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

ARKANSAS POWER AND LIGHT COMPANY

DOCKET NO. 50-368

ARKANSAS NUCLEAR ONE, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 20
License No. NPF-6

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

   A. The application for amendment by the Arkansas Power & Light Company (the licensee) dated October 31, 1980 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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2. Accordingly, Facility Operating License No. NPF-6 is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and by revising and adding paragraphs as follows:

   (1) Revise paragraph 2.C.(2) to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 20, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

   (2) Add paragraphs 2.C.(5) and 2.C.(6) to read as follows:

   (5) The Arkansas Power & Light Company shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following:

   1. Provisions establishing preventive maintenance and periodic visual inspection requirements, and

   2. Integrated leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

   (6) The Arkansas Power & Light Company shall implement a program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

   1. Training of personnel,

   2. Procedures for monitoring, and

   3. Provisions for maintenance of sampling and analysis equipment.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the
Technical Specifications

Date of Issuance: March 3, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 20

FACILITY OPERATING LICENSE NO. NPF-6

DOCKET NO. 50-368

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Section	Page
Table 3.3-10	3/4 3-40
Table 4.3-10	3/4 3-41
3/4 4-5	3/4 4-5
B 3/4.3.3.6	B 3/4 3-3
B 3/4.4.4.4	B 3/4 4-2
Table 6.2-1	6-4
6.4.1	6-5

INSTRUMENTATION

POST-ACCIDENT INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.6 The post-accident monitoring instrumentation channels shown in Table 3.3-10 shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With the number of OPERABLE post accident monitoring channels less than required by Table 3.3-10, either restore the inoperable channel to OPERABLE status within 30 days, or be in HOT SHUTDOWN within the next 12 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.6 Each post-accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-10.

TABLE 3.3-10

POST-ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE
1. Containment Pressure	2
2. Containment Radiation Monitors*	2
3. Pressurizer Pressure	2
4. Pressurizer Water Level	2
5. Steam Generator Pressure	2/steam generator
6. Steam Generator Water Level	2/steam generator
7. Refueling Water Tank Water Level	2
8. Containment Sump Water Level	2
9. Emergency Feedwater Flow Rate	1/steam generator
10. Reactor Coolant System Subcooling Margin Monitor	1
11. Pressurizer Safety Valve Acoustic Position Indication	1
12. Pressurizer Safety Valve Tail Pipe Temperature	1

*This requirement may be satisfied by the use of portable radiation monitors equivalent in number to the minimum channels required OPERABLE until such time as the Category B portions of Item 2.1.8.B of NUREG-0578 must be implemented for ANO-2.

TABLE 4.3-10

POST-ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Containment Pressure	M	R
2. Containment Radiation Monitors*	M	R
3. Pressurizer Pressure	M	R
4. Pressurizer Water Level	M	R
5. Steam Generator Pressure	M	R
6. Steam Generator Water Level	M	R
7. Refueling Water Tank Water Level	M	R
8. Containment Sump Water Level	M	R
9. Emergency Feedwater Flow Rate	M	R
10. Reactor Coolant System Subcooling Margin Monitor	M	R
11. Pressurizer Safety Valve Acoustic Position Indication	M	R
12. Pressurizer Safety Valve Tail Pipe Temperature	M	R

* This requirement may be satisfied by the use of portable radiation monitors, and by substituting a source check for the channel check and by substituting an instrument calibration for the channel calibration until such time as the Category B portions of Item 2.1.8.B of NUREG-0578 must be implemented for ANO-2.

INSTRUMENTATION

CHLORINE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.3.3.7 Two independent chlorine detection systems, with their alarm/trip setpoints adjusted to actuate at a chlorine concentration of $\leq$ 5 ppm, shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one chlorine detection system inoperable, restore the inoperable detection system to OPERABLE status within 7 days or within the next 6 hours initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation.

b. With no chlorine detection system OPERABLE, within 1 hour initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.7 Each chlorine detection system shall be demonstrated OPERABLE by performance of a CHANNEL CHECK at least once per 12 hours, a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per 18 months.

REACTOR COOLANT SYSTEM

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.4 The pressurizer shall be OPERABLE with a water volume of $\leq$ 910 cubic feet (equivalent to $\leq$ 82% of wide range indicated level) and both pressurizer proportional heater groups shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

(a) With the pressurizer inoperable due to water volume $\geq$910 cubic feet, be in at least HOT SHUTDOWN with the reactor trip breakers open within 12 hours.

(b) With the pressurizer inoperable due to an inoperable emergency power supply to the pressurizer heaters, either restore the inoperable emergency power supply within 72 hours or be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.4.1 The pressurizer water volume shall be determined to be within its limits at least once per 12 hours.

4.4.4.2 The pressurizer proportional heater groups shall be determined to be OPERABLE:

(a) At least once per 12 hours by verifying emergency power is available to the heater groups, and

(b) At least once per 18 months by verifying that the summed power consumption of the two proportional heater groups is $\geq$ 150 KW.

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.5.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.

b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

3/4.3.3.6 POST-ACCIDENT INSTRUMENTATION

The OPERABILITY of the post-accident instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Plants to Assess Plant Conditions During and Following an Accident," December 1975, and NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short Term Recommendations."

3/4.3.3.7 CHLORINE DETECTION SYSTEMS

The OPERABILITY of the chlorine detection system ensures that sufficient capability is available to promptly detect and initiate protective action in the event of an accidental chlorine release. This capability is required to protect control room personnel and is consistent with the recommendations of Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," February 1975.

3/4.3.3.8 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumention is inoperable, the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

3/4.3.4 TURBINE OVERSPEED PROTECTION

This specification is provided to ensure that the turbine overspeed protection instrumentation and the turbine speed control valves are OPERABLE and will protect the turbine from excessive overspeed. Protection from turbine excessive overspeed is required since excessive overspeed of the turbine could generate potentially damaging missiles which could impact and damage safety related components, equipment or structures.

3/4.4 REACTOR COOLANT SYSTEM

REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.4.1 Both reactor coolant loops and both reactor coolant pumps in each loop shall be in operation.

APPLICABILITY: As noted below, but excluding MODE 6.*

ACTION:

MODES 1 and 2:

a. With one reactor coolant pump not in operation, STARTUP and/or continued POWER OPERATION may proceed provided THERMAL POWER is restricted to $\leq$ **% of RATED THERMAL POWER and the setpoint for the Linear Power Level - High trip has been reduced to the value specified in Specification 2.2.1 for operation with three reactor coolant pumps operating.

b. With two reactor coolant pumps in opposite loops not in operation, STARTUP and/or continued POWER OPERATION may proceed provided THERMAL POWER is restricted to $\leq$ **% of RATED THERMAL POWER and the setpoint for the Linear Power Level - High trip has been reduced to the value specified in Specification 2.2.1 for operation with two reactor coolant pumps operating in opposite loops.

c. With two reactor coolant pumps in the same loop not in operation, STARTUP and/or continued POWER OPERATION may proceed provided the water level in both steam generators is maintained above the Steam Generator Water Level-Low trip setpoint, the THERMAL POWER is restricted to $\leq$ **% of RATED THERMAL POWER, and the setpoint for the Linear Power Level - High trip has been reduced to the value specified in Specification 2.2.1 for operation with two reactor coolant pumps operating in the same loop.

*See Special Test Exception 3.10.3

**These values left blank pending NRC approval of ECCS analyses for operation with less than four reactor coolant pumps operating.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Vessel Code.

3/4.4.4 PRESSURIZER

A steam bubble in the pressurizer ensures that the RCS is not a hydraulically solid system and is capable of accommodating pressure surges during operation. The steam bubble also protects the pressurizer code safety valves against water relief. The steam bubble functions to relieve RCS pressure during all design transients.

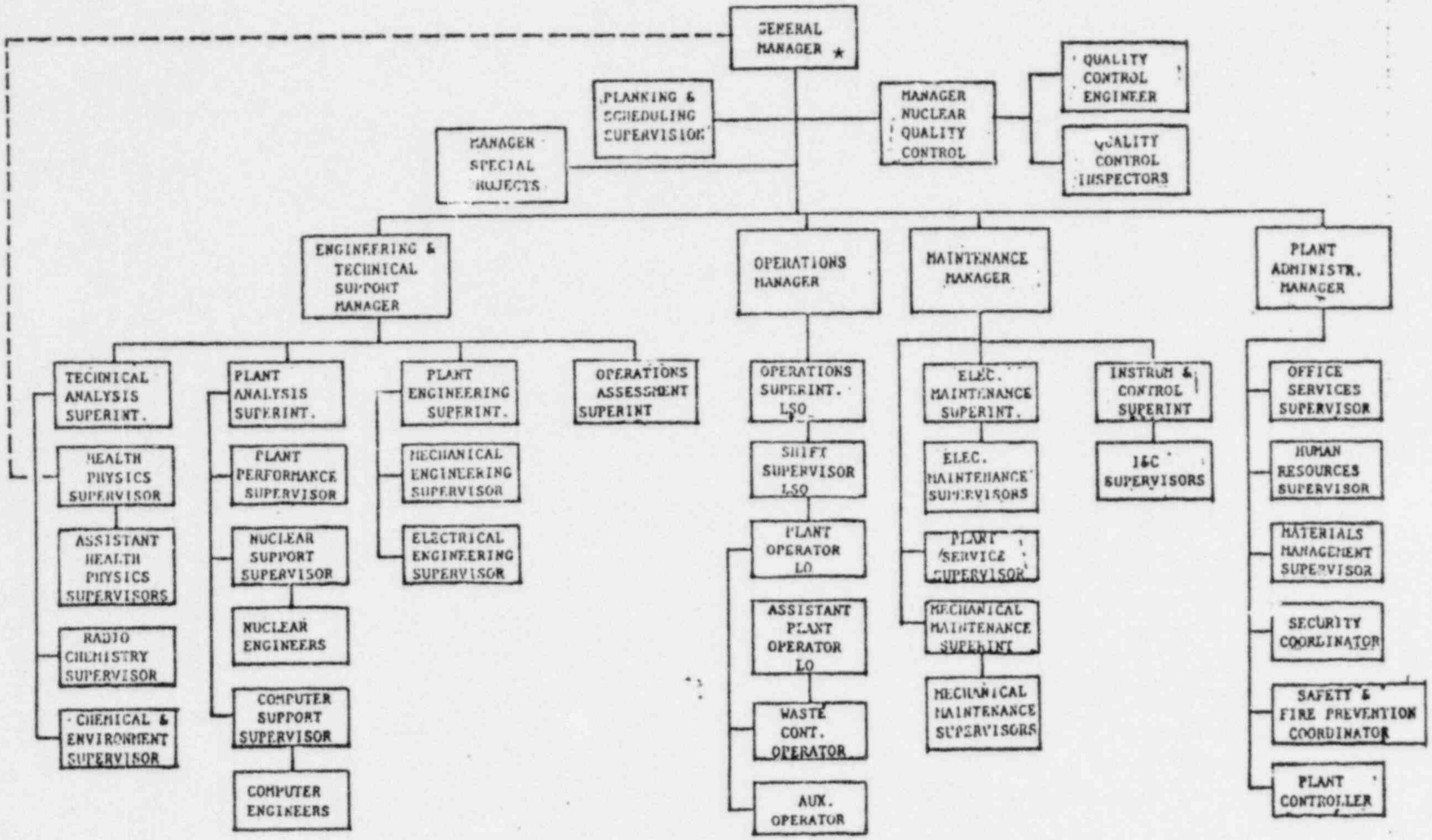
The requirement that 150 KW of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss-of-offsite power condition to maintain natrual circulation at HOT STANDBY.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 0.5 GPM per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 0.5 GPM per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

ARKANSAS POWER & LIGHT COMPANY
ARKANSA NUCLEAR ONE



CODE: LSO-SENIOR OPERATOR LICENSE REQUIRED
LO- OPERATOR LICENSE REQUIRED
*ONSITE RESPONSIBILITY FOR FIRE PROTECTION PROGRAM

Figure 6.2-2 Functional Organization for Plant Operation

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION#

LICENSE CATEGORY	APPLICABLE MODES	
	1, 2, 3 & 4	5 & 6
SOL	1	1*
OL	2	1
Non-Licensed	2	1
Shift Technical Advisor	1	None Required

*Does not include the licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling, supervising CORE ALTERATIONS.

#Shift crew composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2-1.

6.3 UNIT STAFF QUALIFICATIONS

6.3.1. Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for (1) the Health Physics Supervisor who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, and (2) the Shift Technical Advisor who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents.

6.4 TRAINING

6.4.1. A retraining and replacement training program for the unit staff shall be maintained under the direction of the General Manager and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix "A" of 10 CFR Part 55.

6.4.2 A training program for the Fire Brigade shall be maintained under the direction of the General Manager and shall meet or exceed the requirements of Section 27 of the NFPA Code - 1975, except for Fire Brigade training sessions which shall be held at least quarterly.

6.5 REVIEW AND AUDIT

6.5.1 PLANT SAFETY COMMITTEE (PSC)

FUNCTION

6.5.1.1 The Plant Safety Committee shall function to advise the Plant Manager on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The Plant Safety Committee shall be composed of the:

Chairman:	Manager of Special Projects
Member:	Operations Manager
Member:	Maintenance Manager
Member:	Engineering and Technical Support Manager
Member:	Administrative Manager
Member:	Technical Analysis Superintendent
Member:	Plant Analysis Superintendent
Member:	Plant Engineering Superintendent
Member:	Health Physics Supervisor
Member:	Nuclear Software Expert*

The General Manager shall designate in writing the Alternate Chairman in the absence of the PSC Chairman.

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* See page 6-5a

*If one of the above members of the Plant Safety Committee meets the qualification requirements for this position, the requirement to have this member is satisfied. This membership may be filled by two appropriately qualified individuals who shall ballot with a single combined vote. Generic qualifications for this membership shall be as follows:

One Individual

The Nuclear Software Expert shall have as a minimum a Bachelor's degree in Science or Engineering, Nuclear preferred (in accordance with ANSI N18.1). In addition, he shall have a minimum of four years of technical experience, of which a minimum of two years shall be in Nuclear Engineering and a minimum of two years shall be in Software Engineering. (Software Engineering is that branch of science and technology which deals with the design and use of software. Software Engineering is a discipline directed to the production and modification of computer programs that are correct, efficient, flexible, maintainable, and understandable, in reasonable time spans, and at reasonable costs). The two years of technical experience in Software Engineering may be general software experience not necessarily related to the software of the Core Protection Calculator System. One of these two years of experience shall be with certified computer programs.

Two Individuals

One of the individuals shall meet the requirements of the Nuclear Engineering portion of the above. The second individual shall have a Bachelor of Science degree (digital computer speciality) and meet the Software Engineering requirements of the above.

The membership (the Nuclear Software Expert or the Digital Computer Specialist) shall be knowledgeable of the Core Protection Calculator System with regard to:

a. The software modules, their interactions with each other and with the data base.

b. The relationship between operator's module inputs and the trip variables.

c. The relationship between sensor input signals and the trip variable.

d. The design basis of the Core Protection Calculator System.

e. The approved software change procedure and documentation requirements of a software change.

f. The security of the computer memory and access procedures to the memory.