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Ref: License No. NPF-89

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April 9, 2001

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES) UNIT 2
DOCKET NOS. 50-446
OPERATION ABOVE THE LICENSED MAXIMUM THERMAL POWER LEVEL

Gentlemen:

The attached report discusses violation of Section 2.C.(1) of the Comanche Peak Steam Electric Station Operating License for Unit 2, "Maximum Power Level." This report is submitted pursuant to the requirements of Section 2.E of the license.

Should you require additional information regarding this event, please do not hesitate to contact Obaid Bhatty at (817) 897-5839 to coordinate this effort.

D029

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This communication contains no new licensing basis commitments regarding CPSES Unit 2.

Sincerely,

C. L. Terry

By: MR Blevins
M. R. Blevins
Deputy to Senior VP & Principal Nuclear Officer

OAB/ob

Attachment

c - E. W. Merschoff, Region IV
J. I. Tapia, Region IV
D. H. Jaffe, NRR
Resident Inspectors, CPSES

MARCH 12, 2001 OVERPOWER EVENT

PURPOSE

This report discusses violation of Section 2.C.(1) of the Comanche Peak Steam Electric Station (CPSES) Operating License for Unit 2, "Maximum Power Level." The report is submitted pursuant to the requirements of Section 2.E of CPSES Unit 2 License No. NPF-89.

ABSTRACT

On March 12, 2001, at approximately 10:29 a.m., CPSES Unit 2 license power restriction of 3445 MWth was exceeded. The event was initiated by a failure of a main turbine electro hydraulic control (EHC) fluid pressure switch, which caused a closure of the extraction steam valves to the feedwater heaters. The function of the pressure switch is to actuate upon detection of an actual turbine trip and cause isolation of extraction steam to prevent steam backflow from the feedwater heaters to the turbine. However, without a turbine trip, the loss of extraction steam causes a cooling of the incoming feedwater which results in a power transient. Nuclear Instrumentation indicated approximately 97 percent power and N-16 power indicated over 100 percent for a period of approximately 1.5 minutes, peaking at approximately 105 percent.

DESCRIPTION OF THE EVENT

On March 12, 2001, at 10:23 a.m., Comanche Peak Steam Electric Station Unit 2 was operating at 100 percent power.

On March 12, 2001, at approximately 10:24 a.m., CPSES Unit 2 experienced a loss of extraction steam to feedwater heaters 1 through 4. As a result, heater drain pump flow was lost and the low pressure feedwater heater bypass valve 2-PV-2286 opened as designed to maintain adequate main feed pump suction pressure. The loss of extraction steam and bypassing of all low pressure feedwater heaters reduced feedwater system (EIIS:(SJ)) temperature which in turn caused an increase in reactor power. Plant Operators (utility, licensed) entered the appropriate procedures to manually reduce the turbine load to prevent exceeding the licensed power limit. However, power briefly rose to 105 percent as indicated by N-16 power monitors exceeding 100 percent for a period of approximately 1.5 minutes. Power range nuclear instrumentation did not show an increase above 100 percent. The plant was stabilized at 85 percent reactor power to determine cause and corrective actions.

CAUSE OF THE EVENT

The cause of the extraction steam isolation was determined to be a failed EHC pressure switch, which was age related. There were no equipment failures other than one that initiated the event, and all required systems remained capable of fulfilling their specified safety functions.

CORRECTIVE ACTIONS

The EHC pressure switch was replaced , and at approximately 4:17 a.m., on March 13, 2001, the unit was returned to 100 percent power.

The pressure switch that failed was on the trip fluid header. There are two pressure switches for each unit. Both switches have been replaced for CPSES Unit 2 and both Unit 1 switches will be replaced prior to startup from Unit 1 eighth refueling outage (1RF08).

ANALYSIS OF THE EVENT

The immediate result of the inadvertent isolation of the extraction steam is a loss of main feedwater heating, resulting in a significant decrease in the main feedwater temperature and the resultant excessive heat removal from the Reactor Coolant System by the secondary system. A conservative analysis of such an event is presented in FSAR Section 15.1.1, "Feedwater System Malfunctions that Results in a Decrease in Feedwater Temperature." In that analysis, an instantaneous decrease in the feedwater temperature of 245 degrees F was assumed. Neither operator action nor the operation of the automatic turbine runback system were credited. In that analysis, the reactor power was allowed to increase until the overpower N-16 reactor trip setpoint was exceeded, at which time control rod motion terminated the reactivity excursion. (The peak thermal power was calculated to be in excess of 125 percent rated thermal power (RTP).) The relevant event acceptance criterion (related to compliance with the Departure from Nucleate Boiling Ratio (DNBR) limit) was shown to be satisfied.

In the event of March 12, 2001, the initial decrease in the feedwater temperature was approximately 225 degrees F over a six minute period, and the peak reactor power was limited to less than 105 percent RTP by operator action and the automatic rod control system. Both parameters are within the analysis assumptions and results presented in FSAR Section 15.1.1.

Operations initiated a turbine load reduction as soon as a power increase was noted. Although the rate of power increase was too rapid to preclude the overpower transient, operator actions did mitigate the event and an automatic turbine runback was avoided. The automatic turbine runback is set 3 percent conservatively lower than the N-16 reactor trip described above.

During the post-event recovery, beginning approximately six to seven minutes after the initiating event, the reactor operators decreased the reactor power to approximately 85 percent RTP. The feedwater temperature decreased to 170 degrees F for an hour and was then raised to near 225 degrees F as the plant was stabilized. The plant was maintained in this condition for a number of hours while the cause of the initiating event was identified and corrected and restoration of the main feedwater heaters was completed.

The reactor trip system setpoints are selected to preclude operating with combinations of conditions that may lead to fuel clad failures (assumed to occur if the DNBR limit is exceeded). Because no reactor trip system setpoints were exceeded during the transient, it is concluded that the DNBR limit was not exceeded at any point during the event and subsequent recovery.

Operation with feedwater temperatures significantly below nominal is not considered as an initial condition for any of the accidents and transients presented in FSAR Chapter 15; however, an assessment of the acceptability of operating in this condition for a limited time period during the restoration of the plant systems was performed. No material limitations were identified. Even though a few of the accident and transient evaluations presented in FSAR Chapters 6 and 15 (e.g., steam line break, feedwater line break, and steam generator tube rupture) would be more severe than currently analyzed, the probabilities of occurrence of these events are sufficiently small, and the expected duration of operation at the off-normal conditions is sufficiently short, that the overall impact on plant risk is concluded to be minimal.

Based on the forgoing evaluations, it is concluded that the health and safety of the public was unaffected by this transient.

ADDITIONAL INFORMATION

On March 17, 2001, a similar event occurred at CPSES Unit 1. The cause of this event was also a failed EHC pressure switch. However, a violation of license restriction of 3411 MWth did not occur during this event due to the lower reactor power (93 percent) at the initiation of the event. The corrective actions for this event are similar to the Unit 2 event.