JUL 15 1975

Docket Nos. 50-237 and 50-249

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Commonwealth Edison Company
ATTN: Mr. J. S. Abel
Nuclear Licensing Administrator -
Boiling Water Reactors
Post Office Box 767
Chicago, Illinois 60690

Gentlemen:

The Commission has requested the Federal Register to publish the enclosed Notice of Proposed Issuance of Amendments to Facility License Nos. DPR-19 and DPR-25 for the Dresden Nuclear Power Station Units 2 and 3. The proposed amendments include a change to the Technical Specifications and are in response to your request dated April 11, 1975, which was submitted in reply to our letter dated February 14, 1975.

These amendments incorporate: (1) water temperature limits during any testing which adds heat to the suppression pool, (2) suppression pool water temperature limits requiring manual scram of the reactor, (3) suppression pool water temperature limits requiring reactor pressure vessel depressurization, (4) surveillance requirements to monitor water temperatures during operations which add heat to the suppression pool and (5) external visual examinations of the suppression chambers following operations in which the pool temperatures exceed 160°F.

During our review, we discussed with your staff certain modifications to the proposed change for clarification and completeness. Your staff disagreed with certain of these modifications but indicated they would accept the modifications. These modifications have been made.

Copies of our proposed license amendments with changes to the Technical Specifications, Safety Evaluation and the Federal Register Notice relating to these actions also are enclosed.

Sincerely,
Original Signed by:
Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Reactor Licensing

c/P (1)

Enclosures:
1. Proposed Amendments w/Proposed Tech Spec change
2. Safety Evaluation
3. Federal Register Notice

cc w/enclosures:
See next page

OELD S.A. subj. per note of 7/14
7/15/75

OFFICE ➤	RL:ORB#2	RL:ORB#2	RL:ORB#3	RL:ORB#2	RL:AD/ORs
SURNAME ➤	RMDiggs	RDSilver:tc	CJDeBevec	DLZiemann	KRGoller
DATE ➤	6/25/75	6/25/75	6/26/75	6/30/75	7/15/75

cc w/enclosures:
John W. Rowe, Esquire
Isham, Lincoln & Beale
Counselors at Law
One First National Plaza
Chicago, Illinois 60670

Anthony Z. Roisman, Esquire
Berlin, Roisman and Kessler
1712 N Street, N. W.
Washington, D. C. 20036

Morris Public Library
604 Liberty Street
Morris, Illinois 60451

Chairman, Board of Supervisors
of Grundy County
Grundy County Courthouse
Morris, Illinois 60450

cc w/enclosures and cy of
CE's filing dtd. 4/11/75:
Mr. Leroy Stratton
Bureau of Radiological Health
Illinois Department of Public Health
Springfield, Illinois 62706

Mr. Gary Williams
Federal Activities Branch
Environmental Protection Agency
230 South Dearborn Street
Chicago, Illinois 60604

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-237

DRESDEN NUCLEAR POWER STATION UNIT 2

PROPOSED AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.
License No. DPR-19

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

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated April 11, 1975, 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; and

   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.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 3.B of Facility License No. DPR-19 is hereby amended to read as follows:

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"B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. ."

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Giambusso, Director
Division of Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Change No. to the
Technical Specifications

Date of Issuance:

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DATE ➤						

PROPOSED CHANGE TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-19

DOCKET NO. 50-237

Delete existing pages 108, 125 and 129 and insert the attached pages 108, 108A, 125, 125A, 129 and 129A. The changed areas on the revised pages are shown by marginal lines.

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3.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the operating status of the primary and secondary containment systems.

Objective:

To assure the integrity of the primary and secondary containment systems.

Specification:

A. Primary Containment

1. At any time that the nuclear system is pressurized above atmospheric or work is being done which has the potential to drain the vessel, except as permitted by Specification 3.5.F.3 or 3.5.F.4, the suppression pool water volume and temperature shall be maintained within the following limits.

   a. Maximum water volume - 115,655 ft^3

   b. Minimum water volume - 112,000 ft^3

   c. Maximum water temperature

      (1) During normal power operation - 95°F.

      (2) During testing which adds heat to the suppression pool, the water temperature shall not exceed 10°F above the normal power operation limit specified in (1)

4.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the primary and secondary containment integrity.

Objective:

To verify the integrity of the primary and secondary containment.

Specification:

A. Primary Containment

1.a. The suppression pool water level and temperature shall be checked once per day.

b. Whenever there is indication of relief valve operation or testing which adds heat to the suppression pool, the pool temperature shall be continually monitored and also observed and logged every 5 minutes until the heat addition is terminated.

c. Whenever there is indication of relief valve operation with the temperature of the suppression pool reaching 160°F or more and the primary coolant system pressure greater than 150 psig, an external visual examination of the suppression chamber shall be conducted before resuming power operation.

3.7 LIMITING CONDITION FOR OPERATION

above. In connection with such testing, the pool temperature must be reduced to below the normal power operation limit specified in (1) above within 24 hours.

(3) The reactor shall be scrammed from any operating condition if the pool temperature reaches 110°F. Power operation shall not be resumed until the pool temperature is reduced below the normal power operation limit specified in (1) above.

(4) During reactor isolation conditions, the reactor pressure vessel shall be depressurized to less than 150 psig at normal cooldown rates if the pool temperature reaches 120°F.

2. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing low power physics tests at atmospheric pressure at power levels not to exceed 5 Mw(t).

4.7 SURVEILLANCE REQUIREMENTS

d. A visual inspection of the suppression chamber interior, including water line regions, shall be made at each major refueling outage.

2. The primary containment integrity shall be demonstrated by either Method A or Method B, as follows:

a. Integrated Primary Containment Leak Test (IPCLT)

Bases:

3.7

A. Primary Containment — The integrity of the primary containment and operation of the emergency core cooling system in combination, limit the off-site doses to values less than those suggested in 10 CFR 100 in the event of a break in the primary system piping. Thus, containment integrity is specified whenever the potential for violation of the primary reactor system integrity exists. Concern about such a violation exists whenever the reactor is critical and above atmospheric pressure. An exception is made to this requirement during initial core loading and while the low power test program is being conducted during initial core loading and while the low power test program is being conducted and ready access to the reactor vessel is required. There will be no pressure on the system at this time which will greatly reduce the chances of a pipe break. The reactor may be taken critical during this period; however, restrictive operating procedures will be in effect again to minimize the probability of an accident occurring. Procedures and the Rod Worth Minimizer would limit control worth to less than 1.3%Δk. A drop of a 1.3%Δk rod does not result in any fuel damage. In addition, in the unlikely event that an excursion did occur, the reactor building and standby gas treatment system, which shall be operational during this time, offers a sufficient barrier to keep off-site doses well within 10 CFR 100.

The pressure suppression pool water provides the heat sink for the reactor primary system energy release following a postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat released during primary system blowdown from 1000 psig.

Since all of the gases in the drywell are purged into the pressure suppression chamber air space during a loss of coolant accident, the pressure resulting from isothermal compression plus the vapor pressure of the liquid must not exceed 62 psig, the suppression chamber design pressure. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber. Ref. Section 5.2.3 SAR.

Using the minimum or maximum water volumes given in the specification, containment pressure during the design basis accident is approximately 48 psig which is below the design of 62 psig. Maximum water volume of 115,655 ft^3 results in a downcomer submergence of 4 feet and the minimum volume of 112,000 ft^3 results in a submergence approximately 4 inches less. The majority of the Bodega tests (9) were run with a submerged length of 4 feet and with complete condensation. Thus, with respect to downcomer submergence, this specification is adequate.

Experimental data indicates that excessive steam condensing loads can be avoided if the peak temperature of the suppression pool is maintained below 160°F during any period of relief valve operation with sonic conditions at the discharge exit. Specifica-

(9) Bodega Bay Preliminary Hazards Summary Report, Appendix 1, Docket 50-205, December 28, 1962.

Bases: (cond't)

3.7

tions have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings.

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a relief valve inadvertently opens or sticks open. As a minimum this action shall include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling heat exchangers, (3) initiate reactor shutdown, and (4) if other relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open relief valve to assure mixing and uniformity of energy insertion to the pool.

The maximum temperature at the end of blowdown tested during the Humboldt Bay[10] and

(10) Robbins, C. H., "Tests of a Full Scale 1/48 Segment of the Humboldt Bay Pressure Suppression Containment," GEAP-3596, November 17, 1960.

Bases:

4.7

A. Primary Containment

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature normally changes very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be continually monitored and frequently logged during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be the points of highest stress.

The interiors of the drywell and suppression chamber are painted to prevent rusting. The inspection of the paint during each major refueling outage, approximately once per year, assures the paint is intact. Experience with this type of paint at fossil fueled generating stations indicates that the inspection interval is adequate.

The primary containment preoperational test pressures are based upon the calculated primary containment pressure response in the event of a loss of coolant accident. The peak drywell pressure would be about 48 psig which would rapidly reduce to 25 psig within 10 seconds following the pipe break. Following the pipe break, the suppression chamber pressure rises to 25 psig within 10 seconds, equalizes with drywell pressure and therefore rapidly decays with the drywell pressure decay (12).

The design pressure of the drywell and absorption chamber is 62 psig (12). The design leak rate is 0.5%/day at a pressure of 62 psig. As pointed out above, the pressure response of the drywell and suppression chamber following an accident would be the same after about 10 seconds. Based on the calculated containment pressure response discussed above, the primary containment preoperational test pressures were chosen. Also, based on the primary containment pressure response and the fact that the drywell and suppression chamber function as a unit, the primary containment will be tested as a unit rather than the individual components separately.

The design basis loss of coolant accident was evaluated at the primary containment maximum allowable accident leak rate of 2.0%/day at 48 psig. The analysis showed that with this leak rate and a standby gas treatment system filter efficiency of 90% for halogens, 95% for particulates, and assuming the fission product release fractions stated in TID 14844, the

(12) Section 5.2 of the SAR.

Bases: (cont'd)

4.7

maximum total whole body passing cloud dose is about 8 rem and the maximum total thyroid dose is about 185 rem at the site boundary over an exposure duration of two hours. The resultant doses that would occur for the duration of the accident at the low population distance of 5 miles are lower than those stated due to the variability of meteorological conditions that would be expected to occur over a 30-day period. Thus, the doses reported are the maximum that would be expected in the unlikely event of a design basis loss of coolant accident. These doses are also based on the assumption of no holdup in the secondary containment resulting in a direct release of fission products from the primary containment through the filters and stack to the environs. Therefore, the specified primary containment leak rate and filter efficiency are conservative and provide margin between expected off-site doses and 10 CFR 100 guidelines.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT TO LICENSE NOS. DPR-19 AND DPR-25 AND CHANGE TO THE TECHNICAL SPECIFICATIONS

SUPPRESSION POOL WATER TEMPERATURE LIMITS

COMMONWEALTH EDISON COMPANY

DRESDEN UNITS 2 AND 3

DOCKET NOS. 50-237 AND 50-249

INTRODUCTION

By letter dated April 11, 1975, Commmonwealth Edison Company (CE) requested a change in the Technical Specifications appended to Facility Operating License Nos. DPR-19 and DPR-25 for the Dresden Nuclear Power Station Units 2 and 3 located at Grundy County, Illinois. The proposed change in Technical Specifications was submitted in response to our request to the licensee dated February 14, 1975, and is responsive to the guidelines set forth in our letter. We have made additional modifications to these proposed Technical Specifications to improve the clarity and intent of the specification and its basis. These additional changes were discussed with CE staff members. The proposed change in Technical Specifications defines new temperature limits for the suppression pool water to provide additional assurance of maintaining primary containment function and integrity in the event of extended relief valve operation.

DISCUSSION

The Dresden Units 2 and 3 are boiling water reactors (BWR) which are housed in a Mark I primary containment. The Mark I primary containment is a pressure suppression type of primary containment that consists of a drywell and a suppression chamber (also referred to as the torus). The suppression chamber, or torus, contains a pool of water and is designed to suppress the pressure during a postulated loss-of-coolant accident (LOCA) by condensing the steam released from the reactor primary system. The reactor system energy released by relief valve operation during operating transients also is released into the pool of water in the torus.

Experiences at various BWR plants with Mark I containments have shown that damage to the torus structure can occur from two phenomena associated with relief valve operations. Damage can result from the forces exerted on the structure when, on first opening the relief valves, steam and the air within the vent are discharged into the torus water. This phenomenon is referred to as steam vent clearing. The second source of potential structural damage stems from the vibrations which accompany extended relief valve discharge into the torus water if the pool water is at elevated temperatures. This effect is known as the steam quenching vibration phenomenon.

1. <u>Steam Vent Clearing Phenomenon</u>

   With regard to the steam vent clearing phenomenon, we are actively reviewing this generic problem and in our letter dated February 14, 1975, we also requested each applicable licensee to provide information to demonstrate that the torus structure will maintain its integrity throughout the anticipated life of the facility. Because of apparent slow progression of the material fatigue associated with the steam vent clearing phenomenon, we have concluded that there is not immediate potential hazard resulting from this type of phenomenon; nevertheless, surveillance and review action on this matter by the NRC staff will continue during this year.

2. <u>Steam Quenching Vibration Phenomenon</u>

   The steam quenching vibration phenomenon became a concern as a result of occurrences at two European reactors. With torus pool water temperatures increased in excess of 170°F due to prolonged steam quenching from relief valve operation, hydrodynamic fluid vibrations occurred with subsequent moderate to high relief valve flow rates. These fluid vibrations produced large dynamic loads in the torus structure and extensive damage to torus internal structures. If allowed to continue, the dynamic loads could have resulted in structural damage to the torus itself, due to material fatigue. Thus, the reported occurrences of the steam quenching vibration phenomenon at the two European reactors indicate that actual or incipient failure of the torus can occur from such an event. Such failure would be expected to involve cracking of the torus wall and loss of containment integrity. Moreover, if a LOCA occurred simultaneously with or after such an event, the consequences could be excessive radiological doses to the public.

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In comparison with the steam vent clearing phenomenon, the potential risk associated with the steam quenching vibration phenomenon (1) reflects the fact that a generally smaller safety margin1/ exists between the present license requirements on suppression pool temperature limits and the point at which damage could begin and (2) is more immediate.

EVALUATION

The existing Technical Specifications for the Dresden Units 2 and 3 limit the torus pool temperature to 95°F. This temperature limit assures that the pool water has the capability to perform as a constantly available heat-sink with a reasonable operating temperature that can be maintained by use of heat exchangers whose secondary cooling water (the service cooling water) is expected to remain well below 95°F. While this 95°F limit provides normal operating flexibility, short-term temperatures permitted by operating procedures exceed the normal power operating temperature limit, but accommodates the heat release resulting from abnormal operation, such as relief valve malfunction, while still maintaining the required heat-sink (absorption) capacity of the pool water needed for the postulated LOCA conditions. However, in view of the potential risk associated with the steam quenching vibration phenomenon, it is necessary to modify the temperature limits in the Technical Specifications.

This action was, as discussed in our February 14, 1975 letter, first suggested by the General Electric Company (GE) who had earlier informed us of the steam quenching vibration occurrences at a meeting on November 1, 1974, and provided related information by letters to us dated November 7, and December 20, 1974. The letter of December 20, 1974 stated that GE had informed all of its customers with operating BWR facilities and Mark I containments of the phenomenon and included in those communications GE's recommended interim operating temperature limits and proposed operating procedures to minimize the probability of encountering the damaging regime of the steam quenching vibration phenomenon.

Our implementation of the GE recommended procedures and temperature limits via changes in the Technical Specifications are evaluated in the following paragraphs:

1/ The difference, in pool water temperature, between the license limit(s) and the temperature at which structural damage might occur is the safety margin available to protect against the effects of the phenomenon discussed.

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a. The new short-term temperature limit applicable to all reactor operating conditions requires that the reactor be scrammed if the torus pool water temperature exceeds 110°F. This new temperature limit and associated requirement to scram the reactor provides an additional safety margin below the 170°F temperatures related to potential damage to the torus.

b. For specific requirements associated with surveillance testing, i.e., testing of relief valves, the water temperature shall not exceed 10°F above the normal power operation limit. This new limit applicable to surveillance testing of relief valves and HPCI operation provides additional operating flexibility while still maintaining a maximum heat-sink capacity.

c. For reactor isolation conditions, the new temperature limit is 120°F, above which temperature the reactor vessel is to be depressurized. This new limit of 120°F assures pool capacity for absorption of heat released to the torus while avoiding undesirable reactor vessel cooldown transients. Upon reaching 120°F, the reactor is placed in the cold, shutdown condition at the fastest rate consistent with the Technical Specifications on reactor pressure vessel cooldown rates.

d. In addition to the new limits on temperature of the torus pool water, discussion in the Basis includes a summary of operator actions to be taken in the event of a relief valve malfunction. These operator actions are taken to avoid the development of temperatures approaching the 170°F threshold for potential damage by the steam quenching phenomenon.

CONCLUSION

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date: JUL 15 1975

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UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-237 AND 50-249

COMMONWEALTH EDISON COMPANY

NOTICE OF PROPOSED ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) is considering issuance of amendments to Facility Operating License Nos. DPR-19 and DPR-25 issued to Commonwealth Edison Company (the licensee), for operation of the Dresden Nuclear Power Station Units 2 and 3 (the facilities) located in Grundy County, Illinois.

These amendments would incorporate additional suppression pool water temperature limits: (1) during any testing which adds heat to the pool, (2) at which reactor scram is to be initiated and (3) requiring reactor pressure vessel depressurization. They also would add surveillance requirements for visual examination of the suppression chamber during each refueling and following operations in which the pool temperatures exceed 160 °F and add monitoring requirements of water temperatures during operations which add heat to the pool.

Prior to issuance of the proposed license amendments, the Commission will have made the findings required by the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations, which are set forth in the proposed license amendments.

By AUG 25 1975, the licensee may file a request for a hearing and any person whose interest may be affected by this proceeding may file a request for a hearing in the form of a petition for leave to intervene

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with respect to the issuance of these amendments to the subject facility operating licenses. Petitions for leave to intervene must be filed under oath or affirmation in accordance with the provisions of Section 2.714 of 10 CFR Part 2 of the Commission's regulations. A petition for leave to intervene must set forth the interest of the petitioner in the proceeding, how that interest may be affected by the results of the proceeding, and the petitioner's contentions with respect to the proposed licensing action. Such petitions must be filed in accordance with the provisions of this FEDERAL REGISTER notice and Section 2.714, and must be filed with the Secretary of the Commission, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Docketing and Service Section, by the above date. A copy of the petition and/or request for a hearing should be sent to the Executive Legal Director, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, and to Mr. John W. Rowe, Esquire, Isham, Lincoln and Beale, Counselors at Law, One First National Plaza, Chicago, Illinois 60670, the attorney for the licensee.

A petition for leave to intervene must be accompanied by a supporting affidavit which identifies the specific aspect or aspects of the proceeding as to which intervention is desired and specifies with particularity the facts on which the petitioner relies as to both his interest and his contentions with regard to each aspect on which intervention is requested. Petitions stating contentions relating only to matters outside the Commission's jurisdiction will be denied.

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All petitions will be acted upon by the Commission or licensing board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel. Timely petitions will be considered to determine whether a hearing should be noticed or another appropriate order issued regarding the disposition of the petitions.

In the event that a hearing is held and a person is permitted to intervene, he becomes a party to the proceeding and has a right to participate fully in the conduct of the hearing. For example, he may present evidence and examine and cross-examine witnesses.

For further details with respect to these actions, see the application for amendments dated April 11, 1975, which is available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Morris Public Library, 604 Liberty Street, Morris, Illinois 60451. These license amendments and the Safety Evaluation may be inspected at the above locations and a copy may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Reactor Licensing.

Dated at Bethesda, Maryland, this 15th day of July 1975

FOR THE NUCLEAR REGULATORY COMMISSION
Original Signed by:
Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Reactor Licensing

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