February 15, 2002

Mr. Michael R. Kansler
Senior Vice President and
Chief Operating Officer
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

SUBJECT: INDIAN POINT NUCLEAR GENERATING UNIT NO. 2 - AMENDMENT RE: UPDATED PRESSURE-TEMPERATURE LIMIT CURVES (TAC NO. MB2419)

Dear Mr. Kansler:

The Commission has issued the enclosed Amendment No. 224 to Facility Operating License No. DPR-26 for the Indian Point Nuclear Generating Unit No. 2 (IP2). The amendment consists of changes to the Technical Specifications (TSs) in response to an application by the Consolidated Edison Company of New York, Inc. (Con Edison), the former licensee of IP2, for an amendment to the license dated July 16, 2001. On September 6, 2001, Con Edison's interest in the license was transferred to Entergy Nuclear Operations, Inc. (ENO). By letter dated September 20, 2001, ENO requested that the U.S. Nuclear Regulatory Commission (NRC) continue to review and act on all requests before the Commission which had been submitted before the transfer. Accordingly, the NRC staff has acted upon the request. The request for an amendment of July 16, 2001 was supplemented by ENO on January 11, 2002.

The amendment updates the pressure-temperature (P-T) limit curves for IP2.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly *Federal Register* notice.

Sincerely,

/RA/

Patrick D. Milano, Sr. Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-247

Enclosures: 1. Amendment No. 224 to DPR-26
2. Safety Evaluation

cc w/encls: See next page

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Docket No. 50-247

Enclosures: 1. Amendment No. 224 to DPR-26
2. Safety Evaluation

cc w/encls: See next page
DISTRIBUTION: See attached list **TSs: ML020500498**
Package: ML020520660 *Safety Evaluations submitted on December 11, 2001 (SRXB) and January 25, 2002 (EMCB); no major changes
Accession Number: ML020420477 **See previous concurrence

OFFICE	PDI-1/PM*	PDI-1/PM	PDI-1/LA	SRXB/SC*	EMCB/ASC*	OGC**	PDI-1/ASC
NAME	JMunday for GVissing	JMunday for PMilano	SLittle	FAkstulewicz	BElliott	SUttal	JMunday
DATE	2/15/02	2/15/02	2/15/02	12/11/01	01/25/02	02/13/02	2/15/02

OFFICIAL RECORD COPY

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DATED: February 15, 2002

AMENDMENT NO. 224 TO FACILITY OPERATING LICENSE NO. DPR-26 INDIAN POINT UNIT 2

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ENTERGY NUCLEAR INDIAN POINT 2, LLC

ENTERGY NUCLEAR OPERATIONS, INC.

DOCKET NO. 50-247

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 224
License No. DPR-26

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Consolidated Edison Company of New York (the former licensee) dated July 16, 2001, adopted by Entergy Nuclear Operations, Inc. (the licensee) pursuant to a letter dated September 20, 2001, and as supplemented on January 11, 2002, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-26 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 224, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Joel T. Munday, Acting Chief, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the TSs

Date of Issuance: February 15, 2002

ATTACHMENT TO LICENSE AMENDMENT NO. 224

FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
ix	ix
3.1.A-2	3.1.A-2
3.1.A-3	3.1.A-3
3.1.A-6	3.1.A-6
3.1.A-7	3.1.A-7
3.1.A-8	3.1.A-8
3.1.A-9	3.1.A-9
Table 3.1.A-2 (Page 1 of 2)	Table 3.1.A-2 (Page 1 of 2)
Table 3.1.A-2 (Page 2 of 2)	Table 3.1.A-2 (Page 2 of 2)
Figure 3.1.A-1	Figure 3.1.A-1
Figure 3.1.A-2	Figure 3.1.A-2
Figure 3.1.A-3	Figure 3.1.A-3
Figure 3.1.A-4	Figure 3.1.A-4
Figure 3.1.A-5	Figure 3.1.A-5
Figure 3.1.A-6	Figure 3.1.A-6
3.1.B-1	3.1.B-1
3.1.B-2	3.1.B-2
3.1.B-3	3.1.B-3
3.1.B-4	3.1.B-4
3.1.B-5	3.1.B-5
Figure 3.1.B-1	Figure 3.1.B-1
Figure 3.1.B-2	Figure 3.1.B-2
3.1.C-2	3.1.C-2
3.2-2	3.2-2
3.3-3	3.3-3
3.3-12	3.3-12
Table 4.1-1 (Page 8 of 8)	Table 4.1-1 (Page 8 of 8)
4.3-1	4.3-1
4.3-2	4.3-2
4.18-1	4.18-1

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 224 TO FACILITY OPERATING LICENSE NO. DPR-26

ENTERGY NUCLEAR OPERATIONS, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

DOCKET NO. 50-247

1.0 INTRODUCTION

By letter dated July 16, 2001, the Consolidated Edison Company of New York, Inc. (Con Edison), the former licensee of Indian Point Nuclear Generating Unit No. 2 (IP2), submitted a request for changes to the IP2 Technical Specifications (TSs) (Ref. 1). The requested changes would update the pressure-temperature (P-T) limit curves for IP2 and the cold overpressure protection system (OPS) limits. The fluence calculations are summarized in Appendix B of WCAP-15629 (Ref. 2). On September 6, 2001, Con Edison's interest in the license was transferred to Entergy Nuclear Operations, Inc. (ENO). By letter dated September 20, 2001, ENO requested that the U.S. Nuclear Regulatory Commission (NRC) continue to review and act on all requests before the Commission which had been submitted before the transfer. Accordingly, the NRC staff has acted upon the request. The request for an amendment of July 16, 2001, was supplemented by ENO on January 11, 2002 (Ref. 3).

The proposed changes of the P-T curves and the OPS affect TSs: 3.1.A "Operational Components" and 3.1.B "Heatup and Cooldown" and the associated bases. The TSs and the associated bases would be revised to reflect the proposed P-T curves and the revised cold OPS limits. The January 11, 2002, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The existing P-T limit curves are currently authorized for operation to 18 effective full-power years (EFPY). The request for approval for revised P-T limit curves would allow operation up to 25 EFPY.

2.0 BACKGROUND

The NRC has established requirements in Appendix G of Part 50 to Title 10 of the *Code of Federal Regulations* (10 CFR Part 50, Appendix G), to protect the integrity of the reactor coolant pressure boundary in nuclear power plants. The Appendix to Part 50 requires the P-T limits for an operating plant to be at least as conservative as those that would be generated if the methods of Appendix G to Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) were applied. The methodology of Appendix G to the Code postulates the existence of a sharp surface flaw in the reactor pressure vessel

(RPV) that is normal to the direction of the maximum applied stress. For materials in the beltline and upper and lower head regions of the RPV, the maximum flaw size is postulated to have a depth that is equal to one-fourth of the thickness and a length equal to 1.5 times the thickness. The basic parameter in Appendix G to the Code for calculating P-T limit curves is the stress intensity factor, K_I, which is a function of the stress state and flaw configuration. The methodology requires that licensees determine the reference stress intensity (K_{Ia}) factors, which vary as a function of temperature, from the reactor coolant system (RCS) operating temperatures, and from the adjusted reference temperatures (ARTs) for the limiting materials in the RPV. Thus, the critical locations in the RPV beltline and head regions are the 1/4-thickness (1/4T) and 3/4-thickness (3/4T) locations, which correspond to the points of the crack tips if the flaws are initiated and grown from the inside and outside surfaces of the vessel, respectively. Regulatory Guide (RG) 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2, provides an acceptable method of calculating ARTs for ferritic RPV materials. The methods of RG 1.99, Revision 2, include methods for adjusting the ARTs of materials in the beltline region of the RPV, where the effects of neutron irradiation may induce an increased level of embrittlement in the materials.

The methodology of Appendix G requires that P-T curves must satisfy a safety factor of 2.0 on stress intensities arising from primary membrane and bending stresses during normal plant operations (including heatups, cooldowns, and transient operating conditions), and a safety factor of 1.5 on stress intensities arising from primary membrane and bending stresses when leak rate or hydrostatic pressure tests are performed on the RCS. Table 1 to 10 CFR Part 50, Appendix G, provides the staff's criteria for meeting the P-T limit requirements of Appendix G to the Code and the minimum temperature requirements of the rule for bolting up the vessel during normal and pressure testing operations.

In the license amendment request, Con Edison also requested NRC approval to use two exemption methods, Code Cases N-588 and N-640, that would allow Con Edison to deviate from complying with the requirements in 10 CFR Part 50, Appendix G, for generating the P-T limit curves. These exemptions were submitted in accordance with 10 CFR 50.60(b), which allows licensees to use alternatives to the requirements of 10 CFR Part 50, Appendices G and H, requested pursuant to the provisions of 10 CFR 50.12. The first of these exemptions would allow ENO to use ASME Code Case N-588 as the basis for determining what is the most limiting material in the Indian Point RPVs, and to postulate a circumferential flaw as the limiting type of flaw in the RPV circumferential shell welds. The second of these exemptions would allow ENO to use ASME Code Case N-640, and apply the lower bound static initiation fracture toughness value (K_{Ic}) equation as the basis for establishing the P-T curves in lieu of using the lower bound crack arrest fracture toughness value equation (i.e., the K_{Ia} equation), the method invoked by Appendix G to Section XI of the ASME Code.

3.0 LICENSEE EVALUATION

The heatup and cooldown limitation curves are established to provide assurance of RPV integrity during plant operation. All components of the RCS are designed to withstand the effects of loads resulting from system pressure and temperature changes. These loads are introduced by heatup and cooldown operations, power transients, and reactor trips. In accordance with Appendix G to 10 CFR Part 50, the TSs limit the RCS pressure and temperature changes during heatup and cooldown to be within the fracture toughness requirements to preclude non-ductile failure of the carbon and low alloy RCS materials. These

limits are defined by the P-T curves for heatup and cooldown. Each curve defines an acceptable region for normal operation. These curves are used for operational guidance during heatup and cooldown maneuvering when P-T indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

ASME Code Case N-588, "Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels, Section XI, Division I," allows the use of alternative procedures for defining the orientation of postulated flaws in circumferential welds (i.e., only circumferential flaws need to be considered for circumferential welds) and for calculating the applied stress intensity factors of axial and circumferential flaws.

ASME Code Case N-640, “Alternative Fracture Toughness for Development of P-T Curves for ASME Section XI, Division I,” provides an alternate method for determining the fracture toughness of reactor vessel materials for use in determining P-T limits. This code case allows the use of the critical stress intensity factor K_{Ic} rather than the more restrictive arrest stress intensity factor K_{Ia} required by ASME Code Section XI, Appendix G.

The changes to the calculation methodology for the heatup and cooldown limitation curves based on Code Case N-640 and Code Case N-588 provide sufficient margin in the prevention of non-ductile type fracture of the RPV while maximizing operator flexibility during plant heatup and cooldown. The code cases were developed using knowledge gained through years of industry experience. However, the experience gained in the areas of fracture toughness of materials and pre-existing undetected defects show that some of the previous assumptions used for the calculation of the heatup and cooldown limitations were overly conservative. Therefore, using the methods of the subject code cases in developing the heatup and cooldown limitation curves will continue to provide protection against non-ductile failures of the carbon and low alloy steel components of the RCS.

ASME Code Cases N-588 and N-640 have not been approved in RG 1.147, “Inservice Inspection Code Case Acceptability, ASME Section XI, Division I.” Con Edison requests an exemption in accordance with 10 CFR 50.12 from the requirement of 10 CFR 50.60(a), to comply with 10 CFR Part 50, Appendix G, to allow use of ASME Code Cases N-588 and N-640 in the calculation of heatup and cooldown limitations. IP2 has evaluated the use of these ASME Code Cases and has concluded that the use of the code cases will not present an undue risk to the public health and safety, are consistent with the common defense and security, and special circumstances are present.

Con Edison evaluated the effect of neutron irradiation embrittlement on each beltline material in the IP2 RPV. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Rev. 2. Con Edison determined that the material with the highest ART at the 3/4T(T= reactor vessel beltline thickness) location at 25 EFPY is the Intermediate Shell Plate B-2002-3 with 0.25% copper, 0.60% nickel, and an ART of 145° F. The limiting material for the 1/4T location is the intermediate to lower shell girth (circumferential) weld with an ART of 200 °F. However, the use of Code Case N-588 results in Intermediate Shell Plate B-2002-3 controlling the 1/4T location with an ART of 195° F.

Con Edison has removed surveillance samples from the IP2 RPV. The test results from these samples were transmitted to the NRC. The increase in RT_{NDT} values for the limiting materials

for 25 EFPY were calculated based on these IP2 surveillance capsule results supplemented by surveillance capsule data from Indian Point Unit 3 and H. B. Robinson Unit 2.

In addition to beltline materials, 10 CFR Part 50 Appendix G imposes heatup and cooldown limitations based on the reference temperature for the RPV closure flange region materials. Section IV.A.2 of Appendix G states that when the pressure exceeds 20% of the pre-service system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by bolt preload must exceed the reference temperature of the materials in those regions by at least 120 °F for normal operation and 90 °F for hydrostatic and leak tests. The pre-service system hydrostatic test pressure for the IP2 RPV was 3106 psi. Based on the limiting unirradiated flange region RT_{NDT} of 60 °F Con Edison has determined that imposing a minimum allowable temperature limit of 180 °F when pressure exceeds 621 psig satisfies Section IV.A.2 of 10 CFR Part 50 Appendix G.

4.0 STAFF EVALUATION

For the IP2 RPV, the licensee provided the P-T limit curves for normal operating conditions and pressure testing conditions effective to 25 EFPY. To test the validity of the licensee's proposed curves, the staff performed an independent assessment of the licensee's submittal. The staff applied the methodologies of the 1995 Edition with 1996 Addenda of Appendix G to the Code and 10 CFR Part 50, Appendix G, as modified by the methodologies of ASME Code Cases N-588 and N-640, as the bases for its independent assessment.

The staff's assessment also included an independent calculation of the ART values for both the 1/4T and 3/4T locations of the IP2 RPV beltline regions based on the neutron fluence specified in the submittal for the IP2 RPV effective to 25 EFPY. For the evaluation of the limiting beltline materials, the staff confirmed that the ARTs and P-T limit curves were based on the methodology of RG 1.99, Revision 2.

The staff found agreement with the submitted P-T curves, as calculations confirmed various points on the submitted P-T limit curves were bounded by the indicated temperature. The staff also confirmed that the Con Edison's P-T limit curves included appropriate minimum temperature requirements that were at least as conservative as those required in Table 1 to 10 CFR Part 50, Appendix G, as exempted and modified by the code case methods.

4.1 Vessel Fluence

The proposed maximum fluence value for 25 EFPYs is 1.02×10^{19} n/cm^2 which occurs at 45° in the azimuth. It is stated in the submittal that this value was calculated, no adjustments were made and that the RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," guidance has been followed. Because the licensee used staff approved guidance the numerical value is acceptable for the proposed 25 EFPY. Figures 3.1.A have been calculated to satisfy Appendix G to 10 CFR Part 50, thus are acceptable.

4.2 Technical Specification Changes

TS 3.1.A, Operational Components

When the RCS temperature is less than or equal to 280 °F (from 305 °F), the requirements of specification 3.1.A.4 regarding startup of a reactor coolant pump will be adhered to. The change of the enable temperature from 305 °F to 280 °F resulted from the limit calculations using the new fluence value and applying Code Case N-588. This change reflects the new overpressure protection enable limit temperature and is acceptable because the calculations were performed with a staff approved method as reflected in reference 2.

An editorial change was made regarding the charging pumps from "...can be energized..." to "...capable of injection...". This reflects more precisely the pumps operational state and is acceptable. The proposed Figure 3.1.A-1 for the PORV open pressure setpoint versus temperature includes the difference in elevation between the PORV and the RCS pressure sensors, a 5 °F and 10 psi margins and error correction for nonuniform metal and water temperatures. Language has been added to reflect the above changes. Instrument error and/or pressure bias are not reflected in Figure 3.1.A-1. The first reactor coolant pump is prohibited from starting when the RCS temperature is in the 40 °F range between 249 °F (from 275 °F) and 280 °F (from 305 °F) and the pressurizer level is greater than 30%. The purpose of this restriction is to assure the temperature rise resulting from the transient will not be outside the limits for the actuation of the OPS system.

TS Table 3.1.A-2, Reactor Coolant Pumps

This table reflects the new OPS enable temperature and the limitation for starting of the first RCS pump. In addition Table 3.1.A-2: (1) specifies the RCS pressure restrictions as described in Fig. 3.1.A-6, (2) specifies that three charging pumps may be capable of injecting into the RCS (with the OPS operable) as indicated in Fig. 3.1.A-1, and (3) specifies the combination of safety injection and charging pumps capable of injecting into the RCS with the OPS not operable and RCS temperatures below 280 °F, (illustrated in Figures 3.1.A-2, 3.1.A-3 and 3.1.A-4).

TS 3.1.B.1, Heatup and Cooldown

Changes to this TS include the proposed period of validity to 25 EFPYs, a new definition of the P-T limit curves (in the bases) and reference to the proposed curves in Figures 3.1.B-1 and 3.1.B-2. The P-T curves were calculated in reference 2 with staff approved methods, and therefore, are acceptable.

4.3 Pressurized Thermal Shock (PTS)

Because of the power uprate of 1.4% and the fluence reevaluation the licensee reevaluated the PTS to satisfy the 10 CFR 50.61 requirements. For 32 EFPYs of operation, the intermediate to lower shell peripheral weld (Heat # 34B009) is the critical element i.e. with the highest estimated PT_{PTS} = 246 °F at 32 EFPYs. It is worth noting that the intermediate shell plate (B2003-1) has a 32 EFPY PT_{PTS} = 243 °F, therefore, it is 27 °F from the screening criterion (270 °F) while weld 34B009 is 54 °F from the screening criterion. However, both elements are within the screening criteria, thus, are acceptable.

4.4 Overpressure Protection System

To assure compliance with the Appendix G requirements the OPS is comprised from two power operated relief valves (PORVs) and an actuation curve (pressure vs temperature) such as not to exceed the PT curve for the following events:

- mass addition due to starting of three charging pumps without letdown, and
- heat addition due to starting of a reactor coolant pump with steam generators 40 °F hotter than the reactor coolant.

In addition, the licensee analyzed the mass addition transient resulting from starting of two charging pumps and one safety injection pump. The single failure was assumed that a PORV did not function. PORV cycling may result if the opening pressure is reached. Nitrogen supply is adequate for 10-minute operation, should instrument air pressure not be available. Pressure undershoot in this type of cycling could damage the reactor coolant pumps. In this analysis assuming correct PORV operation pressure undershoot is not a problem. The licensee performed a reanalysis of the pressure and temperature uncertainties. The pressure bias between the PORV and the pressure sensor is built into the P-T curves. However, other uncertainties are not. The results of the analysis indicate that there exists significant margin in the P-T curves and the OPS limits. The methodology and the results satisfy the GL 96-03 requirements, thus, are acceptable.

4.5 Summary

In summary the staff finds that: (1) the fluence used in the evaluation was calculated using the guidance in RG 1.190 and is acceptable, (2) the TS changes reflect the Code Cases N-588 and/or N-640 and the staff approved methodology in WCAP-14040, therefore, are acceptable. Finally, the results of the calculations are correctly reflected in the actual TS changes.

Based on the staff's review and evaluation of ENO's proposed P-T limit curves for IP2, the staff has determined that the proposed P-T limit curves satisfy the requirements of 10 CFR 50.60(a), Appendix G to 10 CFR Part 50, and Appendix G to the 1995 Edition with 1996 Addenda of Section XI of the ASME Code, as exempted by the methods of the analyses in Code Cases N-588 and N-640. Therefore, the staff concludes that the updated P-T limit curves proposed by ENO, will continue to provide an acceptable level of margin and safety, and provide sufficient assurance that the IP2 reactors will be operated in a manner that will protect the RPV against brittle fracture. The proposed curves are, therefore, approved for incorporation into the IP2 TS.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York State official was notified of the proposed issuance of the amendment. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32 and 51.35, an environmental assessment and finding of no significant impact was published in the Federal Register on February 15, 2002 (67 FR 7206).

Accordingly, based upon the environmental assessment, the Commission has determined that issuance of this amendment will not have a significant effect on the quality of the human environment.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

8.0 REFERENCES

1. Letter from A.A. Blind Consolidated Edison Company of New York to US NRC “Indian Point 2 License Amendment Request for Reactor Coolant System Heatup and Cooldown Limitation Curves and Request for Exemption from the Requirements of 10CFR50.60(a) and Appendix G," dated July 16, 2001.

2. WCAP-15629, “Indian Point Unit 2 Heatup and Cooldown Limit Curves for Normal Operation and PTLR Support Documentation” by T.J. Laubham, Westinghouse Electric Company LLC, April 2001.

3. Letter from F. Dacimo, Entergy Nuclear Northeast to US NRC “Response to Request for Additional Information Indian Point 2 License Amendment Request for Reactor Coolant System Heatup and Cooldown Limitation Curves (TAC No.: MB2419)” November 30, 2001

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