Code Qualification Plan for an Advanced Austenitic Stainless Steel, Alloy 709, for Sodium Fast Reactor Structural Applications [1]

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Abstract. Sodium Fast Reactor (SFR) is one of the leading advanced reactor concepts that could provide a low-carbon energy option to a diverse U.S. power sources. Nuclear energy releases zero carbon emissions during electricity production, and thus is essential in reducing CO_2 emissions from the U.S. power sector. SFR also supports other possible missions, including recycling of used fuel for closing the fuel cycle.

Improved structural material performance is one way to improve the economics of SFRs; by increasing thermal efficiency, power output, and design lifetimes of the reactor system. Improved performance and reliability of structural materials could also enable greater safety margins and more stable performance over longer times, and reduce down time of the reactor plant. Advanced materials could also spur improvements in high temperature design methodologies and thereby allowing design simplifications and more flexibility in plant operations. Thus, they could have a significant, positive impact on levelized electricity production cost even if the commodity costs for the advanced materials are higher. Capital cost reduction and improvement in economic return are important incentives for commercial deployments of SFRs.

Alloy 709 is an advanced austenitic stainless steel with enhanced creep strength relative to Code-approved reference construction materials (Type 304 and 316 stainless steels) and that makes it an attractive candidate construction material for SFR structural applications.

In this paper, a qualification plan for developing an ASME nuclear code case is reviewed.

Key Words: Code Qualification, Alloy 709, Advanced alloy, Structural application.

1. Introduction

For the large-scale industrial deployment of advanced fast reactors, there must be improvements in the capital cost and economic return of such reactors. Further, greater safety margins and increased design flexibility will also be required for any new system. Flexibility, safety, and economics have been identified as key needs for advanced nuclear reactors. Advanced materials play an important role in fulfilling these needs.

Improved structural material performance is one way to improve the economics of fast reactors, by potentially allowing both higher operating temperatures (and thus, higher thermal efficiency and power output) and longer lifetimes for components. Improved materials reliability could also result in reduced down time. Superior structural materials will also spur improvements in high temperature design methodology, and thereby allow more flexibility in construction and operation. Advanced materials can have a significant impact on controlling capital construction costs, even if the raw materials are more expensive than traditional steels.

Advancements in materials performance also enable greater safety margins and more stable performance over a longer lifetime.

The family of 20Cr25Ni austenitic alloys has a long history of development and application experience. Optimization and properties evaluation of 20Cr25Ni/Nb austenitic alloy were carried out by the nuclear-related centers in the U.K. in the mid-1950s, and it has been deployed for nuclear fuel cladding application in the British Advanced Gas-cooled Reactor (AGR) fleet since 1962.

The strength of 20Cr25Ni/Nb is relatively low. During the late 1980s and 1990s, Nippon Steel modified the chemical compositions of 20Cr25Ni/Nb to improve its strength and developed the so-called NF 709 austenitic alloy specifically for use in ultra-supercritical boilers [1]. The composition was designed to produce (a) stable austenite devoid of sigma and other intermetallic phases under long-time high-temperature service conditions, (b) creep-strengthening by carbo-nitride M(CN) precipitates in a stable, fine dispersion, (c) sold solution strengthening from Mo and N in solution, (d) excellent steam side corrosion, and (e) good coal ash corrosion resistance.

An ASME Code Case requested for Section I on power boilers was proposed by Mitsubishi Heavy Industries, Ltd. of Japan in 2006. The material received an ASTM austenitic stainless steel grade designation of TP310MoCbN and an UNS number of S31025. It was approved for Section I construction to 816°C (1500°F) in Code Case 2581 in 2007. The code case is restricted to seamless tubes. But it is worthwhile to note that the data requirements for a new elevated temperature material code case in Section III, the nuclear section of the ASME code, are significantly more involved than those for Section I. Guidelines on data requirements for new Section III elevated temperature material code case are provided in Section III, Division 5, Appendix HBB-Y. For example, the allowable stresses for Section I applications are based on extrapolating creep rupture data to 100,000 hours, while those for Section III applications are time dependent and design lifetimes up to 300,000 hours, or even 500,000 hours for some new reactor designs, are required.

This paper discusses some of the issues on the code qualification of Alloy 709.

2. Results from Intermediate Term Testing of Alloy 709

An intermediate term collaborative test program was initiated in 2013 by Argonne National Laboratory, Idaho National Laboratory and Oak Ridge National Laboratory to develop and evaluate Alloy 709 in plate product form. Tasks were performed (i) to optimize steel fabrication process parameters to support industrial scale steel production, (ii) to verify that the observed performance of Alloy 709 from small lots, sub-size specimen, accelerated testing is retained for larger lots, standard size specimen, testing, and (iii) to initiate work on welding processes and weldment testing. The results from this intermediate term test program led to a number of important findings.

From the thermal aging and sodium compatibility studies in terms of tensile properties, hardness measurements, and microstructural analysis, it was concluded that Alloy 709 is a stable material when exposed in liquid sodium at elevated temperatures.

The steel fabrication process parameters were optimized and they led to an improvement of the creep-fatigue performance of Alloy 709 without sacrificing its creep strength as compared with standard fabrication process parameters.

Data from the intermediate term testing continued to show the performance enhancement of Alloy 709 over the reference 316H stainless steel.

A matching Alloy 709 filler metal having optimized chemical composition within the specification of the wrought metal was developed to improve the weldability. Weldments from the optimized Alloy 709 filler metal and a nickel-base filler SFA-5.14 class ERNiCrMo-3 (Alloy 625) were successfully fabricated using automatic gas-tungsten arc welding process.

Based on an assessment of the benefits to design flexibility, safety, and economics over the reference 316H stainless steel, Alloy 709 was recommended for code qualification so that it can be used as a construction material for reactor vessel, piping, and core support structure applications in SFRs.

In order for the designers to leverage the advantages of Alloy 709 in SFR applications, Alloy 709 has to be incorporated in Section III (the nuclear section), Division 5, Subsection HB, Subpart B (HBB) of the ASME Boiler and Pressure Vessel Code. An Alloy 709 Code Case for ASME Section III, Division 5, Class A and Class B applications up to 500,000 h is required to support a 60-year SFR design lifetime. Incorporation of short term (3,000 h) data for very high temperature excursions into the Code Case could provide design analysis flexibility.

3. Capital Cost Reduction and Design Advantages

While the significantly higher strength of Alloy 709 over 316H stainless steel permits the use of thinner walls for many SFR structural components, resulting in the reduction of construction commodity and hence capital cost, there are many other design advantages of Alloy 709 over 316H. One example is the ability of Alloy 709 to resist greater thermal gradients, resulting in the prospect of eliminating costly add-on hardware needed to mitigate the deficiency of 316H and enabling more efficient designs due to the widening of the design envelope.

In addition to the reactor vessel, piping and core support structure applications, there are other potential applications of Alloy 709 within a SFR system. One potential application is for a compact heat exchanger that couples a SFR plant to a supercritical CO_2 Brayton energy conversion system. Alloy 709 is compatible with sodium. Some preliminary data show that the corrosion resistance of Alloy 709 against supercritical CO_2 fluid is higher than 316H. The pressure difference across the material that separates the liquid sodium from the supercritical CO_2 fluid is high and hence the higher creep strength of Alloy 709 over 316H is an advantage in the compact heat exchanger application.

The other potential application of Alloy 709 is for an intermediate heat exchanger. The current reference material for the intermediate heat exchanger in a SFR system is the Grade 91 ferritic-martensitic steel because of the lower thermal expansion coefficient and higher thermal conductivity as compared with the 300 series stainless steels. These properties are important in an IHX design. The conceptual assessment shows that if Alloy 709 is adopted as the reference material for SFR reactor assembly construction, the incentive to use Grade 91 over Alloy 709 for the IHX is limited from an overall performance perspective. An Alloy 709 IHX also removes the need for using ferritic-to-stainless steel dissimilar metal welds in the construction, and hence eliminating a potential concern on structural integrity issues associated with dissimilar metal welds.

4. ASME Code Qualification

A strategic plan developed for ASME Code qualification of Alloy 709 calls for the development of a code case for 300,000 h design lifetime, followed by a code case for 500,000 h. An assessment was made on the data requirement in support of 300,000 h versus

500,000 h design lifetime. It was concluded that only thermal aging and creep rupture data require the so-called "hard" extrapolations and hence additional thermal aging and creep rupture data are required in extending the applicable lifetime from 300,000 h to 500,000 h for Alloy 709. Therefore, all major testing in support of the Alloy 709 Code Case, except the long-term thermal aging and creep rupture tests, will be complete when the data package for the 300,000 h code case has been assembled.

The incorporation of new materials into ASME, Section III, Division 5, Subsection HB, Subpart B, or its forerunner Division 1, Subsection NH, had been challenging as there were no formal guidelines on the data requirement for new materials. Through the DOE Office of Nuclear Energy sponsored multi-laboratory efforts on the development of the Alloy 617 Code Case in support of HTGR/VHTR applications, new guidelines on data requirement for new materials were established. These new guidelines formed the basis for the development of the Section III, Division 5, Appendix HBB-Y, entitled, "Guidelines for Design Data Needs for New Materials." Appendix HBB-Y has been incorporated in the 2015 Edition of the ASME Code.

Following the guidelines provided in Appendix HBB-Y, test plans have been developed to generate the data package in support the Alloy 709 Code Case. The tasks are summarized in Table 1.

Table 1. TESTING TASKS FOR ASME CODE IMPLEMENTATION

Test Group	Properties	No. of spec.	ASTM Standard	Aging temp. (°C)	Test temp. (°C)	Longest creep, stress relaxation, aging or hold time per spec. (h)
A1	Thermal expansion	n/a	E228	n/a	20, 50 to 1000 in 50°C increment	n/a
	Thermal diffusivity	n/a	E2585	n/a	ditto	n/a
	Heat capacity	n/a	E1269	n/a	ditto	n/a
	Density	n/a	B311	n/a	ditto	n/a
	Dynamic Young's modulus and dynamic shear modulus	n/a	E1875	n/a	–195, 20, 50 to 1000 in 50°C increment	n/a
A2	Tensile properties	200	E8, E21	n/a	20, 50 to 900 in 50°C increment	n/a
A3	Tensile aging factor (short term)	128	E8, E21	500 to 900	20, aging temp	3,000, creep
	Tensile aging factor (intermediate term)	32	E8, E21	500 to 900	20, aging temp	25,000, creep
	Tensile aging factor (long term)	32	E8, E21	500 to 900	20, aging temp	100,000, creep
A4	Creep rupture (short term)	135	E139	n/a	600, 625, 650, 675, 700, 750, 775, 800, 850, 900, 950	6,000, creep

Test Group	Properties	No. of spec.	ASTM Standard	Aging temp. (°C)	Test temp. (°C)	Longest creep, stress relaxation, aging or hold time per spec. (h)
	Creep rupture (intermediate term)	45	E139	n/a	550, 600, 650, 700, 725, 750, 800, 850, 900, 950	25,000, creep
	Creep rupture (long term)	36	E139	n/a	550, 600, 650, 700, 750, 800	100,000, creep
A5	Cross-weld - tensile	80	E8, E21	n/a	20, 50 to 900 in 50°C increment	n/a
A6	Cross-weld stress rupture (short term)	22	E139	n/a	600, 650,700, 750, 800, 900	3,000, creep
	Cross-weld stress rupture (intermediate term)	24	E139	n/a	550, 600, 650,700, 750, 800, 900	25,000, creep
	Cross-weld stress rupture (long term)	4	E139	n/a	550, 650	60,000, creep
A7	Stress relaxation	6	E328	n/a	550, 650, 760	10,000, stress relaxation
A8	Creep-fatigue	70	E2714	n/a	550, 650, 760	8, hold time
A9	Fatigue	192	E606, E466	n/a	550, 650, 760, 900	n/a
A10	Fatigue (Subsection NB)	26	E606, E466	n/a	20, 427	n/a
A11	Cross-weld fatigue and creep-fatigue	68	E606, E2714	n/a	550, 650, 760	4, hold time
A12	Constitutive - uniaxial	66	n/a	n/a	500, 550, 600, 650, 700, 760	n/a
A13	Stress rupture - biaxial	12	n/a	n/a	650	6,000, creep
A14	Cold work effect - stress rupture	8	E328	n/a	600, 750	25,000, creep
A15	Cold work effect - creep-fatigue	32	E2714	n/a	600, 760	2.5, hold time
A16	Cold work effect - fracture toughness	40	E1820	n/a	550, 650, 760	n/a
A17	Two-bar thermal ratcheting	40	n/a	n/a	650-760, 550-760, 350-650, 250-550	n/a
A18	SMT creep-fatigue	94	n/a	n/a	550, 650, 760	5, hold time

5. Design Rules and Licensing Issues

During the application of a construction permit by the Clinch River Breeder Reactor (CRBR) Project, the U.S. Nuclear Regulatory Commission (NRC) and its Advisory Committee for Reactor Safeguards identified some structural integrity issues for the CRBR design. Before these issues were resolved, the licensing process was stopped because of the abrupt cancellation of the CRBR project. NRC had also performed a pre-application safety evaluation of the Power Reactor Innovative Small Module (PRISM) liquid-metal reactor design in the mid 1990s and similar issues were raised. The so-called NRC Issues List, consisting of 25 items, has been compiled by O'Donnell and Griffin [2] under the sponsorship of the DOE/ASME Gen IV Materials Project.

Since the days of the CRBR Project, numerous improvements that were made to the ASME Code rules used by the CRBR design have been incorporated into Division 1, Subsection NH, and now Division 5, Subsection HB, Subpart B. During the development of the Next Generation Nuclear Plant (NGNP) Project, the items on the NRC Issues List were assessed by Wright et al. [3] with respect to the use of Alloy 617 as a construction material for an intermediate heat exchanger in a high temperature gas-cooled reactor system with a design outlet temperature of 950°C.

A similar assessment of the NRC Issues List with respect to the use of Alloy 709 as a construction material for core supports, reactor vessel, primary and secondary piping and other elevated temperature metallic components is given in Table 2. Actions to resolve these issues for Alloy 709 are also included in the assessment.

TABLE 2. ASSESSMENT OF THE NRC ISSUES LIST FROM CRBR PROJECT REVIEW WITH RESPECT TO USE OF ALLOY 709 IN SFR APPLICATIONS

Structural integrity issues identified by NRC for CRBR	Assessment of Alloy 709 with respect to reactor vessel, piping and core support structures for sodium fast reactor applications (both pool and loop type)	Required actions
Transition joints (i.e., dissimilar metals)	Code specified approach is to model joint with base metal properties to the weld centerline, then include differences in connecting base metal properties in the weldment stress analysis.	This issue needs to be addressed if such transition joints are present in the vendor design concept.
Weld residual stresses	Not considered in HBB methodology – current approach implies selection of weld wires and welding process produce ductile welds and subsequent load cycling and creep reduce residual stresses.	Post-weld heat treatment is generally not required for stainless steels such as 304H and 316H, and similarly, it is not required for Alloy 709. While weld residual stresses are not explicitly addressed in HBB, the data to be collected on testing recommended for Alloy 709 weldment will include whatever effect weld residual stress might have on the measured properties.
Design loading combinations	Owner/regulator issue - beyond scope of HBB.	This is an action for the reactor vendor.

Structural integrity issues identified by NRC for CRBR	Assessment of Alloy 709 with respect to reactor vessel, piping and core support structures for sodium fast reactor applications (both pool and loop type)	Required actions
Creep-rupture and fatigue damage	This is a valid concern.	Task 10 in DOE/ASME Gen IV Materials Project addressed improvements in the creep-fatigue damage evaluation procedure. The initial focus is on improvements to the rules for 9Cr-1Cr-V. Some initial recommendations have been implemented and other improvements are in work by the ASME Code committees.
Simplified bounds for creep ratcheting	This is a valid concern.	The current simplified methods for creep ratcheting bounds have not been verified for Alloy 709.
Thermal striping	Current HBB rules provide framework for assessment of structural response. Generally, issue is determining thermal hydraulic response. Computational fluid dynamics techniques may be needed.	High cycle fatigue data are needed for fatigue damage evaluation under thermal striping conditions.
Creep-fatigue analysis of Class 2 and 3 piping	The code case for Division 1, Class 2 and 3 elevated temperature component design rules has been replaced by the Class B component design rules in Section III, Division 5. Piping is the only component with a specified creep-fatigue analysis procedure in the Class B rules.	Probably not applicable to reactor vessel, primary piping and core support structures which would likely be designed to Division 5, Class A requirements.
Limits of Case N-253 for elevated temperature Class 2 and 3 components met?	The code case for Class 2 and 3 elevated temperature component design rules has been replaced by the Class B component design rules in Section III, Division 5. Creep-fatigue analysis for piping under Class B only deals with pipe loads and does not default to the creep-fatigue design-by-analysis procedure in HBB.	Probably not applicable to Reactor Vessel, Piping and Core internals which would likely be designed to Class A requirements. However, this is currently a priority item for ASME SG-ETD.
Creep buckling under axial compression – design margins	Code committee responsible for HBB not aware of any generic issues or inconsistencies within the creep buckling rules – particularly for thick walled components. This may be a local crimpling issue for very large diameter, thin-walled vessel.	This is a lower-tier issue. No immediate action is recommended.
Identify areas where Appendix T rules are not met	Appendix HBB-T provides procedures to determine strain range using elastic analysis. If these rules cannot be satisfied, additional rules are provided, based on results of inelastic analyses. However, inelastic analysis requires detailed constitutive models of material behavior under time varying loading conditions.	Development of experimentally verified unified constitutive model for Alloy 709 will be required to provide inelastic analysis tools for expected critical areas of the reactor vessel, piping and core support structures expected to have low design margins in critical locations.
Rules for component supports at elevated temperature	Rules are provided in Section III, Division 5 for Core Support Structures. These rules are essentially the same as those previously in Code Case 201.	No immediate action is required.

Structural integrity issues identified by NRC for CRBR	Assessment of Alloy 709 with respect to reactor vessel, piping and core support structures for sodium fast reactor applications (both pool and loop type)	Required actions
Strain and deformation limits at elevated temperature	Data is not currently available to verify the applicability of the current elevated temperature design rules for strain, deformation and creep-fatigue damage to Alloy 709.	Creep and creep-fatigue testing of Alloy 709 to support validation of current and newly proposed design rules with appropriate conservatism is needed..
Evaluation of weldments	A number of provisions in HBB and related documents assure reliable weld joints. HBB methods exceed current requirements for non-nuclear applications as well as nuclear applications below the creep regime. However, Alloy 709 data to support these rules are not available.	Creep and creep-fatigue testing of Alloy 709 weldments is needed to support validation of current and newly proposed design rules with appropriate conservatism.
Material acceptance criteria for elevated temperature	Data to support a 60-year design life for Alloy 709 is not currently available although it has been approved for limited non-nuclear applications based on extrapolated properties to 100,000 h. The reliability of extrapolating shorter-term data to much longer design lifetimes for Alloy 709 is a valid issue. The ability to demonstrate confidence in using accelerated test data to predict performance for a 60-year design lifetime is important for licensing success.	A two-pronged approach is needed: (1) develop a strategy to generate longer-term data, (2) leverage the advancement in understanding of deformation and failure mechanisms to transition from current empirical practice in the Code to a more science-based approach. Very long term Alloy 709 base metal data, will be generated to quantify the effects of thermal aging on tensile properties.
Creep-rupture damage due to forming and welding	This issue is also covered under Issue 2. Issues related to forming/cold work are addressed in HBB-4000.	This needs to be addressed as part of the Code Case for new HBB Code material.
Mass transfer effects	This is an important area not covered by specific code rules in HBB. This is an Owner-regulator issue. However, carburization, decarburization and the effects of a sodium environment on creep rupture strength, ductility and creep-fatigue damage are important design considerations.	Test programs on mechanical properties testing of sodium-exposed Alloy 709 base metal and weldment specimens have been developed. They include tensile, creep rupture, fatigue, creep-fatigue, CFCG and fracture toughness.
Environmental effects	This is closely related to the above Issue 16.	Required action is described above in Issue 16.
Fracture toughness criteria	Creep and creep-fatigue crack growth evaluation procedures are currently being developed by the ASME BPV III Working Group on High Temperature Flaw Evaluation, based on recommendations generated by Task 8 of the DOE/ASME Gen IV Materials project. The initial goal is to support inservice inspection requirements for current HBB materials.	Testing programs to generate FCG, CCG, CFCG and fracture toughness data from Alloy 709 base metal and weldment specimens in the solution-annealed condition; and CFCG and fracture toughness data from Alloy 709 base metal and weldment specimens thermally aged and sodium-exposed have been developed.
Thermal aging effects	Thermal aging effects on allowable stresses are addressed in HBB. Thermal aging and cyclic behavior are important issues for creep-fatigue evaluation.	Test programs to generate tensile, fatigue, creep-fatigue, CFCG and fracture toughness data from Alloy 709 base metal and weldment specimens in thermally aged condition have been developed..

Structural integrity issues identified by NRC for CRBR	Assessment of Alloy 709 with respect to reactor vessel, piping and core support structures for sodium fast reactor applications (both pool and loop type)	Required actions
Irradiation effects	This is an important area not covered by specific Code rules in HBB. This is an Owner/regulator issue and is primarily relevant for core support structures, primary piping and reactor vessel.	From scoping design studies, 10 to 15 dpa is estimated to be the maximum fast neutron damage for SFR core support structures, piping and reactor vessel. Thus, a modest irradiation program has also been developed to determine the effects of fast neutron irradiation, up to 15 dpa, on tensile, creep rupture, fatigue, creep-fatigue and fracture toughness properties of Alloy 709.
Use of simplified bounding rules at discontinuities	Appendix HBB-T currently contains "simplified" bounding rules for the evaluation of strain limits and creep-fatigue damage at discontinuities. However, these so-called simplified rules are actually quite complex. A new methodology based on elastic-perfectly plastic analysis has been developed which avoids the complexities of the current rules. The new EPP code cases for strain limits and creep-fatigue are in the Code committee approval process.	The key-feature testing to support the EPP code cases for current HBB materials and Alloy 617 needs to be expanded to include Alloy 709 at relevant SFR temperatures.
Elastic follow-up	This is part of Issue 21 as accounting for the effects of elastic follow-up is a significant part of simplified bounding rules.	The required action is as described for Issue 21.
Design criteria for elevated temperature core support structures and welds	The elevated temperature core support rules where creep is significant are based on HBB.	Relevant required actions for Alloy 709 core support structures are the same as discussed elsewhere in this tabulation.
Elevated temperature data base for mechanical properties	Appendix HBB-Y of Division 5 has been recently approved by the ASME Code committees. This appendix provides guidance for the database required to obtain Code approval for the use of a new material in elevated temperature Division 5, Class A (Division 1, Class 1) construction.	Test plans have been developed, following the recommendations of Section III, Division 5, Appendix HBB-Y, to generate required data in support of a Division 5, Alloy 709 Code Case..
Basis for leak-before-break at elevated temperatures	Leak-before-break criteria are beyond the scope of current ASME Code rules. However, pressurized tube tests of austenitic stainless steel and a representative nozzle to sphere test failed by through-wall leakage rather than bursting for all but very short duration tests. The leakage was through the development of localized crack growth linking cavities that developed due to creep.	Addition pressurized tube tests and key-feature pressurized component tests using Alloy 709 are needed to validate leak-before break.

6. Summary

Currently, 304H and 316H stainless steels are approved for the construction of Class A and Class B nuclear components intended for elevated temperature under the rules of ASME Section III, Division 5, Subsection HB, Subpart B. Although allowable stress intensities for 304H and 316H are provided to 1500°F (816°C), the steels are relatively low in strength.

Until recently, this coverage has been adequate to meet the needs of the sodium fast reactor designs produced in the 1970s.

The interest in improving structural material performance stems from its beneficial impact on reactor economics, safety margins, and design flexibility. Advanced structural materials with superior high temperature strength and creep resistance can reduce the quantity of materials that need to be used in reactors, promote higher reactor operation temperature for greater thermal efficiency and power output, and improve safety margins during structural design that allow more flexibility in reactor simplification and construction.

Alloy 709 is an advanced alloy and it is compatible with sodium and has good thermal stability. Alloy 709 has great advantages over 316H on many design aspects. It is a relatively matured alloy due to its use in power boiler applications and hence there is user experience base for Alloy 709. Alloy 709 is an all-around construction material with enhanced material performance that would provide design benefits to advanced SFR systems.

A detailed code qualification plan has been developed for Alloy 709. The plan includes material procurement strategy, development of ASTM specifications, ASME Section IX welding specifications, and detailed testing plan per guidelines from Division 5 Appendix HBB-Y on codification of new materials. Some elements of this plan have been discussed in this paper.

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