April 29, 2003

Mr. R. T. Ridenoure
Division Manager - Nuclear Operations
Omaha Public Power District
Fort Calhoun Station FC-2-4 Adm.
P.O. Box 550
Fort Calhoun, NE 68023-0550

SUBJECT: FORT CALHOUN STATION, UNIT NO. 1 - ISSUANCE OF AMENDMENT (TAC NO. MB6469)

Dear Mr. Ridenoure:

The Commission has issued the enclosed Amendment No. 217 to Facility Operating License No. DPR-40 for the Fort Calhoun Station, Unit No. 1. The amendment consists of changes to the Technical Specifications (TS) in response to your application dated October 8, 2002, and its supplements dated December 3, 2002, and March 4, 2003.

The amendment modifies TS 2.3.a, "Emergency Core Cooling System," to extend the allowed outage time for a single low pressure safety injection pump from the existing 24 hours to 7 days. In addition, the word "pump" has been replaced with the word "train."

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

Sincerely,

/RA/

Alan B. Wang, Project Manager, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-285

Enclosures: 1. Amendment No. 217 to DPR-40
2. Safety Evaluation

cc w/encls: See next page

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OMAHA PUBLIC POWER DISTRICT

DOCKET NO. 50-285

FORT CALHOUN STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 217
License No. DPR-40

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

   A. The application for amendment by the Omaha Public Power District (the licensee) dated October 8, 2002, and its supplements dated December 3, 2002, and March 4, 2003, 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 license 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, Facility Operating License No. DPR-40 is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B. of Facility Operating License No. DPR-40 is hereby amended to read as follows:

   B. Technical Specifications

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 217, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of its date of issuance and shall be implemented within 60 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Stephen Dembek, Chief, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: April 29, 2003

<u>ATTACHMENT TO LICENSE AMENDMENT NO. 217</u>

<u>FACILITY OPERATING LICENSE NO. DPR-40</u>

<u>DOCKET NO. 50-285</u>

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

REMOVE	INSERT
2-20	2-20
2-21	2-21
2-23	2-23
2-23b	2-23b

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 217 TO FACILITY OPERATING LICENSE NO. DPR-40

OMAHA PUBLIC POWER DISTRICT

FORT CALHOUN STATION, UNIT NO. 1

DOCKET NO. 50-285

1.0 INTRODUCTION

By application dated October 8, 2002, and its supplements dated December 3, 2002, and March 4, 2003, Omaha Public Power District (OPPD) requested changes to the Technical Specifications (Appendix A to Facility Operating License No. DPR-40) for the Fort Calhoun Station, Unit No. 1 (FCS). The requested changes would modify Technical Specification (TS) 2.3.a, "Emergency Core Cooling System," to extend the allowed outage time (AOT) for a single low pressure safety injection pump from the existing 24 hours to 7 days. In addition, the word "pump" has been replaced with the word "train."

The supplemental letters dated December 3, 2002, and March 4, 2003, provided additional information that clarified the application, did not expand the scope of the application as originally noticed or revise the proposed TS changes and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on November 12, 2002 (67 *FR* 68740).

2.0 REGULATORY EVALUATION

Since the mid-1980s, the NRC has been reviewing and granting improvements to TSs that are based, at least in part, on probabilistic risk assessment (PRA) insights. In its final policy statement on TS improvements of July 22, 1993 (58 *FR* 39132), the NRC stated that it...

> expects that licensees, in preparing their Technical Specification related submittals, will utilize any plant-specific PSA [probabilistic safety assessment][1] or risk survey and any available literature on risk insights and PSAs....Similarly, the NRC staff will also employ risk insights and PSAs in evaluating Technical Specifications related submittals. Further, as a part of the Commission's ongoing program of improving Technical Specifications, it will continue to consider methods to make better use of risk and reliability information for defining future generic Technical Specification requirements.

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[1]PSA and PRA are used interchangeably herein.

The NRC reiterated this point when it issued the revision to 10 CFR 50.36, “Technical Specifications,” in July 1995 (60 *FR* 36953). In August 1995 (60 *FR* 42622), the NRC adopted a final policy statement on the use of PRA methods in nuclear regulatory activities that encouraged greater use of PRA to improve safety decisionmaking and regulatory efficiency. The PRA policy statement included the following points:

1. The use of PRA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC’s deterministic approach and supports the NRC’s traditional defense-in-depth philosophy.

2. PRA and associated analyses (e.g., sensitivity studies, uncertainty analyses, and importance measures) should be used in regulatory matters, where practical within the bounds of the state-of-the-art, to reduce unnecessary conservatism associated with current regulatory requirements.

3. PRA evaluations in support of regulatory decisions should be as realistic as practicable and appropriate supporting data should be publicly available for review.

In May 1995, the Combustion Engineering Owners Group (CEOG) submitted several Joint Application Reports for the staff’s review. One of the CEOG Joint Application Reports provided justifications for extensions of the TS completion time for the low pressure safety injection (LPSI) system.[2] The justifications for this extension are based on a balance of probabilistic considerations, traditional engineering considerations, including defense-in-depth, and operating experience. Risk assessments for all of the Combustion Engineering (CE) plants are contained in the reports. Arkansas Nuclear One, Unit 2 (ANO-2) had been the lead CE plant for the LPSI system TS changes. The staff performed an in-depth review of the ANO-2 PRA methodology relating to these changes, as the lead plant for all of the CEOG. Since then, the staff has reviewed and approved similar license amendment requests for Palisades, St. Lucie, Palo Verde and Waterford that referenced the CEOG report. OPPD has stated that this change is also based on and consistent with the CEOG report.

3.0 EVALUATION

The staff evaluated the licensee’s proposed amendment to extend the TS completion time (completion time and AOT are used interchangeably herein) for one LPSI train out-of-service from 24 hours to 7 days using insights derived from traditional engineering considerations and the use of PRA methods to determine the safety impact of extending the completion times. In addition, TS 2.3.a has been modified to state LPSI "train" rather than LPSI "pump." This wording is consistent with the intent of the CEOG report and is more conservative as it encompasses the loss of the LPSI system for any reason, not just the pump.

---

[2] CE NPSD-995, “Joint Application Report for Low Pressure Safety Injection System AOT Extension,” May 1995.

3.1 Traditional Engineering Evaluation

The current FCS TSs address the LPSI system as a portion of the emergency core cooling system (ECCS). The two trains of the LPSI system, in combination with the two trains of the high-pressure safety injection (HPSI) system, form two redundant ECCS trains. TS 2.3.a requires two LPSI pumps to be operable. If one or more LPSI pumps are inoperable, the inoperable pump must be returned to operable status within 24 hours or a plant shutdown is required. As noted above, the TS will be revised to refer to LPSI "train" rather than "pump."

The proposed change will allow up to seven days for the licensee to restore operability to an inoperable LPSI train that is the cause of ECCS train inoperability. In some instances, corrective maintenance of the LPSI pump and valves and testing of valves may require taking one train of LPSI out-of-service for more than several days. Thus, repair within the existing completion time cannot be ensured and may result in an unscheduled shutdown or a request for temporary relief to allow continued plant operation. On the basis of the review of maintenance requirements of the LPSI train for CE pressurized water reactors (PWRs), the licensee determined that a seven day completion time would provide sufficient margin to effect most anticipated preventive and corrective maintenance activities and LPSI train valve surveillance tests at power.

The primary role of LPSI trains during power operation is to contribute to the mitigation of a large loss-of-coolant accident (LOCA). The frequency of a large LOCA event is on the order of 10^{-4} per year. In contrast, during Modes 5 and 6, the operability of at least one LPSI train is required at all times for reactor coolant system (RCS) heat removal. Thus, in the broad view, performing preventative and corrective maintenance at power on LPSI trains can contribute to an overall enhancement of plant safety by increasing the availability of the LPSI train for shutdown cooling during Modes 5 and 6, when it is most needed.

Another role of the LPSI system is defining the end state for a design-basis steam generator tube rupture (SGTR) event. In this design-basis event, the HPSI functions to keep the core covered at all times, and the LPSI system is required to effect shutdown cooling (SDC) and thereby terminate the event. SDC is initiated after the break has been isolated and the radioactive releases have been controlled.

In the event that one LPSI train is out-of-service and the second LPSI train fails, the operator can continue to control the SGTR event by steaming of the unaffected steam generator. Even though loss of both LPSI trains is beyond the design basis accident assumptions, this cooling mechanism can be maintained indefinitely, provided condensate is available to the unaffected steam generator. Without considering condensate storage tank refill, the FCS has sufficient inventory to steam the affected steam generator for more than 24 hours. Therefore, having one LPSI train out-of-service should not affect the licensee’s ability to mitigate an SGTR event.

3.2 PRA Evaluation

3.2.1 Tier 1

FCS has TSs that currently allow one LPSI train to be out-of-service for 24 hours. A technical basis for extending this AOT to 7 days for all plants in the CEOG fleet (including FCS) was presented in CE NPSD-995. The calculations presented at that time were based on Revision 1 of the OPPD FCS PSA. OPPD is currently using Revision 5 to the PSA for the FCS. Revision 5 reflects the current plant configuration and includes many updates resulting from the OPPD PSA peer review. A revised set of PSA parameters was generated using the evaluation methods described in CE NPSD-995 to demonstrate the applicability of the CE NPSD-995 results to FCS. Summaries of the results of the updated analyses are:

Table 1
CDF/LERF Impacts of LPSI Train AOT Extension at FCS

PARAMETER	LPSI Train Out of Service for Corrective Maintenance	LPSI Train Out of Service for Preventive Maintenance
ICCDP (7-day AOT)	1.1E-08	2.3E-09
Yearly AOT CDF (Proposed full AOT),/r-yr	7.3E-09	6.9E-09
Delta CDF,/r-yr	5.8E-7	1.2E-7
ICLERP (7-day AOT)	4.6E-10	< 1.0E-10
Yearly AOT LERF (Proposed full AOT),/r-yr	3.0E-10	< 1.0E-10
Delta LERF,/r-yr	2.4E-08	< 1.0E-10

As can be seen from Table 1, the results of the analyses are consistent with the evaluations performed in CE NPSD- 995. These analyses confirm that the change in core damage frequency (CDF) is small and that the single-train AOT incremental conditional core damage probability (ICCDP) is well below the Regulatory Guide (RG) 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," guideline of 5.0E-07. Since the LPSI train is used to respond to low-pressure scenarios, such as those discussed in CE NPSD-995, the increased AOT for a single LPSI train should have an extremely small effect on the large early release frequency (LERF). This is confirmed by the PSA evaluation, which determined that the incremental conditional large early release probability (ICLERP) is less than the RG 1.177 guideline of 5.0E-08.

3.2.2 PSA Quality

The above analysis results are based on Revision 5 of the FCS PSA. In a response dated September 18, 2002, to a staff's request for additional information (RAI), the licensee stated that the original peer review of the FCS individual plant examination (IPE) was conducted in

1992 prior to the issuance of their IPE. The results of that peer review, which included the Level 1, 2, and 3 analyses, are included in the above reference.

In March 1999, the FCS PSA was peer reviewed by a team of PSA engineers from Westinghouse, four other utilities, and a PSA consultant. This peer review was the first conducted in accordance with the CEOG implementation of the nuclear industry peer review process, as documented in Nuclear Energy Institute (NEI) 00-02. The peer review team found the FCS PSA to be effective for assessing planned plant maintenance and operations configurations and evaluating future plant design changes. The PSA was also found to be adequate for other applications when supported by deterministic insights and plant expert panel input. The review did identify some areas of weakness in the PSA that should be considered in any application. The review also identified several areas of strength in the FCS PSA.

The review team found the FCS PSA to be strong in the areas of initiating event identification and containment performance analysis. The licensee had a particularly good treatment of the containment reliability analysis. The reviewers did recommend that the plant dependency analysis be upgraded. As a result of an in-depth investigation of dependencies, one mixed dependency of the auxiliary feedwater pump, FW-54, was identified and corrected. Improvement in the documentation of the dependency matrix was also recommended. This activity was tracked by a configuration control program and was integrated into the Revision 3 PSA model used for the severe accident mitigation alternative (SAMA) assessment.

PSA quality is maintained in accordance with plant procedure PED-SEI-37, "Probabilistic Risk Assessment Configuration Control." This procedure describes the PSA inputs, such as plant modifications and equipment failure history, which are reviewed and compared against the PSA model. The PSA model is typically revised once per operating cycle, and more frequently if warranted by major changes.

3.2.3 External Events

The individual plant examination for external events (IPEEE) was peer reviewed in December of 1993. The results of this peer review were also documented and are available for review at FCS. The impact of the proposed TS extended AOT on the external events evaluation results are described below.

3.2.3.1 Seismic

The FCS design basis requires meeting single failure criteria for seismic events up to the design basis earthquake (0.17g). Existing controls for maintaining the design basis of the plant are in place and are implemented to ensure the design basis is not compromised when removing equipment from service. Such practices help keep the seismic risk-significance of plant configurations low.

Higher seismic hazard levels (>0.1g) are sufficiently low in expected frequency that the probability of such an earthquake during a maintenance activity period of one week is of the order of 1E-6 or less. Seismic hazard levels greater than 0.1g have a frequency of approximately 7E-5/year or lower based on EPRI seismic hazard curve SLFC-93-1421. Hence, seismic hazard levels greater than 0.1g are not expected to have a significant impact on decisions regarding acceptable plant configurations. Nevertheless, knowledge of seismic risk drivers at lower g-levels (<0.1g) is useful in considering the seismic risk-significance of certain anticipated plant configurations. This is due largely to the possible impact on non-design basis equipment that may not be protected by existing seismic controls. For this reason, the impact of <0.1g seismic accelerations is explicitly and routinely quantified by the PSA model as part of the 10 CFR 50.65(a)(4) risk assessment process.

In response to a staff RAI, OPPD stated that the plant modifications/procedural improvements that were credited in the IPEEE have been completed with the exception of the anchoring of flammable material storage cabinets so as to prevent them from falling over during a seismic event. These storage cabinets have not yet been anchored and the licensee is considering another option that would result in the removal of these cabinets from the auxiliary building. OPPD has committed to either anchor or remove these flammable material storage cabinets by December 31, 2004.

For the LPSI system, the magnitude and frequency of seismic events at FCS is not risk significant because the LPSI is not credited in any sequences involving seismic events. In the FCS PSA model, the LPSI is credited for three functions associated with mitigating a loss of RCS inventory:

(1) Supplying low-pressure safety injection water to the RCS during a large LOCA,

(2) Providing a backup to HPSI in supplying safety injection to the hot leg for hot/cold leg injection during a large LOCA, and

(3) Providing shutdown cooling.

Consistent with typical industry practices, large LOCAs are not considered for seismic events due to the extremely low probability of such seismically-induced failures (i.e., due to the piping's robust seismic capacity). As a result, the only potential for crediting the LPSI system for seismic events is for its shutdown cooling function. However, shutdown cooling via the LPSI system is considered unavailable during a seismic event due to the failure of non-safety-related components (e.g., instrument air). Therefore, the LPSI is not credited for any sequences involving seismic events and thus, its potential increased unavailability due to the subject AOT extension will not have an impact on the risks associated with seismic events.

3.2.3.2 Fire

The probability of plant fires is not assessed for distinct plant activities such as LPSI pump maintenance. However, the effect of fire risk upon a TS extended AOT can be evaluated qualitatively. Within the context of the power-operation PSA, the LPSI serves the three functions identified above: (1) injection following a large LOCA, (2) alternate hot leg injection following a large LOCA, and (3) establishment of shutdown cooling. The third function, establishment of shutdown cooling, is the only function applicable to fire PSA, since fires concurrent with large LOCAs are not considered since these scenarios would have an extremely low probability of occurrence.

A review of the fire IPEEE model and results led the staff to the following conclusions. First, many of the fire areas involve consequential failure of shutdown cooling, in which case LPSI pump outages are moot. Second, most of the remaining fire areas have no direct impact upon shutdown cooling. For these fire areas, individual LPSI pump outages are of low risk significance. Finally, there are a few fire areas that result in failure of one LPSI pump, in particular, fires in the LPSI pump rooms. These fire areas are not risk-significant, since they have no effect upon main or auxiliary feedwater and since they degrade, but do not fail, once-through cooling.

The licensee further enhances fire safety by the implementation of the following procedures: Standing Order G-103, "Fire Protection Operability Criteria and Surveillance Requirements," and Standing Order G-58, "Control of Fire Protection System Impairments."

3.2.3.3 External Floods

As in the case of fire, the LPSI function of interest is shutdown cooling and the probability of external floods is not estimated for distinct plant activities, such as LPSI pump maintenance. However, the licensee evaluates the relationship between external floods and an extended LPSI pump AOT qualitatively.

The licensee places the consequences of external floods into two categories: (1) moderate floods that cause a loss of offsite power (LOOP) and fail some structures, systems, and components (SSCs) in the turbine building, and (2) large floods that cause a LOOP and fail numerous risk-significant SSCs. Since the large floods also cause a loss of the LPSI function, the proposed extended LPSI AOT would have no impact from large floods, thus the licensee only evaluated the impacts of moderate floods.

Moderate floods are caused by a rising level of the Missouri River, a process that’s relatively slow and predictable. Consequently, there is generally sufficient time available to place the plant in a stable condition and restore unavailable equipment. There would also be opportunities for establishing alternate means of decay heat removal, such as long-term refilling of the emergency feedwater storage tank or use of the containment spray pumps for shutdown cooling (although, as directed by procedure, this option requires RCS temperature < 120°F, and the pressurizer manway open). According to the licensee, it is expected that FCS will eventually adopt hot shutdown as a high-river level TS end state (refer to CE-NPSD-1186-A, Rev. 00, "Technical Justification for the Risk Informed Modification to Selected Required Action End States for CEOG Member PWRs," Westinghouse Electric Company, October 2001), further reducing the importance of shutdown cooling. The staff agrees with the licensee that, for the above reasons, the proposed extension of the LPSI pump AOT from 24 hours to 7 days will not have a significant impact on external flood risk.

3.3 Tier 2

With respect to the prevention of high-risk equipment outage configurations, the licensee believes that the Maintenance Rule, 10 CFR 50.65(a)(4), process provides sufficient configuration control for LPSI pump outages. The 10 CFR 50.65(a)(4) process includes solving the PRA model with the equipment out-of-service (EOOS) software, so that risk-significant configurations are identified. EOOS also identifies risk-significant initiating events and maintenance activities are restricted accordingly. Since the 10 CFR 50.65(a)(4) process explicitly models configuration changes that could increase the risk-significance of a LPSI pump outage, no additional Tier 2 restrictions are needed.

The licensee's process for on-line maintenance activities is described in Standing Order SO-M-101, "Maintenance Work Control." It applies to both planned and corrective maintenance activities. SO-M-101 describes the responsibilities for conducting the licensee's (a)(4) process, addresses both quantitative and qualitative evaluations of maintenance activities, and provides criteria for determining when consideration of risk management actions is warranted. Examples of typical risk management actions are provided. SO-M-101 also provides a trigger for including the Plant Review Committee in the review of maintenance activities. Risk assessments for power operation are both quantitative and qualitative, the quantitative portion being supported by EOOS.

Additional guidance for the risk assessment process is provided in FCSG-19, "Performing Risk Assessments." The guidance supports SO-M-101 in that the Standing Order specifies the requirements and FCSG-19 provides detailed guidance for complying with the requirements. The bulk of the detailed guidance deals with the operation of EOOS.

3.5 Changes to Bases Section

The licensee proposed changes to the Bases Section 2.3, "Emergency Core Cooling System," to reflect these TS changes and a previous change describing the justification for the quantity of water required in the safety injection and refueling water tanks. TS 5.20, "Technical Specification (TS) Bases Control Program," assures the continuing accuracy and adequacy of the Bases Sections. Therefore, the Bases changes have had the appropriate administrative controls and reviews performed to assure the accuracy and adequacy of the change. The staff has reviewed these Bases changes and has no objections to them.

3.6 Summary

The staff has evaluated the licensee's proposed changes for compliance with regulatory requirements as documented in this evaluation and has determined that they are acceptable. This determination is based on the following:

1. The traditional engineering evaluation reveals that increasing the availability of the LPSI system for SDC during outages by performing preventive and corrective maintenance at power can contribute to an overall enhancement of plant safety.

2. The staff finds acceptable the PRA model used by the licensee and also concludes that there is minimal impact of the completion time extension for the LPSI system on plant operational risk (Tier 1 evaluation).

3. The review of potentially high risk configurations did not identify the need for any additional constraints or compensatory actions that, if implemented, would avoid or reduce the probability of a risk-significant configuration (Tier 2 evaluation).

4. The licensee has stated that the maintenance rule (10 CFR 50.65) will be the vehicle that controls the actual equipment maintenance cycle by defining unavailability performance criteria for the LPSI system. The AOT extension will allow efficient scheduling of maintenance within the boundaries established by implementing the maintenance rule. The maintenance rule will thereby be the vehicle that monitors the effectiveness of the AOT extension. Application of these implementation and monitoring strategies will help to ensure that extension of the TS AOT for the LPSI system does not degrade operational safety over time and that the risk incurred when a LPSI system is taken out-of-service is minimized.

The staff, therefore, finds that the completion time for one LPSI train may be extended to seven days, with a negligible impact on risk.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (67 *FR* 68740). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 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.

Principal Contributors: M. Wohl
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Date: April 29, 2003