MAY 9 1977

MEMORANDUM FOR: Robert Baer, Chief, Reactor Safety Branch, OT/DOR
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Brian Grimes, Chief, Environmental Evaluation Branch, OT/DOR
Lawrence Shao, Chief, Engineering Branch, OT/DOR

THRU: Darrell Eisenhut, Assistant Director for Operational Technology, DOR

FROM: Dennis L. Ziemann, Group Leader, Systematic Evaluation Program Review Group, OT/DOR

SUBJECT: SYSTEMATIC EVALUATION PROGRAM (SEP) FOR OPERATING NUCLEAR POWER PLANTS

One task of the Systematic Evaluation Program is to prepare definitions for each of the topics in the Comprehensive List of topics to be considered in the program. This task is to be accomplished primarily by the cognizant OT engineers and coordinated by the SEP group technical coordinators.

It is therefore requested that your engineers prepare definitions for each topic for which your branch has responsibility using the guidance and forms provided by your branch SEP coordinator. A copy of the Comprehensive List, revised to reflect consideration of your comments, an instruction sheet, a model definition and a definition form is enclosed. Definitions for many of the topics already have been prepared for the Technical Activities Lists. To simplify this task and to maintain consistency, existing definitions when available should be used as a basis for preparing definitions in the SEP format. Your branch SEP coordinator will provide additional guidance and will assist in coordination with other OT branches when input from more than one branch is required on a single topic.

We request that all definitions be submitted to the SEP coordinators as they are completed and that they all be completed by May 20, 1977.

Original Signed by:
Dennis L. Ziemann
Dennis L. Ziemann, Group Leader
Systematic Evaluation Program
Review Group
Division of Operating Reactors

D-10

DEFINITION

TOPIC:

1. Definition:

2. Safety Objective:

3. Status:

4. References:

Comprehensive List of SEP Items

COMP-LIST 1 (5/5/77)

COMP-LIST 1 is based on and includes the following listings of items:

(1) TSAR-264: Technical Safety Activities Report, DSS update, 264 items

(2) Encl B-324: Memo Eisenhut to Stello (2/15/77) - Prioritization of NRR Technical Generic Activities, 324 items

(3) TFL-118: Task Force Report on the Systematic Evaluation of Operating Nuclear Power Plants, Appendix B, November 1976, 118 items

(4) DOT-102: List of 102 items prepared by DOR in March 1977

COMP-LIST 1 follows the general format of the Standard Review Plan (SRP) with respect to chapter identification.

II SITE CHARACTERISTICS

II-1 Site

A. Exclusion Area Authority and Control

B. Population Distribution

.. Potential Hazards or Changes in Potential Hazards Due to Transportation, Institutional, Industrial, and Military Facilities (Air Blast Protection)

1. Transportation Routes in the Vicinity of the Plants

2. Aircraft Traffic in the Vicinity of the Plants

3. Industrial and Military Facilities in the Vicinity of the Plants

D. Site Vicinity Changes: Mechanism to Detect and Handle in Orderly Manner

II-2 Meteorology

A. Severe Natural Weather Phenomena (Meteorology and Climatology Conditions Used in Design and Operating Bases)

B. Onsite Meteorological Measurements Program Including Availability of Data in Control Room

C. Atmospheric Diffusion Characteristics for Accidents and Incident Calculations

II-3 Hydrology

A. Hydrologic Description

B. Design Basis Flood

1. Flood Protection Requirements

2. Capability of Operating Plant to Cope with Probable Maximum Flood

3. Generalized Stream and Hurricane Flooding Parameters

C. Ultimate Heat Sink (UHS)

1. Considerations of Ice, Low Water, Leak Potential and Underwater Dams

D. Dispersion, Dilution, and Travel Times of Accidental Releases of Liquid Effluents in Surface Waters

E. Groundwater

II-4 Geologic and Seismologic Information

A. Tectonic Province

B. Proximity of Capable Tectonic Structures in Plant Vicinity

C. Historical Seismicity within 200 Miles of Plant

D. Stability of Slopes

E. Dam Integrity

F. Settlement of Foundations and Buried Equipment

III DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

III-1 Classification of Structures, Components and Systems (Seismic and Quality)

A. Seismic Requirements for Auxiliary Systems

B. Fuel Pool Cooling System Classification

C. Classification of Electrical Systems

III-2 Wind and Tornado Loadings

A. Design Wind Protection

B. Tornado Wind and Pressure Drop Protection

C. Effect of Failure of Structures not Designed for a Tornado on Safety of Category I Structures, Systems and Components

D. Tornado Effects on Emergency Cooling Ponds

III-3 Hydrodynamic Loads

A. Effects of High Water Level on Structures

B. Structural Consequences of Failure of Underdrain Systems

C. Inservice Inspection of Water Control Structures

III-4 Missile Generation and Protection

A. Tornado Missiles

B. Turbine Missiles

C. Internally Generated Missiles (Inside and Outside Containment)

D. Site Proximity Related Missiles (Including Aircraft)

E. Missile Overall Effects

III-5 Evaluation of Pipe Breaks

A. Effects of Pipe Break on Structures, Systems and Components Inside Containment, Including Flooding

B. Pipe Breaks Outside Containment

III-6 Seismic Design Considerations

A. Response Spectra

1. Response Spectral Shape

2. Vertical and Horizontal Earthquake Response Spectra

3. Effective Response Spectra "g" Value

B. Analysis Techniques and Design Criteria

C. Seismic Instrumentation

D. Effect of Failure of Non-Category I Structures on the Safety of Category I Structures, Systems and Components

E. Seismic Qualification of Mechanical and Electrical Equipment

III-7 Category I Structures Integrity

A. Inservice Inspection, Including Prestressed Concrete Containments with Either Grouted or Ungrouted Tendons

B. Load Combinations

C. Evaluation of Secondary Stresses in Concrete Containments

D. Reactor Cavity Design Criteria

E. Pre-stressed Concrete Delamination Criteria

F. Qualification for Cement Grouting

G. Containment Structural Integrity Tests

H. Containment Penetrations

III-8 Reactor Vessel Internals Integrity

A. Loose Parts Monitoring (Including Neutron Noise Analysis)

B. Channel Box Integrity

C. Control Rod Drive Mechanism Integrity

D. Irradiation Damage

E. Stainless Steel Design Fatigue Basis

F. Control of the Use of Sensitized Stainless Steel

G. Core Support Integrity

III-9 Support Integrity

A. Reactor Vessel (Including Subcooled LOCA Loads)

B. Steam Generator and Reactor Coolant Pump

C. Other Class 1, 2, 3 and MC Components

D. Torus

E. Pump and Piping Restraints

F. Inservice Inspection

G. Snubber

H. Corrosion

I. Stress Corrosion Cracking of High Strength Supports/Restraints

J. Fracture Toughness of Supports

III-10 Pumps and Valves Integrity

A. Operability

B. Testability

C. Inservice Inspection

D. Passive Mechanical Valve Failures

E. Thermal-Overload Protection for Motors of Motor-operated Valves

F. Motor-operated Valves in ECCS Accumulator Lines

G. Pump Flywheel Integrity

H. Manual Valve Position Requirements

I. Surveillance Requirements on BWR Recirculation Pumps and Discharge Valves

III-11 Component Integrity

A. Effect of Faulted Conditions on Component Integrity

B. Operability of Components Under Accident Loads

III-12 Environmental Qualification of Safety Related Equipment

A. Temperature, Pressure, Humidity

B. Chemistry

C. Radiation

IV REACTOR

IV-1 Fuel System Design

A. Bowing

B. Peak-Clad Interaction (PCI)

C. Creep

D. Burst

E. Densification

F. Failure Detection

G. Seismic and LOCA Loads on Fuel Assemblies

H. Behavior of Reactor Fuel Under Abnormal Conditions

I. Qualification of New Fuel Geometries (Including Fuel Surveillance)

J. Enhanced Fission Gas Release at High Burnup

K. LWR Steady State and Transient Fission Product Release

L. Behavior of Control Materials

M. LWR Fuel Stored Energy

N. GAP Conductance

IV-2 Nuclear Design

A. Core Power Distribution

B. Reactivity Coefficients

C. Control Rod Patterns

D. Control Worths/NVT Depletion

E. Criticality

F. Rod Insertion Limit Analysis

G. Behavior of Burnable Poison Rods (Including In-reactor Growth)

H. Control Element Assembly Interlocks in CE Reactors

I. Nuclear Uncertainties

J. Westinghouse Constant Axial Offset Control

K. Incore Monitoring

IV-3 Thermal Hydraulic Design and Performance

A. Departure from Nucleate Boiling Ratio/Critical Heat Flux Ratio (DMBR/CHCF)

B. Linear Heat Generation Rate (LHGR)

C. Thermal-Hydraulic (T/H) Stability (BWR)

D. Flow Distribution

E. Bypass Leakage

F. Operation With Less Than All Loops In Service

G. Hold Down and Lift-off Forces

IV-4 Reactivity Control Systems

A. Functional Design of Reactivity Control System (Operating, Shutdown, and Refueling Modes)

B. Protection Against Single Failures In Reactivity Control Systems

IV-5 BWR Jet Pumps

A. Operating Indications

B. Impingement Loads

IV-6 Acoustic and Neutronic Noise Monitoring

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V REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

V-1 Compliance with Codes and Standards (10 CFR 50.55a)

V-2 Applicable Code Cases

V-3 Overpressurization Protection

A. Design Basis Transient

B. Protection System Design Criteria

C. Safety/Relief Valve Requirements (Including Dynamic Loads)

D. Prompt Relief Trip (BWR)

V-4 Piping and Safe End Integrity

A. Fracture Toughness

B. Flaw Evaluation Criteria

C. Stress Corrosion Cracking in PWR Piping

D. Stress Corrosion Cracking in BWR Piping

E. Control of Delta-Ferrite in Stainless Steel Welds

F. Austenitic Stainless Steel Piping

G. Bi-metallic Welds

V-5 Reactor Coolant Pressure Boundary (RCPB) Leakage, Detection

V-6 Reactor Vessel Integrity

A. Fracture Toughness

B. Operating Limitations

C. Materials Surveillance

D. Inservice Inspection

E. Feedwater Nozzle Cracking (BWR)

F. Transient Analyses

G. Appendix G Compliance, Thermal Shock to Reactor Vessel

V-7 Reactor Coolant Pump Overspeed

A. BWR Recirculation Pump Overspeed During a LOCA

B. PWR Pump Overspeed During a LOCA

V-8 Steam Generator (SG) Integrity

A. Secondary Water Chemistry Control

B. Tube Repair

C. S. G. Tube Leakage Limits

D. S. G. Plugging Criteria

E. Inservice Inspection

F. Denting in Steam Generator tubes

G. Steam Generator Design Requirements

H. Flow Induced Vibration of High Temperature Heat Exchanger Tubes

I. Corrosion of S. G. Tubes

V-9 Reactor Core Isolation Cooling System (BWR)

V-10 Residual Heat Removal (RHR) System

A. RHR Heat Exchanger Tube Failures

B. RHR Reliability

V-11 High Pressure/Low Pressure Interface

A. Requirements for Isolation of High and Low Pressure Systems

B. RHR Interlock Requirements

V-12 Reactor Water Cleanup System (BWR)

A. Chemistry of Boiling Water Reactor Primary Coolant

V-13 Flow Instabilities and Waterhammer Including Instrumentation and Control System Reliability

VI ENGINEERED SAFETY FEATURES

VI-1 Organic Materials and Post Accident Chemistry

VI-2 Containment Functional Design

A. Pressure-Suppression Type BWR Containments

1. Mark I Pool Dynamics - LOCA and SRV Discharge Load

2. Mark I Containment Structural Evaluation

B. Subcompartment Pressure Loads

1. Reactor Cavity Subcompartment Analyses to Determine Reactor Vessel Support Capability

2. Asymmetric Pressure Lodas in Containment Subcomparments

3. ECCS Compartment Design

C. Ice Condenser Containment

D. Mass and Energy Release to Containment

E. Minimum Contaiment Back Pressure Analysis for ECCS

F. Effects on Containment External Design Pressure From Containment Depressurization by Inadvertent Spray Operation

VI-3 Containment Pressure and Heat Removal Capability

A. Effective Operation of Containment Sprays in a LOCA

B. Pressure in Containment Following LOCA

C. Containment Post-LOCA Temperature

D. Passive Heat Sinks

VI-4 Secondary Containment Functinal Design

A. Determination of Bypass Leakage in Dual Containments

VI-5 Containment Isolation System (Including use of Purge Valves During Operation)

A. Isolation Criteria for Sump Suction Lines

B. Containment Isolation Provisions for Fluid Systems

C. PWR Containment Ventilation Systems

D. Containment Purging - Potential Release Due to Open Purge Lines, Closure of Purge Valves

VI-6 Combustible Gas Control

A. Hydrogen Mixing Capability in Containment Post LOCA Operation to Determine Adequacy of Containment External Design Pressure

B. Containment Air Dilution (CAD) Systems (Including Radiological Analysis of CAD System)

C. Consequences of the Inadvertent Release of Hydrogen Charging Lines for the Main Generator Cooling of Reactor Coolant Chemical Control Systems, etc.

D. Effects of Explosions

E. Sharing Hydrogen Recombiners

F. Design Requirements of Hydrogen Recombiners

G. Containment Air Monitoring (CAM)

VI-7 Containment Leak Testing

VI-8 Emergency Core Cooling System

A. Emergency Core Cooling System Performance

1. ECCS Re-evaluation to Account for Increased Vessel Head Temperature

2. Effects of Core Flow on BWR LOCA Analysis

3. ECCS Bypass

4. Upper Plenum Injection

5. On-line Testability of ECCS Actuation System and Component Availability

6. Core Spray Nozzle Effectiveness

B. ESF Switchover from Injection to Recirculation Mode

C. ECCS Single Failure Criterion and Requirements for Locking Out Power to Valves

1. Lack of Independence of Interlocks on ECCS Valves

2. Appendix K - Electrical Instrumentation and Control System Branch (EICSB) Re-reviews

3. Failure Mode Analysis - ECCS

4. The Effect of PWR Loop Isolation Valve Closure During a LOCA on ECCS Performance

D. Long Term Cooling - Passive Failures (e.g., Flooding of Redundant Components)

E. ECCS Sump Design and Test for Recirculation Mode Effectiveness

F. Accumulator Isolation Valves Power and Control System Design

VI-9 Control Room Habitability

A. Interpretation of GDC-19 "Control Room"

B. Control Room Infiltration

VI-10 Fission Product Removal Systems (e.g., Sprays and Engineered Safety Features (ESF) Filters)

A. Iodine Removal by Sprays

B. Internal Recirculation ESF Charcoal Filter System

C. Secondary Containment

VI-11 Inservice Inspection of Class 2 and 3 Components

VI-12 Main Steam Isolation Valves

A. MSIV Reliability - PWR

B. MSIV Failure Investigation

C. Main Steam Line Isolation Seal System - BWR

D. MSIV Design Basis

VI-13 Selected Engineered Safety Features (ESF) Aspects

A. Testing of Reactor Trip System and Engineered Safety Features (Including Response Time Testing)

B. Shared Safety and Service Systems for Multiple Unit Facilities

C. Effect of Failure in Non-Safety-Related systems

VII INSTRUMENTATION AND CONTROLS

VII-1 Reactor Trip Systems (IEEE-279)

A. Computer Applications - Reactor Protection System (RPS)

B. Multiplexing in Control and Safety Systems

C. Automatic Resetting of Reactor Trip System Trip Bistable Relays

D. Isolation of RPS from Non-safety Systems and Qualifications of Isolation Devices

VII-2 ESF Actuation System

A. General ESF Actuation System Control Logic and Design, Including Bypasses, Reset Features, and Interaction with Transfers Between On-site and Off-site Power Sources

VII-3 Systems Required for Safety Shutdown

A. Safe Shutdown systems

B. Remote Shutdown

C. Method of Bringing a PWR from a High Pressure Conditin to Low Pressure Cooling Assuming the Usae of Only Safety Grade Equipment

VII-4 Saftey Related Control and Instrument Power

A. Loss of Plant Air Systems (Effect on Plant Control and Monitoring)

VII-5 Instruments for Monitoring Radiation and Process Variables During Accidents

VII-6 RC Pump Underfrequency Trip

VII-7 Trip Uncertainty and Setpoint Analysis Review of Operating Data Base

A. Automatic Resetting of Reactor Trip System Trip Bistable Relays

B. Protection System Automatic Trip Point Changes for Operation with Reactor Coolant Pumps Out of Service

VII-8 Separation of Redundant Safety Related Equipment

A. Acceptability of Swing Bus Design on BWR-4 Plants

VIII ELECTRIC POWER

VIII-1 Offsite Power Systems

A. Potential Equipment Failures Associated with Degraded Grid Voltage

B. Loss of Offsite Power Subsequent to Manual Safety Injection Reset Following a LOCA

C. Load Break Switch

VIII-2 Onsite Emergency Power Systems

A. Load Break Switch

B. Emergency Power for Two or More Reactors at the Same Site

C. Diesel Generator (Including Qualification, Reliability, Operation at Low Loads, Lock Out, Fuel Oil, Testing)

VIII-3 Emergency DC Power Systems

A. Station Battery Capacity Test Requirements

B. DC Power System Bus Voltage Monitoring and Annunciation

C. Loss of an Emergency DC Bus

VIII-4 Electrical Containment Penetrations

A. Failure of Containment Penetrations from Electrical Faults Inside Containment During LOCA

IX AUXILIARY SYSTEMS

IX-1 Fuel Storage

A. Spent Fuel Pool Protection

B. Cooling System

C. Criticality of High Density Racks

D. Criticality of New Fuel Dry Storage

E. New Fuel Storage Facility

IX-2 Cranes (Reactor Building, Fuel Handling, Spent Fuel, Heavy Loads Over Fuel)

A. Overhead Crane Handling Systems

B. Cask Drop, Heavy Loads and Cask Handling (Including Structural Effects)

C. Crane Standards

IX-3 Station Service and Cooling Water Systems

A. Adequacy of Physical Separation of Component Cooling Water Systems Which are Vital to the Performance of Engineered Safety Systems Components

B. Auxiliary Cooling Water Systems Including Component Cooling Water

C. Availability of Cooling Water to Primary Coolant Pump

D. Cooling Water Makeup Requirements

E. Safe Shutdown Systems

F. Effect of Water Over-flow from Tanks

G. Circulating Water System Barrier Failure Protection

IX-4 Process Auxiliaries

A. Chemical and Volume Control System

1. Boron Additon System

2. Boron Precipitation (Post LOCA)

3. Need for Improved Equipment

B. Standby Liquid Control System (BWR)

IX-5 Ventilation Systems

A. Spent Fuel Pool Area Ventilation System

B. ESF Equipment Ventilation System

IX-6 Fire Protection

A. Use of Insulation in Containments

IX-7 Communication and Lighting Systems

X STEAM AND POWER CONVERSION SYSTEM

X-1 Feedwater System

A. Auxiliary Feedwater System Pump Drive and Power Diversity

B. Water Hammer in Feedwater Systems (Including Instrumentation and Control System Reliability)

XI RADIOACTIVE WASTE MANAGEMENT

XI-1 Control of Radioactive Liquids (Includes Appendix I)

XI-2 Control of Radioactive Gases (Includes Appendix I and Considerations of Potential Explosions)

XI-3 Solid Radwaste Systems (Includes Appendix I)

XI-4 Radiological (Effluent and Process) Monitoring Systems

XII RADIATION PROTECTION

XII-1 Shielding (Reactor Cavity/Streaming)

XII-2 Radiation Protection Procedures for Nuclear Power Plants (Including ALARA)

XIII OPERATIONS

XIII-1 Conduct of Operations

A. Organization and Qualification

B. Emergency Planning and Off-site Preparedness

C. Operator Actions During Emergency

D. Long-Term Post-Accident Operator Actions

E. Plant Procedures

F. Alarm Requirements

XIII-2 Safeguards/Industrial Security

A. Design Features to control Sabotage

B. Operating One Plant While Others are Under Construction

C. Effects of Explosion on Nuclear Power Plant Structures

XIII-3 Decontamination

XIV STARTUP TESTS

XV ACCIDENTS AND TRANSIENTS

XV-1 Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow and Inadvertent Opening of a Steam Generator Relief or Safety Valve

XV-2 Spectrum of Steam System Piping Failures Inside and Outside of Containment (PWR)

XV-3 Loss of External Load, Turbine Trip, Loss of Condenser Vacuum, Closure of Main Steam Isolation Valve (BWR), and Steam Pressure Regulatory Failure (CLosed)

XV-4 Loss of Non-Emergency A-C Power to the Station Auxiliaries

XV-5 Loss of Normal Feedwater Flow

XV-6 Feedwater System Pipe Breaks Inside and Outside Containment (PWR)

XV-7 Loss of Forced Reactor Coolant Flow Including Trip of Pump and Flow Controlled Malfunctions

XV-8 Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break

XV-9 Uncontrolled Control Rod Assembly Withdrawal From a Subcritical or Low Power Startup Condition

XV-10 Uncontrolled Control Rod Assembly Withdrawal at Power

XV-11 Control Rod Misoperation (System Malfunction or Operator Error)

XV-12 Startup or an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR CORE FLOW RATE

XV-13 Chemical and volume Control System Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant (PWR)

XV-14 Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position

A. Ability of Off-gas and Steam Line Monitors to Detect and Isolate in the Event of Fuel Damage

XV-15 Spectrum of Rod Ejection Accidents (PWR)

XV-16 Spectrum of Rod Drop Accidents (BWR)

XV-17 Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory

XV-18 Inadvertent Opening of a PWR Pressurizer Safety/Relief Valve or a BWR Safety/Relief Valve

XV-19 Radiological Consequences of Failure of Small Lines Carrying Primary Coolant Outside Containment

XV-20 Radiological Consequences of Steam Geneator Tube Failure (PWR)

XV-21 Radiological Consequences of Main Steam Line Failure Outside Containment (BWR)

XV-22 Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary

XV-23 Waste Gas System Failure

XV-24 Radioactive Liquid Waste System Leak or Failure (Release to Atmosphere)

XV-25 Postulated Radioactive Releases Due to Liquid-Containing Tank Failures

XV-26 Radiological Consequences of Fuel Damaging Accidents (Inside and Outside Containment)

XV-27 Spent Fuel Cask Drop Accidents

XV-28 Anticipated Transients Without Scram

XV-29 Multiple Tube Ruptures in B&W Steam Generators with Concurrent Loss of Condenser Function

XV-30 Loss of All A-C Power

XVI TECHNICAL SPECIFICATIONS

XVI-1 Comparison to Standard Tech. Specs (STS)

XVI-2 Review Calculation Bases - Quantitative vs. Qualitative

XVI-3 Operation with Equipment Out of Service (LCOMS)

XVI-4 Compatibility with Design of Plant

XVI-5 Action to be Taken in Response to Fuel Failures (Level of Activity Before Action is Required; Monitoring)

XVI-6 Instrument Trip Setpoints in Standard Technical Specifications

XVI-7 Nuclear Uncertainties and Tech Specs

XVI-8 Analyses and Reduction of In-core Measurements

XVI-9 Maintenance and Inspection of Plants

XVI-10 Allowable ECCS Equipment Outage Periods

XVI-11 Site Related Technical Specification and Emergency Operation

XVI-12 Filter Technical Specifications

XVI-13 Overpressurization - PWR Equipment Operating Restrictions During Water Solid Conditions

XVI-14 Surveillance Requirements Including Trip Setpoints and Time Delay Relay Setpoints

X II OPERATIONAL QA PROGRAM

GUIDANCE FOR DEVELOPING DEFINITIONS

1. Prepare a definition for each topic and subtopic on the SEP Comprehensive List. However, if subtopics are clearly included in the definition of other subtopics or the major topic, a specific definition need not be prepared for that item. Such items will be considered with the topic for which a definition is written and, therefore, will be deleted from the comprehensive list. Combination of topics in a definition are encouraged.

   Definitions should be prepared in accordance with the attached model format, 1 topic/page.

   EXAMPLE: VI-7 Containment Leak Testing

2. Blank forms are provided for the definition, they need not be typed as long as the hand written draft is clearly legible.

3. The descriptions should be brief, and to the extent practical should provide a brief scope of the review.

4. The safety objective should clearly identify why the topic is reviewed.

5. Status should indicate applicability (i.e., PWR's, BWR's, all, etc.) and an estimated date of completion if the topic is an ongoing study or an ongoing generic review.

6. Include references for further clarification of the topics (e.g., Rules and Regulations, Regulatory Guides, Codes, Standards, BTP, SRP's).

7. Completed definitions should be returned to the respective SEP Technical Coordinator as they are completed.

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DEFINITION

TOPIC:

1. Definition:

2. Safety Objective:

3. Status:

4. References: