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GNRO-2011/00019

March 22, 2011

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: Request for Additional Information Regarding
Extended Power Uprate
Grand Gulf Nuclear Station, Unit 1
Docket No. 50-416
License No. NPF-29

REFERENCES:
1. Email from A. Wang to F. Burford dated February 23, 2011 GG EPU PRA Request for Additional Information Probabilistic Risk Assessment Licensing Branch (ME4639) (Accession Number ML110540712)
2. License Amendment Request, Extended Power Uprate, dated September 8, 2010 (GNRO-2010/00056, Accession Number ML102660403)

Dear Sir or Madam:

The Nuclear Regulatory Commission (NRC) requested additional information (Reference 1) regarding certain aspects of the Grand Gulf Nuclear Station, Unit 1 (GGNS) Extended Power Uprate (EPU) License Amendment Request (LAR) (Reference 2). Attachment 1 provides responses to the additional information requested by Probabilistic Risk Assessment Licensing Branch.

No change is needed to the no significant hazards consideration included in the initial LAR (Reference 2) as a result of the additional information provided. There are no new commitments included in this letter.

If you have any questions or require additional information, please contact Jerry Burford at 601-368-5755.

I declare under penalty of perjury that the foregoing is true and correct. Executed on March 22, 2011.

Sincerely,

MAK/FGB/dm

Attachments:

1. Response to Request for Additional Information, Probabilistic Risk Assessment Licensing Branch

cc: Mr. Elmo E. Collins, Jr.
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Attachment 1

GNRO-2011/00019

Grand Gulf Nuclear Station Extended Power Uprate

Response to Request for Additional Information

Probabilistic Risk Assessment Licensing Branch

Response to Request for Additional Information
Probabilistic Risk Assessment Licensing Branch

By letter dated September 8, 2010, Entergy Operations, Inc. (Entergy) submitted a license amendment request (LAR) for an Extended Power Uprate (EPU) for Grand Gulf Nuclear Station, Unit 1 (GGNS). The U.S. Nuclear Regulatory Commission (NRC) staff by correspondence dated February 23, 2011 (Accession Number ML110540712) has determined that the following additional information related to Probabilistic Risk Assessment Licensing Branch is needed for the NRC staff to complete their review of the amendment. Entergy's response to each item is also provided below.

RAI # 1

Many of the evaluations used to determine success criteria for different systems used the modular accident analysis program (MAAP) software. The submittal states that a file was used that contain workarounds for the latest MAAP Part 21 errors that have been identified for MAAP versions 4.0.6 and 4.0.7. MAAP5 is the latest version of MAAP and includes numerous upgrades to the code. Please justify how any inadequacies related to the applicable software version impacts the timings and success criteria evaluation listed in Attachment 13, Appendix E.

Response

The Modular Accident Analysis Program (MAAP) is the most widely used severe accident analysis tool in the world. The code incorporates 30 years of severe accident research and development and is supported by EPRI with a users group of over 50 organizations. Numerous comparisons to integral and separate effects tests have been performed along with comparisons to a variety of other analysis tools. Use of the code is supported by a detailed Applications Guidance Document developed by EPRI and provided to the NRC. Various NRC staff members have participated in discussions over the Applications Guidance Document and have provided favorable feedback. The current Users Manual along with the Applications Guidance Document includes detailed descriptions of benchmarking calculations. The Applications Guidance Document clearly describes code limitations and provides the users with appropriate methods to address the model limitations.

As with all software codes, MAAP users and EPRI identify limitations or errors in the code between code revisions and in the course of applying the software. These issues are investigated by EPRI and users are provided with appropriate workarounds to address such issues until a corrected version of the code can be released. The Part 21 error mentioned in this RAI is summarized in the bullets below:

- In December 2009 EPRI performed a Part 21 reportability evaluation of identified software errors in the MAAP4 software code and concluded that the safety hazard of the errors is indeterminate due to a lack of detailed information on how the software has actually been applied on a plant specific basis. Accordingly, EPRI, under 10 CFR 21 § 21.2(b), notified licensees so that the purchasers or affected licensees could evaluate these errors pursuant to 10 CFR 21 § 21.2(a).

- There are two (2) separate issues identified in this notification. These involve: 1) potential for an under-prediction of break flow in some LOCA analyses; and 2) incorrect containment response when the HPCI and/or RCIC turbine systems are operating. As noted by EPRI, the errors only apply to BWR MAAP versions 4.0.6 and 4.0.7 and both errors are to be corrected in version 4.0.8.

- All BWR MAAP4 analyses using version 4.0.6 or 4.0.7 are potentially impacted. A detailed description of the error along with a verified work-around was provided to all MAAP users by EPRI.
- These types of code errors have occurred at a very low frequency. The MAAP code is maintained under a very strict Appendix B QA program including an independent assessment of all code changes by a 3rd party reviewer.

MAAP Version 4.0.6 was used for both the GGNS pre-EPU and EPU calculations. The workaround provided to users by EPRI included the specific lines of code to incorporate into MAAP input decks or INCLUDE files. GGNS has correctly implemented the EPRI-supplied workaround (in a MAAP "INCLUDE" file) in both the pre-EPU and EPU MAAP calculations.

MAAP5 has been developed by EPRI and represents advances in modeling. The majority of changes to MAAP4 involve enhancements to the PWR primary system thermal-hydraulic model. The MAAP Users Group has created a transition plan for all users to begin development of MAAP5 parameter files. Testing of the code is ongoing and it is anticipated that the users will complete their transition over the next 3-4 years.

RAI # 2

The submittal states in Attachment 13 page 5:

> The GGNS PRA Human Reliability Analysis utilizes two methods to calculate the human error probabilities (HEP): HCR/ORE correlation and the Caused-Based approach. The Cause-Based method is not affected by allowable operation action time. The method used is determined by choosing the highest probability from the two methods.

Table 4.1-11 lists HEPs that have significant reduction in allowable operator action times and were calculated using the cause-based approach. By using the cause-based approach, decreases in allowable operation action time did not change the HEP probability. Since the delta risk assessment for EPU is highly sensitive to HEP due to decreased operator response time, please explain the applicability of using a methodology that is not sensitive to operator response times. For those HEPs evaluated by the cause based approach that have a decrease in operator action time post-EPU, please confirm that the HCR/ORE correlation method produced a less conservative result.

Response

The statement quoted from the GGNS EPU report was not intended to convey that the Cause-Based Decision Tree (CBDT) human reliability analysis (HRA) method is not influenced by changes in timing. The CBDT method is influenced by changes in operator action timing windows but over broader time frames compared to the HCR/ORE method. As such, the GGNS PRA uses both the HCR/ORE and CBDT methods when calculating individual human error probabilities (HEPs) and then employs in the PRA models the higher HEP from the two methods for a given operator action.

A quantitative HEP sensitivity study has been performed in support of this RAI response. As discussed in a teleconference with NRC staff on February 23, 2011, this sensitivity study approach uses the HCR/ORE method for the HEPs impacted by the EPU. The HCR/ORE method is used for both the pre-EPU and the EPU risk model quantifications for this sensitivity study.

The HEPs calculated for this sensitivity study are summarized in Table 2-1. Observations and conclusions are provided following Table 2-1.

Table 2-1
SUMMARY OF HEPs FOR RAI#2 USING HCR/ORE METHOD(1)

Operator Action(1)		Allowable Action Time		HEP (using HCR/ORE Method)	
Event ID	Description	Current Power (CLTP)	EPU	Current Power (CLTP)	EPU
B21-FO-HEDEP2-I	OPERATOR FAILS TO MANUALLY DEPRESSURIZE VESSEL WITH NON-ADS VALVES	45 min	38 min	1.00E-04	2.70E-04
B21-FO-HEDEP2-L	FAILURE TO MANUALLY DEPRESSURIZE VESSEL WITH NON-ADS VALVES (<2HRS)	240 min	224 min	1.00E-05	1.00E-05
C41-FO-HE1PMP-S	HUMAN ERROR: FAILURE TO MANUALLY INITIATE SLC (ONE PUMP OPERATION)	15 min	13.1 min	3.80E-04	4.80E-04
E12-FO-HESDC-O	OPERATOR FAILS TO PROPERLY ALIGN FOR SHUTDOWN COOLING	360 min	313 min	1.00E-05	1.40E-05
E12-FO-HESPC-M	OPERATOR FAILS TO MANUALLY ALIGN FOR SUPPRESSION POOL COOLING	420 min	353 min	1.00E-05	1.00E-05
E12-FO-HEV3S-O	OPERATOR FAILS TO PROPERLY ALIGN LPCI THRU SHUTDOWN COOLING LINES	15 min	13.1 min	1.70E-01	2.60E-01
E22-FO-DFEATHPCS	OPERATOR FAILS TO DEFEAT HPCS INTERLOCK AND START HPCS IN AN ATWS	20 min	17.4 min	7.20E-04	1.40E-03
E51-FO-HEISOL8-G	OPERATOR FAILS TO MANUALLY ISOLATE RCIC SYSTEM	12 min	10.5 min	3.20E-02	5.00E-02
E51-FO-HETRPBYP	HUMAN ERROR FAIL TO BYPASS RCIC TEMPERATURE TRIPS (EOP Attachment 3)	50 min	43.5 min	4.10E-03	4.30E-03
INHIBIT	FAILURE OF OPERATOR TO INHIBIT ADS/HPCS DURING AN ATWS	765 sec	757 min	3.00E-05	3.20E-05
LEV/PWR_CONTROL	OPERATOR FAILS TO CONTROL LEVEL AND POWER DURING ATWS	20 min	17.4 min	4.10E-04	6.30E-4
M41-FO-AVVCNT-Q	OPERATOR FAILS TO VENT CONTAINMENT	600 min	498 min	2.50E-05	2.50E-05
N11-FO-HEMODSW-G	OPERATOR FAILS TO TURN THE MODE SWITCH TO SHUTDOWN	15 min	12.6 min	1.00E-05	1.00E-05
N21-FO-HELVL9-I (ATWS)	HUMAN ERROR: FAILURE TO RESTART REACTOR FEED PUMPS FOLLOWING LEVEL 9 TRIP	30 min	26.1 min	1.20E-03	1.80E-03

Table 2-1
SUMMARY OF HEPs FOR RAI#2 USING HCR/ORE METHOD[1]

Operator Action[1]		Allowable Action Time		HEP (using HCR/ORE Method)	
Event ID	Description	Current Power (CLTP)	EPU	Current Power (CLTP)	EPU
N21-FO-HELVL9-I (Trans)	HUMAN ERROR: FAILURE TO RESTART REACTOR FEED PUMPS FOLLOWING LEVEL 9 TRIP	22 min	19.1 min	3.30E-03	5.70E-03
N21-FO-HEPCS-G (ATWS)	HUMAN ERROR FAIL TO PROPERLY ALIGN THE PCS FOR INJECTION	15 min	13.1 min	1.30E-04	2.00E-04
N21-FO-HEPCS-G (Trans)	HUMAN ERROR FAIL TO PROPERLY ALIGN THE PCS FOR INJECTION	15 min	12.6 min	1.30E-04	2.40E-04
P41-FO-HESWXT-G (LOCA)	OPERATOR FAILS TO MANUALLY ALIGN FOR SSW CROSS-TIE SYSTEM	20 min	17.4 min	8.90E-02	1.30E-01
P51-FO-CMSTART-T	FAILURE TO START STANDBY SERVICE AIR COMPRESSOR	60 min	50 min	2.50E-05	2.50E-05
P64-FO-HE-G	OPERATOR FAILS TO ALIGN FIREWATER SYSTEM FOR INJECTION	150 min	142 min	5.70E-01	6.70E-01
P64-FO-HE-G (Long Term)	OPERATOR FAILS TO ALIGN FIREWATER SYSTEM FOR INJECTION	480 min	456 min	1.10E-02	1.10E-02
R21-FO-HEBOPTRM	OPERATOR FAILS TO ALIGN ALTERNATE POWER TO BOP BUSSES	60 min	50 min	4.50E-04	8.60E-04
R21-FO-HEESFTRM	OPERATOR FAILS TO TRANSFER TO ALTERNATE TRANSFORMER	60 min	50 min	4.50E-04	8.60E-04
X2-ATWS	OPERATOR FAILS TO DEPRESSURIZE DURING ATWS	20 min	17.4 min	9.80E-05	1.40E-04
X3	X3--DEPRESURIZATION VIA RCIC	90 min	75 min	8.40E-03	1.80E-02
NRS-ALTPW&BOT	FAILURE TO ALIGN ALTERNATE POWER AND CONNECT AIR BOTTLES TO SRVS	Note (2)	Note (2)	1.00E-06	1.10E-06
NRS-ALTPW&BYP	FAILURE TO ALIGN ALTERNATE POWER AND BYPASS RCIC TEMP TRIPS	Note (2)	Note (2)	1.90E-06	3.70E-06
NRS-ALTPW&DEP	FAILURE TO ALIGN ALTERNATIVE POWER AND DEPRESSURIZE	Note (2)	Note (2)	1.00E-06[3]	1.00E-06[3]

Table 2-1
SUMMARY OF HEPs FOR RAI#2 USING HCR/ORE METHOD[1]

Operator Action[1]		Allowable Action Time		HEP (using HCR/ORE Method)	
Event ID	Description	Current Power (CLTP)	EPU	Current Power (CLTP)	EPU
NRS-ALTPWR&FPW	FAILURE TO ALIGN ALTERNATE POWER AND ALIGN FPW	Note (2)	Note (2)	5.00E-06	9.50E-06
NRS-ALTPW&Y47	FAILURE TO ALIGN ALTERNATE POWER AND INSTALL ALTERNATIVE SSW ROOM COOLING	Note (2)	Note (2)	1.00E-06[3]	1.00E-06[3]
NRS-BYP&BOT	FAILURE TO BYPASS RCIC TEMPERATURE TRIPS AND CONNECT AIR BOTTLES TO SRVS	Note (2)	Note (2)	5.40E-06	5.70E-06
NRS-BYP&COND	FAILURE TO BYPASS RCIC TEMPERATURE TRIPS AND ALIGN CONDENSATE INJECTION	Note (2)	Note (2)	7.50E-06	7.80E-06
NRS-DHRLT	FAILURE TO INITIATE SPC AND CONTAINMENT SPRAY	Note (2)	Note (2)	1.00E-06[3]	1.00E-06[3]
NRS-LSS&FWS	FAILURE TO RESET LSS PANEL AND ALIGN FIRE WATER	Note (2)	Note (2)	2.90E-04	3.40E-04
NRS-PCS&BYP	FAILURE TO RESTORE FEEDWATER AND BYPASS RCIC TEMP TRIPS	Note (2)	Note (2)	7.10E-06	1.30E-05
NRS-PCS&CRD	FAILURE TO RESTORE FEEDWATER AND START CRD	Note (2)	Note (2)	1.40E-06	2.40E-06
NRS-PCS&RC&DEP	FAILURE TO RESTART FEEDWATER, TRIP RCIC AND DEPRESSURIZE	Note (2)	Note (2)	1.30E-05	2.80E-05
NRS-PCS&CS	FAILURE TO RESTORE FEEDWATER AND ACTUATE CONTAINMENT SPRAY	Note (2)	Note (2)	1.00E-06[3]	1.00E-06[3]
NRS-PCS&RCIC	FAILURE TO RESTORE FEEDWATER AND TRIP RCIC	Note (2)	Note (2)	1.10E-05	2.30E-05
NRS-PCS&RCICL8	FAILURE TO RESTART FEEDWATER AND TRIP RCIC	Note (2)	Note (2)	2.60E-04	5.60E-04
NRS-PCSL8&BYP	FAILURE TO RESTORE FEEDWATER AND BYPASS RCIC TEMP TRIPS	Note (2)	Note (2)	1.80E-04	3.10E-04
NRS-PCSL8&COND	FAILURE TO RESTORE FEEDWATER AND ALIGN CONDENSATE INJECTION	Note (2)	Note (2)	6.00E-06	1.00E-05

Table 2-1
SUMMARY OF HEPs FOR RAI#2 USING HCR/ORE METHOD(1)

Operator Action(1)		Allowable Action Time		HEP (using HCR/ORE Method)	
Event ID	Description	Current Power (CLTP)	EPU	Current Power (CLTP)	EPU
NRS-PCSL8&DEP	FAILURE TO RESTORE FEEDWATER AND MANUALLY DEPRESSURIZE	Note (2)	Note (2)	5.40E-06	1.50E-05
NRS-PCSL8&HPCS	FAILURE TO RESTORE FEEDWATER AND START HPCS	Note (2)	Note (2)	8.00E-05	8.00E-05
NRS-RCICL8&DEP	FAILURE TO TRIP RCIC ON LEVEL 8 SIGNAL AND MANUALLY DEPRESSURIZE	Note (2)	Note (2)	3.20E-06	1.30E-05
NRS-SPC&DEP	FAILURE TO START SPC AND MANUALLY DEPRESSURIZE	Note (2)	Note (2)	1.00E-06(3)	1.00E-06(3)
NRS-Y47&BYP	FAILURE OF SSW VENTILATION AND MANUAL DEPRESSURIZE	Note (2)	Note (2)	1.60E-06	1.60E-06
NRS-Y47&FPW	FAILURE OF SSW VENTILATION AND ALIGN FPW	Note (2)	Note (2)	2.20E-04	2.60E-04

Note to Table 2-1:

(1) Not all HEPs in the GGNS PRA model are included in this table:
- This table includes only those post-initiator operator action HEPs that are calculated using an HRA method and have operator action timing windows changed by the EPU.
- Events that are not calculated using HRA methods (e.g., Offsite AC recovery, equipment recovery probabilities, AC convolution adjustment events) are not included in this table (the same probabilities as used in the GGNS EPU LAR risk assessment for such events are used in this sensitivity).
- Actions with timing windows that are not changed by the EPU are left at their base value (unchanged by the EPU) and are not summarized here.
- Actions with 1.0 HEPs in the base PRA are also not listed here (the 1.0 HEP is unchanged by the EPU).

(2) Dependent HEPs adjusted using the same methodologies used in the GGNS base PRA.

(3) The default minimum value used in the Grand Gulf model for the dependent HEPs is 1.0E-06. Where values of the dependent HEPs are <1.0E-06 the default value of 1.0E-06 is reported in the table.

Table 2-2 presents a summary of the EPU Risk Assessment CDF and LERF results from Table 5.7-1 of the GGNS EPU LAR Attachment 13:

Table 2-2
EPU LAR Results Using GGNS Base PRA HEP Approach[(1)]

Plant Configuration	CDF (1/yr)	LERF (1/yr)
Pre-EPU	2.68E-06	1.44E-07
EPU	2.91E-06	1.48E-07
Delta Risk	2.30E-07	4.30E-09

Note to Table 2-2:
(1) The GGNS base PRA (and the GGNS EPU LAR risk assessment) uses both the HCR/ORE and CBDT methods when calculating individual human error probabilities (HEPs) and then employs in the PRA models the higher HEP from the two methods for a given operator action.

Inserting the HEP values of Table 2-1 into the GGNS pre-EPU and EPU internal events PRA models (the same models as used in the March 2010 GGNS EPU LAR risk assessment) and quantifying the PRA models produces the following results:

Table 2-3
Sensitivity Case Results Using HCR/ORE Method[(1)]

Plant Configuration	CDF (1/yr)	LERF (1/yr)
Pre-EPU	2.29E-06	1.07E-07
EPU	2.76E-06	1.34E-07
Delta Risk	4.71E-07	2.73E-08

Note to Table 2-3:
(1) This sensitivity quantification uses only the HCR/ORE human reliability analysis (HRA) method for calculating individual human error probabilities (HEPs) for actions with timing impacted by the EPU. Both the pre-EPU and EPU risk model quantifications use this approach in this sensitivity study.

As can be seen from Tables 2-2 and 2-3, the HEP approach taken in the GGNS base PRA and in the GGNS EPU LAR risk assessment (i.e., higher of HCR/ORE or CBDT calculated HEP value used) results in a higher total CDF and total LERF than the sensitivity case.

In this sensitivity study, which used only the HCR/ORE method for operator actions with timings impacted by the EPU, the total CDF and total LERF results (see Table 2-3) are lower than the GGNS EPU LAR base case (i.e., Table 2-2). However, the sensitivity study delta

CDF and delta LERF are higher than the base case. This HEP sensitivity study results in delta risk values of 4.70E-7/yr and 2.73E-8/yr for CDF and LERF, respectively.

The delta CDF and delta LERF for both the GGNS EPU LAR risk assessment and for this sensitivity case remain within Region III ("Very Small Changes in Risk") of the NRC Regulatory Guide 1.174 risk criteria. The results from this quantitative sensitivity study do not change the conclusions of the GGNS EPU risk assessment.

RAI # 3

The submittal states in Attachment 13 Page 100:

> The fire PRA model was rerun for this EPU risk assessment using the same changes incorporated into the internal events PRA with the knowledge that the results would not necessarily reflect the most up to date model of the Grand Gulf plant.

Please explain in more detail why the results would not reflect the most up to date model of the GGNS. Identify any modeling discrepancies that would significantly alter the three percent change in fire core damage frequency (CDF).

Response

The statement quoted from the GGNS EPU report was not intended to convey that the results and conclusions of the GGNS EPU fire risk profile assessment are not applicable to the EPU condition. The statement in question was intended to convey that the fire PRA (FPRA) model used in the GGNS EPU risk assessment is based on the GGNS Individual Plant Examination of External Events (IPEEE) fire analysis and a previous version of the GGNS system fault tree and accident sequence structures; an update of the GGNS IPEEE FPRA model to integrate it with the latest GGNS PRA revision was not performed as part of the GGNS EPU risk assessment.

The GGNS IPEEE-based FPRA model was initially developed in the mid-1990s to respond to Generic Letter 88-20, Supplement 4 (i.e., the IPEEE submittal). This model was later integrated with updated GGNS PRA internal events models (Rev. 2) in the 2004 time frame. Subsequently in 2007, GGNS implemented an update to its Internal Events PRA models (Rev. 3); the GGNS IPEEE-based FPRA was not integrated with the Rev. 3 internal events models. GGNS does not currently maintain an FPRA model developed using more recent FPRA methods (e.g., NUREG/CR-6850).

The GGNS FPRA used in the GGNS EPU LAR risk assessment is the IPEEE-based model described above. The GGNS FPRA models approximately 125 fire scenario initiators. The system fault trees and accident sequence models in the GGNS IPEEE-based FPRA are primarily the same as in Rev. 3 of the GGNS internal events PRA.

Table 3-1 provides a summary of the major changes to the Internal Event PRA system fault trees and accident sequence models after the integration with the FPRA model to create the Internal Events PRA model (circa 2007) used as input for the EPU LAR risk evaluation. The internal events PRA update changes were associated with a standard periodic update and PRA model enhancements. There were no significant changes to the PRA methodologies. The changes to the internal event PRA model would not alter the conclusions of the FPRA in support of the EPU risk evaluation (i.e., RG 1.174 "very small" changes in risk). Fire PRA risk is dominated by fire-induced equipment failures. As such, fire PRA results are less impacted by changes in operator actions timings than the internal events PRA results.

Table 3-1
Summary of Changes to GGNS Revision 2 Model to Create Revision 3 Model[1,2]

Updated plant specific data
Updated plant specific and generic initiator frequencies
Added new initiators (e.g., Loss of Service Transformer, Break (LOCA) Outside of Containment)
Changes to LOSP modeling including LOSP due to transient or LOCA initiator and new industry data used for LOSP recovery analysis
Separated loss of PCS initiator into Closure of MSIVs initiator and Loss of PCS due to other causes initiator
Updated ISLOCA analysis
Updated Common cause analysis
Updated human reliability analysis
Included modeling for loss of ECCS pumps due to containment failure
Revised instrument air system modeling to incorporate new Plant Air compressors
Revised modeling of CRD - less credit for CRD
Added more detailed modeling for failure to scram
Added more detail to power conversion model

Note to Table 3-1:
(1) GGNS PRA Rev. 2 system fault trees and accident sequences are used in the GGNS Fire PRA model used to support the GGNS EPU LAR fire risk profile evaluation.
(2) GGNS PRA Rev. 3 system fault trees and accident sequences are used to support the GGNS EPU LAR internal events risk evaluations.

RAI # 4

The submittal states in Attachment 13 Page 102:

> EPU equipment replacements are judged to be installed using anchorages that are similar to the existing equipment anchorages.

Please confirm that EPU equipment replacements will be installed using anchorages that are seismically acceptable for the particular equipment.

Response

The statement from the GGNS EPU report is intended as a simplifying assumption for the performance of the risk assessment. Specific design details regarding anchorages details were not obtained and reviewed for the risk evaluation. However, GGNS confirms that this risk assessment assumption is consistent with the design approach that has been implemented.

Replacement components are designed commensurate with their function and design requirements. Seismic Category I equipment is designed to maintain its functionality during and after a seismic event. Seismic Category II/I equipment is designed such that, should it fail during a seismic event, it would not fail in a manner that would adversely impact the function of any safety related equipment.

The seismic risk profile at a plant is overwhelmingly dominated by loss of offsite power scenarios given the typically low seismic capacity of offsite power (i.e., transmission systems.) BOP equipment is dependent on offsite power and has little to no risk influence in seismic-induced accident sequences.