equipment per RG 1.177. This guideline is specified in Section 2.4 of RG 1.177.

1.4 SCOPE

This section addresses the requirements of RG 1.200, Revision 3 Section 3.1 which directs the licensee to define the treatment of the scope of risk contributors (i.e., internal initiating events, external initiating events, and modes of power operation at the time of the initiator). Discussion of these risk contributors are as follows:

- Internal Events (IE) – The Hatch PRA model used for this analysis includes a full range of internal initiating events for at-power configurations. The IE model is further discussed in Section 1.5.
- Internal Flooding (IF) – The Hatch PRA model used for this analysis includes flooding scenarios. The IF model is further discussed in Section 1.5.
- Low Power Operation – The intent is for the unit to remain at power during the completion time. Since the RHR is used for shutdown cooling, there is some risk involved with going into lower modes; however, that is not quantified or discussed any further in this assessment.
- Shutdown / Refueling – Hatch does not have a shutdown PRA model, but instead relies upon deterministic methodology to assess defense-in-depth of key safety functions. The intent is for the unit to remain at-power for the duration of the extended CT. Hatch TS 3.4.7, 3.4.8 and 3.9.7 have separate requirements associated with shutdown cooling when the unit is not online. Note that RHR is the key heat removal method in shutdown.
- Internal Fires – The Hatch PRA model used for this analysis contains an as-built, as-operated Fire PRA model. The Fire PRA model is discussed further in Section 1.5.
- Seismic - Hatch has a Seismic PRA. However, the results are not sensitive to maintenance on the RHR pump and therefore the quantified results are not used. The sensitivity is further discussed in Section 1.5.
- Other External Events - Other external event risks (including external flooding and high winds) were assessed in the Hatch Other External Events Screening calculation [Ref. 6] and screened from the PRA.

1.5 Hatch PRA MODELS

This section addresses the requirements of Section 3.1 of RG 1.200, Revision 3 [Ref. 1] which directs the licensee to identify the portions of the PRA used in the analysis.

The PRA analysis uses the Rev. 8 Phoenix One-Top Multi-Hazard Model contained in SNC calculation RIE-PHOENIX-U02 [Ref. 5]. This model has the required quantification and support files set up to calculate either zero-maintenance or average maintenance risks. It also implements several model enhancements identified during PHOENIX development and therefore represents the most accurate model of record available. As described in RG 1.177, subsequent issues identified with the model would most likely impact the base and configuration specific models equally, therefore the delta risk calculations for a one-time TS change should not be impacted. If a permanent change were being requested, model issues could impact the overall CDF and LERF and would need to be addressed further.

The Revision 8 Phoenix OTMHM model of record contains internal events, internal flooding, internal fire and seismic hazards. All other hazards screened out as being very low risk. The model can be evaluated one hazard at a time or with all hazards activated. Each hazard model has been peer reviewed against the ASME peer review standard, and all of the F&Os have been closed. A review of the quantification and uncertainty notebooks for each hazard model did not find any assumption or uncertainty that would impact the results of this evaluation.

2.0 RISK ANALYSIS

This section evaluates the plant-specific risk associated with the proposed TS change, based on the risk metrics of CDF, ICCDP, LERF, and ICLERP.

2.1 ASSESSMENT OVERVIEW AND ASSUMPTIONS

2.1.1 <u>Overview</u>

This analysis is performed for unavailability of 2D RHR Pump. The PRA analysis involves identifying the system and components or maintenance activities modeled in the PRA which are most appropriate for use in representing the extended CT configurations and comparing the results to the baseline. Table 2.1-1 lists the base risk metrics for the Full Power Internal Events (FPIE) PRA, internal flooding PRA, and the Fire PRA (FPRA).

Table 2.1-1

HATCH CDF AND LERF BASE RISK METRICS

Hazard(s)	Risk (1/yr)
OTMHM CDF	5.78E-05
OTMHM LERF	3.76E-06

The general configuration for the extended CT is Hatch at-power on both units with the 2D RHR Pump out of service. The risk impact is for Unit 2. The planned maintenance is expected to focus on repairing the pump within the requested extended CT. Concurrent maintenance work will be carefully managed during the extended CT, through the use of the Configuration Risk Management Program.

The PRA model was quantified using the base "average test and maintenance" PRA model with the 2D RHR Pump maintenance and random failure events set to 1.0. The average test and maintenance model represent baseline assumed maintenance frequencies for all components except for Technical Specification violations that are normally excluded in the disallowed maintenance (mutually exclusive) logic in the base PRA model. As a conservative measure, maintenance events for equipment that is protected per site processes during the RHR 2D outage were left at their normal values. Adjustments for common cause factors associated with the RHR 2D pump are also included.

Table 2.1-2

EXTENDED CT CONFIGURATION REPRESENTATION

BASIC EVENT / GATE	DESCRIPTION	VALUE
CC-RS-4_U2	1/4, P4SS2E11C002D	1.0
CC-RS-49_U2	1/4, P4SR2E11C002D	1.0
MNUNRS_TRNB_U2	2D RHR Pump Maintenance	T
CC-RS-7_U2	2/4, P4SS22E11C0022C P4SS22E11C0022D	8.92E-03
CC-RS-9_U2	2/4, P4SS2E11C0022D P4SS1E11C002A	8.92E-03
CC-RS-10_U2	2/4, P4SS1E11C002D P4SS1E11C002B	8.92E-03
CC-RS-12_U2	3/4, P4SS2E11C002C P4SS2E11C002D P4SS2E11C002A	1.05E-03
CC-RS_13_U2	3/4, P4SS2E11C002B P4SS2E11C002D P4SS2E11C002C	1.05E-03
CC-RS-14_U2	3/4, P4SS2E11C002D P4SS2E11C002A P4SS2E11C002B	1.05E-03
CC-RS-15_U2	4/4, P4SS2E11C002B P4SS2E11C002D P4SS2E11C002A P4SS2E11C002C	1.67E-03
CC-RS-52_U2	2/4, P4SR2E11C002C P4SR2E11C002D	5.10E-03
CC- RS-54_U2	2/4, P4SR2E11C002A P8R2E11C002D	5.10E-03
CC- RS-55_U2	2/4, P4SR2E11C002D P4SR2E11C002B	5.10E-03
CC- RS-58_U2	3/4, P4SR2E11C002C P4SR2E11C002D P4SR2E11C002B	9.26E-04
CC- RS-57_U2	3/4, P4SR2E11C002C P4SR2E11C002A P4SR2E11C002D	9.26E-04
CC- RS-59_U2	3/4, P4SR2E11C002A P4SR2E11C002D P4SR2E11C002B	9.26E-04
CC- RS-60_U2	4/4, P4SR2E11C002C P4SR2E11C002A P4SR2E11C002D P4SR2E11C002B	1.71E-03

2.1.2 <u>Quantification Truncation</u>

To address limitations with each hazard requiring a different truncation level, each hazard was calculated individually and then combined into a single aggregated results file. To generate both the base and case risk, each hazard was quantified at the truncation levels below, to ensure that the basic events for the 2D RHR pump were present.
Internal Events CDF – 1E-12
Internal Events LERF – 1E-12

Internal Flooding CDF – 1E-12
Internal Flooding LERF – 1E-14
Internal Fire CDF – 1E-11
Internal Fire LERF – 1E-12
Seismic CDF – 1E-11
Seismic LERF – 1E-11

2.1.3 Calculation Approach

The proposed technical specification change involves unavailability of the 2D RHR Pump. The revised CDF and LERF values for the CT configurations are obtained by re-quantifying the base PRA model with all of the identified events set as shown in Table 2.1-2.

The evaluation of ICCDP and ICLERP for this condition is determined as shown below: The ICCDP associated with RHR Pump 2D OOS for a new CT is given by:

$$ICCDP_{2d} = (CDF_{2d} - CDF_{BASE}) \times CT_{NEW} \qquad [\textit{Eq. 2-1}]$$

where

CDF_{2d} = the annual average CDF calculated with RHR Pump 2D OOS and other currently OOS equipment assuming the configuration listed in Table 2.1-2 (all quantified hazards)

CDF_{BASE} = baseline annual average CDF with average unavailability for all equipment. This is the CDF result of the baseline PRA (all quantified hazards). Currently OOS equipment other than the RHR Pump 2d was not included in the base case values.

CT_{NEW} = the new extended CT (in units of years)

Note: ICCDP is a dimensionless probability.

Risk significance relative to ICLERP is determined using equations of the same form as noted above for ICCDP.

Since this evaluation is for a one-time Tech Spec CT allowance, the ICCDP and ICLERP are the only meaningful metrics as there is no permanent change in plant risk after this one-time CT extension.

2.1.4 Common Cause Adjustments

RG 1.177 contains specific directions on adjusting the CCF events in a model due to a failed component. The RHR pump failure events are in common cause groups of four for both failure on demand and failure to run. Thus, the common cause events that contain pump D have to be changed to the alpha values for that combination. The common cause basic event values are calculated in Cafta using the formula Qt*CCFM*1 for failure to start and Qt*CCFM*24 hours for failure to run. Qt times the demands or run hours is the overall random failure to start or failure to run probability. CCFM is the common cause multiplier based on the alpha method. The values were obtained from the Hatch data calculation H-RIE-IEIF-U00-007 attachment 4.

For RHR pump failure to start

CCFM, 2 of 4 FTS = 8.92E-03. The three basic events for 2/4 containing pump 2D are increased from 8.232E-06 to 8.92E-03.

CCFM, 3 of 4 FTS = 1.05E-03. The three basic events for 3/4 containing pump 2D are increased from 9.63E-07 to 1.05E-03.

CCFM, 4 of 4 FTS = 1.67E-03. The basic event for 4/4 pumps was increased from 1.54E-06 to 1.67E-03.

For RHR pump failure to run

CCFM, 2 of 4 FTR = 5.10E-03. The three basic events for 2/4 containing pump 2D are increased from 1.56E-06 to 5.10E-03.

CCFM, 3 of 4 FTR = 9.26E-04. The three basic events for 3/4 containing pump 2D are increased from 2.85E-06 to 9.26E-04.

CCFM, 4 of 4 FTR = 1.71E-03. The basic event for 4/4 pumps was increased from 5.25E-07 to 1.71E-03.

2.2 OTMHM Quantification

The relevant inputs from the PRA models to Equation 2-1 (and the equivalent for LERF) are shown in Table 2.2-1 below.

Table 2.2-1

OTMHM RISK ASSESSMENT INPUT PARAMETERS AND RESULTS FOR UNIT 2

Input Parameter	**Value**
CDF_{BASE}	5.78E-05
CDF_{2D}	8.63E-05
Delta CDF	2.85E-05
ICCDP for 15 day LCO	**1.17E-06**
$LERF_{BASE}$	3.76E-06
$LERF_{2D}$	3.83E-06
Delta LERF	7.00E-08
ICLERP for 15 day LCO	**2.88E-09**

<u>Compensatory Measures Discussion</u>

Risk insights from this configuration were examined by comparing the change in Birnbaum values between the base and configuration specific importance rankings. The events and components that become more important are associated with the redundant ECCS pumps and the containment hardened vent system, which is an alternate heat sink if RHR is not available.

RG 1.177 requires a Tier 2 examination of other components that, in combination with the component already out of service, could result in a risk significant configuration. For the RHR 2D pump, the components below had risk increases greater than a factor of three.

Components that become more significant due to the configuration:

Item	Base RAW	Case RAW	Factor Increase
2P52F1171	2.17	17.82556	8.219726
2P52F1187	2.17	17.82556	8.219726
2T48F082	2.76	20.29294	7.364984
2T48F081	2.78	20.39179	7.337937
2R25S067	4.67	21.84714	4.680566
2R25S006	3.48	13.86468	3.988584
2T48D346	14.33	46.91086	3.273964

The above components do not normally have maintenance performed on them while the plant is at power, so no additional restrictions are needed.

RG 1.177 also requires a Tier 3 examination of the (a)(4) Maintenance Rule configuration risk impact. The 2D RHR pump was input into the on-line configuration risk management (CRM) program as a “what-if” evaluation. The Hatch CRM program calculates both the instantaneous and integrated risk and CRM risk levels are based on integrated risk levels. The components already out of service prior to the RHR pump discovery were left out of service for this evaluation to ensure the calculation is conservative. With the ‘B’ pump out of service, the increase in risk is minimal. The CRM program uses the same hazard models that were used for this evaluation, and since the a(4) process evaluates planned work as well as current configurations, it will identify any potential high-risk conditions during the extended CT. The a(4) process of assessing and managing that risk will adequately control the evolution and risk management actions will be generated as necessary.

2.3 EXTERNAL EVENTS

2.3.1 Assessment of Relevant Hazard Groups

The purpose of this portion of the assessment is to evaluate the spectrum of external event challenges to determine which external event hazards should be explicitly addressed as part of the Condition 3.7.1.A extension risk assessment.

Internal events, internal flooding, internal fires, and seismic are quantitatively addressed as described in the previous sections.

The impact due to seismic and other hazard groups are addressed here. It is noted that it is unnecessary to evaluate the low-power and shutdown contribution to the base CDF and LERF since the change being proposed involves performance of the repair while at-power. It should be noted that use of the RHR Pumps is required for shutdown cooling and failure of a pump necessitates entry into other LCOs during shutdown.

.

2.3.3 Other External Hazards Evaluation and Conclusions

A plant-specific evaluation of an extensive set of other external hazards (including high winds and external flooding) was performed in SNC calculation H-RIE-OEE-U00 [Ref. 6]. The results have been previously submitted to the NRC for the Hatch 50.69 license amendment request (LAR) (ADAMS Accession Number ML18158A583) and subsequent RAI responses (ML19197A097).

That evaluation has been performed using the criteria in ASME PRA Standard RA-Sa-2009 and concluded that all other external hazards can be screened from applicability at Hatch. Therefore, there is no significant other external hazards risk contribution for this application.

2.4 RESULTS COMPARISON TO ACCEPTANCE GUIDELINES

The results indicate a one-time extension up to 15 days would not exceed the ICCDP and ICLERP risk limits. Additional compensatory measures would potentially reduce risk further, such as protected equipment and abstaining from entry into the diesel generator 14-day LCO or other activities that impact diesel generator availability. The additional compensatory measures are not accounted for in the quantification.

2.5 UNCERTAINTY ASSESSMENT

The purpose of this section is to disposition the impact of Probabilistic Risk Assessment (PRA) modeling epistemic uncertainty for Condition 3.5.1.A CT extension assessment. The baseline internal events PRA, internal flooding PRA, and fire PRA (FPRA) models document assumptions and sources of uncertainty and these were reviewed during the model peer reviews. The approach taken is, therefore, to review these documents to identify the items which may be directly relevant to Condition 3.5.1.A CT extension assessment, discuss the results, and to provide dispositions. As discussed in Section 1.5, no key assumptions or sources of uncertainty were identified that uniquely impact this application.

3.0 TECHNICAL ADEQUACY OF PRA MODEL

This section provides information on the technical adequacy of the Hatch Nuclear Plant Probabilistic Risk Assessment (PRA) models. The Hatch PRA maintenance and update processes and technical capability evaluations provide a robust basis for concluding that the PRA is suitable for use in risk-informed licensing actions, specifically in support of the requested extended CT for TS Condition 3.5.1.A. The OTMHM is comprised of various hazards PRA models that can be quantified simultaneously or individually. Each hazard (internal events, internal flooding, fire, and seismic) has been peer reviewed. The most up-to-date assessments of PRA technical adequacy (including peer review status, F&O closure status, scope, fidelity, capability, and maintenance/update practices) was provided to the NRC previously for the Hatch 50.69 LAR (ADAMS Accession Number ML18158A583) and subsequent RAI responses (ML19197A097); and also the NFPA-805 LAR (ML18096A955) and subsequent RAI responses (ML19280C812). Additionally, those submittals contain the most up-to-date description of the other external hazards assessment.

4.0 SUMMARY AND CONCLUSIONS

This analysis evaluates the acceptability, from a risk perspective, of a change to the Hatch Unit 2 TS Condition 3.5.1.A for a one-time increase of the CT from 7 days to 15 days when the RHR Pump 2D is inoperable.

The analysis examines a range of risk contributors including internal events, internal flooding, fire, seismic, shutdown risk and other external hazards. The configuration was quantified using the Phoenix OTMHM model and compared to the base risk to obtain delta CDF and LERF values.

4.1 PRA QUALITY

The PRA quality has been assessed and determined to be adequate for this risk application, and the PRA technical adequacy has also been addressed in recent NRC submittals.

To summarize,

- Scope – Hatch PRA modeling is highly detailed, including a wide variety of initiating events, modeled systems, operator actions, and common cause events. The PRA has the necessary scope to appropriately assess the pertinent risk contributors.
- Fidelity –The Hatch PRA models are the most recent evaluation of the risk profile. The PRA reflects the as-built, as-operated plant, with the exception of previously noted items.
- Standards – The PRA has been reviewed against the ASME/ANS PRA Standard and the PRA elements are shown to have the necessary attributes to assess risk for this application.
- Peer Review - The PRA has received a peer review. Based on addressing the peer review results and subsequent gap analyses to the current standards, the PRA is found to have the necessary attributes to assess risk for this application.
- Appropriate Quality – The PRA quality is found to be appropriate to assess risk for this application.

4.2 QUANTITATIVE RESULTS VS. ACCEPTANCE GUIDELINES

This analysis demonstrates with reasonable assurance that the proposed TS change is within the current risk acceptance guidelines in RG 1.177 for one- time changes.

4.3 CONCLUSIONS

This analysis demonstrates the acceptability, from a risk perspective, of a change to the Hatch TS Condition 3.5.1.A to increase the CT from 7 days to 15 days when the RHR Pump 2D is unavailable.

A PRA technical adequacy evaluation was also performed consistent with the requirements of ASME/ANS PRA Standard and RG 1.200, Revision 2. Additionally, a review of model uncertainty and outstanding changes was performed with this application. None of the identified sources of uncertainty were significant enough to change the conclusions from the risk assessment results presented here.

5.0 REFERENCES

[1] Regulatory Guide 1.200, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk Informed Activities,” Revision 3, December 2020.

[2] Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk- Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 3, January 2018.

[3] Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," Revision 2, January 2021.

[4] ASME/ANS RA- Sa-2009, February 2009. “Addenda to RA-S-2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications,”

[5] H-RIE-PHOENIX-U01, “One Top Model for PHOENIX Configuration Risk Management Program.”

[6] H-RIE-OEE-U00 – “Hatch Other External Events Screening”

[7] PRA-BC-H-21-003 – “RHR Pump 2D Emergent Technical Specification Change”