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ARKANSAS POWER & LIGHT COMPANY
CAPITOL TOWER BUILDING/P. O. BOX 551/LITTLE ROCK, ARKANSAS 72203/(501) 377-3525

T. GENE CAMPBELL
Vice President - Nuclear

April 28, 1988

CERTIFIED MAIL
RETURN RECEIPT REQUESTED

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

ATTN: Director, Office of Enforcement

SUBJECT: Arkansas Nuclear One - Unit 1
Docket No. 50-313
License No. DPR-51
Reply to Notice of Violation and Proposed Imposition
of Civil Penalty Relating to Elevated Reactor Building
Temperature at Arkansas Nuclear One, Unit 1

Gentlemen:

The Arkansas Power and Light Company (AP&L) has reviewed your correspondence of March 14, 1988 (ØCNAØ3881Ø) which transmitted the Notice of Violation and Proposed Imposition of Civil Penalty resulting from the elevated temperature inside the Arkansas Nuclear One, Unit 1 Reactor Building.

Following that review, AP&L is submitting the attached response to the Violation Notice in accordance with 10CFR2.201, along with check #04-8300 in the amount of One Hundred Thousand Dollars (\$100,000). This remittance is payment in full for the subject imposition of civil penalty.

Your aforementioned correspondence pointed to concerns relative to the level of safety consciousness within the AP&L organization. While more specifics are contained within the attached response, let me assure you that the safe

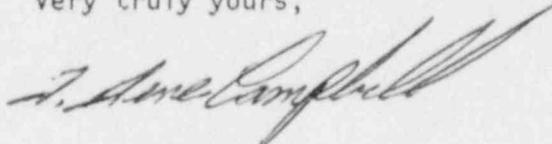
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April 28, 1988

operation of our nuclear facilities remains AP&L's highest priority. Every effort is being made to ensure that this policy is reflected throughout the AP&L nuclear program.

Very truly yours,



T. Gene Campbell

TGC/MT/lg

Enclosure

cc: Mr. Robert D. Martin, Administrator
U. S. Nuclear Regulatory Commission
Region IV
611 Ryan Plaza Drive, Suite 1000
Arlington, TX 76011

REPLY
to
NOTICE of VIOLATION
and
PROPOSED CIVIL PENALTY

On March 14, 1988, the Nuclear Regulatory Commission (NRC) issued to the Arkansas Power and Light Company (AP&L) a Notice of Violation (NOV) and Proposed Imposition of Civil Penalty resulting from actions relating to the elevated temperature inside the Arkansas Nuclear One, Unit 1 (ANO-1) reactor building.

The ANO-1 temperature concern came under the most recent discussion following an exchange of information relating to elevated ANO-1 reactor building temperatures between the resident NRC inspector and the AP&L staff. During initial discussions with the NRC Staff, AP&L relayed the results of our preliminary investigation which indicated the elevated temperatures had been identified and evaluated at the time of initial startup in 1974. The results of those initial 1974 evaluations did not indicate that the elevated temperatures posed an adverse impact upon safe operations. Additional reviews conducted in support of discussions with the NRC in August of 1987 supported these conclusions.

A subsequent, special inspection by the NRC was conducted during the period of August 18-20, 1987. As a result of communications between NRC and AP&L management, AP&L then developed an extensive Justification for Continued

Operation (JCO) evaluating the technical and safety concerns relating to the elevated temperatures and presented this to the NRC on August 28, 1987. Following the completion of review of the JCO by the NRC, a Safety Evaluation Report (SER) was issued by the staff on October 15, 1987. The SER concluded that there were no immediate safety concerns that would preclude the plant from continuing to operate. An Enforcement Conference, held in Arlington, Texas on December 18, 1987, discussed the findings of the August, 1987, inspection and the actions undertaken by AP&L to address the elevated temperature concerns. After evaluating the information presented, the NRC issued the March 14, 1988, NOV. The NOV finds that AP&L violated Criterion XVI, Appendix B, 10CFR Part 50 in that AP&L did not promptly correct the high temperature condition, a condition the staff viewed as adverse to quality. The following information is in reply to that notice:

1. Admission or Denial of the Alleged Violation

AP&L agrees that a violation of NRC regulations occurred. The principle deficiency impacting AP&L's overall quality effort resulted from a weakness in our design process. This weakness did not allow our program to ensure that an identified deviation, such as the ANO-1 reactor building temperature, would be fully integrated into all aspects of designs and evaluations or be adequately reflected in AP&L's design basis documentation. As a result, the elevated reactor building temperatures were not reflected in the Safety Analysis Report (SAR) and other design documents nor documented in formal evaluations required to support such changes.

However, AP&L does not believe that our initial evaluation of the impact upon plant operability and safety resulting from the elevated temperature allowed a specific potential condition adverse to quality or public health and safety to materialize. Considerable efforts were undertaken in 1974 when the concern was first identified during initial start-up, to determine the root cause of the higher temperature and pursue certain modifications to attempt to reduce those temperatures. These efforts failed to produce significant reductions of the containment temperatures. Consequently, AP&L specifically requested the architect/engineer to evaluate potential effects of the higher temperatures on equipment and systems. Although, we have no record of a direct response to this request, the available documentation from that time period indicates that the potential effects of the elevated temperatures were considered and determined not to adversely effect the safe operation of ANO-1. Topical reports and other information were also available in 1974 which indicated that the higher initial temperatures inside the ANO-1 reactor building would not invalidate the pertinent safety analyses. Thus, the situation was reviewed and evaluated in a manner consistent with good engineering judgement and determined to be acceptable and not a potential condition adverse to quality or public health and safety. Ultimately, this 1974 engineering judgement was reaffirmed by the substantial reanalyses conducted as part of the JCO effort during August of 1987.

The work performed during the JCO also supported the fact that the higher reactor building temperature did not exceed the design basis of the plant and affected systems. AP&L does, however, agree that the

failure to adequately reflect this deviation in the Unit's Safety Analysis Report (SAR) and other design basis documentation was the result of a central weakness in our design control process. This weakness was manifested in our difficulty in properly coordinating participating design organizations to ensure that design interfaces correctly translated and accounted for the identified deviation during subsequent plant modifications.

Two additional concerns noted in the NOV, while not part of the cited violation, discussed the issues of reportability based upon conditions outside of design basis and the resulting operation of acoustic monitors beyond their qualified life. The remaining comments in this section are in response to these issues.

Although the reactor building temperature issue was the subject of several conversations with NRC staff and management, as noted in the letter transmitting the NOV, AP&L did not formally report this issue per 10CFR50.72. Your letter, however, does not correctly reflect the basis for AP&L's reportability determination. Based upon the regulatory guidance available and the definitions provided in 10CFR50.2, along with engineering judgement regarding the potential effect of elevated reactor building temperatures on safety analysis results as well as equipment performance and qualification, it was AP&L's conclusion that this was not a condition outside the design basis.

AP&L is strongly committed to compliance with 10CFR50.72 regarding formal reports and also attempts to keep NRC informed on a less formal basis of other items which we believe may be of interest. The decision not to make a formal report in this instance was based upon our interpretation of the governing regulation. In addition, as noted above, we were aware that NRC was actively involved in discussions with AP&L relative to this issue. In order to assure continued compliance with 10CFR50.72, we plan to initiate additional discussions with the Region IV staff with the objective of clarifying our understanding of the reporting criteria at issue in this instance.

Concerning the specific issue of acoustic monitor qualification, current testing documentation supports a qualified life for these components of more than three years at an ambient temperature of 150°F. Since the ambient temperature in which the acoustic monitors were required to operate was approximately 142°F and none of these qualified components have exceeded a three (3) year installed life, we do not concur that ANO-1 has operated with unqualified acoustic monitor pre-amplifiers in place. This information was presented at the December, 1987, Enforcement Conference and it was our understanding that the staff was in agreement with AP&L's position.

2. Reasons for the Violation, if Admitted

AP&L concurs that a weakness in the previously existing administrative programs governing design interfaces was the root cause which resulted

in our failure to integrate the reactor building temperature anomaly into all phases of the design control program. The AP&L quality organization, as well as management, had identified problems with the design change process prior to the August, 1987 inspection. AP&L management was, at that time, taking steps aimed at improving the design control and plant modification processes by a systematic elimination of those identified weaknesses.

3/4. Corrective Steps That Have Been Taken And The Results Achieved;
Corrective Steps That Will Be Taken To Avoid Further Violations

Immediately following the August 18-20, 1987 inspection by the NRC, AP&L mounted an extensive effort which culminated on August 28, 1987 with the presentation to the staff of a Justification for Continued Operation (JCO). This JCO was a comprehensive program of evaluation and analysis which addressed such areas as a re-evaluation of safety analyses to consider elevated temperature, re-analysis of reactor building structural integrity, re-analysis of affected environmentally qualified components, evaluation of non-EQ equipment and an evaluation of the elevated temperature effects on equipment and systems outside of the reactor building. Additionally, the JCO also evaluated the potential impact on Unit 2 (ANO-2) and identified a plan for further actions to be taken immediately, as well as near and long term. The conclusions reached following this significant effort were that the use of elevated temperatures in bounding analyses failed to exceed design

parameters, no equipment inside the ANO-1 containment had exceeded its qualified life and that no equipment was identified as unable to perform its required function.

The net result of these conclusions is that the operation of ANO-1 with the elevated reactor building temperatures did not result in a significant reduction in the level of safety associated with the original design parameters. Following the submission of the JCO, Startup Test Deficiency Reports were reviewed to confirm that there had been an adequate resolution of identified deficiencies involving failure to meet design requirements. The review of these documents for neither ANO-1 nor ANO-2 has identified additional concerns.

AP&L utilized a mid-cycle outage previously scheduled for October, 1987 on ANO-1 to conduct walkdowns, performance tests and inspections, as well as install temporary temperature monitoring instrumentation in the ANO-1 reactor building. In addition, attention was given to the identification of any measures which could be taken to provide incremental reductions to reactor building temperature. The results of these efforts included a validation of input assumptions used in some of the JCO analyses, an improvement in the performance of the reactor building coolers following some added maintenance and the installation of modifications to the chilled water system to enable additional chiller capacity to be added, should it become necessary at some future time.

The data gathered from temporary instrumentation installed during the mid-cycle outage has been utilized to develop new heat loads in the ANO-1 reactor building and those heat loads were, in turn, used to develop a mathematical model to predict return air temperatures inside the reactor building under varying conditions. This model has been employed to analyze the effectiveness of various modifications and long term actions.

In January, 1988, AP&L submitted a long term action plan to reduce the reactor building return air temperature. This action plan identified actions to be taken and additional options to be evaluated between then and the end of the ninth refueling outage (1R9). Specific actions to be performed during the eighth refueling outage (1R8) (Fall, 1988) include replacement of the two chilled water pumps with pumps of higher capacity, re-balancing/modifying the HVAC system to maximize air flow, and repairing selected, damaged Reactor Coolant System (RCS) insulation identified during the mid-cycle outage walkdowns. Additional options under consideration for 1R9 include additional chiller capacity, an additional reactor building air handling unit and replacement of portions of the RCS insulation.

The specific future actions taken during 1R9 will be determined during the summer of 1989 and will be largely dependent upon how effective the 1R8 modifications have been. The ultimate goal for reactor building return air temperature is to achieve a 120°-125° temperature by the end of the 1R9 refueling outage. AP&L believes that the long term action

plan for ANO-1 reactor building temperature reduction is sound from an engineering standpoint and represents a comprehensive evaluation of options and alternatives which will result in an acceptable reduction of temperatures in a timely and responsive manner. In the interim, the ANO environmental qualification program has been updated to reflect the August, 1987, JCO reanalysis to address equipment replacement during the period that efforts are ongoing to reduce the reactor building temperatures.

Several programmatic initiatives were undertaken by AP&L prior to the August 1987 inspection which were aimed at improving the quality of our design reviews and design change process so as to enhance our ability to identify design discrepancies and fully incorporate necessary changes/modifications into our design program. As stated previously, AP&L had recognized a weakness in this area and had undertaken efforts to strengthen this aspect of its nuclear program well before August, 1987. One of the first major steps was the development and implementation of a program to improve the design reviews and documentations required by 10CFR 50.59. A 10CFR50.59 policy was developed under Plant Safety Committee/Safety Review Committee direction consistent with NRC requirements and clearly defined the scope of the review, as well as the responsibilities of the parties involved. The new policy also required 10CFR50.59 reviewers to meet the experience requirements of ANSI/ANS-3.1-1981 for their discipline,

and that they successfully complete a competency-based training program as well as certified by the Plant Safety Committee or Safety Review Committee. The program also requires that qualified 10CFR50.59 reviewers requalify every two (2) years.

The increased emphasis on 10CFR50.59 reviews indicated a need to further improve the quality of the Safety Analysis Reports (SAR). In response to that identified need, AP&L management established a 1987 Nuclear Operations Goal to upgrade the SAR for both Unit 1 and Unit 2. The project was developed with four (4) major objectives. These are to improve the accuracy of the SARs in support of the new 50.59 policy, provide consistent descriptions within the SAR, improve grammatical content and overall quality of the SAR and add the appropriate NRC commitments that fall within 50.59 to the SAR. The upgraded SARs are scheduled for submittal to the NRC in 1988. The benefits and results of the upgrade project which we expect to realize include the identification and incorporation into the SARs of facility changes or deviations from expected operating conditions that were not adequately addressed previously, a more comprehensive review process for future facility changes due to greater detail within the SARs, the correction of conflicting and confusing information and an improvement in the SAR update process because of lessons learned from this effort.

Due to the identification of weaknesses in our previous design interfaces, improvements in both the design and plant modification processes have been initiated. These improvements have addressed not only programmatic changes but have also included changes to the design

engineering organization as well. To improve the efficiency of interfaces, the majority of corporate engineering has been reassigned to Nuclear Operations, reporting directly to the Vice President, Nuclear; engineering disciplines have undergone some consolidation to reduce the number of interfaces required; and nuclear engineering has been relocated into the design engineering organization in order to ensure that safety analyses are in the mainstream of design engineering efforts. The engineering organization located on-site has been restructured as a major department in an effort to maximize the efficiency of that interface into the design engineering organization. These efforts have been aimed at correcting those weaknesses which were attributed to poor intra-organizational communications and interfaces.

Building upon the organizational re-alignment, programmatic changes in the design process have been initiated concerning the development of design change packages (DCP) to avoid any confusion as to the need for each DCP to document the associated design basis; to address failure modes and effects considerations; to address operational modes and functional requirements and also to receive a 10CFR 50.59 review performed in accordance with the formal 50.59 program previously discussed. DCPs will also receive an independent review by a third party utilizing system expertise and considerations, and there is to be increased emphasis on the role of lead engineers for DCP overall coordination and responsibility.

Supporting these improvements to the design process are efforts to develop individual design engineering personnel via a customized

training program utilizing INPO criteria for job task analysis based training. Significant engineering manhours have been expended on training in a variety of topics. Additional supporting improvements include significant upgrades in the quality and content of various design documents. Considerable effort has been, and continues to be, expended to address such documentation as the cable information system, piping and support design specifications, reactor protective system setpoint analysis, the equipment data base resident in the Station Information Management System (SIMS), the vendor technical manual project, safe shutdown capability calculations, isometric drawing update project and the fire protection program manual. In addition, a formal Design Basis Documentation Project has been undertaken to systematically evaluate and upgrade the various types of design basis documentation for ANO.

The improved integration of our quality organization into the design process has been re-evaluated as well. This has resulted in steps to develop quality action plans for DCPs to enhance the early identification of key interface points for the quality organization in DCP development and the production of a specific action plan identifying QA/QC actions relative to each DCP.

This effort should provide for quality organization input at key steps along the developmental path of a DCP instead of applying that input at the end of the process, thereby improving the efficiency and effectiveness of the impact of the quality organization on DCP development.

As integral to those efforts in improving the design process, AP&L has evaluated the plant modification process to determine where improvements can be made to this function. The impetus for this effort, as for the whole design process, came from concerns identified by AP&L management and the AP&L quality organization. In responding to those concerns, a task force was established to develop a definitive, prioritized list of weaknesses and to provide recommendations and alternatives for correcting these weaknesses.

Stemming from this effort, and in conjunction with organizational changes made to the design engineering organization, a re-organization of Nuclear Operations resources created an on-site Plant Modifications Department. As with the other changes to the nuclear organization, these re-alignments were aimed at resolving past concerns related to effective interface and communications. In this case, the objective was to improve this aspect where it involved design development and installation of a design into the plant.

The major programmatic improvements which have accompanied the organizational re-alignment include the development of a Plant Modification Manual, the development and approval of a Plant Modification Process Implementation Plan and the development of Implementing Procedures. These improvements have significantly upgraded the process by requiring more advanced planning and coordination, and more in-depth design development of plant

modifications. These efforts have also resulted in effective positive control of the process and require greater accountability for plant modifications.

To address concerns relating to internal as well as external reporting of significant items, AP&L is implementing a central corrective action system called the Condition Reporting System. This system fosters an integrated approach to reporting deficiencies at ANO. The mechanics of how the Condition Reporting System will function and its relationship to other reporting channels at ANO were presented during a Management Meeting on February 4, 1988. The Condition Reporting System will specifically provide for the identification and disposition of deviations from design documentation.

The Condition Reporting System will continue to track conditions noted to ensure that data in the system is comprehensive. This integrated database will become a significant enhancement to ANO trending tools.

As part of the Condition Reporting System project, a control system for Condition Reports has been developed to provide further assurance that changes are tracked until closeout, and then, that historical documentation is properly retained. Additional enhancements, including escalation of overdue actions, management interface at the department head level via a signoff of closure of an item and the flexibility of the oversight role of the Plant Safety Committee, were addressed to resolve internal concerns.

The core objectives that AP&L has had in mind throughout the considerable programmatic and organizational re-evaluations, which have been discussed here, are to both improve our overall performance and to also improve plant personnel's, as well as AP&L management's responsiveness to potential problem areas. AP&L has fully appreciated and taken steps to strengthen control of the design issues which are the central subject of both our concerns and those of the NRC. AP&L has approached these activities with an eye toward implementing actions which will cause our nuclear organization to respond to plant related deficiencies in such a manner that not only resolves the issue of concern, but also forces the root cause to be identified and addressed, thereby preventing future evolutions of the same concern from developing.

5. The Date When Full Compliance Will Be Achieved

AP&L believes that, with regard to this violation, we are currently in compliance with the applicable regulations. The JCO which was presented on August 28, 1987 documented that there was not a reduction in the level of safety relating to the operation of ANO-1 with elevated reactor building temperatures. The commitments contained in the JCO for actions which AP&L would take to reduce those temperatures, as a matter of prudence, are expected to be completed by the end of the 1R9 refueling outage.

The organizational changes which were discussed are implemented, as are the procedure changes associated with improvements to the design and

plant modifications processes. The full implementation of the Condition Reporting System is anticipated by May 1988. The SAR upgrade project is scheduled for completion during 1988. Those efforts which will increase and further develop the overall safety-consciousness of the AP&L nuclear organization, such as individual development as well as further refinements in design basis documentation and programmatic improvements resulting from operational experience, will continue to be ongoing endeavors.