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

August 28, 2012

LICENSEE: Entergy Operations, Inc.

FACILITY: Grand Gulf Nuclear Station

SUBJECT: SUMMARY OF TELEPHONE CONFERENCE CALL HELD ON JUNE 8, 2012, BETWEEN THE U.S. NUCLEAR REGULATORY COMMISSION AND ENTERGY OPERATIONS, INC., CONCERNING REQUESTS FOR ADDITIONAL INFORMATION PERTAINING TO THE GRAND GULF NUCLEAR STATION LICENSE RENEWAL APPLICATION (TAC. NO. ME7493)

The U.S. Nuclear Regulatory Commission (NRC or the staff) and representatives of Entergy Operations, Inc. (Entergy) held a telephone conference call on June 8, 2012, to discuss and clarify the staff's requests for additional information (RAIs) concerning the Grand Gulf Nuclear Station license renewal application. The telephone conference call was useful in clarifying the intent of the staff's RAIs.

Enclosure 1 provides a listing of the participants and Enclosure 2 contains a listing of the RAIs discussed with the applicant, including a brief description on the status of the items.

The applicant had an opportunity to comment on this summary.

Nathaniel Ferrer, Project Manager
Projects Branch 1
Division of License Renewal
Office of Nuclear Reactor Regulation

Docket No. 50-416

Enclosures:
As stated

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TELEPHONE CONFERENCE CALL
GRAND GULF NUCLEAR STATION
LICENSE RENEWAL APPLICATION

LIST OF PARTICIPANTS
JUNE 8, 2012

PARTICIPANTS	**AFFILIATIONS**
Nate Ferrer	U.S. Nuclear Regulatory Commission (NRC)
Seung Min	NRC
Pat Purtscher	NRC
Ching Ng	NRC
Ben Parks	NRC
Ted Ivy	Entergy Operations, Inc. (Entergy)
Andy Taylor	Entergy
Alan Cox	Entergy
Jack Lingenfelter	Entergy
Wayne Bichlemier	Entergy
David Wootten	Entergy
Stan Batch	Entergy
Kirk Ehren	Entergy
Bill Nichols	Entergy

REQUESTS FOR ADDITIONAL INFORMATION (SET 25 AND 26)
LICENSE RENEWAL APPLICATION
JUNE 8, 2012

The U.S. Nuclear Regulatory Commission (NRC or the staff) and representatives of Entergy Operations, Inc. (the applicant) held a telephone conference call on June 8, 2012, to discuss and clarify the following requests for additional information (RAIs) concerning the Grand Gulf license renewal application (LRA).

Draft RAI 4.2.1-1

Background. LRA Section 4.2.1 addresses the applicant's reactor vessel fluence calculations. LRA Section 4.2.1 states that the fluence is calculated based on a time-limited assumption defined by the operating term, which indicates that the applicant identified the reactor vessel neutron fluence calculations as a time-limited aging analysis (TLAA).

Issue. LRA Table 4.1-1 does not identify the neutron fluence calculation as a TLAA. In addition, the LRA does not address applicant's TLAA disposition of the neutron fluence calculations in terms of the dispositions described in 10 CFR Part 54.21(c)(i), (ii) and (iii).

Request.

a. Clarify why LRA Table 4.1-1 does not identify the neutron fluence calculation as a TLAA even though the fluence is calculated based on a time-limited assumption defined by the operating term.
b. If the fluence calculation is identified as a TLAA, describe the TLAA disposition of the neutron fluence calculation in terms of the dispositions described in 10 CFR Part 54.21(c)(i), (ii) and (iii). Additionally, revise LRA Section 4.2.1, Table 4.1-1 and Section A.2.1.1 to include a relevant TLAA disposition, consistent with the response.

Discussion: The applicant stated that it was unclear about the description of fluence in request (a). The staff noted that sufficient background and issue descriptions were provided and will reword the request (a) as follows:

> Request.
>
> a. Clarify why LRA Table 4.1-1 does not identify the neutron fluence calculation as a TLAA.

The staff will issue the revised question as a formal RAI.

Draft RAI 4.2.1-2

Background. LRA Section 4.2.1 addresses the peak neutron fluence values (E > 1 MeV) for 54 EFPY based on planned extended power uprate (EPU) power level beginning with Cycle 19. The predicted peak neutron fluence value is 4.44E+18 n/cm^2 at the vessel inner surface of the lower-intermediate shell and axial welds (i.e., Shell Plate 2 location).

The LRA also states that the neutron fluence for the reactor pressure vessel (RPV) beltline region was determined using the General Electric-Hitachi (GEH) method for neutron flux calculation documented in report NEDC-32983P-A and approved by the NRC. The LRA further states the GEH method adheres to the guidance provided in RG 1.190.

During the audit, the staff noted that Reference 1, which is addressed below, describes the GEH method for applicant's fluence calculations. Reference 1 also refers to Reference 2, which describes another fluence calculation method (MPM method) that the applicant used.

Reference 1: GE Hitachi, Project Task Report, 0000-0104-5984-R0, Revision 0, "Entergy Operations, Inc. Grand Gulf Nuclear Station Extended Power Uprate," Task T0313, RPV Flux Evaluation, October 2009.

Reference 2: MPM-809633, "Grand Gulf Extended Power Uprate Neutron Transport Analysis," August 2009.

In addition, the GEH report referenced above indicates the following information:

(1) The total fluence values at different effective full power years (EFPYs) were calculated by adding the corresponding post-EPU fluence to the pre-EPU fluence. The post-EPU fluence and total fluence values are calculated and reported in the GEH report, while the MPM-809633 report calculated the pre-EPU fluence.

(2) Section 3.4.2, "Observations," of the GEH report indicates that calculated post-EPU flux values for core shroud welds H1, V1, V2, V3 and V4 were found to be significantly lower than pre-EPU flux values derived from MPM-809633. This section also states that because the pre-EPU flux values were not calculated by GEH, the reason for this large difference at these locations is unknown.

Issue. Based on the information described above, the staff noted the following concerns.

a. LRA Sections 4.2.1 and A.2.1.1 (UFSAR supplement for reactor vessel fluence) do not identify the methodology described in MPM-809633 as one of the methods that have been used to calculate the neutron fluence. In addition, the staff is not clear as to which fluence calculation methods are included in the current licensing basis.
b. The LRA does not provide information regarding how significantly the post-EPU flux values for shroud welds H1, V1, V2, V3 and V4 based on the GEH methodology are lower than the pre-EPU flux values derived from MPM-809633. The staff is not clear why the post-EPU flux values, which are significantly lower than the pre-EPU flux values, are acceptable.
c. The LRA does not provide information to confirm how the fluence calculation methods of the applicant comply with Regulatory Guide (RG) 1.190.
d. The LRA does not describe the results of the measurement benchmarking of the fluence calculation methods with the plant-specific dosimetry data such as the first-cycle or test capsule dosimetry data as addressed in BWRVIP-86, Revision 1, Section 5.4, "Plan for Ongoing Vessel Dosimetry." It is noted that the staff issued its safety evaluation for BWRVIP-86, Revision 1 by letter dated October 20, 2011.

Request.

a. Justify why LRA Sections 4.2.1 and A.2.1.1 (UFSAR supplement for reactor vessel fluence) do not identify the methodology described in MPM-809633 as one of the methods that have been used to calculate and project the reactor vessel neutron fluence. Alternatively, revise LRA Sections 4.2.1 and A.2.1.1 to identify and include the MPM method in the LRA, as appropriate.
b. As part of the response, clarify what methods for fluence calculations are included in the current licensing basis. If the LRA does not identify all the fluence calculation methods

that constitute the current licensing basis, justify why the LRA does not identify all the fluence calculation methods.

c. Provide additional information regarding how the fluence calculation methods have been incorporated into the current licensing basis (e.g., whether through 10 CFR 50.90 or 50.59 process). As part of the response, provide information to demonstrate that such methods are consistent with Regulatory Guide (RG) 1.190.
d. Provide the following information related to the flux and fluence calculations.
   1. Provide the pre-EPU (MPM method) and post-EPU (GEH method) flux values for core shroud welds H1, V1, V2, V3 and V4. If existent, describe any other location that involves the higher flux differences between the two calculation methods than these five core shroud welds and provide the associated flux values.
   2. Clarify why the post-EPU flux values (GEH method), which are significantly lower than the pre-EPU flux values (MPM method), are acceptable in terms of the adequacy of the fluence calculations and the compatibility between the fluence calculation methods.
   3. Provide the pre-EPU (MPM method) and post-EPU (GEH method) peak flux values for the inner surfaces of Shell Plates 1 and 2 at the cycle when the EPU is planned to start. If any significant difference exists between these flux values for either of the shell plates, justify why the significant difference is acceptable.
   4. If pre-EPU fluence values obtained using the MPM method were combined with post-EPU fluence values obtained using the GEH method to determine total, post-EPU and end-of-life-extended (EOLE) fluence values, please describe the treatment of uncertainty associated with this technique, and explain how it conforms to the guidance contained in RG 1.190. If the neutron fluence values were combined, and the uncertainty treatment is not believed to adhere to RG 1.190, please justify the acceptability of this approach.
e. Provide additional information to confirm whether the fluence calculation methods have been benchmarked with the ongoing vessel dosimetry, consistent with Section 5.4 of BWRVIP-86, Revision 1. In addition, provide information to confirm whether the fluence calculations using the implemented methods are consistent with the vessel dosimetry data.

Discussion: The applicant indicated that the question is clear. The staff will issue the question as a formal RAI.

Draft RAI 4.7.3-1

Background. In LRA Section 4.7.3, the applicant states that a fluence analysis was performed of components included in the design specification 22A4052 at extended power uprate (EPU) operating conditions for 60 years plant life. The LRA further states the design specification 22A4052 for the reactor vessel internals components includes requirement beyond the ASME design requirements for austenitic stainless steel base metal components exposed to greater than 1×10^{21} nvt (> 1 MeV) or weld metal greater than 5×10^{20} nvt (> 1 MeV). After location-specific fluence levels were determined, the applicant concludes that the internal core support structure components meet the irradiation criteria in the design specification at EPU operating conditions for 60 years plant life.

Issue. SRP-LR Section 4.7.3.1.2 indicates that for a TLAA disposition pursuant to 10 CFR 54.21(c)(1)(ii), the applicant shall provide a sufficient description of the analysis and document the results of the reanalysis to show that it is satisfactory for the 60-year period. Without this information, the staff cannot evaluate the adequacy of the TLAA.

Request.

a. Justify why only the internal core support structure components were evaluated against the irradiation criteria in the design specification 22A4052.
b. Identify the 40-year fluence levels of the reactor vessel internals components; identify and justify the projected 60-year fluence levels; and identify the design requirements from both the design specification 22A4052 and the ASME code. Justify that the analysis based on the EPU operating conditions has been projected through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

Discussion: The applicant stated that it was unclear what additional information was being requested for justification in request (b). The staff noted that the information being requested in (a) and the first portion of (b) is what the staff needs to complete the review. The staff will reword the request section as follows:

> Request.
>
> a. Justify why only the internal core support structure components were evaluated against the irradiation criteria in the design specification 22A4052.
> b. Identify the 40-year fluence levels of the reactor vessel internals components; identify and justify the projected 50-year fluence levels; and identify the design requirements from both the design specification 22A4052 and the ASME code.

The staff will issue the revised question as a formal RAI.

Draft RAI B.1.11-2

Background. In the LRA Section B.1.11, "BWR Vessel Internal Program," the applicant stated that "detection of aging effects" program element has been enhanced to manage loss of fracture toughness due to neutron irradiation embrittlement and thermal aging embrittlement for those reactor vessel internal (RVI) components that are made from cast austenitic stainless steel (CASS), martensitic stainless steel, precipitation hardened martensitic (PH) stainless steel, or alloy X-750 materials (subject materials). The enhancement states that:

> The susceptibility to neutron or thermal embrittlement for reactor vessel internal components composed of CASS, X-750 alloy, precipitation-hardened (PH) martensitic stainless steel (e.g., 15-5 and 17-4 PH steel), and martensitic stainless steel (e.g., 403, 410,431 steel) will be evaluated. Portions of the susceptible components determined to be limiting from the standpoint of thermal aging susceptibility, neutron fluence, and cracking susceptibility (i.e., applied stress, operating temperature, and environmental conditions) will be inspected, using an inspection technique capable of detecting the critical flaw size with adequate margin. The critical flaw size will be determined based on the service loading condition and service-degraded material properties. The initial inspection will be performed either prior to or within 5 years after entering the period of extended operation. If cracking is detected after the initial inspection, the frequency of re-inspection will be justified based on fracture toughness properties appropriate for the condition of the component. The sample size will be 100% of the accessible component population, excluding components that may be in compression during normal operations.

Issue. The staff has identified the following issues with the enhancements:

a. In LRA Table 3.1.2-2, only those RVI components made from CASS materials are listed as being managed for loss of fracture toughness. LRA Table 3.1.2-2 does not identify loss of fracture toughness as an aging effect requiring management (AERM) for any RVI components made from martensitic stainless steel, precipitation hardened martensitic (PH) stainless steel or alloy X-750 materials.
b. The enhancement on the BWR Vessel Internals Program does not specify which type of inspection technique or techniques will be used to inspect components manufactured from the subject materials (i.e., CASS, martensitic stainless steel, PH stainless steel, or X-750 nickel alloy). The GALL report identifies that VT-1 visual techniques (including enhanced VT-1 techniques) and volumetric techniques are examples of acceptable inspection methods to detect cracking and indirectly manage loss of fracture toughness in these types of components.
c. It is not evident what criteria will be used to a) select the limiting and expansion components manufactured from the subject materials as being susceptible to thermal and/or neutron embrittlement, b) determine the expansion criteria that triggers expansion of the inspections, and (c) determine the scope of the inspection of the expansion components if expansion is triggered. This issue is consistent with the statement in NUREG-1800, Revision 2 (SRP-LR), Branch Position RLSB-1, which states that provisions on expanding the sample size when degradation is detected in the initial sample should also be included.
d. The enhancement does not define how loss of fracture toughness will be managed in the susceptible, but inaccessible components manufactured from the subject materials when evidence of cracking has been detected in the accessible components.

Request. Based on the points identified in the "Issue" section of this RAI, the staff requests the following information:

a. Are there any components made from martensitic stainless steel, precipitation hardened martensitic (PH) stainless steel or alloy X-750 materials in the BWR vessel internals that are exposed to greater than 1.0E+17 n/cm^2 (E > 1 MeV)? If the answer is yes, then add new line items to the AMR Table 3.1.2-3. If the answer if no, then correct the enhancement to reflect the materials present in the plant.
b. Specify the inspection technique(s) to be used to detect cracking in the enhanced inspections.
c. Discuss the criteria that will be used to determine the components that will be initially inspected. Clearly identify and justify how the lead or limiting susceptible components for loss of fracture toughness due to thermal aging and neutron embrittlement will be chosen. Address the considerations of NUREG-1800, Branch Technical Position RLSB-1, Section A.1.2.3.4, item 4 regarding sampling based condition monitoring plans.
d. Describe the criteria for expansion of the scope of inspections if degradation is found in the primary enhanced inspections. Describe the inspection techniques to be used and the scope for the expansion beyond the primary enhanced inspections.
e. Describe how inaccessible components will be addressed:
    1. If determined to be highly susceptible to embrittlement (limiting component).
    2. If not limiting component, but degradation is detected in limiting component.
f. Revise the UFSAR Supplement A.1.11 as necessary to reflect all changes to the enhancement.

Discussion: The applicant stated that request (d) was unclear because the sample size for the program will be 100 percent of the accessible component population. The staff noted that based on the sample size for the program, request (d) was not necessary. The staff will reword the request section as follows:

> Request. Based on the points identified in the "Issue" section of this RAI, the staff requests the following information:
>
> a. Are there any components made from martensitic stainless steel, precipitation hardened martensitic (PH) stainless steel or alloy X-7S0 materials in the BWR vessel internals that are exposed to greater than 1.0E+17 n/cm2 (E > 1 MeV)? If the answer is yes, then add new line items to the AMR Table 3.1.2-3. If the answer is no, then correct the enhancement to reflect the materials present in the plant.
> b. Specify the inspection technique(s) to be used to detect cracking in the enhanced inspections.
> c. Discuss the criteria that will be used to determine the components that will be initially inspected. Clearly identify and justify how the lead or limiting susceptible components for loss of fracture toughness due to thermal aging and neutron embrittlement will be chosen. Address the considerations of NUREG-1800, Branch Technical Position RLSB-1, Section A.1.2.3.4, item 4 regarding sampling based condition monitoring plans.
> d. Describe how inaccessible components will be addressed:
>   1. If determined to be highly susceptible to embrittlement (limiting component).
>   2. If not limiting component, but degradation is detected in limiting component.
> e. Revise the UFSAR Supplement A.1.11 as necessary to reflect all changes to the enhancement.

The staff will issue the revised question as a formal RAI.

Draft RAI B.1.11-3

Background. In GALL, Section XI.M9 for monitoring and trending, the program recommends acceptable documents where additional guidelines for evaluation of crack growth in stainless steels, nickel alloys, and low-alloy steels can be found. During the period of extended operation, core shroud welds and base materials may be exposed to neutron fluence values of 1.0E+21 n/cm^2 (E > 1 MeV) or greater. BWRVIP-100-A report, "Updated Assessment of the Fracture Toughness of Irradiated Stainless Steel for BWR Core Shrouds," provides the BWRVIP's updated fracture toughness data for the irradiated stainless steel materials. For stainless steel materials exposed to neutron fluence equal to or greater than 1.0E+21 n/cm^2 (E >1 MeV), the generic core shroud linear elastic fracture mechanics analyses in Appendix C of the BWRVIP-100-A report used a lower K_{IC} fracture toughness value for the subject materials than the corresponding value reported for these materials in Appendix C of the BWRVIP-76 report, "BWR Vessel and Internals Project BWR Core Shroud Inspection and Flaw Evaluation Guidelines."

Issue. In LRA Appendix C, the applicant also states that BWRVIP-76-A is credited for management of cracking in the Grand Gulf core shroud components. The staff is concerned that, for the "monitoring and trending" activities of stainless steel RVI components, the applicant may not be using the more conservative fracture toughness value for stainless steel materials reported in the BWRVIP-100-A report.

Request. Clarify whether the "monitoring and trending" bases for evaluation of cracks in stainless steel RVI components have been reconciled to use the more conservative lower bound fracture toughness value reported for these materials in BWRVIP-100-A. If not, justify the use of a less conservative lower bound fracture toughness value for those RVI components that are made from stainless steel (i.e., use the value reported in BWRVIP-76-A for stainless steel core shroud components or in other applicable NRC-approved BWRVIP reports for other stainless steel RVI components).

Discussion: The applicant stated that it was unclear what bases the request section was referencing. The staff was referring to the acceptance criteria bases and will reword the request as follows:

> Request. Clarify whether the acceptance criteria for evaluation of cracks in stainless steel RVI components will use the more conservative lower bound fracture toughness value reported for these materials in BWRVIP-100-A. If not, justify the use of a less conservative lower bound fracture toughness value for those RVI components that are made from stainless steel (i.e., use the value reported in BWRVIP-76-A for stainless steel core shroud components or in other applicable NRC-approved BWRVIP reports for other stainless steel RVI components).

The staff will issue the revised question as a formal RAI.

Draft RAI 4.2.3-1

Background. Generic Letter (GL) 92-01, Revision 1 and Regulatory Guide (RG) 1.99, Revision 2 address the information related to reactor vessel structural integrity required of all licensees for beltline materials. The staff has maintained that the licensees should provide comparable information for all extended beltline materials as part of the license renewal process. With the increase in neutron fluence associated with license renewal, three additional plates are now above the fluence threshold (> 1E+17 n/cm2, E > 1 MeV) and must be considered as extended beltline materials. Note (2) of license renewal application (LRA) Table 4.2-2 indicates that since information is not available for the actual measured copper content for the three plates of Shell Course 1, the maximum allowable copper content was obtained from the vessel design specification (i.e., copper content of 0.12 percent).

Issue. RG 1.99, Rev. 2 specifically considers best estimate values for the material as acceptable, which will normally be the mean of measured values for a given plate. If such values are not available, then upper limiting values given in the material specification are acceptable. The RG does not mention the design specification. Conservative estimates of the chemistry (mean plus one standard deviation) based on generic data may be used if justification is provided.

Request.

a. Provide the part of the design specification for Shell Course 1 that describes the required copper content and the material specification that was in effect when the reactor vessel for Grand Gulf was built.

b. Describe the documented basis for the copper content of Shell Course 1 plates, such as available certified material test records, quality control documents, and/or other data that might be used to justify the assumed copper content.

Discussion: The applicant indicated that the question is clear. The staff will issue the question as a formal RAI.

Draft RAI B.1.23-2

Background. LRA Section B.1.23, "Inservice Inspection," states that, "ISI Program Summary Reports between 2004 and 2010 reveal compliance and provide evidence that the program is effective for managing aging effects in accordance with the ASME Boiler Pressure Vessel Code Section XI."

GALL AMP XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," "Detection of Aging Effects" program element states that "The extent and schedule of the inspection and test techniques prescribed by the program are designed to maintain structural integrity and ensure that aging effects are discovered and repaired before the loss of intended function of the component." In addition, "Monitoring and Trending" program element states that, "For Class 1, 2, or 3 components, the inspection schedule of IWB-2400, IWC-2400, or IWD-2400, respectively, and the extent and frequency of IWB-2500-1, IWC-2500-1, or IWD-2500-1, respectively, provides for timely detection of degradation."

Issue. The staff noted that Event Notification Report No. 47880 dated April 30, 2012, indicates that the applicant detected an unacceptable indication in one of the residual heat removal (RHR) system to reactor pressure vessel nozzles (weld area of N06B-KB nozzle) during the current refueling outage. The defect has a size of 0.9 inches in length and 0.5 inches in depth. Nominal wall thickness of the weld is 1.3 inches.

The staff needs clarification regarding how this plant-specific operating experience affects the effectiveness of the applicant's aging management program (e.g., detection of aging effects and directing corrective actions in a timely manner).

Request.

a. Clarify whether the defect detected in the RHR nozzle is age-related. If it is, and based on the size of the defect, provide justification that the applicant's proposed ISI program is still effective in timely detection of aging effects (i.e., whether inspection intervals are adequate to prevent unacceptable flaw propagation).
b. Clarify when the previous UT examination was performed on the subject RHR weld and provide the examination results. Describe any corrective actions and extent of condition performed. Provide justification that the current inspection schedule for all affected components is adequate for timely detection of aging effects.

Discussion: The applicant stated that it was unclear what corrective actions the request was referencing. The staff was referring to corrective actions for any past unacceptable indications as well as the corrective actions for the indication described in the event notification report. The staff will reword the request section as follows:

> Request.
>
> a. Clarify whether the defect detected in the RHR nozzle is age-related. If it is, and based on the size of the defect, provide justification that the applicant's

proposed Inservice Inspection (ISI) program is still effective in timely detection of aging effects (Le., whether inspection intervals are adequate to prevent unacceptable flaw propagation).

b. Clarify when the previous UT examination was performed on the subject RHR weld and provide the examination results.
c. Describe any corrective actions and extent of condition performed from previous examinations or as a result of the recent unacceptable indication. Provide justification that the current inspection schedule for all affected components is adequate for timely detection of aging effects.

The staff will issue the revised question as a formal RAI.

August 28, 2012

LICENSEE: Entergy Operations, Inc.

FACILITY: Grand Gulf Nuclear Station

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The applicant had an opportunity to comment on this summary.

/RA/

Nathaniel Ferrer, Project Manager
Projects Branch 1
Division of License Renewal
Office of Nuclear Reactor Regulation

Docket No. 50-416

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Memorandum to Entergy Operations Inc. from N. Ferrer dated August 28, 2012

SUBJECT: SUMMARY OF TELEPHONE CONFERENCE CALL HELD ON JUNE 8, 2012,BETWEEN THE U.S. NUCLEAR REGULATORY COMMISSION AND ENTERGY OPERATIONS, INC., CONCERNING REQUESTS FOR ADDITIONAL INFORMATION PERTAINING TO THE GRAND GULF NUCLEAR STATION LICENSE RENEWAL APPLICATION (TAC. NO. ME7493)

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