UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

October 9, 2024

The Honorable Christopher T. Hanson
Chair
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

SUBJECT: SUMMARY REPORT – 718th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, SEPTEMBER 4-5, 2024

Dear Chair Hanson:

During its 718th meeting, September 4 through 5, 2024, which was conducted in person and virtually, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters. The ACRS completed the following correspondence:

MEMORANDA

Memoranda to Dr. Mirela Gavrilas, Executive Director for Operations, U.S. Nuclear Regulatory Commission (NRC), from Scott W. Moore, Executive Director, ACRS:

- Documentation of Receipt of Applicable Official NRC Notices to the Advisory Committee on Reactor Safeguards for September 2024, dated September 12, 2024, Agencywide Documents Access and Management System (ADAMS) Accession No. ML24255A080, and

- Regulatory Guides, dated September 13, 2024, ADAMS Accession No. ML24255A072.

HIGHLIGHTS OF KEY ISSUES

a. X-energy Principal Design Criteria Topical Report

The following was discussed during the designated portion of the meeting as well as during the planning and procedures portion of the full committee meeting.

The X-energy Subcommittee met on August 21, 2024, to discuss X-energy's Xe-100 Topical Report (TR), "Principal Design Criteria," Revision 3, and the associated NRC staff safety evaluation (SE). During these meetings, the Committee had the benefit of discussions with the staff and representatives of X-energy, as well as documents cited in X-energy's TR and the staff SE. X-energy plans to pursue the risk-informed, performance-based licensing approach described in Nuclear Energy Institute (NEI) 18-04 (ADAMS Accession No. ML19241A336) and NEI 21-07 (ADAMS Accession No. ML21250A378). These industry documents have been endorsed by the NRC in Regulatory Guides (RG) 1.233 (ADAMS

Accession No. ML20091L698) and 1.253 (ADAMS Accession No. ML23269A222), respectively. The NEI and regulatory guide documents emphasize a risk-informed approach (i.e., Licensing Modernization Project (LMP)) for the selection of licensing basis events (LBEs), classification of systems, structures, and components (SSCs), and the evaluation of defense-in-depth (DiD). They also include expectations for the preparation of principal design criteria (PDC) that begin with the application of RG 1.232, "Guidance for Developing Principal Design Criteria for Non-Light Water Reactors," (ADAMS Accession No. ML17325A611). This RG provides reactor designers with guidance on how to establish PDC that ensure safety while accommodating the innovative features of advanced reactor designs. Relevant to the Xe-100, this includes generic modular high temperature gas-cooled reactor design criteria (MHTGR-DC).

X-energy developed the Xe-100 PDC primarily by adapting these MHTGR-DC, with changes reflecting design-specific details and the licensing process described by RG 1.233 and RG 1.253. Most of these changes were vernacular in nature. For example, per NEI 18-04, safety-related (SR) and non-safety-related with special treatment (NSRST) functions and SSCs are referred to as "safety significant." As such, wherever "important to safety" appeared in the MHTGR-DC, it was replaced with "safety significant." Another artifact of applying RG 1.253 is that X-energy presented PDC as Required Functional Design Criteria (RFDC) and Complementary Design Criteria (CDC). Both RFDC and CDC are distinguished as addressing the functions provided by safety-related (SR) and non-safety related with special treatment (NSRST), respectively. In addition, they introduced "Owner Controlled Design Criteria" (OCDC) for non-safety related with no special treatment (NST) functions and SSCs.

Another departure from RG 1.232 was deletion of "single failure" in favor of reliability criteria, an option acknowledged in RG 1.233. A concern raised by an ACRS member regarding this change was that the applicant may find reliability characteristics, particularly uncertainties, difficult to quantify for some SSCs, such as those with limited industrial applications. As such, application of single failure criterion might be a necessary alternative.

RG 1.232 specifies that the MHTGR-DC apply when certain design characteristics are met. These are the use of TRISO fuel, graphite as a primary structural material for the core and reflector, and helium gas as the primary coolant, all of which are credited in the Xe-100. In addition, RG 1.232 states that designs applying the MHTGR-DC should have operational characteristics, addressing anticipated operational occurrences (AOOs) and design-basis events (DBEs) like MHTGR designs. While specific details regarding operational characteristics and the Xe-100's response to AOOs and DBEs will be an outcome of X-energy's implementation of NEI 18-04, the NRC staff acknowledged that preliminary information provided by X-energy has been sufficient to support their review.

X-energy's PDC 16, "Functional Containment Design," departed from RG 1.232 language by replacing "multiple barriers" with a RFDC specific to fuel particle and pebble "barriers" and a CDC specific to the helium pressure boundary. The deletion of the term "multiple barriers" was discussed, as the revised PDC 16 replaces a statement of principle (provide multiple barriers) with design details for an approach that provides multiple barriers consistent with prior design MHTGRs. The Subcommittee concluded that the deletion of the overall statement of principle was acceptable since, though not specifically stated in the PDC text, the principle is met.

Related to PDC 16, X-energy plans to use a proprietary TRISO particle fuel referred to as TRISO-X. From prior engagement between X-energy and the staff on X-energy's Fuel Qualification Methodology, X-energy will evaluate any discrepancies between their fuel and the TRISO fuel specifications and performance data from the U.S. Department of Energy's Advanced Gas Reactor (AGR) program (as presented in EPRI-AR-1) to confirm its capability as a functional containment.

The staff's Limitations and Conditions (L&Cs) provide closure for the Xe-100 PDCs. The specific L&Cs convey requirements to X-energy related to subsequent development of the Xe-100 safety case and engagement with the staff. In contrast to how General Design Criteria influence the development and review of LWR applications, an apparent outcome of applying RG 1.232 is documentation of specific safety case requirements that may not naturally arise from other topical or safety analysis report reviews. Early agreement between X-energy and the staff on such details will provide the best assurance for the predictability and efficiency of staff design reviews.

The opinion of the Subcommittee is that the staff's SE report is sufficiently complete to recommend that it be issued. It is also recommended that this writeup serve as the record of the subcommittee meeting and an ACRS letter report not be prepared.

The Committee agreed with the Subcommittee's recommendation.

b. Seabrook Alkali-Silica Reaction (ASR) Update

The Committee heard from representatives of the Office of Nuclear Reactor Regulation (NRR) and Region I, including inspectors, about the status of the progression of the ASR phenomena at the Seabrook station and NRC oversight. In addition, the Committee heard from members of
C-10, a public interest group that has been following this issue since the discovery of ASR at Seabrook.

NRR provided its views on the review of the inspection results and the licensees corrective actions, as well as future plans by the licensee to address this issue.

The staffs and C-10's presentations may be found at ADAMS Accession No. ML24270A193.

During the planning and procedures session of this meeting, the discussion of proposed additional topics the next steps for the Seabrook ASR issue was broached. The Committee agreed that the staff would target the next meeting in about 6 – 9 months with the proposed focus to be on the status of the licensee's (NextERAs) evaluation of the progression of ASR and proposed corrective actions.

Member Sunseri recused himself from these discussions due to a potential conflict of interest.

c. Discussions during the Planning and Procedures Session

1. The Committee discussed the Full Committee (FC) and Subcommittee (SC) schedules through February 2024 as well as the planned agenda items for Full Committee meetings.

Member Roberts noted that the Limerick digital instrumentation and controls license amendment request meeting would be delayed from its currently scheduled date of October 16, 2024. No date has yet been established for the new meeting.

2. The ACRS Executive Director led a discussion of significant notices issued by the Agency since the last FC meeting in July 2024. The Executive Director documented this activity in a memorandum dated September 12, 2024, ADAMS Accession No. ML24255A080.

3. The Executive Director also led a discussion of three draft RGs/RGs regarding possible review by the Committee. The Executive Director documented this activity in a memorandum dated September 13, 2024, ADAMS Accession No. ML24255A072.

4. Member Martin led a discussion about the Committee's review of the X-energy PDC TR and the staff's associated safety evaluation. The Committee agreed on the summary that appears under "Highlights of Key Issues" in this summary report.

5. Member Roberts led a discussion of the eVinci PDC TR.

   On August 21, 2024, the eVinci Design Center Subcommittee met with representatives of Westinghouse Electric Company LLC (Westinghouse/WEC) and the NRC staff to review topical report EVR-LIC-RL-001-P/NP, "Principal Design Criteria Topical Report," and the associated staff draft safety evaluation. This TR contains a brief overview of the eVinci microreactor design, a summary of how the PDC were developed, and the PDC selected for the eVinci microreactor. The draft safety evaluation includes four L&Cs.

   As a heat-pipe reactor, the eVinci design is different from the advanced reactor types that were used to develop the specific design criteria defined in Appendices B and C of RG 1.232. Westinghouse primarily used the more generic advanced reactor design criteria (ARDC) in Appendix A of RG 1.232, with some of the more specific criteria deemed applicable due to use of sodium as the heat transfer fluid within the heat pipes (mapping to sodium fast reactor criteria of Appendix B) and use of helium as the cover gas within the reactor canister (mapping to modular high temperature gas reactor criteria of Appendix C). The result is a hybrid set of design criteria that show traceability to all the specified ARDC.

   The L&Cs in the draft SER: (1) relate to the preliminary state of the eVinci plant design and the need to complete the RG 1.233 design process (also called the LMP process) to fully justify the PDC and identify any additional PDC; and (2) acknowledge that the NRC is evaluating microreactor licensing changes and some of the PDC may not be fully applicable until the NRC completes this process.

   Two potential issues were discussed at the meeting, and both were found to be adequately covered by the applicant and/or the L&Cs:

   a. It was not clear why no new PDC were required, given that RG 1.232 requires PDC to address unique design features and use of heat pipes for energy transfer is a unique concept relative to reactor technologies that were considered when RG 1.232 was written. The applicant explained during the meeting that they organized the PDC by function, not by SSC, and the functions associated with eVinci are essentially the same as any reactor technology. The applicant further clarified that unique methods that may be needed to implement those functions would be determined via their requirements management system and do not require PDC to be defined. Additionally, SER L&C No. 1 observes that the design is preliminary and the LMP process requires determination of the need for new PDC as the safety analysis is developed; compliance with this L&C should ensure that any new PDC necessary to reflect the heat pipe design will be specified as the safety analysis progresses.

b. The applicant had changed several of the RG 1.232 design criteria to delete specific direction for redundancy or defense-in-depth, in favor of the LMP methodology for determining reliability objectives. One specific change discussed at the subcommittee meeting related to criterion No. 16 (functional containment). In this case, the specific RG 1.232 direction for functional containment to consist of “multiple barriers internal and/or external to the reactor and its cooling system” was revised to delete the specific direction for multiple barriers, on the basis that the LMP methodology will determine whether multiple barriers were needed. It was noted that this change appears to conflict with recommendation No. 4 from the 2018 ACRS letter on functional containment (ADAMS Accession No. ML18108A404), which states “A functional containment should include multiple barriers as defense-in-depth features that should be minimally dependent upon each other and diverse in nature.” The applicant stated that, regardless of the wording change made to this design criterion, their functional containment will have multiple barriers consistent with the ACRS recommendation. Additionally, SER L&C No. 2 states that acceptance of such changes is “based on the assumption that the LMP process will be implemented during licensing and will address the topics of defense-in-depth, redundancy, independence, and/or diversity”. Since the LMP process requires consideration of uncertainty and the need for defense in depth to account for such uncertainty and given the important role of functional containment in mitigating uncertainties, the applicant would likely be unable to justify a functional containment that does not have multiple barriers. Based on the applicant’s intent to include multiple barriers and the assumptions stated in L&C No. 2, the reworded criterion No. 16 appears to be acceptable.

The opinion of the Subcommittee is that the staff’s SE report is sufficiently complete to recommend that it be issued. It is also recommended that this writeup serve as the record of the subcommittee meeting and an ACRS letter report not be prepared.

The Committee agreed with the Subcommittee recommendations.

6. Member Ballinger led a discussion of the ACRS activities associated with the increased enrichment draft rulemaking package to include:

    a. November 19, 2024, Subcommittee meeting on RG 1.183, revision 2, topics: pathway specific source term using MELCOR (Electric Power Research Institute Modular Accident Analysis Program (MAAP) runs), and updated to 2023 source term presentation,

    b. December 18 through 20, 2024, Subcommittee meeting on draft rule language and package;

    c. December 20, 2024, Subcommittee meeting on entire RG 1.183, revision 2,

d. January 2025 Subcommittee meeting on transition break size, FFRD guidance documents, and

e. Letter report on draft rule language during February 2025 FC meeting.

7. Members Roberts and Bier led a discussion of the TerraPower Natrium Human Factors Engineering TR.

   This TR provides elements of the human factor engineering processes to be used during development of the Natrium safety analysis. It is incorporated by reference in Section 11.2 of the Natrium PSAR. Members Roberts and Bier concluded that Committee review of this topical report in advance of the Committee's review of the PSAR is not necessary, for two reasons: (1) there is nothing innovative or novel in the content of the topical report, which describes a graded approach to human factors engineering; and (2) the content of PSAR section 11.2 provides context for how the process will be applied, leading to a more informed review by the Committee. Therefore, Members Roberts and Bier recommend that the NRC staff be informed that the ACRS does not intend to review this topical report.

   The Committee agreed with the recommendation.

8. Member Ballinger led a discussion about the details concerning the site visit by several members to the General Electric-Hitachi fuel fabrication facility scheduled for September 17, 2024. The ACRS support staff will provide additional information on logistics to support the trip.

   The group will travel to the facility on September 16, tour the facility on September 17, and return that evening or on September 18.

   The following persons are scheduled to attend:

<u>Members</u>	<u>Staff</u>
Ron Ballinger	Scott Moore
Matt Sunseri	Rob Krsek
Bob Martin	Derek Widmayer
Tom Roberts	Christopher Brown
Craig Harrington	Mike Greenleaf, RGN II
Greg Halnon	
Scott Palmtag	
Steve Schultz - Consultant	

9. Member-At-Large Petti led a discussion of potential review of a Kairos TR entitled, "Safety Analysis Methodology for the Kairos Power Fluoride Salt-Cooled High-Temperature Test Reactor [KP-FHR]."

   The TR describes the methodology used to evaluate the KP-FHR Maximum Hypothetical Accident (MHA) and other postulated events that are bounded by the MHA. The report provides modeling for the source term, descriptions of postulated events, and a description of the KP-SAM code and how the code is applied to each of the postulated events. Given the Committee's new approach for deciding on whether to review a TR and because the report deals directly with the source term and the MHA construction, Member-At-Large Petti recommended reviewing the TR. Member Petti will lead the source term part of the review but asks Members Martin and Palmtag to look at the safety analysis piece carefully.

The Committee agreed with the recommendation.

10. Member Petti also led a discussion of the Kairos TR entitled, "Core Design and Analysis Methodology."

    The TR describes the methods used to design and analyze the pebble bed core in the KP-FHR using Serpent 2 (core design) and STAR-CCM+ (thermal, fluid and discrete element modeling) and a number of Kairos-developed wrapper codes. The TR presents the approach to accounting for uncertainties and capturing biases in the modeling. While there are some unique aspects here related to a pebble bed, given the Committee's new approach for deciding on whether to review a TR, Member-At-Large Petti recommended not reviewing the TR.

    The Committee agreed with the recommendation.

11. Member Sunseri led a discussion about potential review of several guidance documents related to license renewal.

    NRR/Division of New and Renewed Licenses (DNRL) initially contacted ACRS regarding updates to several Subsequent License Renewal guidance documents during the summer of 2022. The documents were being updated to incorporate lessons learned and current practices. Because ACRS was very active in the subsequent license renewal process at the time, Member Sunseri concluded that a meeting to go over the proposed updates was not warranted. Member Sunseri requested that NRR/DNRL contact us when the documents were prepared for public comment presentation.

    In March 2023, the following documents were presented to Member Sunseri for consideration of ACRS review.

    a. NUREG-2191, Volumes 1 and 2, "Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report," July 2017 (ADAMS Accession Nos. ML17187A031 and ML17187A204)

    b. NUREG-2192, "Standard Review Plan for Review of Subsequent License Renewal Applications for Nuclear Power Plants," July 2017 (ADAMS Accession No. ML17188A158)

    c. NUREG-2221, "Technical Bases for Changes in the Subsequent License Renewal Guidance Documents NUREG–2191 and NUREG–2192," December 2017 (ADAMS Accession No. ML17362A126)

    Member Sunseri conducted a review of the proposed changes and concluded that DNRL was indeed largely incorporating lessons learned and current practices. Member Sunseri recommended at the April 2023 Full Committee Procedures and Practices meeting that no ACRS review was necessary prior to release for public comments. Member Sunseri also recommended that the decision for an ACRS review be deferred until after public comments were received and dispositioned. The ACRS concurred with these recommendations.

In July 2024, DNRL provided Member Sunseri with the record of public comments and disposition of those comments. Member Sunseri reviewed the comments and concluded that the updates incorporate current practices and operating experience, lessons learned and staff efficiencies. Industry provided numerous comments, most of which staff has either fully or partially accepted (Member Sunseri went through the disposition of all 200 plus pages of comments). Main areas of disagreement were in high density polyethylene/carbon fiber reinforced polymer repair aging management programs, nondestructive examination associated with gray cast iron pipe, inaccessible cable inspections, some boiling water reactor-IV issues and use of references not previously reviewed by NRC. Member Sunseri concludes that staff has sufficiently justified their positions on the disagreements. Furthermore, Member Sunseri does not consider any of the disagreements to have safety significance.

Member Sunseri does not recommend that the ACRS review the proposed changes due to the fact that we have been actively engaged in the process and that the updates incorporate current practices and operating experience, lessons learned and staff efficiencies.

The Committee agreed with the recommendation.

12. Member Bier led a discussion of potential review of Interim Staff Guidance (ISG) document entitled, “Content of Risk Assessment and Severe Accident Information in Light-Water Power Reactor Construction Permit Applications.”

    The staff has developed an ISG on content of risk assessment and severe accident information in light-water power reactor construction permit (CP) applications. The staff developed this guidance in response to a request from potential construction permit applicants on needed content for risk assessment and severe accident information. A draft of the ISG was distributed to the Members via email on July 11, 2024. The ISG includes a “Hazard-Specific Information” section that covers all of the associated subsections in the American Society of Mechanical Engineers (ASME) Standard, and it includes meeting the Commission’s quantitative health objectives. Lead Member Bier along with Member Dimitrijevic recommend accepting the staff’s request that the Committee consider possible review of the ISG after resolution of public comments but before final issuance.

    The Committee agreed with the recommendation.

13. Vice Chair Halnon led a discussion of potential review of Holtec’s Risk Significance Determination Methodology Topical Report for the SMR-300 design.

    This report describes the methodology that HOLTEC has developed to identify candidate risk-significant SSCs using the SMR-300 probabilistic safety assessment (PSA). This methodology uses alternative risk significance criteria other than those given in RG 1.200 (ADAMS Accession No. ML20238B871). The RG 1.200 discusses ‘significant’ in terms of relative risk criteria and defines the basic events that have a Fussell-Vesely (FV) importance greater than 0.005 or a risk-achievement worth (RAW) greater than 2 as ‘significant’.

Since the relative importance measures in RG 1.200, RAW and FV, are based on the relative risk associated with the operating fleet of reactors, they do not account for the lower risk profile of the passive SMR-300 design. Applying the relative risk criteria outlined in RG 1.200 to SMR-300 would elevate the significance of SSCs that do not have commensurate contribution to risk in the SMR-300 design.

For the SMR-300 design, Holtec is directly addressing the ratio limitations of the RAW and FV traditional importance measures by implementing an alternative methodology that adjusts these ratio limits based on the estimated risk level to ensure that measurable contributors to risk are identified regardless of the risk profile. This is consistent with other advanced reactor designers’ approach to applying risk significance.

There is a larger policy issue that should be considered outside of this TR. The risk importance factors (FV and RAW) certainly will show that very important SSCs and human actions provide for low absolute risk numbers ($<10^{-6}$), however the relative risk may remain high and of concern to the overall risk profile of the plants. A RAW of > >2 for SSCs will be normal and sole use of absolute risk may possibly screen out very important systems and actions. Further discussion on this topic is warranted, but for this Topical Report, Vice Chair Halnon (in consultation with Member Dimitrijevic) recommends no further review.

For the subject of review of TR, the Committee agreed with the recommendation not to review the document.

It was also agreed that regarding the generic issue of use of PRA for advanced reactors which have a very low risk profile (DiD issues, etc.), interaction with the staff in the future is warranted. Members Bier and Dimitrijevic will provide topics to discuss with the staff (NRR and RES). ACRS support staff will provide research and provide information (ACRS letters, etc.) on how other advanced reactor designs use risk information to inform risk significance determinations.

Member Roberts added that the October 1, 2024, subcommittee meeting on how the staff uses risk information in reviewing the size of emergency planning zones should also be relevant to the generic topic.

14. Member-At-Large Petti led a discussion of the plans to produce the report on the triennial review of NRC’s safety research program including lead member assignments as follows:

    Note: The target time frame for production of the report is now November 2024.

Topic	**SC/FC mtg date**	**ACRS Member Lead**
Integration of Source Term Activities in Support of Advanced	SC: February 17, 2022 FC: March 2-4, 2022	Petti

Reactor Initiatives	ACRS Letter: April 4, 2022	
Digital Twins Information Briefing	FC: May 4, 2022	Bier
Update on NRC Materials Harvesting Activities	FC: October 6, 2022	Sunseri
Level 3 Probabilistic Risk Assessment	SC: June 22, 2022, and October 19, 2023 FC: November 1, 2023	Dimitrijevic
Implementing the NRC's Artificial Intelligence (AI) Strategic Plan Fiscal Years 2023-2027	SC: November 15, 2023	Bier
High Burnup Fuel Source Term Accident Analysis	SC: November 16, 2023	Petti
Research Information Letter on Fuel Fragmentation, Relocation and Dispersal at High Burnup	FC: December 1, 2021	Ballinger
Advanced Manufacturing	FC: July 6, 2022	Sunseri
How machine learning is influencing Non-Destructive Examination and Inservice Inspection	FC: March 6, 2024	Ballinger
Non-Light Water Reactor code development update	FC: April 3, 2024	Martin
High Energy Arc Faults	FC: March 7, 2024	Roberts
Risk assessment and human factors for non-light water reactors	FC: July 10, 2024	Dimitrijevic and Bier

Input from the highlighted areas is needed by September 13, 2024, and should address the scope and breadth of the research activities, timeliness, and the agility of the staff's research in supporting licensing decisions.

It was agreed that the staff will place information on the SharePoint site (letters, presentations, etc.) and communicate location to Members.

15. Chair Kirchner and Technical Assistant Krsek led a discussion of recently updated or issued member guidance and templates. They discussed the development of the most recent template for opening statements for members to use for meetings. A new ACRS Guidance and Template folder on the Committee's SharePoint site has been created and contains the most current word version of guidance and templates will be kept for member and ACRS support staff use.

16. Chair Kirchner led a discussion and review of chapter memoranda for the NuScale US460 standard design approval application.

    The Committee reviewed and provided input for the following chapter reviews:

    a. Chapter 7, “Instrumentation and Controls” (Member Roberts lead),

    b. Chapter 9, “Auxiliary Systems” (Member Sunseri), and

    c. Chapter 12, “Radiation Protection” and (Member-At-Large Petti).

17. Executive Director Moore stated that there were no reconciliations for this meeting.

18. During the discussion of proposed additional topics, the next steps for the Seabrook ASR issue was broached. The Committee agreed that the staff would target the next meeting in about six to nine months with the proposed focus to be on the status of the licensee’s (NextERAs) evaluation of the progression of ASR and proposed corrective actions.

19. There was a closed portion of the meeting held to discuss personnel and administrative issues.

20. The following topics are on the agenda of the 719th ACRS FC meeting which will be held on October 2 through 4, 2024:

    - Pressurized Water Reactor Owners Group (PWROG) TR on use of direct fracture toughness for evaluation of reactor pressure vessel integrity, and

    - Palisades nuclear plant restart information briefing.

Sincerely,

Halnon, Gregory signing on behalf of Kirchner, Walter on 10/09/24

Walter L. Kirchner
Chair

October 9, 2024

SUBJECT: SUMMARY REPORT – 718th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, SEPTEMBER 4-5, 2024

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