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CHAPTER 13 - CONDUCT OF OPERATIONS

13.1 ORGANIZATIONAL STRUCTURE OF APPLICANT

13.1.1 Role and Responsibility of KHNP

Korea Hydro & Nuclear Power Company (KHNP) ensures and maintains technical ability, operating organization and manpower appropriate for operating Yonggwang Nuclear Power Plant Unit 3&4(YGN 3&4) without endangering the health and safety of the public and plant personnel.

KHNP has total responsibility for the construction and operation of YGN 3&4. The planning, scheduling, and performance of the initial tests and plant operations are under the direct control of KHNP.

Korea Power Engineering Company (KOPEC) has been retained by PHNP to provide architect-engineering services, including engineering, procurement, construction management, and technical direction services for YGN 3&4.

The nuclear steam supply system(NSSS) is provided by Korea Heavy Industries & Construction Co., Ltd. (KHIC) and Asea Brown Boveri Combustion Engineering (ABB-CE).

The turbine-generator(TG) has an electrical output of approximately 1,000MWe. The turbine-generator and auxiliaries are provided by KHIC and General Electric Company.

The balance of plant(BOP) designed by KOPEC and Surgent & Lundy(S&L) is compatible with the above NSSS and TG.

Preoperational testing and initial startup testing are conducted under the direct control of KHNP, with technical assistance from KOPEC and KHIC.

Detailed procedures are prepared by KHNP from design information and procedures provided by KOPEC, KHIC, GE and the manufacturers of the equipment. The results of initial tests and operation are fully documented and reviewed by KHNP personnel qualified by experience and training.

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13.1.2 Related Organization of Nuclear Power Plant Operation

KHNP manages the organization of supervision and technical assistance at the head office in Seoul in order to improve the safety and reliability of YGN 3&4.

The functions of the organization follow :

a. plant operation and maintenance,
b. maintenance of safety and improvement of facilities,
c. training and management of plant personnel,
d. nuclear fuel and in-core management,
e. management of health physics, radiation protection and radioactive materials,
f. establishment and adjustment of a plan for developing and using nuclear power,
g. establishment of QA programs and confirmation of QA activities, and
h. research and development for plant operation and maintenance.

The organization table and division of responsibility of the organization are included in the Regulation of KHNP.

13.1.3 Operating Organization of Plant

13.1.3.1 Operating Organization

KHNP manages the operation organization of YGN 3&4 responsible for the following activities :

a. general management of overall plant operation,
b. general management of plant operation,
c. general management of plant maintenance,
d. management of plant operation, efficiency and training,
e. operation of plant facility and equipment management,
f. management of plant safety, assistance of licensing, technical assistance of operation and maintenance,
g. management of rad-waste, health physics and operation of radiation emergency plan.
h. water treatment for reactor and turbine system.
i. management and maintenance of mechanical equipment.

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j. management and maintenance of electrical equipment,
k. management and maintenance of instrumentation and control equipment,
l. refueling, operation and maintenance of fuel handling equipment, and
m. confirmation of QA activity and reporting of significant defects.

The organization table and division of responsibility of the organization are included in the Regulation of KHNP.

13.1.3.2 Responsibility and Authority

During normal plant operation, Director General is responsible for all plant activities. In his absence, or in an emergency situation, the operation office Deputy Director General, the maintenance & engineering office Deputy Director General, and the safety & engineering support Team General Manager, and the operation management Team Genaral Manager are responsible for all 775
plant operations in order.

During back shifts, weekends, or holidays, the immediate responsibility falls upon the shift supervisor, followed by the assistant shift supervisor.

13.1.3.3 Operating Shift Crew Composition

Each unit employs six operating shifts to allow for routine training, vacations, and sickness.

ITS Chapter 3 1.3 describes the composition of operating shift crew.

13.1.3.4 Qualification Requirements for Nuclear Power Plant Personnel

The operating, technical, and maintenance support personnel of all nuclear power plant must complete training according to the following :

a. KHNP's Nuclear Training Program
b. Minimum Qualification Requirements of ANSI/ANS-3.1-1993

License requirements for the personnel of nuclear power plant must meet the following :

a. Presidential Decree Chapter 8, pursuant to Article 84 of Nuclear Safety Act of Republic of Korea 757

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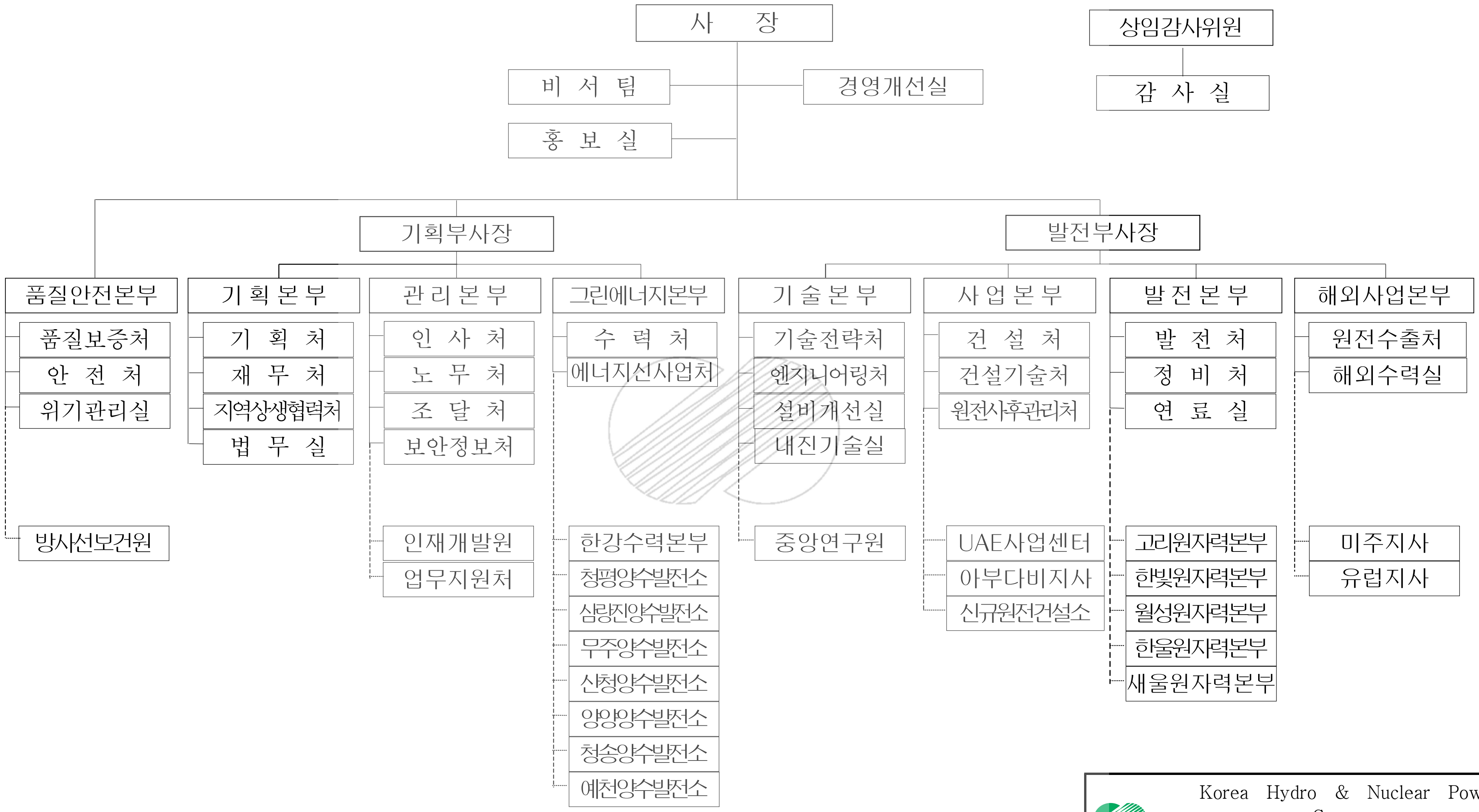
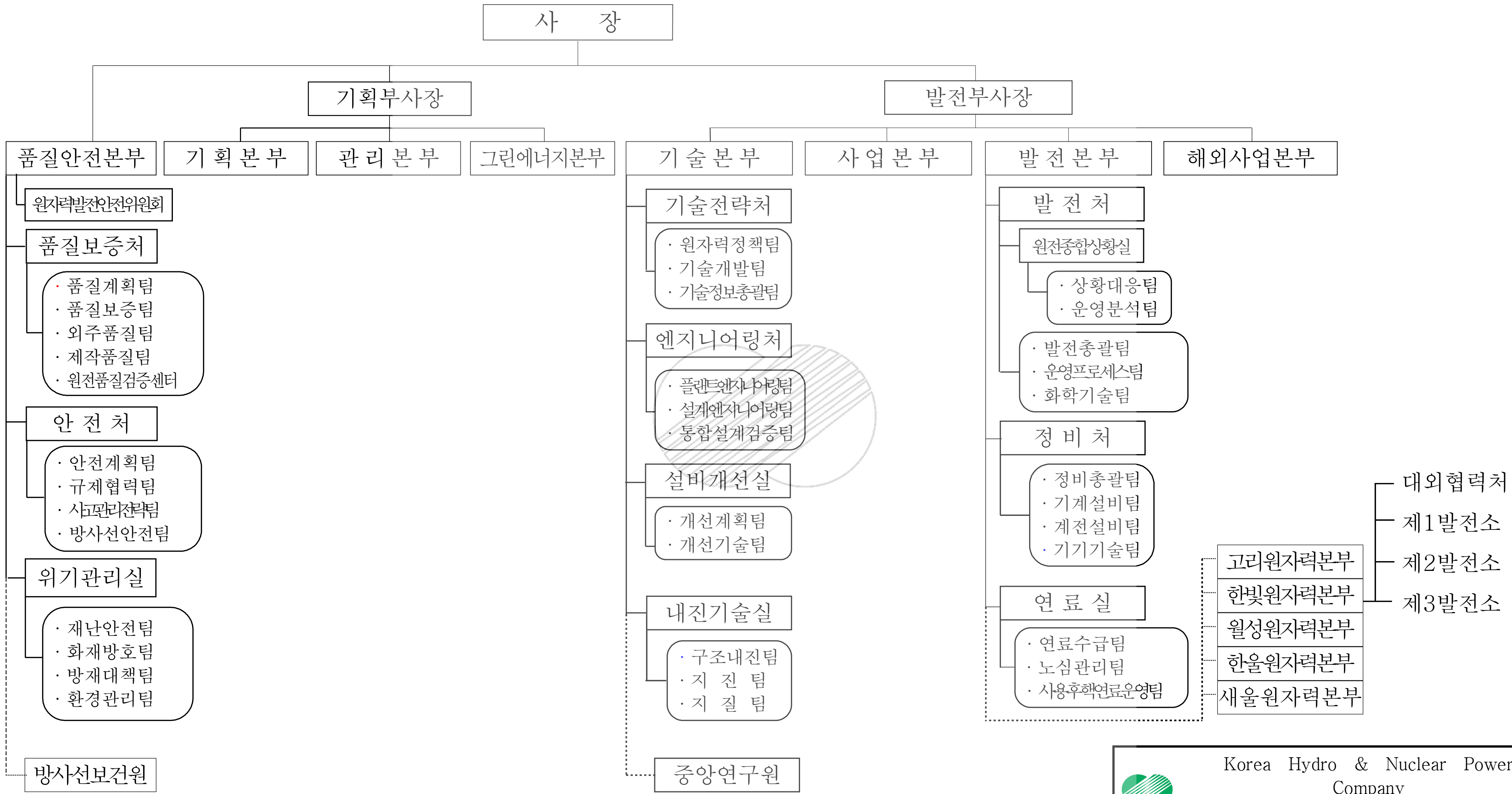


Korea Hydro & Nuclear Power Company
HBN 3 & 4 FSAR

Organization Chart Korea Hydro & Nuclear Power Company

Figure 13.1-1

Amendment 773
2017.3.22



Korea Hydro & Nuclear Power Company
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Organization in the field of the Nuclear Power

Figure 13.1-2

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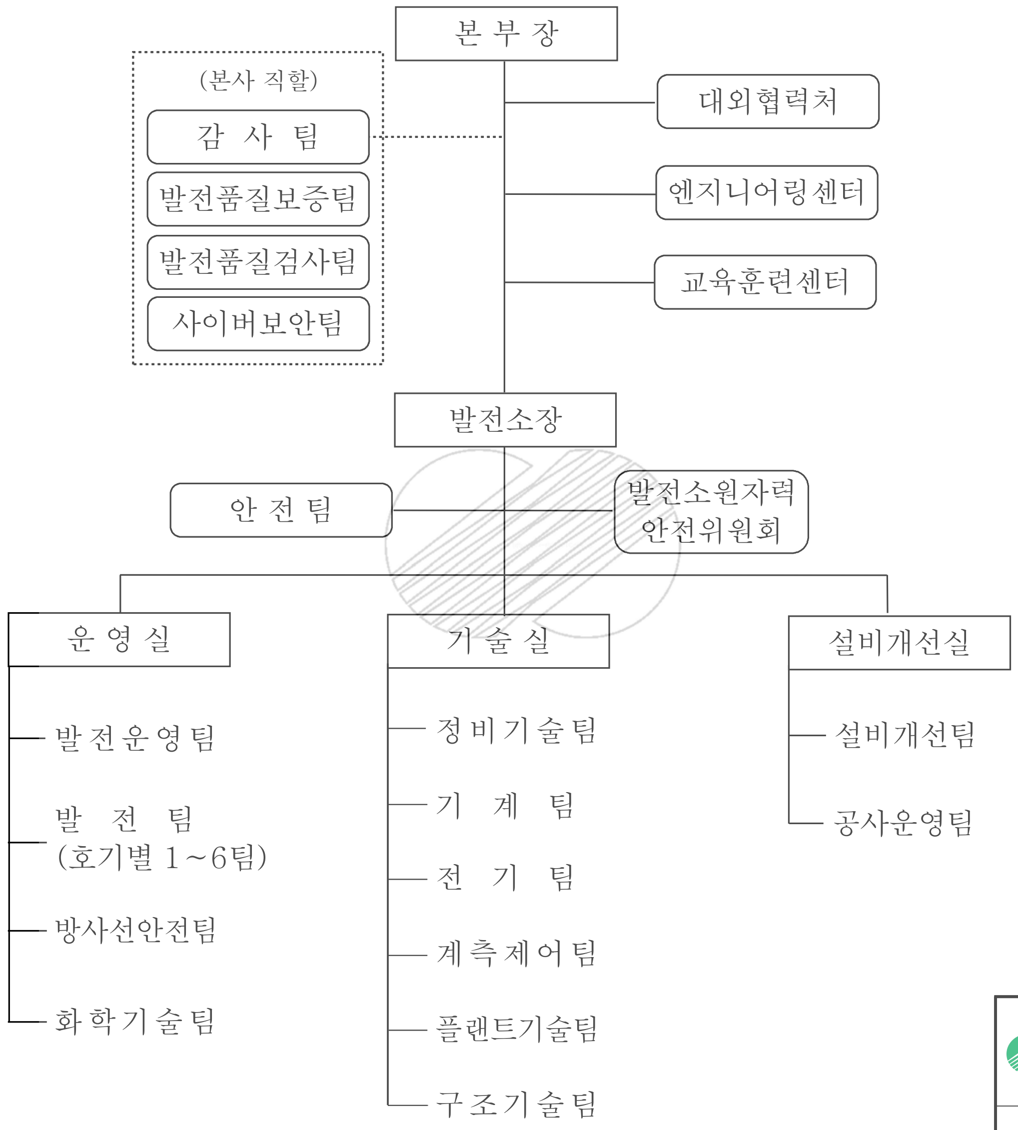

Korea Hydro & Nuclear Power Company
HBN 3 & 4 FSAR

Site of Hanbit 3&4 Organization

Figure 13.1-3

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13.2 TRAINING

13.2.1 Plant Staff Training Program

The training program is designed to provide plant personnel with requisite knowledge and skills, and enable them to practice safety improvements and plant efficiency. Individual training needs, contents, and levels are established by careful examination of the trainees experience, previous training, and job requirements. Responsibility for administration and evaluation of the training program planned or conducted by KEPCO's Nuclear Training Center rests with the Center's general manager. Responsibility for administration and evaluation of the training program planned or conducted by the plant rests with the plant manager. In case the training program planned by KEPCO's Nuclear Training Center is conducted at the plant, the Training Center's general manager can delegate responsibility for administration and evaluation of the training program to the plant manager for efficient progress. Effectiveness of the training program is evaluated through analysis of the training courses, estimation of the training effects, and the 181
performance of employees in carrying out their assigned duties.

13.2.1.1 Program Description

The training program consists of Nuclear Recruits' Basic Training, Plant Personnel Duty Training, General Employee Training, Fire Protection Training, and Project Contract Training. Bellow is a description of the training program.

13.2.1.1.1 Nuclear Recruits' Basic Training

13.2.1.1.1.1 Basic Nuclear Training

The basic nuclear training program is an assembly training course for all nuclear recruits who work in a technical department. This program is divided into two courses.

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a. Course I : Basic Nuclear Theory

This is a 8 week course that provides basic knowledge and nuclear theory related to the overall power plant. Duration of the training course can be adjusted within 20% according to the entry level of the recruits. The topical outline of the course is as follows :

- o Reactor theory and control
- o Thermal hydraulics engineering
- o Electronics / Instrumentation & Control fundamentals
- o Chemistry fundamentals
- o Radiation fundamentals
- o Mechanics / Electricity fundamentals

b. Course II : Basic Nuclear Systems

This is a 10 week course conducted to provide a knowledge of nuclear power plant systems, power plant design, and overall power plant operation after completion of the basic nuclear theory course. The training is conducted by distinguishing common items from particular items of the nuclear power plant. The total duration of the training course can be adjusted by no more than 20%, according to the entry level of the recruits. The topical outline of the course is as follows : 181

- o Reactor Equipment
- o Reactor Auxiliary Equipment
- o Reactor Safety Equipment
- o Turbine and Steam Generator Equipment
- o Generator and Auxiliary Equipment
- o Control and Protection Equipment
- o Plant Electrical System
- o Administration of Techniques

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13.2.1.1.1.2 Onsite Familiarization Training

This training program is conducted at KEPCO's Headquarters or plant site for 16 weeks. It covers sections for which individual training is more effective than assembly training for nuclear recruit's assigned to onsite jobs. The total duration of the training course can be adjusted within a 20% range according to the entry level of the recruits. To improve training efficiency it may be divided and then conducted before, after, or during the assembly training of 13.2.1.1.1.1. The contents of the training course can be adjusted according to the entry level of the recruits and prearranged on site. The main contents of the training course are as follows :

- Organization and general administration in techniques
- Understanding operation through cooperation with operator
- Acquiring practical experience through the rotation of services among plant departments

13.2.1.1.2 Plant Personnel Duty Training 181

This training program is designed to provide operating, maintenance, and support personnel with the skills, knowledge, and abilities necessary to perform job assignments. The courses, contents, trainees, and duration of the training program are quite flexible according to the needs and emphasis of the plant that are periodically conducted by the annual KEPCO training program. The contents of the training program are as follows :

13.2.1.1.2.1 Training for Main Control Board Operators

This training program is designed to provide practical knowledge for 10 weeks. Duration of the training course can be adjusted within a 20% range according to the entry level of the applicants. The topical outline of the course is as follows :

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- o Reactor theory
- o Radiation management
- o Operation practice
- o Control room familiarization
- o Nuclear act
- o Fuel handling
- o Reactor operation and control
- o Reactor structure and design
- o Mitigating core damage
- o Administrative procedure

13.2.1.1.2.2 Electrical Maintenance Personnel Training

This training program on electrical maintenance is designed to provide theory and practical knowledge in general nuclear electricity and the main electric facilities for at least 1 week. The topical outline of the course is as follows :

- o Electricity in general 181
- o Electric facilities

13.2.1.1.2.3 Mechanical Maintenance Personnel Training

This training program is designed to provide theory and practical knowledge on general mechanics and main mechanical facilities for at least 1 week. The topical outline of the course is as follows :

- o Mechanics in general
- o Mechanical facilities

13.2.1.1.2.4 Instrumentation and Control Personnel Training

This training program is designed to provide theory and practical knowledge in instrumentation & control of general and main facilities for at least 1 week. The topical outline of the course is as follows :

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- o General instrumentation & control
- o Instrumentation & control facilities

13.2.1.1.2.5 Plant Computer Personnel Training

This training program is designed to provide special knowledge in the operation of computer hardware and software as it is related to computer operation and maintenance. The program is designed to last for at least 1 week. The topical outline of the course is as follows :

- o Computer hardware
- o Computer software

13.2.1.1.2.6 Personnel Training in Chemistry

This program for chemists is designed to provide general knowledge and practical analysis in chemistry for at least 1 week. The topical outline of the course is as follows : 181

- o General practice on chemistry
- o Analysis practice on chemistry
- o Primary water chemistry control
- o Secondary water chemistry control
- o Radio-Chemistry

13.2.1.1.2.7 Radiation Control Personnel Training

This program is designed to provide theory and practical knowledge in radiation control for at least 1 week. The topical outline of the course is as follows :

- o Health physics theory
- o Environmental radiation management
- o Radiation material management
- o Radiation emergency management

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13.2.1.1.2.8 Incore Management Personnel Training

This program is designed to provide theory and practical knowledge in incore management for at least 1 week. The topical outline of the course is as follows :

- o Reactor theory
- o Nuclear design
- o Incore management

13.2.1.1.2.9 Supervisor/Manager Training

This program is applicable to supervisor/manger including technical staff personnel. The program provides the practical knowledge, manegement skill and supervisoring their departments in the performance of assigned responsibilities. It is conducted in the management develoment couses or participating at the concerned workshop and the expert meeting at home and abroad. The topical outline of the course is as follows : 181

- o Manegement skill (Leadership, Supervisory, Business control, etc.)
- o Reguration, policy and strategy of the company
- o Advanced practical knowledge

13.2.1.1.3 General Employee Training

All persons regularly employed by KEPCO to work at nuclear power plants are instructed in the following areas at the KNTC and/or plant.

- o Radiological Health and Safety
- o Emergency Plan
- o Fire Protection and Security
- o Quality Assurance

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All persons having unescorted access to the plant areas must complete training courses in (1) elementary health physics (2) radiation protection techniques for entering radiation work areas and controlled zones (3) pertinent sections of the site emergency plan. When persons who have not completed these training courses enter plant areas, they are escorted by an employee who has received the pertinent training.

13.2.1.1.4 Fire Protection Training

a. Fire brigade personnel must complete a training course covering the following topics :

- o Identification of fire hazards and associated fires types that occur in the plant.
- o Identification of areas where breathing apparatus is required, regardless of the size of the fire.
- o Familiarization with plant layout, including ingress and egress routes for each area. 181
- o Identification of the locations of installed and portable fire-fighting equipment in the plant.
- o Proper use of communication, lighting, ventilation, and emergency
  breathing equipment.
- o Proper use of available fire fighting equipment and correct methods of fire fighting for each type of fire.
- o Indoctrination in the fire protection plan which shall include fire brigade responsibilities.
- o Methods of fire fighting inside buildings and tunnels.
- o Design and maintenance of fire detection, suppression, and extinguishing systems.
- o Fire protection techniques and procedures.

Refresher training will be scheduled by the plant manager.

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b. Fire brigade drills

Fire brigade drills shall be performed in the plant to promote effective teamwork. Fire brigade drills shall be conducted using the following guidelines :

- Drill scenarios shall be prepared to establish the objectives of each drill.
- Each fire brigade shall be drilled at least quarterly.
- Each drill shall be evaluated to determine how well the training objectives are met.
- Drills shall include the following :
  - Assessment of fire alarm effectiveness, time required to notify and assemble fire brigade, selection, replacement and use of equipment, and fire fighting strategies.
  - Simulated use of fire fighting equipment required to cope with the situation and the type of fire selected for the drill.
  - Assessment of brigade leader's direction of the effort. 181
  - Assessment of each member's knowledge of fire fighting strategy, procedures, and use of equipment.

The plant manager shall be responsible for scheduling, conducting and documenting the fire brigade drills.

c. Instruction for all employees

Each plant employee shall receive instruction on the fire protection plan, and implementation instructions, evacuation routes from his normal place of duty, and procedures for reporting fires.

13.2.1.1.5 <u>Project Contract Training</u>

This program is designed to develop or enhance the skills, knowledge, and ability of personnel to perform job assignments.

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Implementation, courses, contents, and duration of the training program are flexible according to the contract conditions of the project.

13.2.2 Retraining and Replacement Training

13.2.2.1 Licensed Operator Requalification Training

The training program for the licensed (senior) operators will meet the requirements of the Atomic Energy Laws. This training program will keep the licensed (senior) operator familiar with plant design changes, and proficient in the execution of all procedures and the application of technical standards. This training consists of assembly training, on-the-job training, and operator evaluation.

13.2.2.1.1 Assembly Training

The training program is conducted for a continuous period that doesn't to exceed two years, and consists of a minimum of 50 hours of classroom training each year. Training is scheduled for designated groups, including main control room operators. These groups are relieved from regular duties for the training sessions. Each licensed (senior) operator in the main control room is assigned to a complete training course covering the following topics : 181

a. Theory and principles of reactor operation
b. General and specific plant operating characteristics
c. Plant instrumentation and control systems
d. Plant protection systems
e. Engineered safety systems
f. Normal, abnormal, and emergency operating procedures
g. Radiation control and safety
h. Technical specifications
i. Government regulations
j. Special subjects that are requested for the plant
k. Mitigating core damage

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overlooked. This training consists of three elements:

a. Lecture series
b. On-the-job training
c. Evaluation

13.2.2.1.1 Lecture Series

The training program is conducted for a continuous period not to exceed two years and consists of a minimum of 50 hours of classroom training a year. A minimum of 1 week or a maximum of 2 weeks of the training program is conducted at KNTC during the first 6 months of each year.

The balance of training required is conducted in the last 6 months of each year, if needed. Training is scheduled for the designated groups including the other main control room operators. These groups are relieved from regular duties for the training sessions. Each license holder in the main control room is assigned to a group and is scheduled for training.

The requalification program covers the following subjects:

a. Theory and principles of reactor operation
b. General and specific plant operating characteristics
c. Plant instrumentation and control systems
d. Plant protection systems
e. Engineered safety systems
f. Normal, abnormal, and emergency operating procedures
g. Radiation control and safety
h. Technical specifications
i. Applicable portions of government regulations
j. Special subjects that are requested for the plant, if any

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l. Transient and accident analysis

13.2.2.1.2 On-The-Job Training

Each licensed operator manipulates the plant controls and each licensed senior operator manipulates the controls and directs the activities of individuals. These manipulations keep each licensed (senior) operator familiar with the knowledge and skills of reactivity control systems through at least 10 reactivity control manipulations in any combination of reactor start ups and reactor shut downs by using the control panel of the facility involved or by using a simulator for a continuous period not to exceed two years. Each licensed (senior) operator is cognizant of facility design changes, procedure changes, and facility license changes. Each licensed (senior) operator reviews the contents of all abnormal and emergency procedures on a regularly scheduled basis.

13.2.2.1.3 Evaluation 181

At the end of each training program, the licensed (senior) operators will participate in an annual examination and must receive a score of more than 70% in any category in which he is trained. A grade of less than 70% requires further study and reexamination on the same subject.

13.2.2.1.4 Responsibility

The general manager of the KEPCO Nuclear Training Center is responsible for the collection and overall operation of the training program for the licensed (senior) operator. If a part or the whole of the training program is conducted at a plant location, the director and plant manager are responsible for assembly training, quizzes, and documentation for this part of the training program. The director and plant manager designate qualified individuals to be responsible for these duties.

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the overall operator requalification training program at the KEPCO Nuclear training center according the guidelines of this procedure.

He designates qualified individuals to be responsible for the preparation and presentation of lectures, quizzes, scheduling of training, annual written examination, performance evaluations, and documentation of the requalification training program, as required

13.2.2.1.5 Records

Records which include sufficiently detailed, updated information to adequately document the participation record of each license holder in requalification training are maintained. The records contain copies of written examinations administered, the answers given by the licensee, results of evaluations, and documentation of any additional training administered in which an operator or senior operator has exhibited deficiencies.

13.2.2.2 Refresher Training for Unlicensed Personnel

A refresher training program for unlicensed operators is conducted for a continuous period not to exceed two years and consists of a minimum of 50 hours of classroom training a year. The training groups are relieved from regular duties for the training sessions according to the training plan. The subjects in the refresher program are similar to that of the licensed operator program.

13.2.2.3 Replacement Training

KEPCO has a continuing responsibility to supply qualified personnel to fill vacancies in the supervisory, operating, instrument, and maintenance groups. It is the policy of KEPCO to promote qualified men into job vacancies from candidates next in line. This policy is to be implemented for replacement

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13.2.2.2 Refresher Training for Unlicensed Personnel

A refresher training program for unlicensed operators is conducted for a continuous period that doesn't exceed two years, and includes a minimum of 50 hours of assembly training each year. The training groups are relieved from regular duties for the training sessions according to the training plan. The subjects in the refresher program for unlicensed personnel are similar to that of the licensed operator program.

13.2.2.3 Replacement Training

It is the policy to promote qualified candidates to job vacancies. This policy is to be implemented for replacement personnel in the plant. The station staff, under the direction of the plant manager, will be responsible for the implementation of this on-the-job training program and will also maintain the proficiency of the replacement personnel. 181

13.2.3 Records

13.2.3.1 General Records

Records of employee qualifications, experience, and previous training are maintained in a standardized arrangement by the authorized department in line with the officially recognized data. The records are maintained in current and accurate status and are controlled according to their availability.

13.2.3.2 Plant Records

Records of training history for plant personnel are maintained by the authorized department of the plant. These records are comprised of completed training courses and correspondence confirming whether training requirements have been met or not.

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personnel in the YGN 3&4.

The program described in Subsection 13.2.1.1.1 will be used as the replacement training program, if it is appropriate. In addition, the on-the-job training program consists of classroom lecture reviews and on-the-job participation. The station staff, under the direction of the plant manager, will be responsible for the implementation of this on-the-job training program and will also maintain the proficiency of the replacement personnel.

13.2.3 Records

13.2.3.1 KEPCO Records

Official records of employee qualifications, experience, and training are maintained in the official KEPCO Personnel History Record (PHR) by the General Affairs Department (GAD). The GAD provides, in a standardized arrangement, the information officially recognized in recording and supporting employee status. The GAD records are maintained in current and accurate status and are controlled as to availability. The material recorded is restricted to items for which authenticity has been confirmed through established procedures; e.g., official KEPCO forms or signed statements from the employee, management representatives, etc.

13.2.3.2 Plant Records

Records of employee participation in training activities leading to promotion at a higher level of competence are maintained by plant training subcommittees. Records supporting requests for senior operator and operator licenses are maintained in the plant master files. These records include training courses attended, retraining classes, and other information necessary to ensure that training requirements have been met. Some of these records are duplicated in the GAD.

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13.2.3.3 Training Program Evaluations

Training programs conducted at the KEPCO Nuclear Training Center and associated plants enhance the knowledge and skills required by each plant staff member, including the reactor operator. Efficiency of the training program is evaluated through analysis of the training courses, surveys of the training effects, and other numerous kinds of examinations taken after the training course, including written examinations.

13.2.4 Training Records and Evaluations

13.2.4.1 Training Records

Individual training records which include training stats and level of the quality for each plant staff are maintained. The committed training programs which cannot get an evaluation are excluded.

13.2.4.2 Evaluations

In principle, training evaluations of collective training program are made with written examination. Other training programs are evaluated with interview, written examination, inspection, test, record review, and report. 181

13.2.4 Reference Documents

The following list of documents was used reference for the development of Nuclear Power Plant Personnel Training Programs :

- KOREA Nuclear Act and Rules
- KEPCO Nuclear Personnel Training System
- 10 CFR 50, "Licensing of Production and Utilization Facilities"
- 10 CFR 55, "Operators'Licenses"
- 10 CFR 20, "Standards for Protection Against Radiation"
- ANSI/ANS 3.1 1993, "Selection, qualification and training of personnel for Nuclear Power Plants"

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13.3 EMERGENCY PLANNING

The Hanbit 3&4 Emergency Plan is described in detail in a separately published 809
book.

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13.4 REVIEW AND AUDIT

The quality assurance manual for the operation phase will be established by applying KEPIC QAP and ANSI/ANS 3.2 "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants, 1994 Edition" in accordance with Nuclear Safety Act(Article 21, 4), Rule of Technical Standards for Nuclear Facilities(Article 67), Criteria of Quality Assurance for Nuclear Facilities, and be submitted with the application for the operating license. 736

The review and audit program consists of onsite review, independent review and audit in accordance with ANSI/ANS 3.2-1994

13.4.1 Onsite Review

The onsite review will be performed by the Plant Nuclear Safety Committee(PNSC) reviewing overall plant operation. The PNSC has the responsibility to review all plant administration, maintenance, 736
and operations as related to safety and environmental aspects. A detailed description of the PNSC is provided in the technical specifications.

The PNSC is composed of the plant manager, all senior plant technical supervisory personnel, and those department managers responsible for operation, maintenance, and technical activities of the unit. Collectively, they possess the type and degree of expertise required to properly review proposed changes to systems, procedures, and unplanned events that affect nuclear safety. The PNSC meets at least once every three month and maintains written 204
minutes of each meeting. The proposed changes considered by the PNSC involve routine operational matters which will not require a review by the KHNP 736
Nuclear Review Board (KNRB). However, the minutes of the meetings and any significant proposed changes or tests are submitted to the KNRB for approval.

13.4.2 Independent Review

The KNRB provides an independent review of plant operating activities and details on the KNRB operation are described in the technical 736
specifications. The KNRB is composed of experienced top-level KHNP officers

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and outside experts who have sufficient competence in a specialty related to plant operation and safety. This body will not have direct responsibility for routine plant operation but will exercise overall control and provide guidance for the safe reliable operation of the plant. 736

Any potential hazard associated with plant operations shall be identified and the nuclear safety of all proposed changes shall be assured by the KNRB. Proposed changes or tests that involve changes in the technical specifications or which pose an unreviewed safety question are not carried out until authorized by the KNRB. The board meets at least once per calendar quarter during the initial year of unit operation following full loading and at least once per six months thereafter. 736

"Delete" Details of the activities and duties of the KNRB are described in KNRB operating procedure 736

13.4.3 Audit

A comprehensive system of planned and documented audits is conducted to verify compliance with all aspects of the administrative controls and quality assurance program. The audits shall be performed by the quality assurance organization in accordance with ANSI/ANS 3.2-1994 and with a frequency commensurate with their safety significance and encompass the following: 736

a. Conformance of facility operation to applicable license conditions and technical specifications

b. Training and qualifications of the operating staff

c. Results of actions taken to correct deficient items that affect nuclear safety 736

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d. Quality assurance program

e. Facility emergency plan

f. Facility security plan

g. Fire protection program

h. Implementation of inspection on the fire protection equipment and plan by external experts

i. Radiological environmental monitoring plan

j. Offsite dose calculation manual

k. Process control program and implementing procedures for processing of radioactive wastes 736

l. Items required by radioactive effluent controls program

m. Any area considered appropriate by the KNRB

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The Quality Assurance Division has the responsibility of auditing any activity or documentation affecting the quality of a safety-related item. Audits are performed by quality assurance engineers or other qualified persons designated by the quality assurance manager. These audits comply with the requirement for audit provided by ASME NQA-1 and KEPIC QAP, and are performed at the plant or at the contractor, vendor, or consultant source locations as required. The details of the audits is described in a separately published QA Manual. Written reports of the audits are reviewed by the independent review committees and by appropriate members of management, 736

Appropriate and timely follow-up actions, "Delete" is taken to ensure the overall effectiveness of the review and audit program.

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13.5 PLANT PROCEDURES

All safety-related operations will be conducted using written and approved procedures. Procedures will be reviewed periodically and will be revised as necessary to ensure proper and safe operation of the plant. Operating personnel will be thoroughly trained to ensure familiarity with the appropriate procedures.

13.5.1 Administrative Procedures

13.5.1.1 Conformance with Regulatory Guide 1.33

Procedures will be prepared in accordance with ANSI N18.7, "Administrative Controls for Nuclear Power Plant". The applicable portions of Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)," will be used for guidance.

13.5.1.2 Preparation of Procedures

Cognizant plant department managers are responsible for initiating, preparing, and controlling plant procedures consistent with their responsibilities, and for ensuring that work is performed in accordance with the latest applicable documents. The procedures are reviewed and approved by the Plant Nuclear Safety Committee (PNSC). The plant manager promulgates all the procedures after the committee's approval.

13.5.1.3 Description of Administrative Procedures

a. Procedures for Shift Supervisors and Operators

These procedures are "Plant Operation Organization and Responsibility" and "Shift and Relief Turnover." They define the authority and

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responsibilities for reactor operators and senior reactor operators. These procedures also define the standards for operation and protection of the reactor and its related control equipment including the reactor control and protection system, rod control system, and nuclear instrumentation system.

b. Special Procedures

These procedures shall be issued when written instructions are required to change the scope of a project or to provide procedures for completion. These procedures are reviewed by the PNSC and are self-cancelling when the job is completed.

c. Equipment Control Procedures

Equipment control procedures are written to provide control over the status of plant equipment, of purchased material, and of nonconforming material. Such procedures will include the following:

o Change of instrument setpoint
o Work authorization
o Control of purchased material, equipment, and services
o Handling, storage, and shipment of materials
o Nonconforming materials, parts, components, or operations

d. Control of Maintenance and Modification Procedures

Maintenance of safety-related equipment will be accomplished in accordance with written procedures. Such procedures will include the following:

o Work stop order

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- o Maintenance work control
- o Gas cuting and welding control
- o Welding procedure verification control

Modification of equipment in safety-related systems is also accomplished in accordance with written procedures, such as "System Equipment Modification Control."

e. Master Surveillance Testing Schedule

The master surveillance testing schedule for safety-related systems will be established in accordance with the plant technical specification surveillance requirement (see ~~Chapter 16~~ ITS).

f. Log Book Usage and Control Procedures

These procedures describe the kinds of log books and the format of the log book entry, and will be covered in "Operating Log Books."

g. Temporary Procedures

These procedures are issued as required to provide detailed instructions for specific jobs that are of a specific duration and of a one-time-only nature. These procedures are reviewed and approved by the PNSC.

13.5.2 Operating and Maintenance Procedures

13.5.2.1 Operating Procedures

a. Purpose

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Describes the objective to be accomplished.

b. Reference

Lists the documents referenced in the procedures

c. Prerequisites

Identifies the activities (tests, inspections, calibrations, valve lineup conditions, etc.) that must be completed prior to operation of the system or plant.

d. Precautions

Lists the precautions to be observed or followed.

e. Limitations

Lists the limitations and parameters or conditions within which the plant or system must operate.

f. Procedures

Provides step-by-step procedures.

g. Appendices

Provides appendices when applicable.

Operating procedures are to be provided for the following categories:

a. General Plant Operation Procedures

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These procedures describe how to bring the plant from cold shutdown or hot standby condition to power operations, how to change load, and finally, how to bring the plant down to cold shutdown or hot standby conditions.

b. System Operating Procedures

These procedures describe the steps required to put the individual system into service, or take the system out of service. These procedures also instruct operators how to manipulate the system for several normal conditions as required.

c. Abnormal and Alarm Procedures

Abnormal and alarm procedures instruct operators how to respond to system abnormal conditions. The alarm procedures are classified according to their alarm window position index (panel, line, and row numbers). This allows operators to easily refer to the specific alarm procedure. As for alarm systems, they are designed to give visual (light) and audible (sound) alarms for each window. The visible alarms are classified into two categories: red trips and white alerts. Every visual alarm is initiated by a unique protective system and is accompanied by a high frequency buzz noise alarm to remind the operator to take action. When the alarm clears, the annunciator system acknowledges with a low frequency buzz.

d. Emergency Procedures

These procedures instruct the operators how to handle the plant in emergency situations such as the following:

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- o Reactor trip
- o Loss of offsite power
- o S/G tube rupture
- o Loss of feedwater
- o Loss of coolant
- o Station blackout

e. Temporary Procedures

These procedures are required by the PNSC to provide detailed instructions for specific tests of operations of safety-related systems.

13.5.2.2 Other Procedures

Other procedures are provided for the following categories:

a. Plant Radiation Protection Procedures

These procedures are designed to limit and control radiation exposures and the spread of contamination as well as to meet the requirements of USNRC 10 CFR 20 as modified by the NSSC and the ALARA (as low as 582
reasonably achievable) philosophy. These procedures shall be followed by all plant personnel.

Procedures will include the following:

- o Responsibilities
- o Radiation exposure limits
- o Access control
- o Protective clothing
- o Personnel monitoring

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- o Radiological surveys and records
- o Contaminated equipment control
- o Radioactive shipments
- o Radiation incidents - procedure and reporting
- o Radioactive material handling

b. Emergency Preparedness Procedures

These procedures are provided to implement the provisions of the Emergency Plan (see Section 13.3). They include organization, assignment of responsibilities, instructions to employees, procedures for emergencies, and the mobilization of offsite assistance when necessary. Procedures in this area are detailed in the Emergency Plan and shall be followed by all plant personnel.

c. Instrument Calibration and Test Procedures

These procedures provide detailed step-by-step methods for instrument calibration and test, test intervals, and their acceptance criteria. These procedures include the following:

- o Reactor protection instrument test and calibration
- o Area radiation monitoring system calibration
- o Process radiation monitoring system calibration
- o Nuclear instrumentation monitoring system calibration
- o Calibration of test instrumentation and devices

d. Chemical-Radiochemical Control Procedures

These procedures provide instructions on various chemical and radiochemical analysis and counting techniques. They also define the intervals for taking samples and apply to the work performed by

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chemical and radiation protection technicians. The following procedures are included in this area:

- o Chemical analysis
- o Calibration and operation of chemical instrumentation
- o Radiochemical analysis procedures
- o Calibration and operation of radiochemical instrumentation
- o Chemical and radiochemical solutions
- o Test forms
- o Waste quality limits
- o Chemical cleaning

e. Radioactive Waste Management Procedures

These procedures include the following:

- o Solid waste handling and storage
- o Radwaste release control
- o Decontamination
- o Counting room equipment control

f. Maintenance and Modification Procedures

These procedures provide detailed instructions for performing maintenance and modifications on safety-related systems or equipment.

g. Material Control Procedures

These procedures include the following:

- o Material storage control
- o Material receiving and distribution control

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- o Material purchasing specification control
- o Material purchasing control
- o material identification control

h. Plant Security Procedures

Plant security procedures provide for the implementation of the Security Plan (see Section 13.6)

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13.6 INDUSTRIAL SECURITY

This section presents KEPCO's plans for physical protection of YGN 3&4. The detailed security plan is presented in the physical security plan, which is in compliance with ROK security requirements. The physical security plan development uses 10 CFR 73.55 as guidance where 10 CFR 73.55 covers subjects not addressed by ROK security requirements. The physical security plan is withheld from public disclosure.

13.6.1 Preliminary Planning

The YGN 3&4 security plan complies with the intent of NRC Regulatory Guide 1.17, June 1973, "Protection of Nuclear Power Plants Against Industrial Sabotage," and its referenced standard ANSI N18.17-1973, "Industrial Security for Nuclear Power Plants."

13.6.1.1 Personnel Selection

A background investigation will be made for all plant personnel prior to assignment to a position allowing access without escort. Also, all plant employees will be examined by a licensed physician to evaluate personality and physical characteristics.

All plant personnel will be evaluated by their manager at least annually. Plant personnel will be under daily observation by their supervisor and any suspicious actions or tendencies will be immediately reported to the plant manager and manager of security.

Each member of the plant organization will be trained in plant security requirements and will be briefed periodically on security matters.

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13.6.1.2 Plant Design and Layout

The plant design and layout have been prepared considering security guidelines, including Regulatory Guide 1.17 and ANSI N18.17. Features unique to the YGN 3&4 site were considered in the plant design and placement of physical barriers.

The detailed security plan provides figures and/or drawings indicating the following:

a. Owner-controlled area
b. Parking lots
c. Roads that can be used for surveillance
d. Fenced protected area
e. Private property markers
f. Protected area isolation zone
g. Security access control points
h. Protected and vital area lighting
i. Intrusion monitoring, detection, and perimeter alarm system locations
j. Areas containing vital equipment
k. Alarm station location
l. Security force description

Design features and equipment arrangements are provided, consistent with other safety requirements, to reduce the opportunity for successful industrial sabotage of vital equipment. To the extent feasible, these features include measures to protest against undetected intentional acts that could impair equipment performance, such as automatic indication of inoperability.

13.6.1.3 Physical Barriers

Physical barriers for protected and vital area meets the intent of ANSI N18.17-1973.

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The protected area is enclosed by a perimeter fence (No. 11 American Wire Gauge or heavier) topped by at least three strands of barbed wire on an angled bracker (facing out). The fence is at least [redacted] in height. A clear isolation zone is maintained on both sides of the protected area fence.

All plant equipment designed as "vital equipment" is located in vital areas that is all within the protected area. All practical efforts are expended to prevent locating nonvital equipment in vital area. Vital areas are protected to limit unauthorized intrusion.

13.6.1.4 Access Points into Vital Areas

All doors and gates to vital areas are normally closed and have an alarm system indicating unauthorized intrusion. Detailed information on locking devices and intrusion alarms is included in the final physical security plan along with a description of the methods for personnel identification and entry authorization.

13.6.1.5 Alarm Systems

Detailed information for alarm systems has been included in the final physical security plan.

13.6.1.6 Protection of Security Systems

Security systems such as alarm stations, communication systems, and locks are protected against tampering. Protection is facilitated by use of two alarm stations. Detailed information is presented in the final physical security plan.

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13.6.1.7 Security Plan Development

The YGN 3&4 Nuclear Power Plant staff will develop the detailed security plan for YGN 3&4. Qualified consultants will be utilized as required. During the project phase, the KEPCO Nuclear Construction Department will provide review and input to the plant design and layout to facilitate security provisions.

13.6.2 Security Plan

The physical plant security plan describes security measures used to minimize the potential for industrial sabotage including access control, surveillance of vital equipment, and plans for responding to security threats.

13.6.2.1 Access Control

All personnel must show identification to the security guard on duty in the guard office prior to entering the site. Access to the site must be through the guard office, except for vehicle entry. KEPCO personnel must show their KEPCO photo identification card (only).

Visitors and drivers of vehicles must show their photo identification cards, issued by the government, or their KEPCO contractor's pass. Personnel without proper identification must have their identity verified prior to entering the site. Individuals without proper identification, who cannot be verified, are not allowed on site. These individuals may use the telephone in the guard office. All personnel must pass through the security card-key control system with an identification card in the guard office prior to entering and leaving the
92 site. All personnel and vehicles are subject to an inspection of briefcases, bags, tool boxes, packages, etc., prior to entry and exit.

Personnel authorized to be on the reactor site are identified by badges. Badges are marked to clearly show the areas to which access is permitted.

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Plant operators are instructed to challenge all unidentified persons. In addition, plant operators are instructed to challenge plant employees who are not in their normal work areas.

Administrative control of all activities within the restricted area is the responsibility of the plant superintendent. Access to the restricted area is limited by a security fence with perimeter lighting and a controlled entrance gate. Entry to the restricted area is through this gate.

Entry to operating areas such as the control room within the restricted area is limited to those persons authorized for entry by qualified supervisory personnel.

Access to critical plant areas is controlled by screening of visitors and use of identification badges.

The communication systems available to the security force for use in summoning assistance from the local police or military police include independent telephone circuits and a two-way radio connection. At the start of each shift, telephone circuits are tested by the security force and the radio is tested by operating personnel.

All alarm systems are functionally tested for operability and reliability at least once a week.

13.6.2.2 Control of Personnel by Categories

A list is maintained of all manufacturers' representatives, company employees, and others who require and are permitted access to plant property on a regular basis. These individuals are given appropriate badges as stated on the list. Other visitors are admitted onto plant property only after authorization from a member of the plant supervisory staff. Visitors are classified into four categories as follows:

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1. Sightseers and other occasional visitors

2. Salesmen and visitors requiring admittance to the plant office only

3. Manufacturers' service representatives, contractors, and their employees

4. Company employees other than regular plant staff.

Category 1 visitors are issued a badge that indicates that they must be escorted. They are provided with an escort.

Category 2 visitors are issued a badge identifying them as being authorized to visit the office only and are directed to the office after receiving authorization from a member of the plant staff. Should it be necessary for them to visit other areas of the plant, they are provided with an escort. Anyone entering the controlled area must wear a radiation monitoring device.

Category 3 visitors who are visiting the plant for the first time are directed to the office. On subsequent visits, they may be issued appropriate identification and monitoring devices at the gate. If they have not received radiation protection training, they are required to have an escort at all times. If they have satisfied the health physics engineer that they have the necessary training, and if approved by the plant superintendent or his representative, they are issued monitoring devices and identification badges and do not require an escort. On subsequent visits, they may be issued appropriate badges and monitoring devices at the gate or reception area.

Category 4 visitors identify themselves and, depending on their reason for visiting the plant and the extent of their radiation training, are treated in the same manner as Category 1 or 3 visitors.

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13.6.2.3 Access Control During Emergencies

Upon hearing of an emergency, the public security personnel and policemen on duty at the access portal lock all doors to ensure controlled entry and exit. Visitors who are onsite are escorted to the access portal. Plant employees report to predesignated stations from which they are dispatched as needed to combat the emergency.

13.6.2.4 Surveillance of Vital Equipment

The unit operator continuously monitors the status of plant systems and equipment by means of annunciators, indicating lights, indicators, and recorders. New equipment or material is inspected on delivery. Operating logs and computer printout data are periodically examined for changes in equipment performance. Most equipment is in continuous operation and any change is immediately detected by the operator. Standby and emergency equipment is periodically tested on a routine basis as required by the technical specifications. Assistant unit operators inspect equipment and spaces at least once each shift. In addition, the assistant shift supervisors, shift supervisors, and other supervisory personnel knowledgeable in plant conditions make frequent unscheduled inspection tours through the plant. The combination of these efforts provides reasonable assurance that unauthorized, physical changes in the status of components or equipment do not go undetected for long periods.

Key operating log sheets and selected recorder tracings are reviewed daily by the plant operations section. Abnormal changes observed are called to the attention of the plant superintendent and the appropriate supervisors for investigation and corrective action, if required. This operational audit serves to ensure early detection of physical changes that would have a significant bearing on plant performance.

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13.6.2.5 Potential Security Threats

A closed and continuing line of communication is maintained with the Chulla-Nam-Do Police. If an intruder does not immediately leave the area or attempts to breach the security fence, the Chulla-Nam-Do Police are immediately notified and their assistance is requested to remove the intruder. There are two separate lines of communication available for notification.

All actual intrusion attempts or attempted acts of sabotage are reported to the NSSC within 24 hours. 582

13.6.2.6 Administrative Procedures

All intrusion attempts are investigated, and the intruder, placed under continuous surveillance, is advised that he is in a restricted area and should leave immediately. If the intruder does not comply, the Chulla-Nam-Do Police are notified as stated in Subsection 13.6.2.5.

The following security-related records are maintained:

a. A visitors log

b. Results of all tests, inspections, and maintenance performed on physical barriers and communication links

c. A list of all intrusion attempts and action taken

The security force's performance of their duties is audited regularly by the security supervisor. Results of the audits are kept on file in the administrative office.

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13.7 TECHNICAL SPECIFICATION CONVERSION

This section contains requirements that were removed from the Technical Specifications during the conversion from the standard format Technical Specifications to the Improved Standard Technical Specifications. This conversion endorsed in NEI 96-06 and Guidelines for conversion of Technical Specification. This conversion was approved by the NSSC on (later) and placed in revision (later) of the YGN 3 and 4 FSAR. 582

13.7.1 Improved Standard Technical Specification

US-NRC developed the NUREG-1432 Revision 1, which was published in April 1995. NUREG-1432 contains the improved Standard Technical Specification(STS) for Combustion Engineering plants. KHNP has translated the NUREG-1432, revision 1, into korean language, and submitted the Topical Report number "한전기술-0005" to the NSSC. The Topical Report named Improved Standard Technical Specification(ISTS) was approved in November 1999 by NSSC. 582

The ISTS is divided into three parts, Operation, Radiation and Environmental control, and Administrative control of nuclear power facility. Limiting Conditions for Operation(LCO) of the ISTS are the lowest functional capability or performance levels of equipment required for safe operation of the facility. Each LCO of the ISTS has Bases, respectively.

Limiting Conditions for Operation of ISTS must be established for each item meeting one or more of the following criteria:

a. Criterion 1 : Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

b. Criterion 2 : A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

c. Criterion 3 : A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either

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assumes the failure of or presents a challenge to the integrity of a fission product barrier.

d. Criterion 4 : A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

13.7.2 Conversion to Improved Technical Specification

KHNP has commenced the conversion of Technical Specification from standard format to Improved Standard format in January 2000. This conversion endorsed in NEI 96-06 and Korean version Improved Technical Specifications Conversion Guidance which was developed by KHNP Technical Specifications Task Force.

The project to convert to ITS is divided into two phases, the Submittal Phase and Implementation Preparation Phase. The Submittal Phase extends from the start of the project through the submittal to the NSSC of a license amendment 582
request for conversion of the YGN 3&4 CTS to the ITS. The Submittal Phase includes the development of YGN 3&4 ITS. The Implementation Preparation Phase begins with the submittal of the license amendment request to the NSSC and 582
ends on the effective date of the ITS.

The objective of the Submittal Phase is to prepare a license amendment request for submittal to the NSSC. This license amendment request addressed 582
the overall conversion of the YGN 3&4 CTS to the ITS. Various documents were created in support of this objective. The documents to be included in the license amendment request are:

a. Summary of Technical Specification Conversion

b. ITS and Bases

c. CTS Discussion of change(DOC)

d. Deviation from the Applicable ISTS - Justification Documents(JD)

These documents are ordered in the application on a chapter or section basis to allow the application to be easily divided by the KINS for review. All

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documents discussing the conversion of a particular chapter or section should be grouped together.

Removed LCOs or Surveillance Requirements from the CTS should be relocated to FSAR, Programs and Manuals, or Procedures. Most of the removed LCOs were relocated to FSAR Table 13.7, and controlled with license documents.

13.7.3 Control of Relocated Items

Table 13.7 provides those limitations upon plant operations which are part of the licensing basis for the station but do not meet the criteria for continued inclusion in the Technical Specifications.

It also provides information which supplement the Technical Specifications such as specific plant setpoints for Technical Specification equipment.

System requirements and Test Requirements are the implemented the same as Technical Specifications. However, System requirements and Test Requirements are treated as plant procedure and are not part of the Technical Specifications. Therefore the following exceptions apply: 322

a. Violations of the Test Requirements in a System Requirements are not reportable as conditions prohibited by, or deviations from, the Technical Specifications, unless specifically required by FASR table 13.7.

b. Power reduction or plant shutdowns are not required to comply with the actions of the Test Requirements.

c. Violation of System Requirements or Test Requirements shall be treated the same as plant procedure violations.

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YGN 3&4 FSAR

TABLE 13.7-1(Sh 1 of 33)

REACTIVITY CONTOL SYSTEMS - BORATION SYSTEMS

SYSTEM REQUIREMENTS	TEST REQUIREMENTS	FREQUENCY	APPLICABILITY
1. FLOW PATHS - SHUTDOWN As a minimum, one of the following boron injection flow paths shall be OPERABLE: a. If only the spent fuel pool is OPERABLE, a flow path from the spent fuel pool via a gravity feed connection and a charging pump to the reactor coolant system. b. If only the refueling water tank is OPERABLE, a flow path from the refueling water tank via either a charging pump, a high-pressure safety injection pump, or a low-pressure safety injection pump to the reactor coolant system	1. At least one of the required flow paths shall be demonstrated OPERABLE by verifying that each valve(manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.	31 days	MODES 5 and 6

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TABLE 13.7-1(Sh 2 of 33)

REACTIVITY CONTOL SYSTEMS - BORATION SYSTEMS

SYSTEM REQUIREMENTS	TEST REQUIREMENTS	FREQUENCY	APPLICABILITY
2. FLOW PATHS - OPERATING At least two of the following three boron injection flow paths shall be OPERABLE:	1. At least two of the required flow paths shall be demonstrated be OPERABLE:		MODES 1, 2, 3, and 4.
a. A gravity-feed flow path from either the refueling water tank or the spent fuel pool through the RWT gravity feed isolation valve and a charging pump to the reactor coolant system.	a. By verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.	31 days	
b. A gravity-feed flow path from the refueling water tank through the RWT gravity-feed/safety injection system isolation valve and a charging pump to the reactor coolant system.	b. When the reactor coolant system is at normal operating pressure by verifying that the flow path delivers at least [redacted] for one charging pump and [redacted] for two charging pumps to the reactor coolant system.	18 months	
c. A gravity-feed flow either the refueling water tank or the spent fuel pool through the boric acid filter bypass valve, utilizing gravity feed and a charging pump to the reactor coolant system.			

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REACTIVITY CONTROL SYSTEMS - CHARGING PUMPS

SYSTEM REQUIREMENTS	TEST REQUIREMENTS	FREQUENCY	APPLICABILITY
1. SHUTDOWN At least one charging pump[1]) or one high-pressure safety injection pump or one low-pressure safety injection pump in the boron injection flow path required OPERABLE and capable of being powered from an OPERABLE emergency power source	No additional surveillance requirements other than those required by In-service Test Program	Refer to IST Program	MODES 5 and 6
2. OPERATING At least two charging pumps shall be OPERABLE.	No additional surveillance requirements other than those required by In-service Test Program	Refer to IST Program	MODES 1,2,3, and 4.

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REACTIVITY CONTOL SYSTEMS - BORATED WATER SOURCES

SYSTEM REQUIREMENTS	TEST REQUIREMENTS	FREQUENCY	APPLICABILITY
1. SHUTDOWN As a minimum, one of the following borated water sources shall be OPERABLE:	The required borated water sources shall be demonstrated OPERABLE:		MODES 5, 6
a. The spent fuel pool with:	a. Verifying the follows:	7 days	
1. A minimum useful contained borated water volume of [redacted] 2. A boron concentration of between [redacted] and 3. [redacted] emperature between [redacted]	1. Boron concentration of the water, and 2. The contained borated water volume of the refueling water tank or the spent fuel pool.		
	b. Verifying the refueling water tank temperature when it is the source of borated wate [redacted] am [redacted] mperature is outside the [redacted] to [redacted] range.	24 hours	
b. The refueling water tank with:			
1. A minimum useful contained borated water volume of [redacted] 2. A boron concentration of between [redacted] and 3. [redacted] emperature between [redacted]	c. Verifying the spent fuel pool temperature when it is the source of borated water and irradiated fuel is present in the pool.	24 hours	

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REACTIVITY CONTOL SYSTEMS - BORATED WATER SOURCES

SYSTEM REQUIREMENTS	TEST REQUIREMENTS	FREQUENCY	APPLICABILITY
2. OPERATING Each of the following borated water sources shall be OPERABLE:	Each of the required borated water sources shall be demonstrated OPERABLE:		MODES 1,2,3,4
a. The spent fuel pool with: 1. A minimum useful contained borated water volume as specified in Figure 13.7-2, 2. A boron concentration of between [redacted] and 3. A s[redacted]perature between [redacted]	a. Verifying the follows: 1. Boron concentration of the water, and 2. The contained borated water volume of the refueling water tank and the spent fuel pool.	7 days	
	b. Verifying the refueling water tank temperature [redacted] amb[redacted]perature is outside the [redacted] range.	24 hours	
b. The refueling water tank with: 1. A minimum useful contained borated water volume as specified in Figure 13.7-2, 2. A boron concentration of between [redacted] and 3. A s[redacted]perature between [redacted]	c. Verifying the spent fuel pool temperature when irradiated fuel is present in the pool.	24 hours	

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INSTRUMENTATION - INCORE DETECTORS

SYSTEM REQUIREMENTS	TEST REQUIREMENTS	FREQUENCY	APPLICABILITY
The incore detector system shall be OPERABLE with:	The incore detector system shall be demonstrated OPERABLE:		When the incore detector system is used for monitoring: a. AZIMUTHAL POWER TILT, b. Radial peaking factors, c. Local power density, d. DNB margin.
a. At least 75% (34 of 45) of all incore detector locations OPERABLE, and 75% (170 of 225) of all detectors OPERABLE, with at least one detector OPERABLE in each quadrant at each of five levels ; and	a. By performance of a CHANNEL CHECK within 24 hours prior to its use if the system has just been returned to OPERABLE status of if 7 days or more have elapsed since last use and at least once per 7 days thereafter when required for monitoring the AZIMUTHAL POWER TILT, radial peaking factors, local power density, or DNB margin:	7 days	
b. A minimum of six tilt estimates, with at least one at each of three levels.	b. By performance of a CHANNEL CALIBRATION operation which exempts the neutron detectors but includes all electronic components. The fixed incore neutron detectors shall be calibrated prior to installation in the reactor core.	18 months	
An OPERABLE incore detector location shall consist of a fuel assembly containing a fixed incore detector assembly with a minimum of three OPERABLE fixed incore detectors. (continued)			

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INSTRUMENTATION - INCORE DETECTORS

SYSTEM REQUIREMENTS	TEST REQUIREMENTS	FREQUENCY	APPLICABILITY
However, when incore detector locations are OPERABLE within 75% and 60% (from 33 to 27), and detectors OPERABLE within 75% and 60% (from 169 to 135), and other system requirements are met, the incore detector system can be used for monitoring the AZIMUTHAL POWER TILT, local power density, and DNB margin with implementing the increased uncertainty due to the reduction of operable incore detectors into the COLSS. In this case, the uncertainty increases due to the reduction of operable incore detectors shall be 2% for local power density and 1% for DNB margin, and this operation is limited to 7 consecutive days.			

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INSTRUMENTATION - SEISMIC INSTRUMENTATION

SYSTEM REQUIREMENTS	TEST REQUIREMENTS	FREQUENCY	APPLICABILITY
The seismic monitoring instrumentation shown in Table 13.7-2 shall be OPERABLE.	1. Each of the seismic monitoring instruments shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 13.7-3.	Refer to the Table 13.7-3	At all times
	2. Each of the seismic monitoring instruments actuated during a seismic event shall have a CHANNEL CALIBRATION performed within 5 days. Data shall be retrieved from actuated instruments and analyzed to determine the magnitude of the vibratory ground motion.	within 5 days after actuated	At all times

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INSTRUMENTATION - LOOSE-PART DETECTION INSTRUMENTATION

SYSTEM REQUIREMENTS	TEST REQUIREMENTS	FREQUENCY	APPLICABILITY
The loose-part detection system shall be OPERABLE with all sensors specified in Table 13.7-4	Each channel of the loose-part detection system shall be demonstrated OPERABLE by performance of:		MODES 1 and 2.
	a. a CHANNEL CHECK,	24 hours	
	b. a CHANNEL FUNCTIONAL TEST, and	31 days	
	C. a CHANNEL CALIBRATION.	18 months	

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INSTRUMENTATION - TURBINE OVERSPEED PROTECTION

SYSTEM REQUIREMENTS	TEST REQUIREMENTS	FREQUENCY	APPLICABILITY
At least one turbine overspeed protection system shall be OPERABLE.	The required turbine overspeed protection system shall be demonstrated OPERABLE:		MODE 1, 2, and 3
	a. By cycling, each of the following valves from the running position and observing valve closure: 1. Four high-pressure turbine stop valves. 2. Four high-pressure turbine control valves. 3. Six low-pressure turbine combined intermediate valves.	92 days	
	b. Each of the 14 extraction steam non-return check valves shall be cycled from the closed position	Within 7 days prior to entering MODE 3 from MODE 4,	
	c. By direct observation, verify freedom of movement of the 14 extraction steam non-return check valve weight arms.	31 days	

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INSTRUMENTATION - TURBINE OVERSPEED PROTECTION

SYSTEM REQUIREMENTS	TEST REQUIREMENTS	FREQUENCY	APPLICABILITY
(continued)	(continued)		
	d. By performance of a CHANNEL CALIBRATION on the turbine overspeed protection instrumentation.	18 months	
	e. At approximately 40-month intervals* by disassembling at least one of each of the above valves and performing a visual and surface inspection of valve seats, disks, and stems and verifying no unacceptable flaws or corrosion.	40-month	
	* During refueling or maintenance shutdowns coinciding with the inservice inspection schedule required by Section XI of the ASME Code for reactor components.		

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REACTOR COOLANT SYSTEM - AUXILIARY SPRAY

SYSTEM REQUIREMENTS	TEST REQUIREMENTS	FREQUENCY	APPLICABILITY
Both auxiliary spray valves shall be OPERABLE.	The auxiliary spray valves shall be verified to have power available to each valve.	24 hours	MODES 1, 2, 3, and 4
	The auxiliary spray valve shall be cycled.	18 months	

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REACTOR COOLANT SYSTEM - CHEMISTRY

SYSTEM REQUIREMENTS	TEST REQUIREMENTS	FREQUENCY	APPLICABILITY
The reactor coolant system chemistry shall be maintained within the limits specified in Table 13.7-5.	The reactor coolant system chemistry shall be determined to be within the limits by analysis of those parameters at the frequencies specified in Table 13.7-6.	Refer to the Table 13.7-6	At all times

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REACTOR COOLANT SYSTEM - PRESSURIZER HEATUP/COOLDOWN LIMITS

SYSTEM REQUIREMENTS	TEST REQUIREMENTS	FREQUENCY	APPLICABILITY
The pressurizer temperature shall be limited to: a. A maximum heatup rate of per hour, and b. A maximum cooldown rate of per hour	The pressurizer temperatures shall be determined to be within the limits during system heatup or cooldown.	30 minutes	At all times

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REACTOR COOLANT SYSTEM - SAFETY DEPRESSURIZATION SYSTEM (SDS)

SYSTEM REQUIREMENTS	TEST REQUIREMENTS	FREQUENCY	APPLICABILITY
Each SDS flow path shall be OPERABLE comprised of : a. VRC-101 closed and VRC-103 closed and power removed. b. VRC-102 closed and VRC-104 closed and power removed.	Each SDS flow path shall be demonstrated OPERABLE by: a. Verifying the follows: 1. Verifying that each SDS flow path valve is closed with correct valve position indication in the control room. 2. Verifying that power to each SDS flow path valve VRC-103 and VRC-104 is removed. b. Verifying the follows: 1. Cycling each SDS flow path valve through at least one complete cycle from the control room.	 7 days 18 months	MODES 1, 2, 3, and 4

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EMERGENCY CORE COOLING SYSTEMS - ECCS Subsystem

SYSTEM REQUIREMENTS	TEST REQUIREMENTS	FREQUENCY	APPLICABILITY
Each ECCS subsystem shall be demonstrated OPERABLE:	By performing a flow balance test, during shutdown, following completion of modifications to the ECCS subsystems that alter the subsystem flow characteristics and verifying the following flow rates: HPSI System - Single Pump The sum of the injection line flow rates, excluding the highest flow rate, is greater than or equal to [redacted] LPSI System - Single Pump 1. Injection Loop 1, total flow equal to [redacted] 2. Injection Legs 1A and 1B when tested individually, with the other leg isolated, shall be within [redacted] of each other.	following completion of modifications to the ECCS subsystems	MODES 1, 2, 3

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EMERGENCY CORE COOLING SYSTEMS - ECCS Subsystem

SYSTEM REQUIREMENTS	TEST REQUIREMENTS	FREQUENCY	APPLICABILITY
(continued)	LPSI System - Single Pump (continued) 3. Injection Loop 2, total flow equal to [redacted] 4. Injection Legs 2A and 2B when tested individually, with the other leg isolated, shall be within [redacted] of each other. Simultaneous Hot-Leg and Cold-Leg Injection - Single Pump 1. Hot Leg, flow equal to [redacted] 2. Cold Leg, flow equal to [redacted]	(continued)	(continued)

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PLANT SYSTEMS - STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

SYSTEM REQUIREMENTS	TEST REQUIREMENTS	FREQUENCY	APPLICABILITY
The temperature of the secondary coolant in the steam generators shall be greater than when the pressure of the secondary coolant in the steam generator is greater than	The pressure in the secondary side of the steam generators shall be determined to be less than when the temperature of the secondary coolant is less than	12 hours	At all times

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PLANT SYSTEMS - SNUBBERS

SYSTEM REQUIREMENTS	TEST REQUIREMENTS	FREQUENCY	APPLICABILITY
All hydraulic and mechanical snubbers shall be OPERABLE. The only snubbers excluded from this requirement are those installed on non-safety-related systems and then only if their failure or failure of the system on since they are installed would have no adverse effect on any safety-related system.	Each snubber shall be demonstrated OPERABLE by performance of the inservice inspection program.	Refer to the Table 13.7-7	MODES 1, 2, 3, and 4. MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES.

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PLANT SYSTEMS - SEALED SOURCE CONTAMINATION

SYSTEM REQUIREMENTS	TEST REQUIREMENTS	FREQUENCY	APPLICABILITY
Each sealed source containing radioactive material either in excess of [redacted] of beta-and/or gamma-emitting material or [redacted] of alpha-emitting material shall be free of greater than or equal to [redacted] of removable contamination.	1. Each sealed source shall be tested for leakage and/or contamination by: a. The licensee, or b. Other persons specifically authorized by the NSSC. 2. The test method shall have a detection sensitivity of at least [redacted] per test sample. 3. A report shall be prepared and submitted to the NSSC on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to [redacted] of removable contamination.	Each category of sealed sources (excluding startup sources and fission detectors previously subjected to core flux) shall be tested at the frequencies described below. a. Sources in use - At least once per 6 months for all sealed sources containing radioactive material: 1. With a half-life greater than 30 days (excluding tritium (H-3), and 2. In any form other than gas.	At all times

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PLANT SYSTEMS - SEALED SOURCE CONTAMINATION (continued)

SYSTEM REQUIREMENTS	TEST REQUIREMENTS	FREQUENCY	APPLICABILITY
(continued)	(continued)	b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous 6 months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use. c. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source or detector.	

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PLANT SYSTEMS - SHUTDOWN COOLING SYSTEM

SYSTEM REQUIREMENTS	TEST REQUIREMENTS	FREQUENCY	APPLICABILITY
Two independent shutdown cooling subsystems shall be OPERABLE, with each subsystem comprised of:	Each shutdown cooling subsystem shall be demonstrated OPERABLE:		MODES 1, 2 and 3
a. One OPERABLE low pressure safety injection pump, and	a. Verifying the follows:	7 days	
	1. Verifying that each SCS suction isolation valve is closed with correct valve position indicated in the control room.		
b. An independent OPERABLE flow path capable of taking suction from the RCS hot leg and discharging coolant through the shutdown cooling heat exchanger and back to the RCS through the cold leg injection lines. (SI-651, 655 and 656 shall be closed and SI-653 and SI-654 shall be close with power removal at the local motor starters of equipment numbers 3/4-441-ESI-012C and-012D)	2. Verifying that power to SI-653 and SI-654 is removed.		
	b. During shutdown, by establishing shutdown cooling flow from the RCS hot legs, through the shutdown cooling heat exchangers, and returning to the RCS cold legs.	18 months	

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PLANT SYSTEMS - SHUTDOWN COOLING SYSTEM

SYSTEM REQUIREMENTS	TEST REQUIREMENTS	FREQUENCY	APPLICABILITY
(continued)	(continued) c. During shutdown, by testing the automatic and interlock action of the shutdown cooling system connections from the RCS. The shutdown cooling system suction valves shall not open when RCS pressure is greater than [redacted] The shutdown cooling system suction valves located outside containment shall close automatically when RCS pressure is treater than [redacted] The shutdown cooling system suction valve located inside containment shall close automatically when RCS pressure is greater than [redacted]	18 months	

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ELECTRICAL POWER SYSTEM- ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

SYSTEM REQUIREMENTS	TEST REQUIREMENTS	FREQUENCY	APPLICABILITY
Primary and backup containment penetration conductor overcurrent protective devices associated with each containment electrical penetration circuit shall be OPERABLE.	1. All containment penetration conductor overcurrent protective devices shall be demonstrated OPERABLE: a. Verifying the follows: 1. By verifying that the medium-voltage (13.8 kV) circuit breakers are OPERABLE by selecting, on a rotating basis, at least 10% of the circuit breakers, and performing the following: a) A CHANNEL CALIBRATION of the associated protection relays, and b) An integrated system functional test which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breakers and control circuits function as designed. c) For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.	18 months	MODES 1, 2, 3, and 4

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ELECTRICAL POWER SYSTEM- ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

SYSTEM REQUIREMENTS	TEST REQUIREMENTS	FREQUENCY	APPLICABILITY
(continued)	2. By selecting and functionally testing a representative sample of at least 10% of each type of lower voltage circuit breakers. Circuit breakers shall be selected on a rotating basis for functional testing, Testing of these circuit breakers shall consist of injecting a current with a valve equal to 300% to 600% of the sensor rating for the long-time delay trip element and 250% of the sensor rating for the short-time delay or instantaneous trip element, and verifying that the circuit breaker operates within the time delay band width or instantaneous time for that current specified by the manufacturer. Pickup values may vary within ±10%. Molded-case circuit-breaker testing shall also follow this procedure except that generally to more than two trip elements, time delay and instantaneous, will be involved. The instantaneous element on molded-case breakers shall be tested by injecting 150% of instantaneous trip current for a frame size of 400 amperes or less verifying that the circuit breaker trips instantaneously with no apparent time delay.		

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ELECTRICAL POWER SYSTEM- ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

SYSTEM REQUIREMENTS	TEST REQUIREMENTS	FREQUENCY	APPLICABILITY
(continued)	Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested. 3. By selecting and functionally testing a represen-tative sample of each type of fuse on a rotating basis. Each representative sample of fuses shall include at least 10 percent of all fuses of that type. The functional test shall consist of a nondestructive resistance measurement test which demonstrates that the fuse meets its manufacturer's design criteria. Fuses found inoperable during these func-		

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ELECTRICAL POWER SYSTEM - ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

SYSTEM REQUIREMENTS	TEST REQUIREMENTS	FREQUENCY	APPLICABILITY
(continued)	tional tests shall be replaced with OPERABLE fuses prior to resuming operation. For each fuse found inoperable during these functional tests, an additional representative sample of at least 10 percent of all fuses of that type shall be functionally tested until no more failures are found or all fuses of that type have been functionally tested.		
	b. At least once per 60 months or less if according to manufacturer's recommendations by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.	60 months or less	

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ELECTRICAL POWER SYSTEM - MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION

SYSTEM REQUIREMENTS	TEST REQUIREMENTS	FREQUENCY	APPLICABILITY
The thermal overload protection and bypass devices, integral with the motor starter of each valve used in safety-related systems (including containment isolation valves in non-safety-related systems) shall be OPERABLE.	The above-required thermal overload protection shall be demonstrated OPERABLE: a. By the performance of a CHANNEL FUNCTIONAL TEST of a representative sample of at least 25% of all thermal overload devices which are calibrated at least once per 6 years.	18 months	Whenever the motor-operated valve is required to be OPERABLE.

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REFUELING OPERATIONS - DECAY TIME

SYSTEM REQUIREMENTS	TEST REQUIREMENTS	FREQUENCY	APPLICABILITY
The reactor shall be subcritical for at least 100 hours.	The reactor shall be determined to have been subcritical for at least 100 hours by verification of the date and time of subcriticality.	Prior to movement of irradiated fuel in the reactor pressure vessel	During movement of irradiated fuel in the reactor pressure vessel.

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TABLE 13.7-1(Sh 29 of 33)

REFUELING OPERATIONS - COMMUNICATIONS

SYSTEM REQUIREMENTS	TEST REQUIREMENTS	FREQUENCY	APPLICABILITY
Direct communications shall be maintained between the control room personnel and personnel at the refueling station.	Direct communications between the control room and the refueling station shall be demonstrated during CORE ALTERATIONS.	Within 1 hour prior to the start of and at least once per 12 hours	During CORE ALTERATIONS

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TABLE 13.7-1(Sh 30 of 33)

REFUELING OPERATIONS - REFUELING MACHINE

SYSTEM REQUIREMENTS	TEST REQUIREMENTS	FREQUENCY	APPLICABILITY
The refueling machine shall be used for movement of fuel assemblies and shall be OPERABLE over the entire hoisting range with a minimum capacity of [redacted] dry weight and a fuel only overload cutoff limit of less than or equal to [redacted] plus the wet weight of the grapple.	The refueling machine used for movement of fuel assemblies shall be demonstrated OPERABLE by lifting a load test of at least [redacted] dry weight and demonstrating an automatic load cutoff when the refueling machine load exceeds [redacted] plus the wet weight of the grapple.	Within 72 hours prior to the start of such operations	During movement of fuel assemblies within the refueling cavity

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TABLE 13.7-1(Sh 31 of 33)

REFUELING OPERATIONS - CRANE TRAVEL (FUEL BUILDING)

SYSTEM REQUIREMENTS	TEST REQUIREMENTS	FREQUENCY	APPLICABILITY
1. The fuel building crane shall be physically restricted from traveling over the spent fuel pool.	1. The fuel building crane interlocks and physical stops which prevent crane travel over spent fuel assemblies shall be demonstated OPERABLE.	Within 7 days prior to crane use and at least once per 7 days thereafter during crane operation	With fuel assemblies and water in the storage pool.
2. Loads carried by the spent fuel handling machine or the monorail crane over the spent fuel pool and the heights at which they may be carried over racks containing fuel shall be limited in such a way as to preclude impact energies over [redacted] [redacted] if the loads are dropped.	2. For the monorail crane, the potential impact energy due dropping the crane's load shall be determined to be less than or equal to [redacted] Impact energy is equal to the weight of the lifted object times the maximum height above the top of the spent fuel racks. The maximum lift height of the following gate should be restricted no more then 4 ft above top of the spent fuel racks. a. Refueling Canal Gate b. Cask Loading Pit Gate	Prior to moving each load over racks containing fuel	

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TABLE 13.7-1(Sh 32 of 33)

REACTOR PROTECTIVE INSTRUMENTATION - DIVERSE PROTECTION SYSTEM (DPS)

SYSTEM REQUIREMENTS	TEST REQUIREMENTS	FREQUENCY	APPLICABILITY
As a minimum, the diverse protection system (DPS) channels and bypasses of Table 13.7-9 shall be OPERABLE with RESPONSE TIMES as shown in Table 13.7-10	Each diverse protection system (DPS) channel shall be demonstrated OPERABLE by the performance of :		MODES 1, 2
	a. a CHANNEL CHECK	12 hours	
	b. The following CHANNEL FUNCTIONAL TEST shall be done with the specified frequency for the PZ[redacted]ressure Channel [Allowable Value of ≤[redacted] [redacted] and for the S/G Low Level Channel (Allowable Value of ≥[redacted][1] WR)		
	1. The CHANNEL FUNCTIONAL TEST shall include the entire channel except the Control Element Drive Mechanism Control System output contactors which remove motive power to the Reactor Trip Switchgear System.	92 days	

1) Percent of the distance between steam generator upper- and lower-level wide-range instrument nozzles.

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TABLE 13.7-1(Sh 33 of 33)

REACTOR PROTECTIVE INSTRUMENTATION - DIVERSE PROTECTION SYSTEM (DPS) (continued)

SYSTEM REQUIREMENTS	TEST REQUIREMENTS	FREQUENCY	APPLICABILITY
(continued)	2. The CHANNEL FUNCTIONAL TEST shall include the entire channel, including the Control Element Drive Mechanism Control System output contactors which remove motive power to the Reactor Trip Switchgear System.	18 months	(continued)
	c. a CHANNEL CALIBRATION	18 months	
	d. a RESPONSE TIMES	18 months on a STAGGERED TEST BASIS	

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TABLE 13.7-2 (Sh. 1 of 2)

SEISMIC MONITORING INSTRUMENTATION

INSTRUMENTS AND SENSOR LOCATIONS	MINIMUM INSTRUMENT OPERABLE
1. Triaxial Accelerometers	
a. Tendon Gallery Floor, 58′level	1
b. Containment Shell at Operating Floor Elevation, 142′-10″ level	1
c. Primary Auxiliary Bldg Base, 48′level	1
d. Containment Shell, 104′-9″ level	1
e. Component Cooling Water Surge Tank 01TA Support, 165′ level	1
f. Free Field, 100′ level	1
2. Triaxial Peak Accelerograph	
DELETE	
3. Solid State Accelerograph Recorders	
a. Main Control Room Area, 144′level(Tendon Gallery Floor)	1
b. Main Control Room Area, 144′level(Containment Shell at Operating Floor Elevation)	1
c. Main Control Room Area, 144′level(Primary Auxiliary Bldg Base)	1
d. Main Control Room Area, 144′level(Containment Shell)	1
e. Main Control Room Area, 144′level(Component Cooling Water Surge Tank 01TA Support)	1
f. Main Control Room Area, 144′level(Free Field)	1
4. Solid State Playback Unit and Analyzer	
a. Main Control Room Area, 144′level	1

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TABLE 13.7-2 (Sh. 2 of 2)

SEISMIC MONITORING INSTRUMENTATION

INSTRUMENTS AND SENSOR LOCATIONS	MINIMUM INSTRUMENT OPERABLE
5. Seismic Switches	

DELETE

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TABLE 13.7-3 (Sh. 1 of 2)

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS AND SENSOR LOCATIONS	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. Triaxial Accelerometers			
a. Tendon Gallery Floor, 58′level	NA	R	SA
b. Containment Shell at Operating Floor Elevation, 142′-10″level	NA	R	SA
c. Primary Auxiliary Bldg Base, 48′ level	NA	R	SA
d. Containment Shell, 104′-9″ level	NA	R	SA
e. Component Cooling Water Surge Tank 01TA Support, 165′level	NA	R	SA
f. Free Field, 100′ level	NA	R	SA

Note)Performance test should be conducted once every five years in accordance with KINS/RG-N04.06 802

2. Triaxial Peak Accelerograph

DELETE

3. Solid State Accelerograph Recorders			
a. Main Control Room Area, 144′level	M	R	SA
b. Main Control Room Area, 144′level	M	R	SA
c. Main Control Room Area, 144′level	M	R	SA
d. Main Control Room Area, 144′level	M	R	SA
e. Main Control Room Area, 144′level	M	R	SA
F. Main Control Room Area, 144′level	M	R	SA

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TABLE 13.7-3 (Sh. 2 of 2)

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS AND SENSOR LOCATIONS	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
4. Solid State Playback Unit and Analyzer			
a. Main Control Room Area, 144′level	NA	R	SA
5. Seismic Switches			

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DELETE

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TABLE 13.7-4(Sh. 1 of 1)
LOOSE PARTS SENSOR LOCATIONS

Channel	Component	Location
V-101	Reactor Vessel	Lower head outside ICI nozzle area, 0' *
V-102	Reactor Vessel	Lower head outside ICI nozzle area, 180' *
V-103	Reactor Vessel	Upper lead outside CEDM nozzle area, 15' *
V-104	Reactor Vessel	Upper lead outside CEDM nozzle area, 195' *
V-105	Steam Generator 1	Secondary side at a place between two manways 0' **
V-106	Steam Generator 1	Secondary side at a place between above two economizer feedwater nozzles, 0' **
V-107	Steam Generator 1	Primary-side lower head at hot leg nozzle 180' **
V-108	Steam Generator 1	Primary-side lower head at manway between cold leg nozzle, 0' **
V-109	Steam Generator 2	Secondary side at a place between two manways, 0' **
V-110	Steam Generator 2	Secondary side at a place above two economizer feedwater nozzles, 0' **
V-111	Steam Generator 2	Primary-side lower head at hot leg nozzle 180' **
V-112	Steam Generator 2	Primary-side lower head at manway between cold leg nozzle, 0' **

* Reactor Vessel Hot Leg 1 Area is 0" Reference

** Steam Generator Cold Side Manway Area is 0" Reference Total LPMS Channels = 12

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TABLE 13.7-5(Sh. 1 of 1)

REACTOR COOLANT SYSTEM CHEMISTRY

PARAMETER	STEADY-STATE LIMIT	TRANSIENT LIMIT
Dissolved Oxygen*	≤ 0.10 ppm	≤ 1.00 ppm
Chloride	≤ 0.15 ppm	≤ 1.50 ppm
Fluoride	≤ 0.15 ppm	≤ 1.50 ppm

* Limit not applicable with T_{cold} less than or equal to

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TABLE 13.7-6(Sh. 1 of 1)

CHEMISTRY LIMITS SURVEILLANCE REQUIREMENTS

PARAMETER	SAMPLE AND ANALYSIS FREQUENCY
Dissolved Oxygen*	At least once per 72 hours
Chloride	At least once per 72 hours
Fluoride	At least once per 72 hours

* Not required with T_{cold} less than or equal to

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TABLE 13.7-7 (Sh. 1 of 6)
SNUBBER TEST REQUIREMENTS

Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and FSAR Table 13.7-1.

a. Snubber Types

As used in this specification, type of snubber shall mean snubbers of the same design and manufacturer, irrespective of capacity.

b. Visual Inspections

Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these categories (inaccessible and accessible) may be inspected independently according to the schedule determined by Table 13.7-8. The visual inspection interval for each type of snubber shall be determined based upon the criteria provided in Table 13.7-8 and the first inspection interval determined using this criteria shall be based upon the previous inspection interval as established by the requirements in effect before this revision.

c. Visual Inspection Acceptance Criteria

Visual inspections shall verify that: (1) there are no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are functional, and (3) fasteners for attachment of the snubber to the component and to the snubber anchorage are functional. Snubber to the component and to the snubber anchorage are functional. Snubbers which appear inoperable as a result of visual inspections shall be classified as unacceptable and may be reclassified acceptable for the purpose of establishing the next visual inspection interval, provided that: (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers irrespective of type on that system that may be generically susceptible; and (2) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per Table 13.7-7 "f". When a fluid port of a hydraulic

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TABLE 13.7-7 (Sh. 2 of 6)
SNUBBER TEST REQUIREMENTS

snubber is found to be uncovered, the snubber shall be declared inoperable and cannot be determined OPERABLE via functional testing unless the test is started with the piston in the as-found setting, extending the piston rod in the tension mode direction.

d. Transient Event Inspection

An inspection shall be performed of all hydraulic and mechanical snubbers attached to sections of systems that have experienced unexpected, potentially damaging transients as determined from a review of operational data. A visual inspection of the systems shall be made within 6 months following such an event. In addition to satisfying the visual inspection acceptance criteria, freedom-of-motion of mechanical snubbers shall be verified using at least one of the following: (1) manually induced snubber movement; or (2) evaluation of in-place snubber piston setting; or (3) stroking the mechanical snubber through its full range of travel.

e. Functional Tests

During the first refueling shutdown and at least once per 18 months thereafter during shutdown, a representative sample of snubbers shall be tested using one of the following sample plans. The sample plan shall be selected prior to the test period and cannot be changed during the test period. The NSSC shall be notified in writing of the sample plan selected prior to the test period or the sample plan used in the prior test period shall implemented:

1. At least 10% of the total of each type of snubber shall be functionally tested either in-place or in a bench test. For each snubber of a type that does not meet the functional test acceptance criteria of the Table 13.7-7 "f", an additional 10% of that type of snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested; or

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TABLE 13.7-7 (Sh. 3 of 6)
SNUBBER TEST REQUIREMENTS

2. A representative sample of each type of snubber shall be functionally tested in accordance with Figure 13.7-1. "C" is the total number of snubbers of a type found not meeting the acceptance requirements of the Table 13.7-7 "f" The cumulative number of snubbers of a type tested is denoted by "N." At the end of each day's testing, the new values of "N" and "C" (previous day's total plus current day's increments) shall be plotted on Figure 13.7-1. If at any time the point plotted falls in the "Reject" region, all snubbers of that type shall be functionally tested. If at any time the point plotted falls in the "Accept" region, testing of snubbers of that type may be terminated. When the point plotted lies in the "Continue Testing" region, additional snubbers of that type shall be tested until the point falls in the "Accept" region or in the "Reject" region, or all the snubbers of that type have been tested. Testing equipment failure during functional testing may invalidate that day's testing and allow that day's testing to resume anew at a later time, providing all snubbers tested with the failed equipment during the day of equipment failure are retested; or

3. An initial representative sample of 55 snubbers shall be functionally tested. For each snubber type which does not meet the functional test acceptance criteria, another sample of at least one-half the size of the initial sample shall be tested until the total number tested is equal to the initial sample size multiplied by the factor, 1 + C/2, where "C" is the number of snubbers found which do not meet the functional test acceptance criteria. The results from this sample plan shall be plotted usint an "Accept" line which follows the equation n=55(1 + C/2).

Each snubber point should be plotted as soon as the snubber is tested. If the point plotted falls on or below the "Accept" line, testing of that type of snubber may be terminated. If the point plotted falls above the "Accept" line, testing must continue until the point falls in the "Accept" region or all the snubbers of that have been tested.

The representative sample selected for the functional test sample plans shall be randomly selected from the snubbers of each type and reviewed before beginning

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TABLE 13.7-7 (Sh. 4 of 6)
SNUBBER TEST REQUIREMENTS

the testing. The review shall ensure as far as practical that they are representative of the various configurations, operating environments, range of size, and capacity of snubbers of each type. Snubbers placed in the same locations as snubbers which failed the previous functional test shall be retested at the time of the next functional test but shall not be included in the sample plan. If during the functional testing, additional sampling is required due to failure of only one type of snubber, the functional test results shall be reviewed at the time to determine if additional samples should be limited to the type of snubber which has failed the functional testing.

f. Functional Test Acceptance Criteria

The snubber functional test shall verify that:

1. Activation (restraining action) is achieved within the specified range in both tension and compression;

2. Snubber bleed, or release rate where required, is present in both tension and compression, within the specified range;

3. For mechanical snubbers, the force required to initiate or maintain motion of the snubber is within the specified range in both directions of travel; and

4. For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement.

Testing methods may be used to measure parameters indirectly or parameters other than specified if those results can be correlated to the specified parameters through established methods.

g. Functional Test Failure Analysis

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TABLE 13.7-7 (Sh. 5 of 6)
SNUBBER TEST REQUIREMENTS

An engineering evaluation shall be made of each failure to meet the functional test acceptance criteria to determine the cause of the failure. The results of this evaluation shall be used, if applicable, in selecting snubbers to be tested in an effort to determine the OPERABILITY of other snubbers irrespective of type which may be subject to the same failure mode.

For the snubbers found inoperable, an engineering evaluation shall be performed on the components to which the inoperable snubbers are attached. The purpose of this engineering evaluation shall be to determine if the components to which the inoperable snubbers in order to ensure that the component remains capable of meeting the designed service.

If any snubber selected for functional testing either fails to lock up or fails to move, i.e., frozen-in-place, the cause will be evaluated and, if caused by manufacturer or design deficiency, all snubbers of the same type subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated in the Table 13.7-7 "e" for snubbers not meeting the functional test acceptance criteria.

h. Functional Testing of Repaired and Replaced Snubbers

Snubbers that fail the visual inspection or the functional test acceptance criteria shall be repaired or replaced. Replacement snubbers and snubbers which have repairs that might affect the functional test result shall be tested to meet the functional test criteria before installation in the unit. These snubbers shall have met the acceptance criteria subsequent to their most recent service, and the functional test must have been performed within 12 months before being installed in the unit.

i. Snubber Service Life Program

The service life of hydraulic and mechanical snubbers shall be monitored to

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TABLE 13.7-7 (Sh. 6 of 6)
SNUBBER TEST REQUIREMENTS

ensure that the service life is not exceeded between surveillance inspections. The maximum expected service life for various seals, springs, and other critical parts shall be determined and established based on engineering information and shall be extended or shortened based on monitored test results and failure history. Critical parts shall be replaced so that the maximum service life will not be exceeded during a period when the snubber is required to be OPERABLE.

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TABLE 13.7-8 (Sh. 1 of 2)
SNUBBER VISUAL INSPECTION INTERVAL

Population or Category (Notes 1 and 2)	Number of Unacceptable Snubber		
	Column A Extend Interval (Notes 3 and 6)	Column B Repeat Interval (Notes 4 and 6)	Column C Reduce Interval (Notes 5 and 6)
1	0	0	1
80	0	0	2
100	0	1	4
150	0	3	8
200	2	5	13
300	5	12	25
400	8	18	36
500	12	24	48
750	20	40	78
1000 or greater	29	56	109

Note 1: The next visual inspection for a snubber population or category size shall be determined based upon the previous inspection interval and the number of unacceptable snubbers found during that interval. Snubbers may be categorized, based upon their accessibility during power operation, as accessible. These categories may be examined separately or jointly. However, the licensee must take and document that decision before any inspection and shall use that decision as the basis upon which to determine the next inspection interval for that category.

Note 2: Interpolation between population or category sizes and the number of unacceptable snubbers is permissible. Use next lower integer for the value of the limit for Columns A, B, or C if that integer includes a fractional value of unacceptable snubbers as determined by interpolation.

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TABLE 13.7-8 (Sh. 2 of 2)
SNUBBER VISUAL INSPECTION INTERVAL

Note 3: If the number of unacceptable snubbers is equal to or less than the number in column A, the next inspection interval may be twice the previous interval but not greater than 48 months.

Note 4: If the number of unacceptable snubbers is equal to or less than the number in column B but greater that the number in column A, the next inspection interval shall be the same as the previous interval.

Note 5: If the number of unacceptable snubbers is equal to or greater than the number in column C, the next inspection interval shall be two-thirds of the previous interval. However, if the number of unacceptable snubbers is less than the number in column C but greater than the number in column B the next interval shall be reduced proportionally by interpolation. That is, the previous interval shall be reduced by a factor that is one-third of the ratio of the difference between the number of unacceptable snubbers found during the previous interval and the number in column B to the difference in the number in columns B and C.

Note 6: The provisions of specification surveillance requirement(SR) 3.0.2 are applicable for inspection intervals up to and including 48 mon

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TABLE 13.7-9(Sh. 1 of 1)
DIVERSE PROTECTION SYSTEM(DPS) INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CANNEL	CANNELS TO TRIP	MINIMUM CANNELS OPERABLE
Ⅰ. TRIP GENERATION			
Pressurier Pressure-High	2[1)]	2	2
Ⅱ. AUXILIARY FEEDWATER(SG-1,2) (AFAS-1,2)			
Steam Generator 1,2 Level-Low	2	2	2

TABLE 13.7-10(Sh. 1 of 1)
DIVERSE PROTECTION SYSTEM(DPS) RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIMES
Ⅰ. TRIP GENERATION	
Pressurizer Pressure-High	≤ 1.15 초[2)]
Steam Generator Level-Low	≤ 1.25 초

1) There are two channels, each of which opens one Control Element Drive Mechanism Moter Generator output contactor arranged in a two-out-of-two logic, thus removing motive power to the Reactor Trip Switchgear System.

2) The response times in Table 13.7-10 are made up of the time to generate the trip signal at the detector (sensor response time) and the time for the signal to interrupt power to the CEA drive mechanism (signal or trip delay time).

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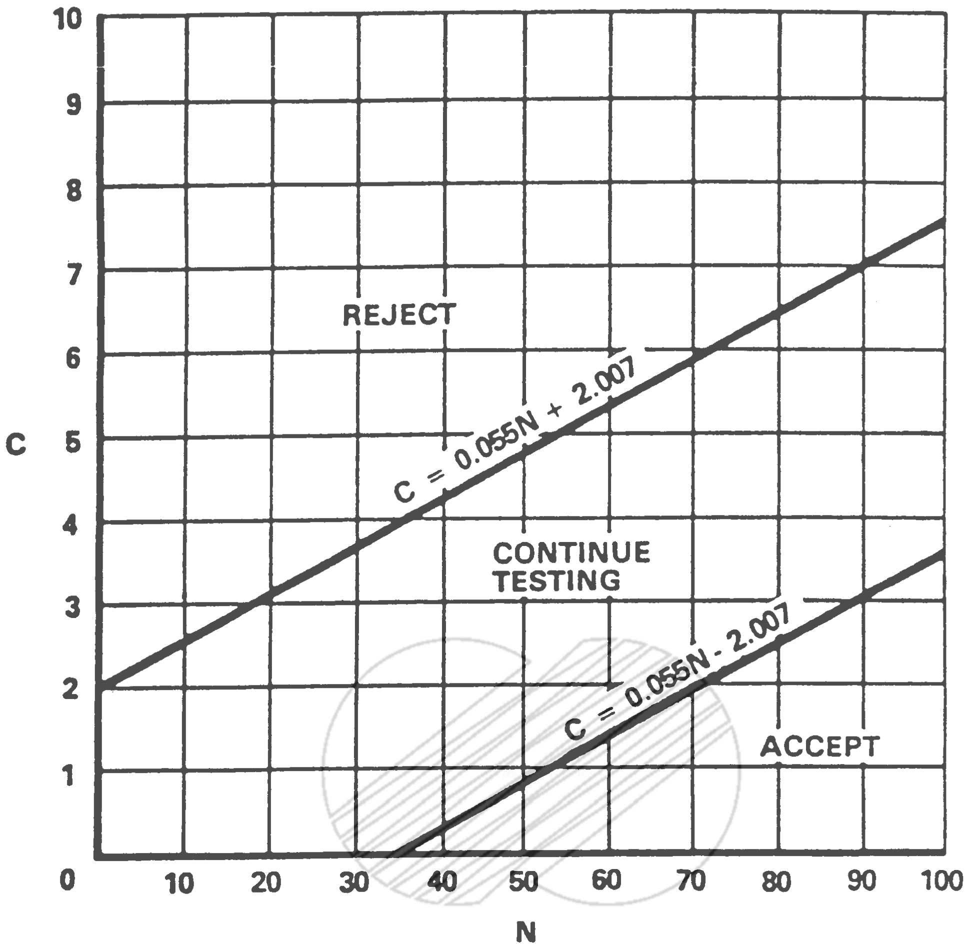

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SAMPLE PLAN FOR SNUBBER FUNCTIONAL TEST

FIGURE 13.7-1

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MINIMUM BORATED WATER VOLUME
FIGURE 13.7-2