UNITED STATES
NUCLEAR REGULATORYCOMMISSION
TECHNICAL TRAINING CENTER

GE TECHNOLOGY MANUAL (R-304B)

This manual is a text and reference document for the General Electric Systems Course. It should be used by students as a study guide during attendence at this course. This manual was compiled by staff members of the Technical Training Division in the Office for Analysis and Evaluation of Operational Data.

The information in this manual was developed or compiled for NRC personnel in support of internal training and qualification programs. No assumptions should be made as to its applicability for any other purpose. Information or statements contained in this manual should not be interpreted as setting official NRC policy. The data provided are not necessarily specific to any particular nuclear power plant, but can be considered to be representative of the vendor design.

Course Information

TTC PHONE SYSTEM

1. Commercial: 423/855-6500
2. Incoming calls for students -- see paragraph on STUDENT MESSAGES
3. Classroom phones are a common internal line and can only be used to call other areas inside the Training Center.
4. Wall phones in the 1st, 2nd, 3rd, and 4th floor student lounge areas can be used for students making outside calls.
5. To make local calls: dial 9 -- local number
6. To make long distance calls: dial 8 – Area Code – Number

Note: *TTC is now on detailed billing for actual telephone usage and all calls are listed on a computer printout. Please limit calls home to no more than 5 minutes, per NRC Manual Chapter Appendix 1501, Part IV.D.5.*

AREA INFORMATION

1. Restaurants -- Eastgate Mall, Brainerd Road area
2. Hospital -- Humana in East Ridge -- Phone: 894-7870
3. Emergency Phone Number -- 911

COURSE RELATED ITEMS

1. Working hours are from 7:30 a.m. to 4:15 p.m. Classroom presentations are from 8:00 a.m. to 4:00 p.m. Lunch break will begin between 11:30 a.m. - 12:00 p.m. at the discretion of the instructor.

2. The Course Director and Course Instructor(s) are available to answer questions before and after class, during the breaks, and during lunch time with prior arrangement. Instructors not in the classroom can be reached via the inside phone. Please call ahead to ensure availability.

3. All course related materials (pencil, paper, manuals, notebooks, and markers) are provided. If there is a need for additional material or administrative service, please coordinate with the Course Instructors.

4. Shipping boxes will be provided to the students for the mailing of course materials (manuals & notebooks). Each student must write their name and address to which the box is to be mailed on a mailing label and tape it to the outside of their box. The TTC staff will affix the proper postage.

5. Student registration for all TTC courses is accomplished through Training Coordinators. The TTC staff does not register students directly.

TTC SECURITY

NRC badges will be required to be worn while at the TTC. Please promptly notify Course Director if badge is lost or misplaced.

STUDENT MESSAGES

There is a printer located in the third floor lounge area. All non-emergency student messages will be sent to this printer. It is the responsibility of the students to check this printer for messages. If there are messages on the printer students are asked to post them on the bulletin board above the printer.

FIRST AID KITS

First Aid Kits are located in the instructors desk of each simulator, in the second floor and third and fourth floor student lounges in the sink cabinets, and the sink cabinet in the staff lounge on the second floor. In addition, each location also has a "Body Fluid Barrier Kit". These kits are to be used in the event of personnel injury involving serious bleeding. Each kit contains two complete packets each with: 1 pair of latex gloves, 1 face shield, 1 mouth-to-mouth barrier, 1 protective garment, 2 antiseptic towelettes, and 1 biohazard disposable bag.

TAX EXEMPTION CERTIFICATES

NOTE: We do not have Tax Exempt Certificates for lodging in Chattanooga, Chattanooga is not one of the localities permitted to use these certificates. For a list of locations which are allowed to use them, see the Federal Travel Directory published monthly by GSA.

Please remember that you, as students, represent the NRC and when you knowingly avoid paying Tennessee State Tax, the results can have a negative effect on the Agency.

If you are not able to obtain adequate lodging and stay within the per diem rate established by GSA, advise your Management Support or DRMA office so the proper authorities can be notified.

Student Information Sheet

PLEASE PRINT THE FOLLOWING INFORMATION:

Course Title: ______________________________

Course Dates: ______________________________

Name: ______________________________
(How you want it to appear on the Training Certificate)

Social Security No: ____/____/______ Job Title: ____________

Phone No: ____________ Mailing Address: ____________
(No P.O. Boxes please)

Motel & Room No. where you are staying: ____________

Emergency Contact: ____________ Phone No: ____________

Estimated Travel Cost (including transportation costs): ____________

Name of Immediate Supervisor: ____________

Name of Division Director: ____________ Name of Division: ____________

Catalog No: R-304-B Grade: ____________
(For Office Use Only)

Please provide the following background information: (Please circle one)

1. Highest Level of Education:

   Doctorate Masters Bachelors Associate Other

2. Subject Matter Specialty:

   Engineering Physical Science Math or Statistics Other Science Other

3. Years of Nuclear Experience: >9 7-9 4-6 1-3 <1

4. Type of Nuclear Experience:

   Commercial BWR
   Commercial PWR RO/SRO Navy Test Reactor Other

5. Years with NRC: >9 7-9 4-6 1-3 <1

6. Previous TTC sponsored training attended:

Course Objectives
(R-304B)

The General Electric technology systems course is designed to provide the student with a comprehensive understanding and working knowledge of the boiling water reactor (BWR/4) commercial steam electric plant. At the end of this course each student should have achieved a basic understanding of the following:

- Nuclear theory, reactivity coefficients, and thermal limits,
- Process systems, purposes, theory of operation, normal system configuration, and safety related flowpaths and/or operations,
- Plant electrical system design and distribution,
- Process instrumentation systems including, logics, selected interlocks, limiters, control and protection functions,
- PRA insights into assessing a change to the level of plant safety/risk as a result of system or component problems or failures.

Course Outline

for R-304B

Day	Title	Chapter
1	TTC Introduction Course Overview Reactor Physics	 1.0-1.6 1.7
2	Review Reactor Vessel System Reactor Vessel Instrumentation System	 2.1 3.1
3	Review Fuel and Control Rods Thermal Limits	 2.2 1.8
4	Review Contorl Rod Drive Reactor Manual Control	 2.3 2.5
5	Review Rod Worth Minimizer System Recirculation System Recirculation Flow Control System	 7.5 2.4 7.2
6	Review Reactor Water Cleanup System Reactor Core Isolation Cooling System Main Steam System Circulating Water System	 2.8 2.7 2.5 11.1
7	Review Electro Hydraulic Control System Condensate and Feedwater System	 3.2 2.6
8	Review Turbine Building Service Water System Turbine Building Closed Loop Cooling Water System Feedwater Control System Introduction to Neutron Monitoring Source Range Monitoring System Intermediate Range Monitoring System Local Power Range Monitoring System	 11.4 11.5 3.3 5.0 5.1 5.2 5.3

9	Review Average Power Range Monitoring System Rod Block Monitor System Traversing Incore Probe System Reactor Protection System	 5.4 5.5 5.6 7.3
10	Review Primary Containment System Secondary Containment System Reactor Building Standby Ventilation System Nuclear Steam Supply Shutoff System Off Gas System	 4.1 4.2 4.3 4.4 8.1
11	Review Introduction to Emergency Core Cooling High Pressure Coolant Injection System Automatic Depressurization System Core Spray System Residual Heat Removal System	 10.0 10.1 10.2 10.3 10.4
12	Review Reactor Building Service Water System Reactor Building Closed Loop Cooling Water System Electrical Systems ECCS System Wrapup	 11.2 11.3 9.1-9.4 4.0,9.0,10.0
13	Review Standby Liquid Control System Fuel Pool Cooling and Cleanup System Process Computer System and Thermal Limits Review Review Session	 7.4 12.1 6.1 & 1.8
14	Final Examination	

TTC R-304-B COURSE EVALUATION SHEET

Course Dates__________to__________

I. Instructions:

In order to improve and maintain the quality and applicability of TTC courses it is necessary to obtain feedback from attending students. Please rate the following subject areas. Amplifying comments are desired but not required. Please place your amplifying comments in the section for written comments. Course evaluation should be identified by student to allow for follow-up and amplification of significant issues or suggestions.

II. Evaluation

	Strongly Disagree	Disagree	Agree	Strongly Agree
1. Stated course objectives were met.	_______	_______	_______	_______
2. Learning objectives were helpful in identifying important lecture concepts.	_______	_______	_______	_______
3. Classroom presentations adequately covered the learning objectives.	_______	_______	_______	_______
4. Classroom exercises and demonstrations were effective in reinforcing previously covered concepts and introducing new concepts.	_______	_______	_______	_______
5. Course manual adequately covered course topics where applicable.	_______	_______	_______	_______
6. Course manual was organized so that it can be used as an effective study guide.	_______	_______	_______	_______
7. Visual aids reinforced the presentation of course materials.	_______	_______	_______	_______
8. Completion of this course will assist me in my regulatory activities.	_______	_______	_______	_______

Signature______________________________

9. Overall course rating (considering merits of this course only):

Unsatisfactory	Marginal	Satisfactory	Good	Excellent
________	________	________	________	________

10. What did you like best or find most helpful about the course?

11. What did you like least about the course?

12. What subjects might be added or expanded?

13. What subjects might be deleted or discussed in less detail?

14. How will this course aid you in your ability to do your job as a regulator?

15. What could be done to make this course more useful in aiding you in your ability to effectively carry out your regulatory activities?

16. Additional comments.

Simulator Demonstration Instruction

A portion of this course includes demonstrations of system controls, indication, and operation on the BWR/4 simulator, located on the fourth floor of this building. The class may be split into two groups for these demonstrations. Due to the physical floor space limitations, it is important to stay in your assigned group.

Simulator demonstrations are planned demonstrations coordinated by the instructor. Students should **not** manipulate any switches on the simulator unless directed to by the instructor. The instructor console and computer area are not normally accessible areas for students. For hardware considerations, please do not place drinks on the simulator panels; ample desk area is available.

If the class is split into two groups for simulator demonstrations, while one group is in the simulator, the other group will be on self study. The self study time is to be used for reading manual chapters, review of material that has been presented, or if desired, review of material on an individual basis with an instructor. If instructor assistance is needed, please use the telephone in the classroom to call one of the course instructors. A telephone number list is attached beside the phone.

TABLE OF CONTENTS

Table of Contents

1.0 INTRODUCTION

Learning Objectives:

1. Explain the basic steam cycle as applied to BWR systems.
2. State which BWR control systems are used for the following important functions:
    a. control of reactor power
    b. control of reactor pressure (normal situations)
    c. control of reactor water level
3. State the type of containment package which is provided and explain the following terms:
    a. Drywell
    b. Suppression Pool
    c. Containment
4. List the four Emergency Core Cooling Systems and state which are high pressure systems and which are low pressure systems.

1.0.1 Introduction

The purpose of the Boiling Water Reactor (BWR) System design is to economically generate electrical power through the use of the direct cycle system design which includes the nuclear fuel and internal structures within the reactor pressure vessel, systems associated with a basic steam cycle, normal auxiliary systems to accommodate the operation requirements of the plant, Engineered Safety Systems to accommodate the safeguard requirements of the plant, and the necessary instrumentation and controls to accommodate operator control of the plant.

The BWR/4 Systems Manual is designed for use as a text for the BWR Technology (R-304B) and BWR Advanced Technology (R-504B) Courses. It can also be used for general reference purposes. The manual was written to reflect the BWR/4 design. The numerical values used in the manual are for a specific BWR/4 design. The reader should bear this in mind when attempting to use the manual as a general reference document.

1.0.2 Manual Organization

This manual has been organized to follow, as closely as possible, the order of the material presented in the above course. General subject areas are classified by chapters. Systems which fall under the general classification are arranged as sections within the chapter. Where applicable each section follows the same format; i.e., introduction, system description, component description, system features and interrelations, BWR differences, summary, and graphics.

1.0.2.1 System Introduction

The system introduction states the system purpose and functional classification. The purpose of the introduction is to orient the reader.

The system description provides the reader with an overview of the system and its components. Attention is focused on major components and their purposes without including the detail found in the component description.

1.0.2.2 Component Description

The components are listed in basic flow path order or block diagram arrangement. Each component is described in appropriate detail with specific set points and capacities often referenced in tables.

1.0.2.3 System Features and Interfaces

The system features and interrelations section includes such items as the operational features and limitations. It also identifies interfaces with other systems.

1.0.2.4 BWR Differences

The differences section identifies major design differences between BWR/2, BWR/3, BWR/4, BWR/5, and BWR/6 product lines.

1.0.2.5 Summary

The summary is designed to key the reader to the major items contained in the chapter. It is important for the reader to recognize that the summary is not a substitute for a comprehensive review of the text material.

1.0.2.6 Graphics

The graphic package is located at the end of each chapter section. The graphics are arranged to follow the text and are referenced in the written portion of each chapter section.

1.0.3 Use of the Manual During Course Presentation

Proper use of the manual during class presentations can greatly aid the student in understanding the material presented. The student should follow the presentation using the figures and diagrams provided. Properly noting minor and major points on these figures should eliminate the necessity for taking comprehensive notes during the lecture.

1.1 COMMERCIAL NUCLEAR POWER PLANTS

1.1.1 Introduction

To understand the BWR power plant, a basic knowledge of the major components and their functions is needed.

1.1.2 Nuclear Power Plants

A nuclear power plant is an arrangement of components and systems used to generate heat. The heat is used to make steam which is converted to electrical power. The principal components of a nuclear power plant are:

* nuclear fuel and moderator
* heat removal system
* control systems
* power conversion systems

1.1.3 Nuclear Fuel and Moderator

Nuclear fuel consists of a mixture of fissile, fissionable, and fertile materials. The essential ingredient is a fissile material, which is a material that readily undergoes nuclear fission when struck by neutrons. The only naturally available fissile material is uranium-235 (U-235), an isotope (or form) of uranium constituting less than 1% of the element as found in nature. Two synthetic fissile materials are plutonium-239 (Pu-239) and uranium-233 (U-233). When neutrons strike uranium-238 (U-238), which constitutes more than 99% of the natural uranium, Pu-239 is formed. For this reason U-238 is called a "fertile" material. The element thorium is also a fertile material, forming U-233 when struck by a neutron.

The three basic fissile materials may be used separately or with one of the fertile materials as fuel for a nuclear reactor.

The most commonly used fuel is uranium, either natural, or enriched in the U-235 isotope.

Fuels may be solid or fluid and they may be used in different material forms: metals, alloys, oxides, or salts. A variety of solid fuel physical shapes is used, including rods, plates, tubes, and other shapes, along with various methods for cladding (containing) the fuel.

A moderator is a substance used in a reactor to slow down neutrons from high to low energy levels. Slowing down increases the probability of continued fission. Moderators commonly used include ordinary water, heavy water, and graphite. Liquid moderators can also serve as the coolant.

1.1.4 Heat Removal System

The heat removal system or cycle removes heat which is generated by the fission process in the reactor core. Heat removal system arrangements include single, double, and triple heat transfer cycles. An example of the single cycle system is the direct cycle boiling water reactor delivering steam to a turbine. Pressurized water reactors use two cycles, with the primary water transferring heat in a steam generator to produce steam for the turbine cycle.

1.1.5 Control Systems

In the general sense of the term, there are numerous control systems on modern reactors. The specific control system of concern here is reactivity control, which is the method by which the reactor core fission process is regulated. The basic method of accomplishing this regulation is to insert a neutron poisoning or absorbing material into the reactor core, thereby preventing those neutrons absorbed in the poison from causing fission in the fuel. There are other methods, some of which are specific to the BWR, which are discussed later in this text.

1.1.6 Power Conversion Systems

In modern reactor power plants, steam turbine generators are used to convert the energy of the steam into electrical power.

1.2 WATER COOLED REACTORS

Water is generally used as a coolant and a moderator for power reactors. Initially it was believed that water could not be permitted to boil in a reactor vessel because of the possibility of cladding burnout. This resulted in the early development of pressurized water reactors. The first pressurized water reactor went critical in 1953 at the AEC National Reactor Testing Station in Idaho.

A different type of water-cooled and water-moderated reactor was started in 1953 with the first experiment to test the theory of boiling water in a reactor vessel and making steam directly. Successive experiments established the principle that boiling was not only acceptable but even advantageous for certain purposes.

It is only natural that water became the preferred primary reactor coolant. Reliability is a key factor and water has many important advantages that do not require extensive experimental programs. Water is a known quantity. It is cheap, and it was readily available when the reactor program was started. It has good heat transfer characteristics which can be extended beyond its normal narrow temperature range by pressurizing the water to inhibit boiling. Furthermore, water does not become significantly activated if kept pure. The induced radioactivity of the coolant is short lived so that maintenance is not hampered greatly.

The corrosive quality of water is known, and the pressurizing intensifies the corrosive action. An important inducement is that water serves as moderator to slow down the neutrons. Its tendency to absorb neutrons can be overcome by enriching the fuel.

The disadvantages of using water as a moderator are: water must be highly pressurized to achieve reasonably high temperatures; pure hot water is highly corrosive and requires that the primary coolant system be constructed of special materials; water at high pressure and saturation temperature will flash to steam if the pressure is rapidly reduced, as in a rupture of the primary loop; and water can chemically react violently under certain temperature conditions with uranium, thorium, and structural materials.

The fundamental similarity in nuclear characteristics of water-moderated reactors is determined basically by the nuclear and thermal properties of light water. Briefly, these similarities can be summarized as follows:

- Enriched fuel is required.
- Relatively low moderator-to-fuel ratios are employed.
- Relatively high excess reactivity is provided.
- Conversion ratios for existing types are low, but this is not an inherent characteristic.
- Power densities are comparatively high.

1.3 BOILING WATER REACTORS

In a boiling water reactor, the coolant is very pure water which boils adjacent to the fuel elements. The resulting steam-water mixture then proceeds to steam separators, where the water is separated from the steam bubbles. The water then goes back to the reactor core and the boiling operation is repeated. The steam which is formed passes from the steam separators, through the steam dryers, and to a turbine located outside the containment.

The major difference in the operating characteristics of a boiling water reactor core from other nuclear systems is a result of steam void production. Water affects both the heat generation and the neutron flux characteristics of a nuclear system because it serves the dual function of coolant and neutron moderator. If this water is allowed to boil, which greatly lowers the density of molecules, there is a significant change in nuclear performance. The boiling water reactor design results in a system that produces reactivity changes varying inversely with the steam void content in the core. This provides an inherent safety feature of the boiling water reactor; that is, a transient power increase will produce more steam voids, reducing reactivity, which reduces power and thus limits the excursion.

The fuel used in a boiling water reactor contains uranium in the form of an oxide. This eliminates the hazard involved in using uranium in metallic form. Moreover, before assembly into fuel elements, the uranium oxide is generally heated and converted into a ceramic material, somewhat like the bricks used to line fireplaces. This form of uranium oxide does not react chemically with the reactor coolant and does not burn in air.

1.4 TYPES OF BOILING WATER REACTORS

In direct cycle BWRs, as shown in Figure 1.4-1, steam leaving the reactor passes directly to the turbine. In an indirect cycle BWR, steam is passed to a primary coolant/secondary steam generator. No economic incentive exists for the latter cycle, although it does possess some advantage in that radioactive particles from the primary coolant normally cannot transfer to the steam used in the turbine generator portion of the plant. Dual cycle plants employ a combination of direct and indirect cycles. The first large utility owned BWR (Dresden 1) employed this dual cycle concept. Current BWRs use only the forced circulation direct cycle because it is more economical.

In a forced circulation direct cycle reactor system, the nuclear fuel generates heat within the reactor vessel and boils the water, producing wet steam that passes through internal steam separators and dryers. The water within the reactor is circulated through the core by two external recirculation pumps. The steam is directed from the reactor to the turbine, entering the turbine steam chest at about 950 psi and 540° F. Steam leaving the high pressure turbine passes through moisture separator units before being admitted to the low pressure turbines. The low pressure turbines exhaust steam is condensed in the main condenser, which also provides system deaeration. The condenser is followed by a full flow demineralizer system through which all condensate and makeup must pass before entering the feedwater heaters.

The demineralizer system removes corrosion products produced in the turbine, condenser, and feedwater piping. It also protects the reactor against condenser tube leaks and removes other sources of impurities which may enter the system in the makeup water. The turbine cycle uses a conventional regenerative feedwater system. The feedwater temperature and the number of feedwater heaters are selected in accordance with normal power plant considerations of turbine cycle performance and economics.

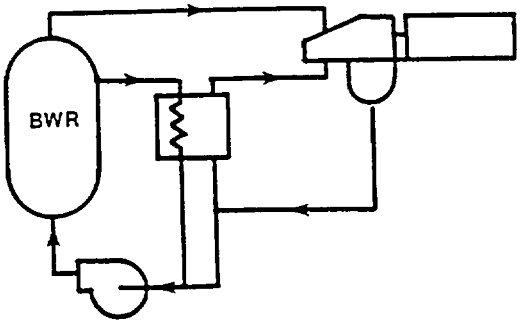
1.4.1 Forced Circulation BWRs

Power density in a BWR core may be increased by using a mechanical pumping system to force the water through the core. This is called a forced circulation BWR. In this design a portion of the coolant in the annulus area between the core shroud and vessel wall is taken outside the vessel in recirculation loops, where it is increased in pressure by means of recirculation pumps. Water at increased pressure is pumped from the two recirculation loops back into the bottom of the reactor pressure vessel via jet pumps. Flow orificing of the fuel support pieces provides desired flow distributions. Water enters the core through the fuel assembly nosepieces and passes upward inside the channels containing the fuel bundles, where it is heated to become a two phase, steam-water mixture. The steam-water mixture leaves the top of the fuel assemblies and enters a plenum area above the core which directs the flow into the steam separators. Here the water is separated by centrifugal action. The rejected water is returned to recirculate through the pumping system. The steam then passes through a dryer where the last traces of water are removed. Dry steam exits through steam outlet nozzles at the top of the vessel body. Feedwater is added to the system through thermally sleeved spargers located in the downcomer annulus. Here the feedwater joins the water rejected by the steam separators before entering the recirculation pumping system.

1.4.1.1 Forced Circulation BWR Control Systems

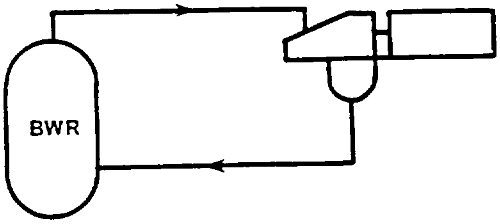
The fluid flow rates and reactivity level in a forced circulation direct cycle BWR require rigid control of steam flow from the reactor, of feedwater flow into the reactor, of recirculation flow through the reactor, and of control rod position. The design of the control systems considers conventional power generation objectives, such as reliability, ease of operation, and response times of the controlling parameter. Beyond the traditional power generation objectives, the control systems must incorporate features specific to reactivity control and nuclear plant safety. These considerations involve effects on moderator temperature, fuel temperature, and void content as a function of steam pressure; steam generation and feedwater input; fuel exposure; and automatic shutdown of the nuclear chain reaction during unsafe or potentially unsafe conditions.

DRESDEN 1 (BWR/1)



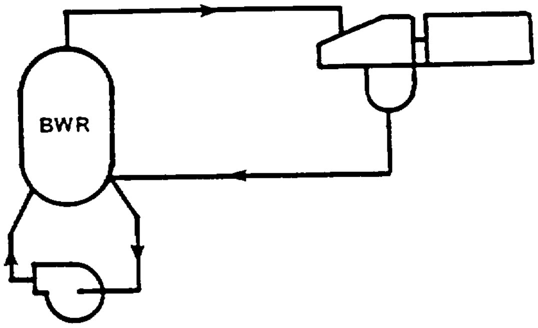
FORCED CIRCULATION, DUAL CYCLE

HUMBOLDT BAY (BWR/1)



NATURAL CIRCULATION, DIRECT CYCLE

BIG ROCK POINT (BWR/1)
OYSTER CREEK (BWR/2)
DRESDEN 2&3 (BWR/3)
BROWNS FERRY (BWR/4)
LASALLE 1&2 (BWR/5)
GRAND GULF 1&2 (BWR/6)



FORCED CIRCULATION, DIRECT CYCLE

FIGURE 1.4–1 BOILING WATER REACTORS IN VARIOUS SYSTEMS

1.5 PRODUCT LINES

Different product lines or classes of BWRs are designated numerically. There are currently six product lines, BWR/1, 2, 3, 4, 5, and 6. The following listing gives some basic information on the different product lines.

GE/BWR PRODUCT LINES

Product Line Number	Year of Introduction	Characteristic Plants and Their Features
BWR/1	1955	Dresden 1, Big Rock Point, Humboldt Bay, KRB - Initial commercial BWRs - First internal steam separation
BWR/2	1963	Oyster Creek - The first turnkey plant - Elimination of dual cycle
BWR/3	1965	Dresden 2 - The first jet pump application - Improved emergency core cooling systems (ECCS)
BWR/4	1966	Browns Ferry - Increased power density 10%
BWR/5	1969	LaSalle - Improved Recirculation System performance - Improved ECCS performance - Mark II Containment
BWR/6	1972	Grand Gulf - Improved core performance - Improved rod control systems - Mark III Containment

This manual is written to the BWR/4 product line. Table 1.5-1 gives a listing of BWR/2 plants through BWR/6s along with some basic information on each plant.

Plant Name	BWR Class	Product Line	Containment *Type	MWT	MWE	Power Density KW/L	N0. Fuel Bundles	N0. Control Rods	IC or RCIC	LPCI or RHR *4	Feed Pump Type	Bypass Capacity	RFC Type	HPCI or HPCS	N0. Relief Valves	Press. Control Initial / Backup	*Type Offgas
Oyster Creek	2	63	Mark I	1930	620	33.6	560	137	IC	-	M	45	MG *5	FWCI	4	EPR/ MPR	AMB
Nine Mile Point #1	2	63	Mark I	1850	620	34.0	532	129	IC	-	M&T	45	MG *5	FWCI	6	EPR/ MPR	Low Temp
Dresden #2	3	65	Mark I	2527	809	41.08	724	177	IC	LPCI	M	45	MG	HPCI	5	EHC/ EHC	AMB
Millstone	3	65	Mark I	2011	690	40.08	580	145	IC	LPCI	M	100 *(3)	MG	FWCI	3	EPR/ MPR	Low Temp
Dresden #3	3	66	Mark I	2527	809	41.08	724	177	IC	LPCI	M	45	MG	HPCI	5	EHC/ EHC	AMB
Monticello	3	66	Mark I	1670	545	40.6	484	121	RCIC	RHR (A)	M	15	MG	HPCI	4	EPR/ MPR	Comp
Quad Cities 1/2	3	66	Mark I	2511	809	40.9	724	177	RCIC	RHR (A)	M	45	MG	HPCI	5	EHC/ EHC	AMB
Pilgrim	3	66	Mark I	1998	655	40.5	580	145	RCIC	RHR (A)	M	26	MG	HPCI	4	EPR/ MPR	AMB
Brown's Ferry 1/2/3	4	67	Mark I	3293	1065	50.7	764	185	RCIC	RHR (A)	T	26	MG	HPCI	13	EHC/ EHC	AMB
Vermont Yankee	4	67	Mark I	1593	514	51.0	368	89	RCIC	RHR (A)	M	100*(3)	MG	HPCI	4	EPR/ MPR	AMB
Duane Arnold	4	67	Mark I	1658	538	51.0	368	89	RCIC	RHR (A)	M	26	MG	HPCI	6	EHC/ EHC	AMB
Peach Bottom 2/3	4	67	Mark I	3293	1065	50.7	764	185	RCIC	RHR (A)	T	26	MG	HPCI	11	EHC/ EHC	Comp
Cooper	4	67	Mark I	2381	778	50.6	548	137	RCIC	RHR (A)	T	26	MG	HPCI	8	Digital (W)	Low Temp
Hatch 1/2	4	67	Mark I	2436	786	51.2	560	137	RCIC	RHR (A)	T	26	MG	HPCI	9	EHC/ EHC	AMB
Brunswick #1	4	67	Mark I (C)	2436	821	51.2	560	137	RCIC	RHR (A)	T	100 *(3)	MG	HPCI	9	EHC/ EHC	AMB
Brunswick #2	4	67	Mark I (C)	2436	821	51.2	560	137	RCIC	RHR (A)	T	26	MG	HPCI	9	EHC/ EHC	AMB

Table 1.5-1 Plant Listing (Part 1)

Plant Name	BWR Class	Product Line	Containment *Type	MWT	MWE	Power Density KW/L	N0. Fuel Bundles	N0. Control Rods	IC or RCIC	LPCI or RHR *4	Feed Pump Type	Bypass Capacity	RFC Type	HPCI or HPCS	N0. Relief Valves	Press. Control Initial / Backup	*Type Offgas
Fitzpatrick	4	67	Mark I	2436	821	51.2	560	137	RCIC	RHR (A)	T	26	MG	HPCI	9	EHC/ BHC	AMB
Enrico Fermi	4	67	Mark I	3293	1093	50.0	764	185	RCIC	RHR (A)	T	26	MG	HPCI	11	EHC/ EHC	AMB
Hope Creek	4	67.5	Mark I	3293	1067	50.7	764	185	RCIC	RHR (C)	T	26	MG	HPCI	11	EHC/ BHC	AMB
Susquehanna 1/2	4	67.5	Mark II (C)	3293	1050	50.0	764	185	RCIC	RHR (A)	T	25	MG	HPCI	16	EHC/ EHC	AMB
Shoreham	4	67.5	Mark II (C)	2436	821	50.0	560	137	RCIC	RHR (A)	T	26	MG	HPCI	9	EHC/ BHC	AMB
Limerick 1/2	4	67.5	Mark II (C)	3293	1065	50.7	764	185	RCIC	RHR (C)	T	26	MG	HPCI	11	EHC/ BHC	AMB
LaSalle 1/2	5	69	Mark II (C)	3293	1078	50.0	764	185	RCIC	RHR (B)	T	25	Valve	HPCS	11	EHC/ BHC	AMB
Hanford #2	5	69	Mark II	3323	1100	50.0	764	185	RCIC	RHR (B)	T	25	Valve	HPCS	18	Digital (W)	Low Temp
Nine Mile Point #2	5	69	Mark II (C)	3323	1100	50.0	764	185	RCIC	RHR (B)	T	25	Valve	HPCS	18	EHC/ BHC	AMB
Grand Gulf	6	72	Mark III (C)	3835	1306	54.1	800	193	RCIC	RHR (B)	T	35	Valve	HPCS	20	EHC/ BHC	Low Temp
Perry	6	72	Mark III	3579	1200	56.0	748	177	RCIC	RHR (B)	M/T	35	Valve	HPCS	19	EHC/ BHC	Low Temp
Clinton	6	72	Mark III (C)	2894	995	52.4	624	145	RCIC	RHR (B)	M/T	35	Valve	HPCS	16	EHC/ BHC	Low Temp
River Bend	6	72	Mark III	2894	995	52.4	624	145	RCIC	RHR (B)	M	10	Valve	HPCS	16	EHC/ BHC	Low Temp

Notes: 1. Mark I : Drywell - Torus (Freestanding Steel Pressure Vessel)
Mark I (C): Drywell - Torus (Concrete with Steel Liner);
Mark II: Over/Under (Freestanding Steel Pressure Vessel);
Mark II (C): Over/Under (Concrete With Steel Liner);
Mark III Drywell - Containment (Freestanding Containment Vessel;
Mark III (C) Drywell - Containment (Concrete Containment With Steel Liner).

2. AMB Recombiner - with Charcoal Beds at Ambient Temperature
Low Temp Recombiner - with Charcoal Beds at Low Temperature(Zero degree f)
Comp · Recombiner - with Comperssor and Storage Tanks for Extended Gas Holdup
3. Equipped with a Select Rod Insert Function
4. (A) Equipped with 2 RHR Loops, LPCI Mode Injects to the Recirculation System
(B) Equipped with 3 RHR Loops, LPCI Mode Injects Directly to the Vessel
(C) Equipped with 4 RHR Loops, LPCI Mode Injects Directly to the Vessel
5. Equipped with 5 Recirculation Loops and No Jet Pumps

Table 1.5-1 Plant Listing (Part 2)

1.6 PLANT LAYOUT

Modern BWR facilities are multiple-unit plants. Greater economy can be realized with this arrangement by sharing certain functions within the facility. The principal buildings and structures associated with each unit of a particular site include a containment building (reactor building), a turbine building, a common control building, a diesel generator building, a common radwaste building, a common intake structure, and natural (or forced) draft cooling towers. Common structures are also provided which house the administrative offices, machine shop and guard house. Location and orientation of typical buildings on site are shown in Figure 1.6-1.

1.6.1 Containment Building

Boiling water reactors use multi-barrier pressure suppression type containments, consisting of a Primary and a Secondary Containment.

The Primary Containment consists of two structures; the drywell which encloses the reactor vessel, a pressure suppression chamber which stores a large volume of water, and a connecting vent system between the drywell and suppression chamber. Their function is to contain the energy and radioactivity released during a loss of coolant accident.

The Secondary Containment or reactor building surrounds the drywell and pressure suppression chamber. It forms a second barrier around the reactor vessel to further impede the release of radioactivity. The reactor building also houses the necessary reactor support systems and the Emergency Core Cooling Systems.

1.6.2 Turbine Building

The turbine building houses the turbine generator, auxiliary systems and balance of plant equipment.

1.6.3 Control Building

The control building is a multistoried structure which houses; the main control room plus control and electrical systems required for safe operation of the plant.

1.6.4 Diesel Generator Building

The diesel generator building contains the emergency diesel generators and their associated equipment in individual rooms within the building.

1.6.5 Radwaste Building

The radwaste building houses various systems provided to process liquid, solid, and gaseous radioactive wastes generated by the plant.

1.6.6 Intake Structure

The intake structure house the equipment providing the heat sink for the plant.

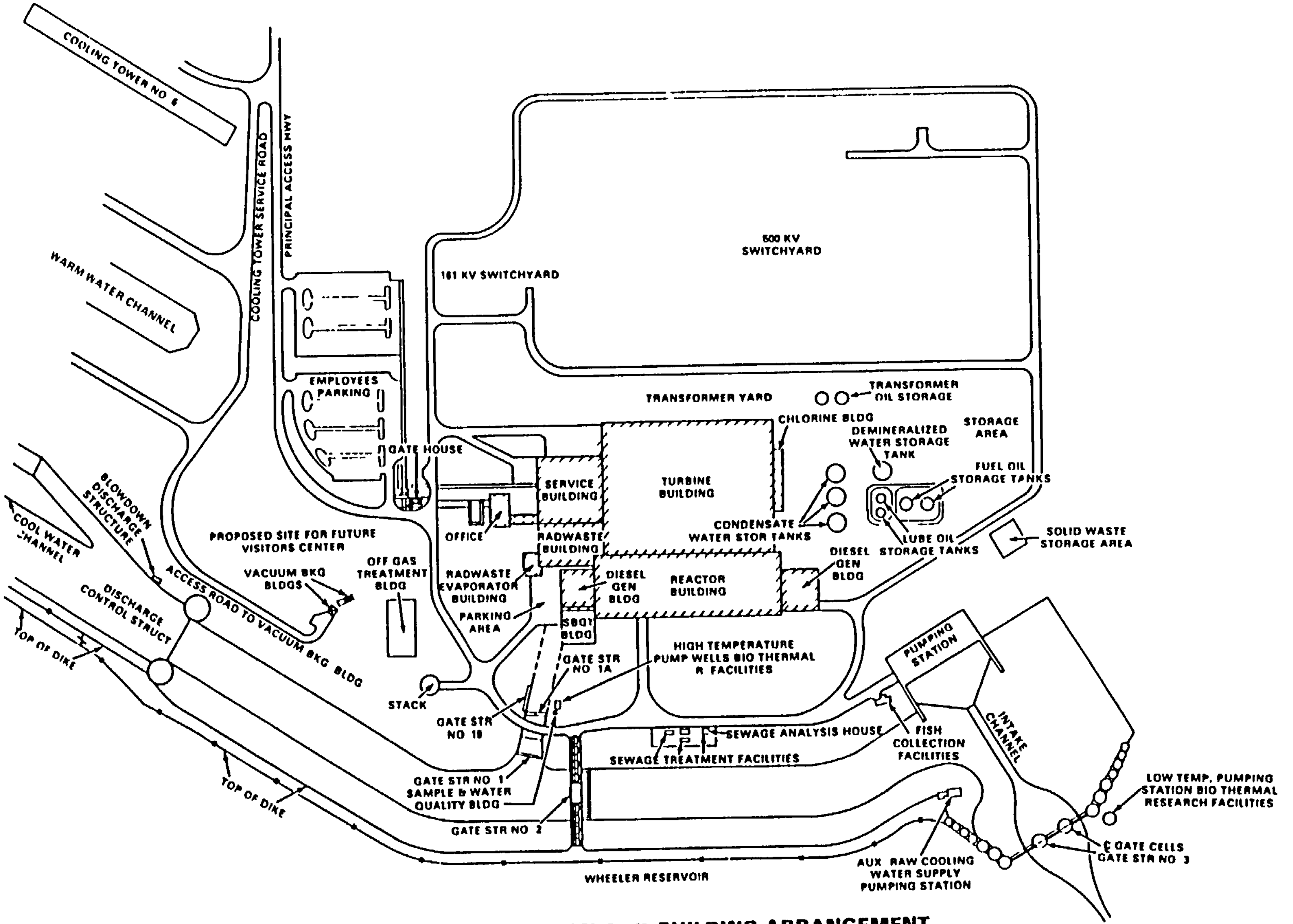


FIGURE 1.6-1 SITE PLAN AND BUILDING ARRANGEMENT

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1.7 REACTOR PHYSICS

Learning Objectives:

1. List the three BWR coefficients of reactivity and state how and why they change with core life and temperature.
2. Define the following reactor physics terms:
   a. Reactivity
   b. K_{eff}
   c. Reactor Period
   d. Shutdown Margin
3. Describe the response of fission product poisons xenon and samarium to changes in reactor power.
4. Describe the initial plant response (via the reactivity coefficients) to changes in significant plant parameters.

1.7.1 Introduction

The purpose of this section is to provide a basic understanding of certain reactor physics concepts relating to BWR technology.

The operation of a nuclear reactor is based upon the successful control of the response of the neutron flux, and hence reactor power, to time dependent changes. Startup and shutdown, as well as power level changes are typical operations which involve transient conditions. Factors which affect the reactor behavior include control rod movements, fuel depletion, fission product poisoning, temperature and pressure changes, and changes in coolant density and density distributions. These, and other concepts important to BWR operation, are discussed in the paragraphs that follow.

1.7.2 Neutron Cycle

The neutron cycle is taken as a complete sequence of steps necessary to produce the neutron chain reaction. To operate at a constant power level, the reactor must be able to sustain the chain reaction. Each neutron in the previous generation must produce a neutron in the next generation. The principle of operating a nuclear reactor is based on neutron economy, which is the accounting of the number of neutrons between successive generations.

Many of the processes within a reactor compete for the neutrons. In the case of a thermal reactor, such as a BWR, thermal neutrons are required to initiate fission; and in turn the fast neutrons produced by the fission reaction lead to the regeneration of thermal neutrons. Events that are included in a neutron cycle are shown in Figure 1.7-1.

1.7.2.1 Multiplication Factor

The expression which describes all the events in the life of a neutron and effectively describes the state of a reactor (critical, subcritical, or supercritical) is called the multiplication factor.

The multiplication factor in a nuclear system is a measure of the change in the fission neutron population from one neutron generation to the subsequent generation. If the multiplication factor for a reactor core (or any nuclear assembly) is less than 1.0, then the system is decaying or dying out and will never be self sustaining. With a multiplication factor greater than 1.0, a nuclear system is producing more neutrons than are being used by the system and is subjected to an increasing chain reaction that must be controlled by some exterior factor.

The stable or useful condition of a nuclear system occurs when the multiplication factor is equal to 1.0.

In a core of infinite extent, there can be no leakage. For such a core the infinite multiplication factor, K_∞, is the ratio of the number of neutrons resulting from fission in each generation to the number of neutrons absorbed in the preceding generation.

In a system of finite size, however, some neutrons are lost by leaking out, and the multiplication factor is called the effective multiplication factor, K_{eff}. K_{eff} is defined as the ratio of the number of neutrons resulting from fission in each generation to the total number lost by both absorption and leakage in the preceding generation. The effective multiplication factor is the product of the nonleakage probability, P, and the infinite multiplication factor.

$$K_{eff} = K_\infty P$$

In reactor operation, K_{eff} is the most significant property with regard to reactor control. At any specific power level or condition of the reactor, K_{eff} is kept as near to the value of 1.0 as possible. At this point in operation, the neutron balance is kept to exactly one neutron completing the life cycle for each original neutron absorbed in the fuel (Figure 1.7-1). When a reactor is operating at a constant power level or condition, the effective multiplication factor is defined in equation form as:

$$K_{eff} = \varepsilon L_f p L_t f \eta$$

These symbols are defined and considered in detail in the following paragraphs.

Fast Fission Factor (ε)

The fast fission factor, ε, is the contribution to neutron multiplication from the fissions that occur at higher than thermal energies. This contribution is from the fast fission in U-235 and U-238. The probability for a fission reaction in U-238 is quite low, but there is so much of this isotope in the reactor core that there is a contribution to the multiplication factor. The fast fission factor is defined as the ratio of the neutrons produced by fissions at all energies to the number of neutrons produced in thermal fission.

Physically,ε is a function of physical spacing of fuel rods, the size of the fuel rod, the moderator and the amount of U-238 and Pu-239 in the core. The idea here is that the longer the neutron remains at a high energy, or the greater the number of target nuclei, the greater the probability of a fast fission. The value of ε is slightly >1.

Fast Nonleakage Factor (L_f)

The fast nonleakage factor, L_f, is the fraction of neutrons that are not lost because of leakage from the core during the slowing down process from fission energies to thermal energies. It is also the probability that a neutron will remain in the core and become a thermal neutron without being lost by fuel leakage.

Physically, L_f is a function of the moderator, core size, and shape. A good moderator presents a large target for collision, decreasing the probability that the neutron will escape. A large core allows more moderator and fuel, thus decreasing the likelihood of leakage.

The value of L_f is slightly less than 1. It is the exact analog of Lt, the thermal nonleakage factor.

Resonance Escape Probability (p)

The resonance escape probability, p, is the probability that a neutron will be slowed to thermal energy and will escape resonance capture. It is also the fraction of neutrons that escape capture during the slowing down process. It is always less than 1.0 when there is any amount of U-238 present in the core, which means that high energy capture by U-238 always removes some of the neutrons.

As the reactor temperature increases, the resonance escape probability decreases in value because of the decrease in the ratio of the water moderating atoms to fuel atoms and the broadening of the resonance capture cross sections. Physically, p is a function of the amount of resonance absorbers in the core and the neutron energy spectrum. Because the amount of resonance absorbers is dependent on enrichment and core age, p also varies with a change in either parameter.

Thermal Nonleakage Factor (L_t)

This factor, L_t, is the fraction of the thermal neutrons that do not leak out of the core during thermal diffusion but remain to contribute to the chain reaction. L_t is also the probability that a thermal neutron will remain and be used in the core.

The value of L_t decreases as the temperature of the core increases because as temperature is increased, the values of absorption cross sections decrease, and the thermal neutrons must travel further before being absorbed.

Physically, L_t like L_f, is a function of the moderator, core size, and core shape. The value of L_t is slightly less than 1.

Thermal Utilization Factor (f)

The thermal utilization factor, f, is the ratio of the probability that a neutron will be absorbed in the fuel to the probability that the neutron will be absorbed in all the material that makes up a core. It is described by the following equation:

$$f = \frac{a \sum^{\text{fuel}}}{a \sum^{\text{fuel}} + a \sum^{\text{other}}}$$

Where Σ_a = macroscopic absorption cross section; the combination of the capture cross section, Σ_c, and the fission cross section, Σ_f, ($\Sigma_a = \Sigma_c + \Sigma_f$).

Physically f is an inverse function of the ratio of the nonfissionable absorbers number densities to the fuel number density. Since the fuel number density decreases slowly over core life, as the absorbers increase, f decreases. The absorbers are functions of core power history, core age, and power level.

Neutron Production Factor (η)

The neutron production factor, η, is the average number of neutrons produced per thermal neutron absorbed in the fuel. Physically, η is a function of enrichment and core age since enrichment varies with core age.

The numerical value of η does not change with core temperature over the range considered for most reactors. There is essentially no change in over the lifetime of the reactor core because of the closeness of the values of η for U-235 and Pu-239.

As the reactor operates for a period of time, and Pu-239 begins to contribute to the neutron economy of the core, the average effect of is expressed by:

$$\eta = \frac{v_{235}\,\Sigma_f^{235} + v_{239}\,\Sigma_f^{239}}{\Sigma_a^{235} + \Sigma_a^{239} + \Sigma_a^{238}}$$

where v is the number of neutrons per fission. A representative value for η is greater than 1.

1.7.2.2 Terms Related to K_{eff}

$K_{excess} = K_{eff} - 1$, and is generally used to describe how much extra fuel is loaded into the core for increased core life.

The shutdown margin (SDM) is the amount by which the reactor is subcritical. Mathematically, this can be expressed as:

$$SDM = 1 - K_{eff} \text{ (for } K_{eff} < 1)$$

The design shutdown margin will be specified for a plant, and tests are conducted periodically to demonstrate that it is met. The value specified normally assumes that the strongest rod is stuck out in the fully withdrawn condition. These tests demonstrate a margin of safety should that event occur.

1.7.3 Reactivity and Factors Affecting Reactivity

When discussing changes in neutron population in a reactor, it is convenient to describe neutron population changes by changes in reactivity. Reactivity is defined as the fractional change in neutron population per neutron generation. If N_0 neutrons exist in one generation, then $N_0 K_{eff}$ neutrons are produced in the next generation. The change in neutron population is $(N_0 K_{eff} - N_0)$ and the fractional change or reactivity $(\Delta K/K)$ is

$$\frac{N_0 K_{eff} - N_0}{N_0 K_{eff}}$$

therefore,

$$\rho = \frac{N_0 (K_{eff} - 1)}{N_0 (K_{eff})} = \frac{(K_{eff} - 1)}{K_{eff}}$$

Beginning with an all rods in condition in a cold clean core, enough reactivity must be added to bring the reactor initially critical. With the reactor critical, $K_{eff} = 1$ and the reactivity is zero. Anything that will cause a change in one of the parameters in the equation for K_{eff} will cause a change in neutron population and a change in reactivity.

The factors that affect the reactivity of a reactor core throughout core life are fission product poisons, fuel depletion, moderator temperature, fuel temperature, steam void fraction, reactor pressure, control rods, and burnable poisons.

Each of these factors must be accounted for in the design of the core so that criticality and full power can be attained and maintained throughout the design lifetime of the core.

1.7.3.1 Reactivity Coefficients

Reactivity coefficients are a means of describing the effect of the multiplication factor (K_{eff}) as a result of positive unit changes in a particular reactor parameter. They are usually expressed in terms of $\Delta K/K$/unit change of parameter variable. There are three such coefficients at work in a BWR operating at power. These are the moderator temperature coefficient, the moderator void coefficient, and the Doppler (fuel temperature) coefficient.

Moderator Temperature Coefficient ($\propto_T$)

The moderator temperature coefficient of reactivity is defined as the change in reactivity produced by a unit change in moderator temperature ($\Delta K/K/\Delta^0$ F). As the temperature of the moderator increases, it becomes less dense. This decreases the amount of neutron moderation and increases the probability that a neutron may leak from the core or undergo nonfission absorption in a control rod or some core structural material. The effects of moderator temperature and fuel burnup on the coefficient are shown in Figure 1.7-2.

The negative slope of the moderator temperature coefficient curves arises from the fact that moderator density varies non-linearly with moderator temperature as illustrated in Figure 1.7-3. The change in density per °F change in temperature increases with increasing temperature. This results in the coefficient becoming more negative with increasing moderator temperature.

Several factors affect the moderator temperature coefficient of reactivity as core burnup increases. The buildup of Pu-240, a strong resonant absorber, increases the chance of resonance absorption and makes the coefficient more negative. During the core life, control rods are withdrawn to balance the decrease in reactivity because of fuel burnup. This causes the effective size of the core to become larger. Decreased fast and thermal leakage tend to make the coefficient less negative. Since local steam voids cause an increase in the distance a neutron travels after it is thermalized, the presence of control rods, which absorb thermal neutrons, makes the coefficient more negative. Consequently, when control rod density (all rods fully inserted = 100% control rod density) decreases at higher burnups, the net effect is that the moderator temperature coefficient becomes less negative.

In a BWR, the moderator temperature coefficient is designed to be negative by establishing a proper moderator to fuel ratio (Figure 1.7-4). However, at low temperatures toward the end of core life, the negative effects of resonance absorption and neutron leakage may not be sufficient to overcome the positive effects of thermal utilization and control rod withdrawal and can result in a small positive temperature coefficient. This, however, is not a safety consideration because the coefficient is small and becomes negative at higher temperatures.

Moderator Void Coefficient ($\propto_v$)

As the moderator boils and the void fraction increases, the moderator density decreases resulting in a reactivity change involving several interacting mechanisms. The void coefficient of reactivity ($\propto_v$) defined as the change in reactivity produced by a unit change in void fraction ($\Delta K/K/\Delta\%V$) is illustrated in Figure 1.7-5. A decrease in moderator density results in less absorption in the moderator and the thermal utilization factor increases. An increased distance between moderator molecules means that the neutrons travel a greater distance while at higher energies and have a greater probability of being captured in resonance regions of uranium or plutonium; thus, the resonance escape probability decreases. In addition, the distance a neutron travels after thermalizing also increases, producing a decrease in thermal nonleakage probability. The large negative effects of increased resonance absorption and, to a lesser degree, increased thermal leakage outweigh the positive effect of less absorption in the moderator and result in a coefficient of reactivity which is strongly negative.

The slopes of the curves in Figure 1.7-5 are negative because resonance capture increases more rapidly at high void fraction than at low void fraction. As previously explained, the increased resonance capture results from the decrease in moderation as the water boils.

Consider two examples: 10% void fraction and 70% void fraction. At 10% void fraction (90% water fraction), a 1% increase in void fraction decreases water fraction from 90% to 89% or roughly 1.1%.

At 70% void fraction (30% water fraction), however, a 1% increase in void fraction decreases the water fraction from 30% to 29% or a change of 3.45%. Thus, at higher void fraction for a percent change in void fraction, there is a greater percent change in moderator fraction than (for a percent change in void fraction) at low void fraction. Therefore, the negative reactivity contribution at higher voids is greater than at lower voids.

The behavior of the void coefficient as core burnup increases is similar to the moderator temperature coefficient. However, the density decrease when voids increase is much larger than the density decrease when moderator temperature increases. Therefore, the void coefficient is more negative than the moderator temperature coefficient. This makes the exact behavior of the void coefficient hard to predict. Depending on core design and exposure, the void coefficient may increase, decrease, or remain approximately constant. In general, the effects of control rod withdrawal usually dominate and the void coefficient becomes less negative as core burnup increases.

Doppler Coefficient of Reactivity (α_D)

The final reactivity coefficient that is of primary importance in safety considerations is the Doppler coefficient of reactivity which is defined as the change in reactivity that results from a unit change in fuel temperature ($\Delta K/K/\Delta\ ^{0}F$) and which accounts for the prompt negative reactivity addition which acts to terminate a power increase in the event of a reactivity excursion. The coefficient is negative, as shown in Figure 1.7-6, because an increase in fuel temperature results in an increase in resonance absorption. To examine the reason for this increase, Doppler broadening must first be discussed.

A resonance peak is a narrow band of neutron energy in which the neutron capture cross section within that band of energy is considerably higher than at other neutron energies. As the temperature of the fuel changes, the thermal motion of the fuel nuclei changes. As fuel temperature increases, the energy band of the resonance broadens and the cross section of the peak decreases as shown in Figure 1.7-7. This broadening is referred to as Doppler broadening. Doppler broadening shifts the energy at which the neutrons are absorbed to higher or lower values depending on the direction of motion of the fuel nuclei with respect to that of the incident neutron. The nuclear properties of the nucleus, however, are not affected; thus, the total cross section over the entire energy band remains constant. Therefore, although the total amount of absorption around a resonance peak does not change (the area under the peak in Figure 1.7-7 is constant), the energy band of the resonance is broadened.

It may appear then that Doppler broadening would have no effect on resonance absorption in a reactor. This is not the case for nuclides that are present in sufficient concentration because of an effect called self shielding. At low temperatures, the resonance peak is very narrow and the capture cross section is very high, resulting in a very large reduction in neutron flux in that energy band in the outer layer of the fuel rod. The interior of the rod sees very little flux in that energy band, so there is very little absorption toward the center of the fuel pin. Thus, the interior of the rod is shielded from the flux of the proper energy by the outer layers of fuel atoms.

Now, consider an increase in fuel temperature. The energy band is wider for neutron absorption in the central higher regions of the rod. The broadened areas of absorption are not shielded by the less broadened fuel surface and the shielding resonance absorption in the fuel rod increases. So one can see that negative Doppler reactivity associated with an increase in fuel temperature arises from the combined effect of Doppler broadening and the fact that a BWR is a heterogenous reactor with lumped fuel.

The slope of the curves in Figure 1.7-6 is positive because as the temperature rises, the broadening of the resonance is less, and it follows a 1/v behavior. In a BWR the Doppler coefficient of reactivity is always negative and always adds negative reactivity when the fuel temperature rises. The Doppler coefficient is more negative at greater void fraction because of the increase in resonance capture with steam voids.

As core age increases, the buildup of Pu-240 in the fuel results in an increase in the total resonance absorption cross section. This will increase self shielding so a reduction in self shielding has a greater effect on the total amount of resonance absorption. Thus, the Doppler coefficient of reactivity becomes more negative with increased burnup as shown in Figure 1.7-8.

Reactivity Coefficient Values

Approximate numerical values for the three reactivity coefficients are as follows:

$\propto_V$ -1 x 10-3 ΔK/K per Δ% voids

$\propto_T$ -1×10^{-4} ΔK/K per Δ °F moderator

$\propto_D$ -1×10^{-5} ΔK/K per Δ °F fuel

From these values it is easy to see that the void coefficient is dominant when the reactor is in the power range (i.e., a substantial percentage of voids).

Reactivity Coefficients - Summary

In summary, this discussion has been a simplified explanation of how various factors affect the reactivity balance in a BWR. Realistically, of course, many of the mechanisms causing change are constantly interacting with other mechanisms. For example, under normal operating conditions, an increase in fuel temperature as a result of control rod withdrawal or a core flow increase is always accompanied by an increase in moderator temperature or void fraction.

1.7.4 Reactor Control

The reactor power level is directly related to the neutron population. The flux is related to neutron level by:

$$\phi = nv,$$

where n is the neutron density (neutrons/cm^3) and v is the neutron velocity (cm/sec).

The velocity is a constant since the discussion concerns only the thermal neutron flux for fissioning.

The reaction rate or power is related to the flux by:

$$\text{power} = \frac{\Sigma f \ \phi}{\text{conversion constant}}$$

The parameters Σ_f, v, and the conversion constant are essentially constant during any given time. Therefore, the power and neutron population are directly related.

With this in mind,

$$P = P_o e^{t/T}$$

is the expression which describes how reactor power varies with time. This means that power increases exponentially for a stable period (T). Period (T) is defined as the time in which neutron flux level changes by a factor of the natural log base (e).

The following sections apply the principles covered above to explain reactor control during a normal startup from a cold shutdown condition.

1.7.4.1 Source Range

The source range covers approximately 10^{-8} % to 10^{-4} % reactor power. The power is controlled by control rod withdrawal to establish the reactor in the critical condition. Further control rod withdrawal then establishes a rate of power increase into the intermediate range to commence a plant heat up.

The operator verifies, before withdrawing any control rods to begin the startup, that there is some minimum count rate indicated on the source range monitor. As he withdraws control rods, a nonfission absorber is removed from the core so that the thermal utilization factor increases, causing K_{eff} to increase. The source range monitor indications show a slow increase in count rate. The neutrons being counted are coming from an incore neutron source and from the phenomenon of an increasing or steady state count rate while subcritical which is subcritical multiplication.

Sources

Sources are present in the reactor so that it is possible to see the approach to critical on the reactor instrumentation. The neutron multiplication can be seen on the instrumentation as the control rods are withdrawn. If there were no sources present, the instrument range would not be sensitive enough to detect a positive, decreasing period until the power was high enough to indicate on the instrumentation. By the time the power was indicating on the instruments, the period could be very short and a startup accident could have occurred.

Natural sources include the photoneutron source ($\gamma + {}^2H_1 \rightarrow {}^1H_1 + {}^1n_0$), spontaneous fission of uranium and plutonium and an alpha reaction with oxygen-18 (${}^4He_2 + {}^{18}O_8 \rightarrow {}^{21}Ne_{10} + {}^1n_0$). These sources are not significant during an initial startup so additional sources must be installed in the core.

The operational source used in an irradiated antimony source in a beryllium sleeve. This gives about 3 x 10^5 n/s/curie (see Section 5.1 for more details). Other sources can be Po-Be, Ra-Be, Pu-Be, or Am-Be.

Subcritical Multiplication

In the subcritical condition, excessive numbers of neutrons are lost by leakage or absorption per generation, so that the chain reaction with fission neutrons is not self sustaining. If left alone, the power level of the core would eventually drop to intrinsic levels which are below the range of indication of the SRM. This is an undesirable condition and to prevent it, the incore neutron sources (described above) are used. Basically (Figure 1.7-9), the neutron source makes up for the excessive loss of fission neutrons. With source neutrons added the chain reaction can be self sustaining with $K_{eff} < 1$, except that it is not dependent solely on fission neutrons. Also a steady state power level can be reached that is within the indication band of the SRM.

Remember that

$$K_{eff} = n_1 / n_0 ; \quad \text{therefore,} \quad n_1 = n_0 K_{eff}$$

If, at any given time, a source could be placed into a subcritical reactor, the instantaneous neutron level would be:

$$n_0 = S_0$$

Assuming that the source put out So neutrons in each generation, the level after one generation would be:

$$n_1 = S_0 + S_0 K_{eff} = S_0(1 + K_{eff}),$$

where S_0 represents the neutrons put out by the source in the next generation and S_0K_{eff} represents those left over from the initial generation.

After the next generation, it is

$$n^2 = S_0 + n_1K_{eff} = S_0(1 + K_{eff} + K_{eff}{}^2),$$

and so on until

$$n_n = S_0(1 + K_{eff} + K_{eff}{}^2 + K_{eff}{}^3 + ... + K_{eff}{}^n).$$

This series converges to the following expression:

$$n = \frac{S_0}{1 - K_{eff}}$$

This will not tell the time it takes to get there, but does give the final level. Clearly, this no longer works when $K_{eff} = 1$.

As an example, take the case in which $K_{eff} = 0.5$ and $S_0 = 100$. A table can then be generated:

Generation	Fission n Produced	Source n Produced	Total Fast n Produced	Fast n Gained
0	0	100	100	100
1	50	100	150	50
2	75	100	175	25
3	87	100	187	12
4	93	100	193	6
*	*	*	*	*
*	*	*	*	*
n	100	100	200	0

n = a large number of generations

Note the following items:

1. If at any time the source is removed, neutron level will decrease.
2. As long as the source is present, neutron level will increase, but by a smaller amount each generation.
3. After a large number of generations, neutron level should be substantially constant, with source neutron production equal to the difference between the number of neutrons that start the generation and the number reproduced by the fission process.

Now consider the new steady state count rate following a step increase in K_{eff}. The immediate response of the core to the K_{eff} step increase is that the number of neutrons lost per generation decreases. Since the number of source neutrons contributed per generation remains constant, the number of neutrons causing fission increases and power level increases. As power level increases, the number of neutrons lost per generation also increases until it again equals the source contribution per generation. At that time a new, higher, steady state power is achieved. Note also, that the rate of decrease of fast neutrons gained decreases slower. Therefore, it takes longer to reach the new steady state power level as K_{eff} gets closer to 1.

To carry the discussion through, next consider what happens when $K_{eff} = 1$. In this case the source neutrons are always in excess of the minimum number of neutrons required for the chain reaction to be self sustaining. The power increase per generation is what the source neutrons contribute.

In actual operation this power increase is very slow, and the fact that the reactor is critical is difficult to detect. In order to have a positive indication of criticality, control rods are withdrawn until the reactor is slightly supercritical. In this case, the operator sees a continuously increasing power level which is not confused with the level increases which were caused by subcritical multiplication. At this point the power level and its rate of increase are important parameters to be monitored by the operator.

Source Range - Summary

The reactor operator withdraws control rods until the reactor is slightly supercritical. Indications of this condition are a continuously increasing power level and a sustained positive period without control rod motion. The reactor operator then adjusts the period by control rod withdrawal to about 100 seconds and allows power to increase into the intermediate range where the plant heat up is started.

1.7.4.2 Intermediate Range

The intermediate range encompasses power levels from approximately 10^{-5} % to about 40 %. The power level is controlled by control rods and the negative feedback effect of the moderator temperature coefficient of reactivity. The purpose of power control here is to control the heat up rate of the plant to prevent undue thermal stresses on plant structural materials. This range of reactor operation is probably the most difficult for the operator. It is here that the reactor operates in the classic sense that is normally taught on test reactors. Here, it obeys the $P = P_o e^{t/T}$ expression.

Recall that the intermediate range is entered with the reactor supercritical. To the unwary operator this can be a problem. The rapidly increasing flux level can give a high level flux scram from the intermediate range monitors (Section 5.2). Attention must be paid to proper range switching.In this range, certain approximations can be used to estimate the period after a rod is pulled. Making appropriate substitutions and mathematical manipulations, yields the following:

$$\frac{n}{n_o} = e^{t/T} \quad \text{and} \quad T = \frac{t}{\text{Ln } n/n_0}$$

This expression gives reactor period for a given neutron increase in a given length of time. Using only the neutron increase and elapsed time, the period can always be determined provided the rate of increase is constant (step change in reactivity). Some quick methods of doing this are:

1. Time to increase by 10%:

$$\text{Ln } \frac{n}{n_0} = \text{Ln } \frac{1.1}{1} = \text{Ln } 1.1 \sim 0.095$$

$$T = \frac{t}{0.095} = t(10.5) \text{ or } \sim t(10)$$

2. Doubling time:

$$\text{Ln } \frac{2}{1} = \text{Ln } 2 \sim 0.69$$

$$T = \frac{t}{0.69} = t(1.45) \sim t(1.5)$$

Thus, it can be seen that power in this range does behave classically. Once criticality is confirmed by a constant period and increasing flux level (actually supercritical), the reactor can be put on a positive stable period.

Intermediate Range - Summary

Power increases to the point of adding heat, and moderator temperature increases. The resulting moderator density decrease adds negative reactivity, and net core reactivity becomes <0. This causes power to turn and begin to decrease. The operator now withdraws a control rod (or rods) until core reactivity is positive and the process is repeated. The rate at which this is done and the magnitude of positive reactivity controls reactor power, which in turn controls the heatup rate. This process is continued until the plant is at its operating temperature.

1.7.4.3 Power Range

For this discussion, the power range is considered to be a power level greater than 1%. The reactor still responds by the $P = P_o e^{t/T}$ expression in the power range. However, this response is difficult to recognize because it is impossible to establish a stable period. There is always some factor resisting a power increase or decrease. Therefore, an attempt to establish a positive period by control rod withdrawal is immediately terminated at some fractionally higher power because of increased voids, or fuel, or moderator temperature increase.

From 1% to approximately 25%, reactor power is controlled to establish plant conditions required for rolling the turbine and picking up initial load on the generator. From 25% to 100%, reactor power is controlled to control the generator load. Reactor control is accomplished through use of control rods and the Recirculation System. The negative reactivity coefficients related to the fuel temperature and core void fraction provide a negative feedback to power changes. The effect of the moderator temperature coefficient ($\propto_T$) is limited here because of the pressure and temperature relationship existing at BWR power range operating pressures and because of the pressure control system. Longer term reactivity changes are caused by fission product poisons (hours) and fuel depletion (weeks).

Control Rods

The purpose of control rods is to control reactor power or to shape thermal neutron flux for optimum fuel burnup and control of peaking factors. To accomplish this, the control rod contains boron-10 (B-10) and absorbs thermal neutrons according to the following reaction:

$$^{10}B_5 + ^1N_0 \rightarrow ^4He_2 + ^7Li_3$$

Relating this to K_{eff}, one finds the thermal utilization factor, f, changes most with control rod motion

$$f = \frac{\delta_a \text{ fuel}}{\delta_a \text{ fuel} + (N^{\text{control rod}} \ \delta a^{\text{control rod}} + ./N^{\text{fuel}})}$$

As the control rod is withdrawn from the core, N control rod decreases, and f increases. Conversely, control rod insertion causes f to decrease.

The effectiveness of a specific control rod for absorbing thermal neutrons is called control rod worth and is measured in units of reactivity ($\Delta K/K$). Two types of rod worths are generally considered.

The first is the differential rod worth (Figure 1.7-10), which is the reactivity per notch of control rod travel. By summing reactivity from all notches through one normal stroke of travel, one generates the integral rod worth curve (Figure 1.7-11). Using this curve, one can calculate reactivity from any consecutive group of notches between full in and full out.

Because an increase in either type of control rod worth causes an increase in the other, one often speaks of only rod worth and specifies the type only where significant.

The worth of a control rod is a direct function of the thermal neutron flux to which it is exposed. Several of the factors affecting control rod worth are discussed below.

1. Core position - Considering radial core position, control rods at the center of the core are exposed to a higher thermal flux than those at the core periphery and, therefore, have a greater worth.

Considering axial rod worth, the differential rod worth is highest as the control rod end travels through the peak axial thermal neutron flux. This peak changes because of voids, and this in turn causes the differential worth peak to shift.

2. Plant condition - cold to hot at 1% power: As the moderator temperature increases, neutron leakage from the fuel cell to the volume around the control rod increases. Thus the control rod is exposed to a higher thermal neutron flux, and the rod worth increases.

3. Plant condition - hot at 1% power to 100% power: During operation in this power band, voids form. The voids, occurring at areas of high thermal neutron flux (which are also areas of high power) depress the thermal neutron flux peak. Therefore, a control rod in this area is exposed to less flux than it would be without voids, and the control rod worth decreases.

4. Position of adjacent control rods - Figure 1.7-12 (upper) shows control rod positions at a given time. Rod A at this time is absorbing neutrons or controlling only the four adjacent fuel cells. Figure 1.7-12 (lower) shows the rod positions at some later time.

Now rod A is absorbing neutrons from more fuel than it was earlier, or its zone of control is increased. It follows then that the rod is exposed to a greater thermal neutron flux and its worth increases. The Rod Worth Mininizer System (Section 7.5) constrains the operator to prescribed control rod withdrawal and insertion sequences so that excessively high control rod worths are not generated in this manner.
Figure 1.7-13 is an example of the use of control rods for radial flux shaping. The nuclear engineer normally is responsible for recommending rod motion for this purpose.

Power Changes Using Recirculation Flow

Assume that the plant is on the 100% load line at 75% reactor power (Section 3.2.3). Recirculation flow is then increased to bring power to 95% (Figure 1.7-14). As the flow increases, voids are swept from the core faster, resulting in a lower void fraction and a positive net core reactivity. Also, the moderator temperature decreases slightly because the moderator is in less contact with the fuel. The response of the core to the positive reactivity is a power increase. The power increase causes the fuel temperature, the moderator temperature, and the void fraction to increase. This continues until the core net reactivity again equals zero. During this transient, the power increase starts immediately after the core net $\Delta K/K>0$; the Doppler coefficient is the first to add negative reactivity; and the magnitude of the reactivity from the void change is greater than that from Doppler or moderator temperature change.

The ability to change power by changing flow enables a BWR to undergo large power changes without disturbing the core power distribution. Power level can be changed at rates well in excess of those obtainable by control rod manipulation while maintaining thermal margins, etc.

Fission Product Absorbers

During the operation of a reactor, a large variety of fission products is being produced. Although most of the fission products have relatively low absorption cross sections, there are some which are very strong parasitic absorbers of thermal neutrons.

These are referred to as fission product absorbers or fission product poisons. Any change in the concentration of these nuclei changes the thermal utilization factor (f) and, therefore, inserts either positive or negative reactivity into the reactor. The concentration of these fission product absorbers depends on fission rate, decay rate of absorbers, decay rate of precursors, and neutron absorption by nuclides producing the fission products. Although the fission product absorbers may have a significant effect on reactivity available for operation, their impact on immediate operation is primarily on the resulting effects of modified ramp rates, power level drifts, or need for compensating control rod movements. Their transient effects are not of first order importance to the evaluation of reactivity transients pertinent to nuclear safety.

Iodine-Xenon Fission Product Absorbers

The most important fission product poison is xenon-135 (Xe-135), which has the exceptionally high absorption cross section of 2.7 x 10^6 barns for thermal neutrons at 68^0 F, an equilibrium reactivity poison effect of the order of -.03 $\Delta K/K$, and the potential for slow transient reactivity effects of the order of -.04 $\Delta K/K$. Because the Xe-135 concentration is dependent on power, it is useful to describe the production and removal mechanisms.

1. Xe-135 production:

Xe-135 has two production terms. One is a direct fission yield (0.2%). The other is an indirect fission yield (5.6%) which comes from the fission product tellurium (Te) and its decay to iodine (I). Iodine decays to xenon as indicated in the following reaction.

$$ {}^{135}Te_{52} \xrightarrow[19\text{ sec.}]{\beta-} {}^{135}I_{53} \xrightarrow[6.7\text{ hr.}]{\beta-} {}^{135}Xe_{54} $$

Because of the relatively short half life of tellurium, it is often dropped from a discussion of Xe-135.

2. Xe-135 removal:

Xe-135 has two removal mechanisms. One is the burnout term, which represents the absorption of a neutron by the Xe-135 as indicated in the following reaction:

$$ {}^{135}Xe_{54} + {}^{1}n_{0} \rightarrow {}^{136}Xe_{54} + \gamma $$

Xe-136 has a low absorption cross section. The other mechanism is radioactive decay as indicated in the following reaction:

$$ {}^{135}Xe_{54} \xrightarrow[9.2\text{ hr.}]{\beta-} {}^{135}Cs_{55} \xrightarrow[2.\text{x}10^{6}\text{ yr.}]{\beta-} {}^{135}Ba_{56}\ \text{(stable)} $$

The following diagram helps to develop a feel for xenon increase or decrease as a function of current and preceding operation:

burnout (+n)

$$ {}^{135}I_{53}\ \xrightarrow[6.7\text{ hr.}]{\text{formation}}\ {}^{135}Xe_{54} $$

decay (9.2 hr.)

From this diagram, one can associate the formation term largely with prior operation and the burnout term +n with current power level. For example, after some hours of operation, then sudden shutdown, the xenon simply cannot decay as fast (9.2 hr) as it is being formed (6.7 hr) for a while. Conversely, a sudden rise in power level equates to an increase in neutron flux (the burnout term) and, for a while, the xenon depletes faster than it can be formed from decay of the iodine backlog created at the lower power level.

When the reactor is first brought to power, the Xe-135 concentration (atoms/cm^3) is slowly built up to an equilibrium. This is due primarily to the relatively long half lives of I-135 (6.7 hr) and Xe-135 (9.2 hr). Because of the high thermal neutron cross section of Xe-135, as the concentration of the isotope increases, so does the macroscopic absorption cross section of the core increase. Operationally, as xenon builds up, other poisons in the core (control material such as control rods) must be removed, thus maintaining criticality. Provided one has enough control material to remove during this xenon buildup, after approximately 40 hours of power operation it reaches equilibrium, a point at which the production of Xe-135 is equal to the removal of Xe-135 by neutron capture plus the loss of Xe-135 through radioactive decay.

The equilibrium value for the xenon concentration is a function of the reactor's neutron flux level (power level). Since an absorber is added to the reactor when xenon is built into the system, its effect on the chain reaction can be described in terms of reactivity. The multiplication factor is lowered primarily through the decrease in the thermal utilization factor.

A change in power causes a transient in xenon concentration. At the end of the transient, which takes about 2 days, the xenon concentration reaches its new equilibrium, assuming that power is left constant after the change.

Suppose a reactor has reached an equilibrium xenon concentration and the reactor is then shut down. When a reactor is shut down, the thermal neutron flux is reduced essentially to zero.

The I-135 decays more rapidly ($t_{1/2} = 6.7$ hr) than does Xe-135 ($t_{1/2} = 9.2$ hr); therefore, after shutdown, the concentration of Xe-135 builds up to a peak. Peak xenon depends directly upon the concentration of Xe-135 and I-135 present in the reactor at the time of shutdown. Xe-135 concentration rises for a period of about 7 to 11 hours after shutdown (Figure 1.7-15). The exact time required to reach a maximum depends upon the Xe-135 and I-135 concentration before shutdown which, in turn, is dependent upon power history. By this time, much of the I-135 has decayed causing the rate of xenon production to be less than the rate of Xe-135 decay, and the Xe-135 concentration decreases.

The greater the power level before shutdown, the greater will be the concentration of iodine in the reactor at the time of shutdown and, thus, the greater will be the increase in xenon concentration after shutdown. If the value of the positive reactivity needed to overcome the negative reactivity due to peak xenon is not available from the control rods, reduced steam voids, and temperature, the reactor cannot be taken critical at this time.

When reactor power is reduced, but kept in the power range, the behavior of xenon is similar to that after shutdown, but the height of the peak is considerably reduced because there is still a significant flux to burn out the xenon. About 40 to 50 hours after a decreasing power maneuver, xenon reaches the new equilibrium value for the new power level (Figure 1.7-16).

When reactor power is increased, xenon concentration initially decreases as burnout increases and production initially remains essentially constant because of I-135 decay. After a few hours (roughly 4 to 6 hours depending on power levels), the xenon concentration reaches a minimum and subsequently increases to the equilibrium level for the new higher power level. This is reached in roughly 40 to 50 hours (Figure 1.7-16).

Promenthium-Samarium Fission Product Absorbers

Next to Xe-135, the most important fission product poison is samarium-149 (Sm-149), a stable isotope with an absorption cross section of 4.08 x 10^4 barns which has an equilibrium reactivity poison effect of the order of -0.01 ΔK/K. It is the end product of the decay chain which follows:

$$^{149}Nd_{60} \xrightarrow[1.7\text{ hr.}]{\beta^-} {}^{149}Pm_{61} \xrightarrow[47\text{ hr.}]{\beta^-} {}^{149}Sm_{62}$$

This occurs with a total fission yield of about 1.4%. Sm-149 is a stable nuclide and is removed only by burnout.

After shutdown, promethium (Pm) will continue to decay to samarium and there is no removal of samarium; thus the samarium concentration increases after shutdown until there is no promethium left in the reactor (Figure 1.7-17). Considering an operating BWR, it takes approximately 500 hours at full power for Sm-149 to build up to an equilibrium concentration. In addition, Sm-149 concentration builds up to a constant value after shutdown but, because of its smaller microscopic absorption cross section, long time constant, low fission yield, and the longer half life of Pm-149, transient samarium absorptions are less important than those of Xe-135. Samarium is often treated as an equilibrium poison.

Coupled Steam Void Feedback

The large negative void reactivity in a BWR causes a flux increase from local Xe-135 burnout to be heavily damped. For example, if the local flux starts to rise in a PWR, the xenon burnout caused by the rise causes the flux to increase further. In a BWR, when the local flux increases, boiling also locally increases and the reactivity feedback reduces the flux. Consequently, the spatial xenon shape cannot easily shift in the radial direction in a BWR because of the damping from steam void reactivity feedback and the BWR is inherently spatially stable to xenon transients. Axial flux and power distribution transients can occur because of spatial xenon shifts in a BWR. However, unless these transients are driven by inappropriate operator movement of control rods, they will always be damped in 16 to 25 hours by the steam void reactivity feedback which acts like an axial shock absorber. (The power can shift downward in a BWR more easily than it can shift upward as a result of the more negative steam void reactivity coefficient in the top of the core).

Fission Product Absorbers - Summary

In summary, it can be stated that fission product absorption has not caused any serious operating problems in BWRs and none are expected. Near the end of a fuel cycle, it is possible that the time required to return to rated power from a shutdown could be limited by xenon. The delay would not be very long in any case and, in fact, return to power is usually limited by other considerations.

After a significant change in power level or power shape, the operator will observe a power drift and an axial power distribution transient as xenon comes into equilibrium. If necessary, following xenon building, the power level may be reduced temporarily by flow control to readjust the rod pattern to permit further power ascension by flow control.

Fuel Depletion and Burnable Poisons

The long term mechanisms affecting reactivity in the power range are fuel depletion and the burn out of the burnable poisons.

Excess reactivity is loaded into the core to allow operation at 100% power for a given length of time. Factors considered in the amount of excess reactivity include fuel cycle length, fission product poisons, voids, and means to avoid flux peaking problems. This excess reactivity is controlled by control rods and by burnable poisons. When considering only the effect of fuel burnup, as the fuel is burned up, the core becomes less reactive, and control rods must be withdrawn to maintain 100% reactor power.

To allow loading more fuel (higher K_{excess}) without increasing the number of control rods or reaching flux peaking problems, the burnable poison gadolinium (Gd) is loaded into the core.

Figure 1.7-18 shows the relationship of burnable poisons and fuel depletion over core life. From points A to B, the core becomes less reactive because of the buildup of Samarium. From points B to C, the core becomes more reactive for two reasons. One is that the burnable poison is burning up. Secondly, Pu-239 is building up. Of the two, the poison burn out is the most significant. At point C, the Pu-239 build up rate has decreased and the poison burn out and fuel burn up are about equal. From point C to the end of the core life, the fuel burn up is the overriding factor and the K_{excess} drops to a point where 100% reactor power cannot continue using normal control means.

Two tests are conducted at various times in core life to ensure that the core K_{excess} is following predictions. One, termed the shutdown margin test, is conducted at the beginning of core life and ensures that the reactor can be made subcritical any time in core life by at least a certain value of K. If point C (Figure 1.7-18) is greater than point A, this difference is called R, and the shutdown margin specification is increased to account for this.

The second test is the reactivity anomaly test. It is conducted periodically over core life and ensures that for a given exposure the reactivity is within 1% of that expected for the critical control rod configuration.

Standby Liquid Control System

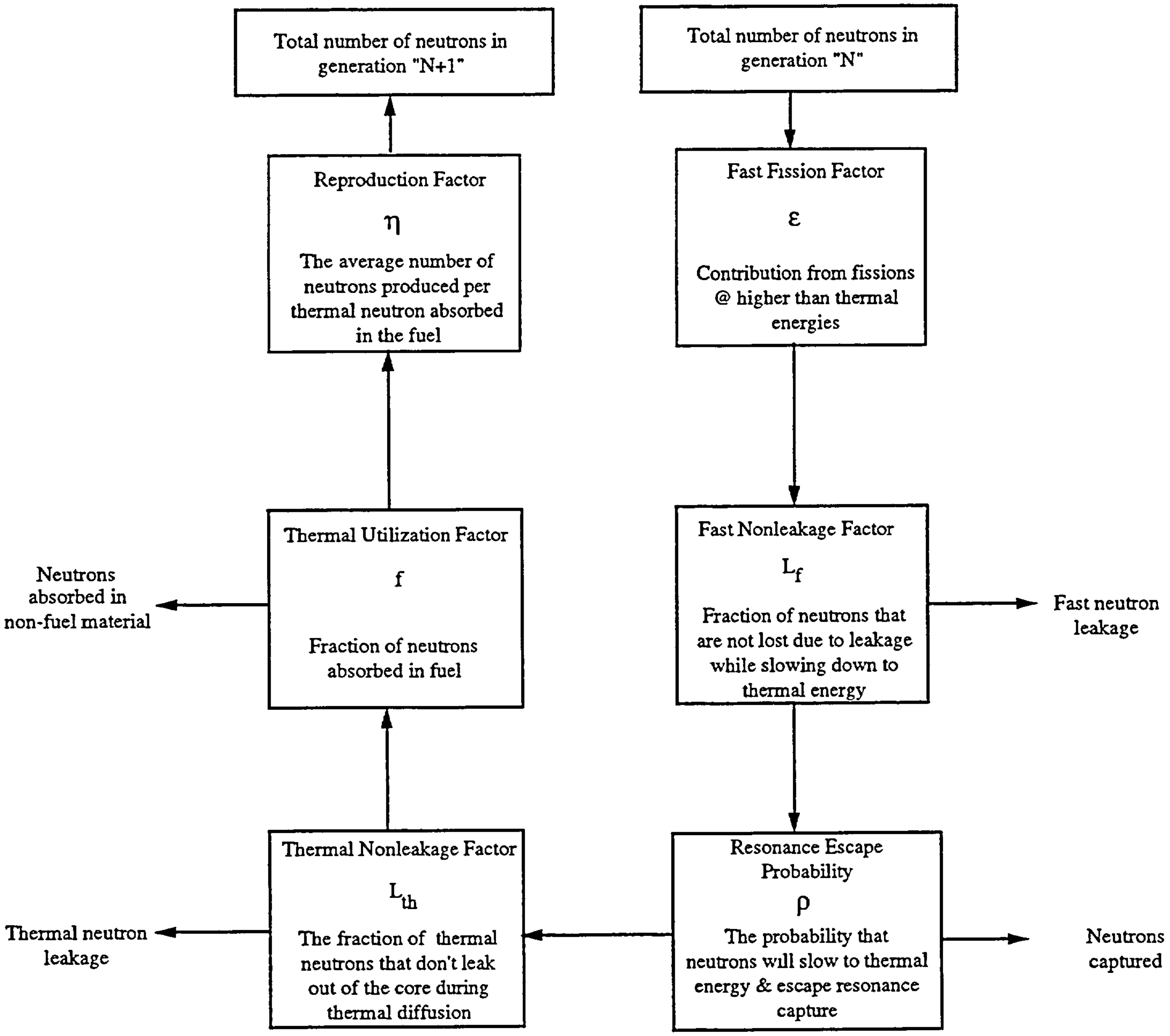
If the operator believes the reactor cannot be shutdown or remain shutdown with the control rods, the Standby Liquid Control (SLC) System (Section 7.4) provides him an alternative means for reactivity control.

The SLC System injects a neutron absorbing poison solution, sodium pentaborate, into the reactor vessel to shut down the reactor independent of any control rod motion, and maintain the reactor subcritical as the plant is cooled to maintenance temperature.

The SLC System can perform its function at any time in core life.

Power Range - Summary

When the plant startup is complete, power range operation is normally steady state. Small power changes required to accommodate grid load changes are normally done by adjusting recirculation flow. Control rods are moved at low power, usually under the direction of the nuclear engineer, to offset fuel depletion or fission product poison reactivity.



$$K_{eff} = \varepsilon \, L_f \, \rho \, L_{th} \, f \, \eta = \frac{\text{Total \# neutrons in "N+1 "generation}}{\text{Total \# neutrons in "N "generation}}$$

When critical : $K_{eff} = 1$

Figrue 1.7-1 Neutron Cycle in a Thermal Reactor

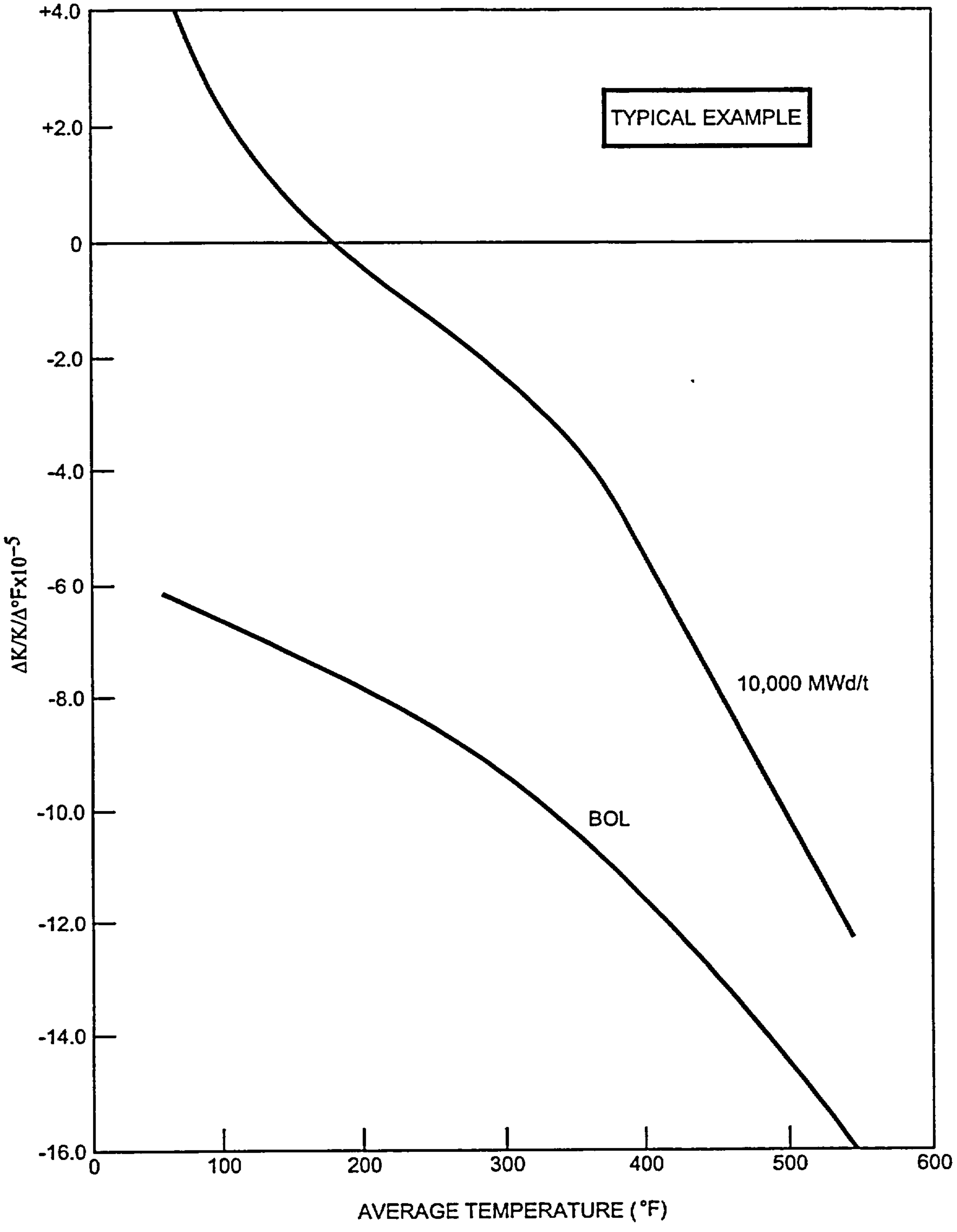


FIGURE 1.7-2 Moderator Temperature Coefficient

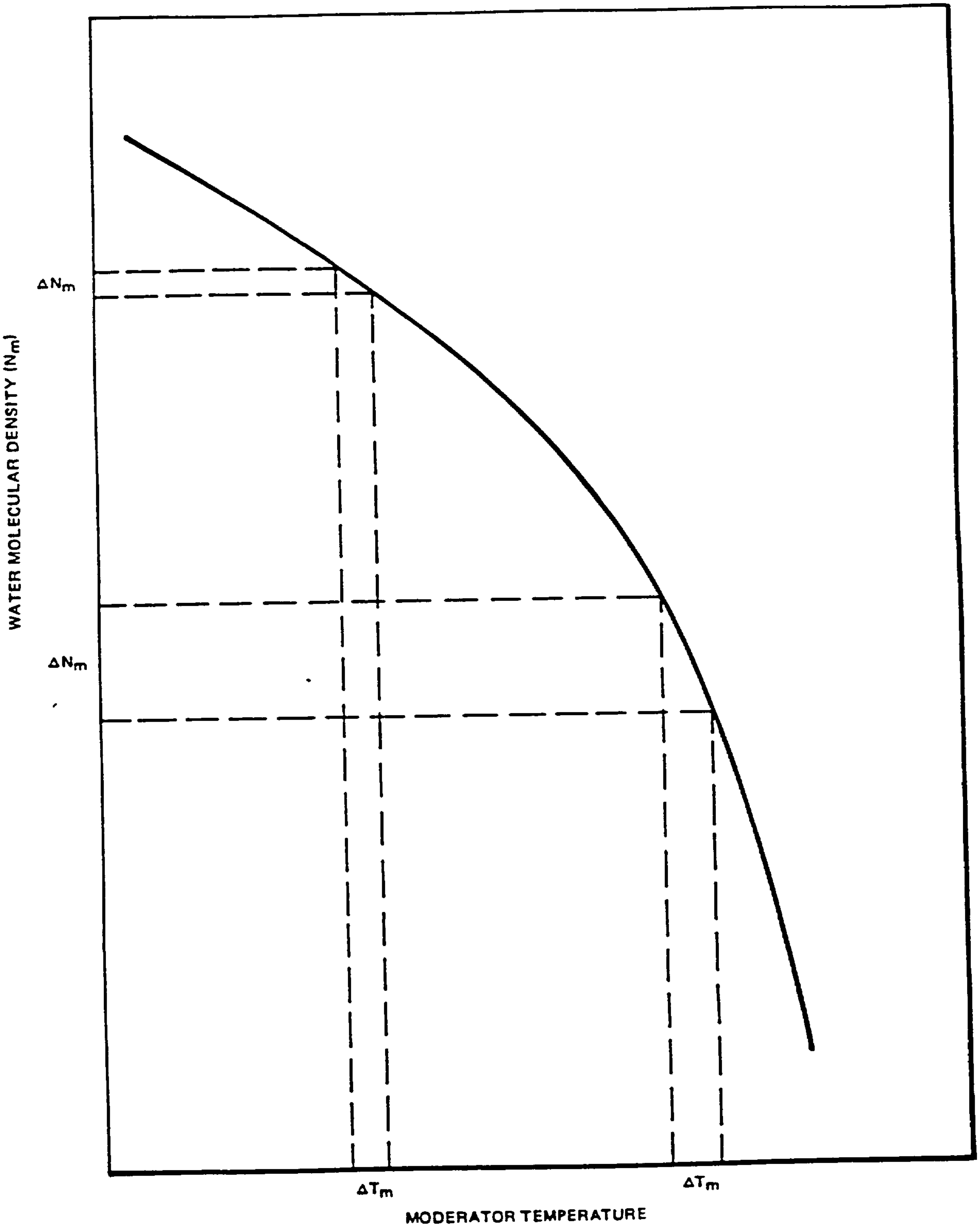


FIGURE 1.7 – 3 WATER DENSITY VERSUS TEMPERATURE

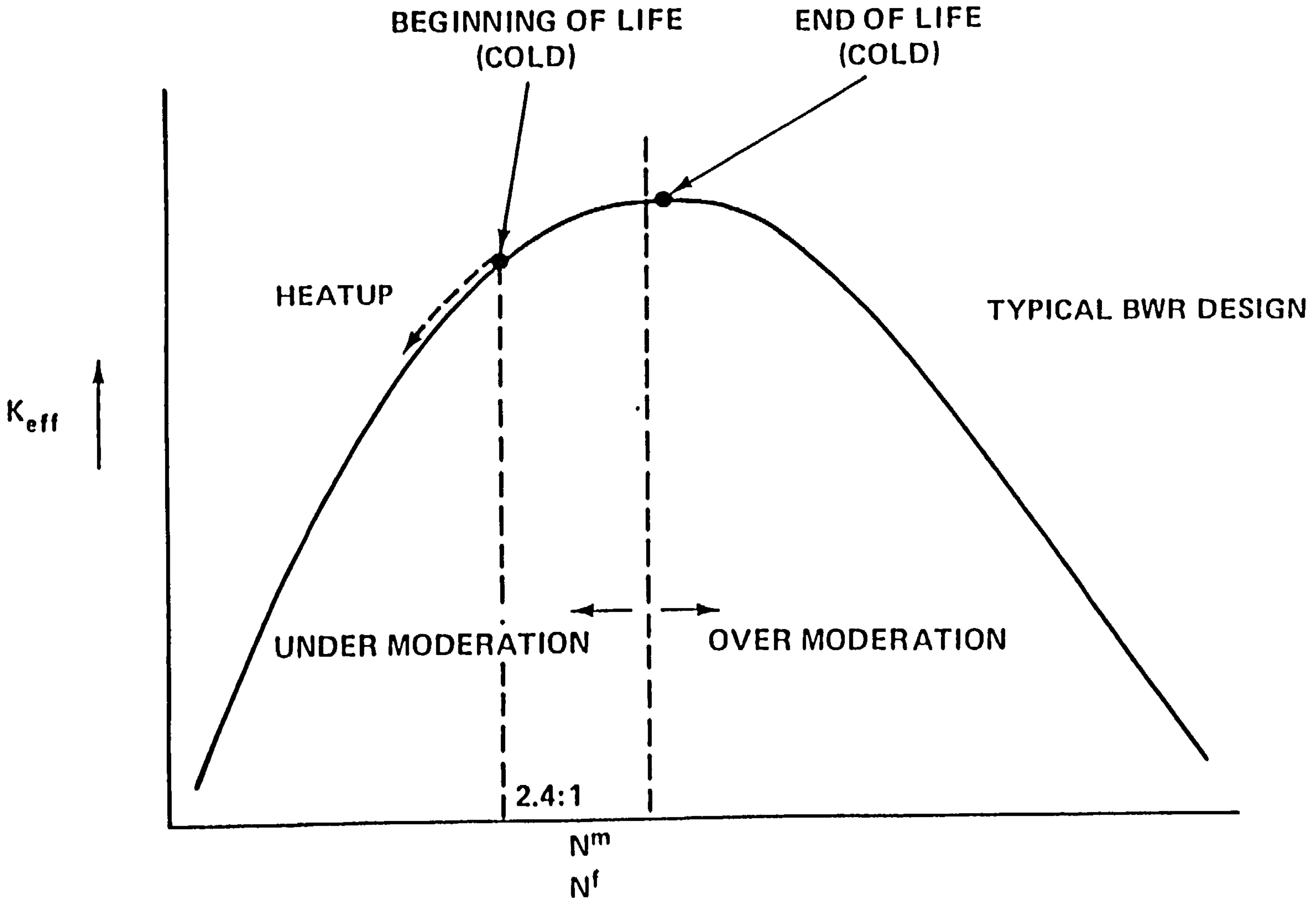


FIGURE 1.7 – 4 OVER/UNDER MODERATION

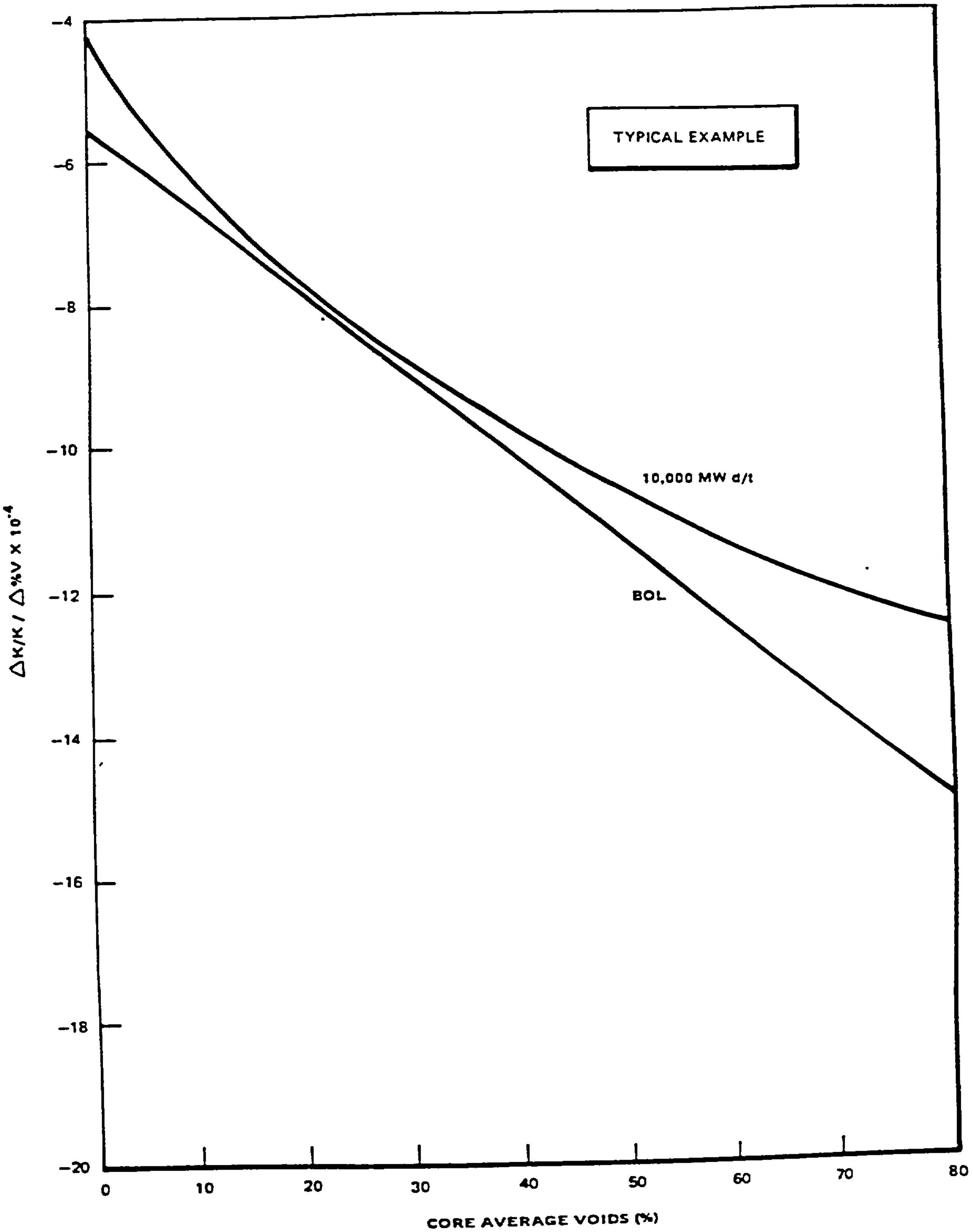


FIGURE 1.7 – 5 VOID COEFFICIENT

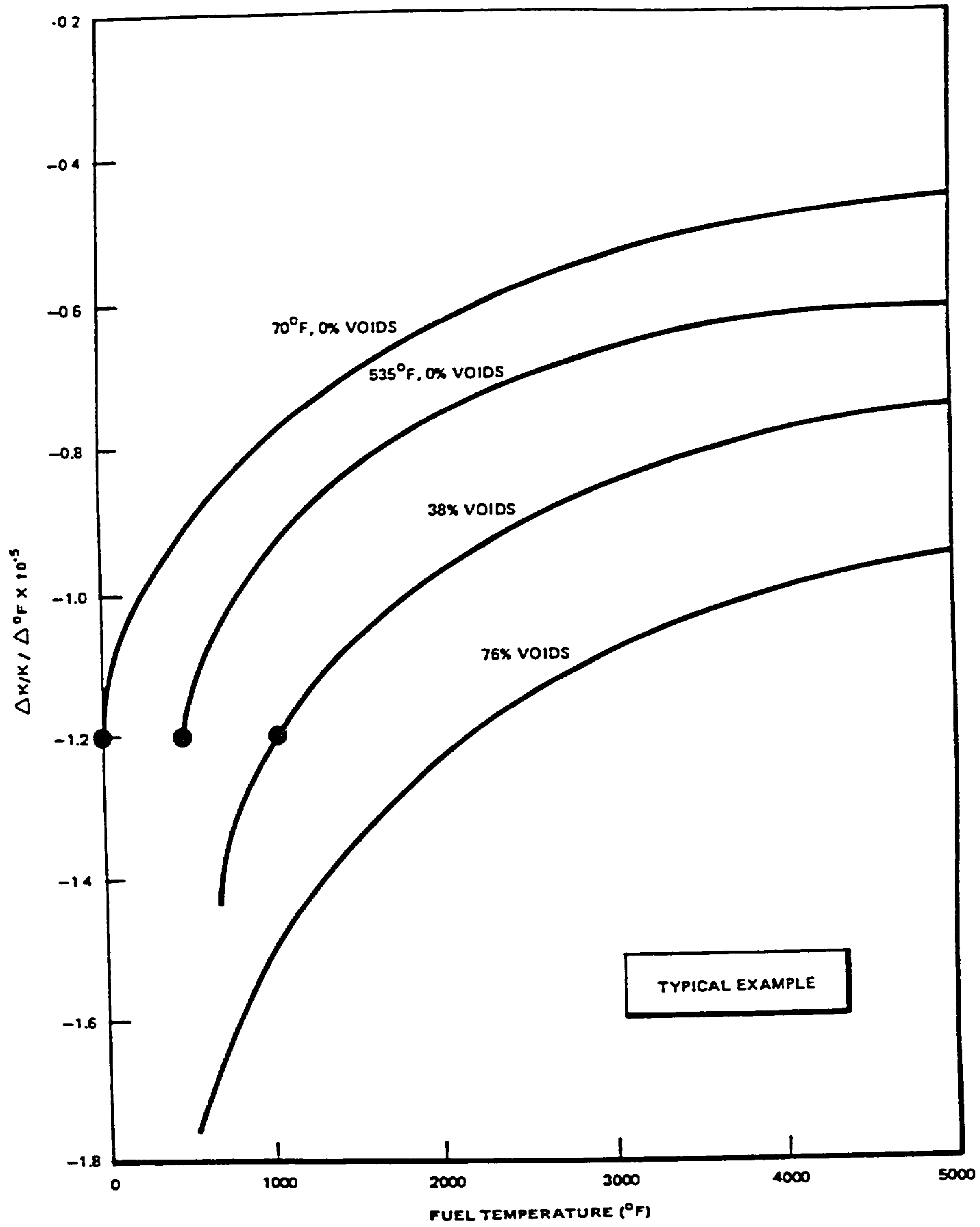
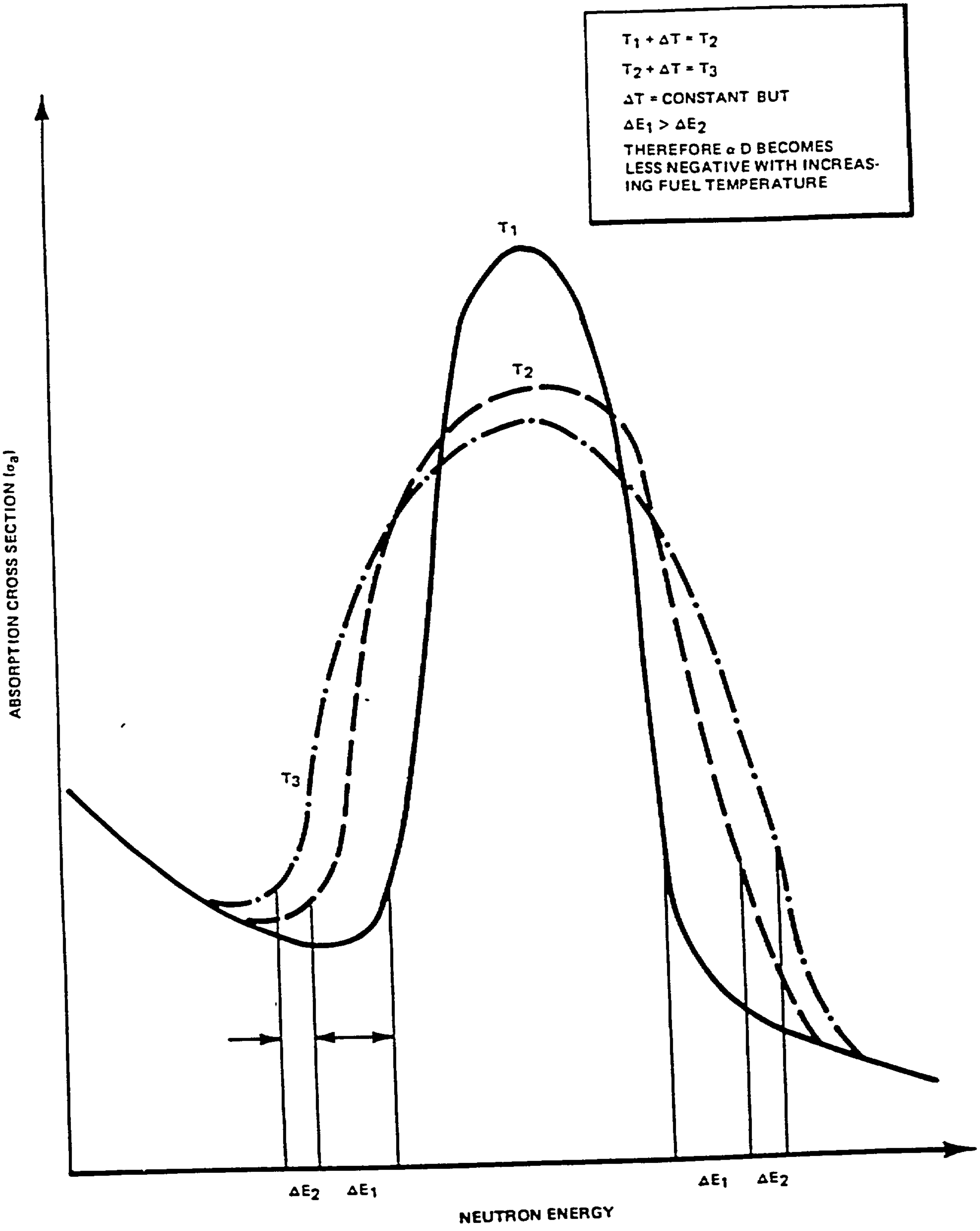
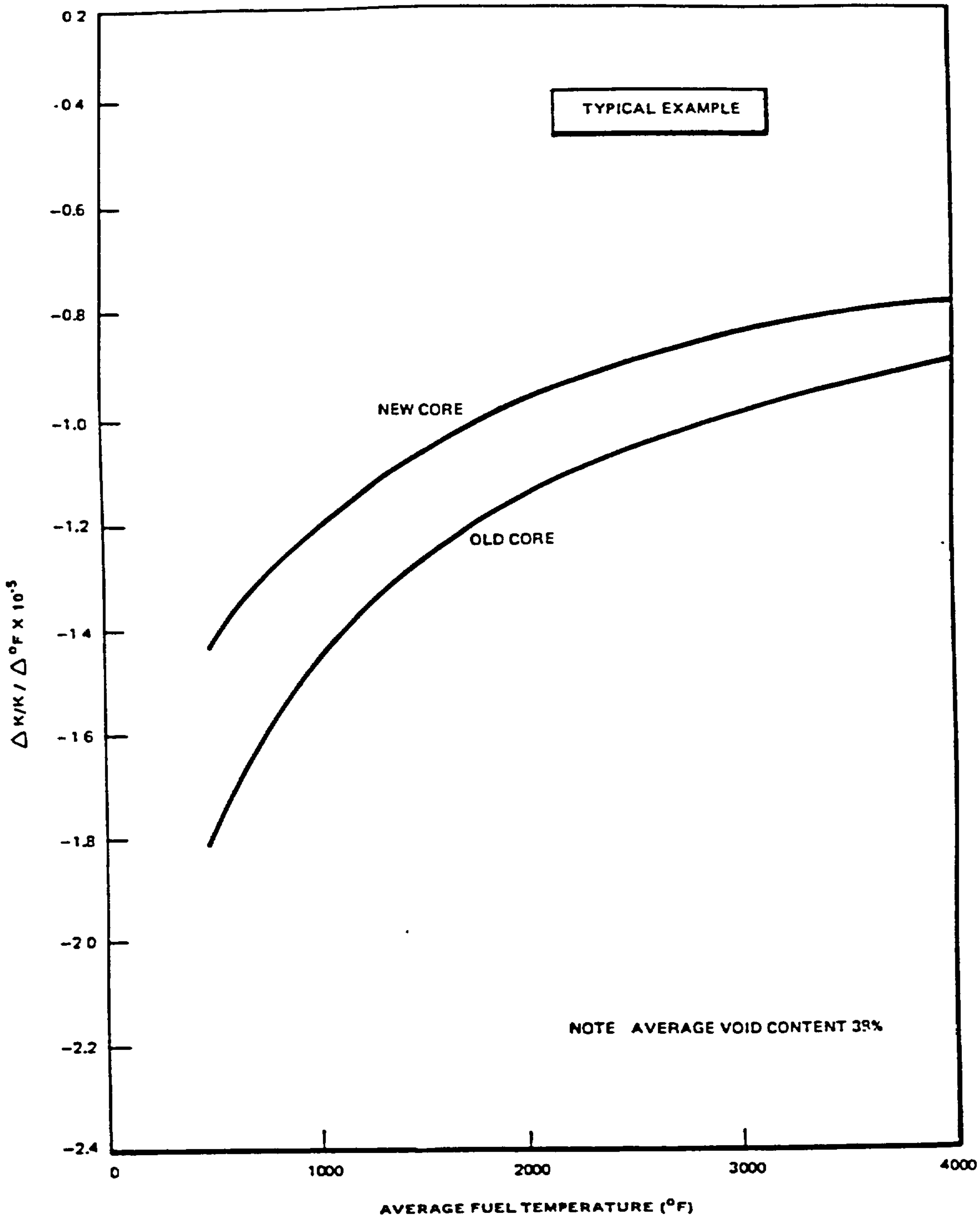


FIGURE 1.7 – 6 DOPPLER COEFFICIENT



$T_1 + \Delta T = T_2$
$T_2 + \Delta T = T_3$
ΔT = CONSTANT BUT
$\Delta E_1 > \Delta E_2$
THEREFORE α D BECOMES LESS NEGATIVE WITH INCREASING FUEL TEMPERATURE

FIGURE 1.7 – 7 EFFECT OF INCREASING TEMPERATURE ON ABSORPTION CROSS SECTION AT RESONANCE PEAK



TYPICAL EXAMPLE

NOTE AVERAGE VOID CONTENT 39%

FIGURE 1.7 – 8 CORE AGE EFFECT ON DOPPLER COEFFICIENT

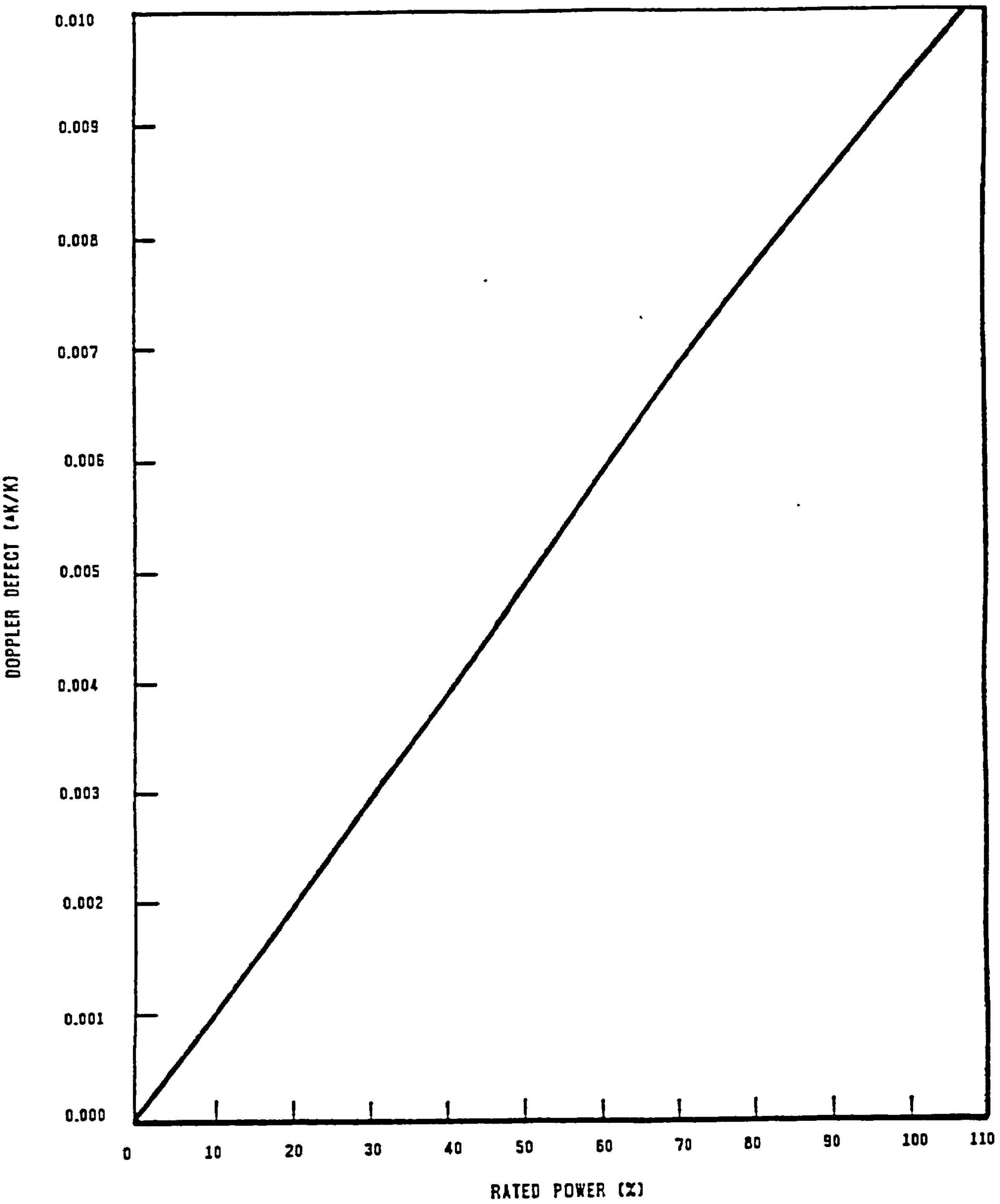
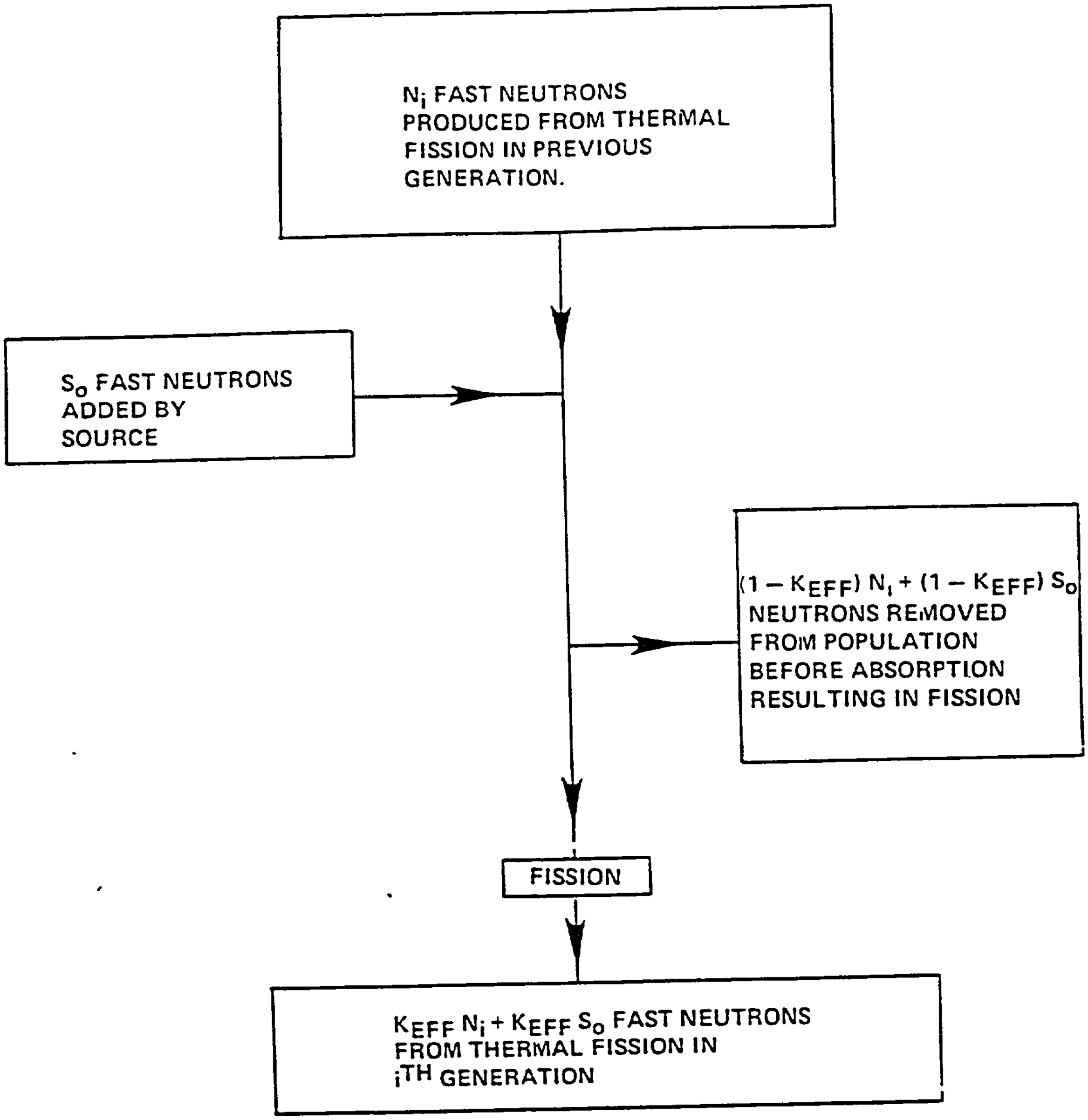


FIGURE 17-8a DOPPLER DEFECT



FOR STEADY STATE CONDITION $N_i = K_{EFF} N_i + K_{EFF} S_o$

REARRANGING THIS EQUATION YIELDS:

$$N_i - K_{EFF} N_i = K_{EFF} S_o$$
$$N_i (1 - K_{EFF}) = K_{EFF} S_o$$

WHICH SAYS THAT THE SOURCE NEUTRONS ARE MAKING UP FOR LOST NEUTRONS TO MAINTAIN A CONSTANT NEUTRON POPULATION.

FIGURE 1.7 – 9 SUBCRITICAL MULTIPLICATION

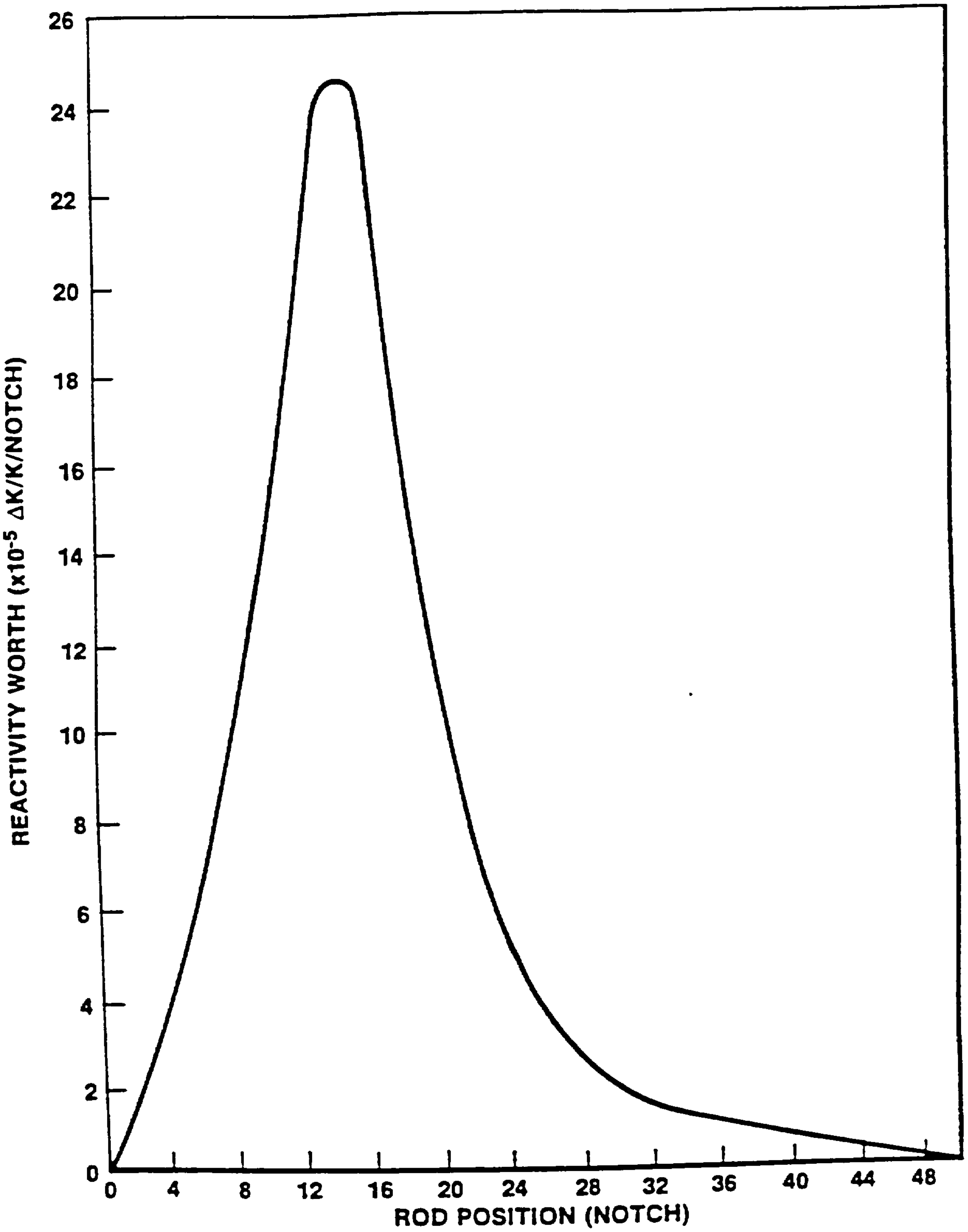


FIGURE 1.7-10 DIFFERENTIAL ROD WORTH

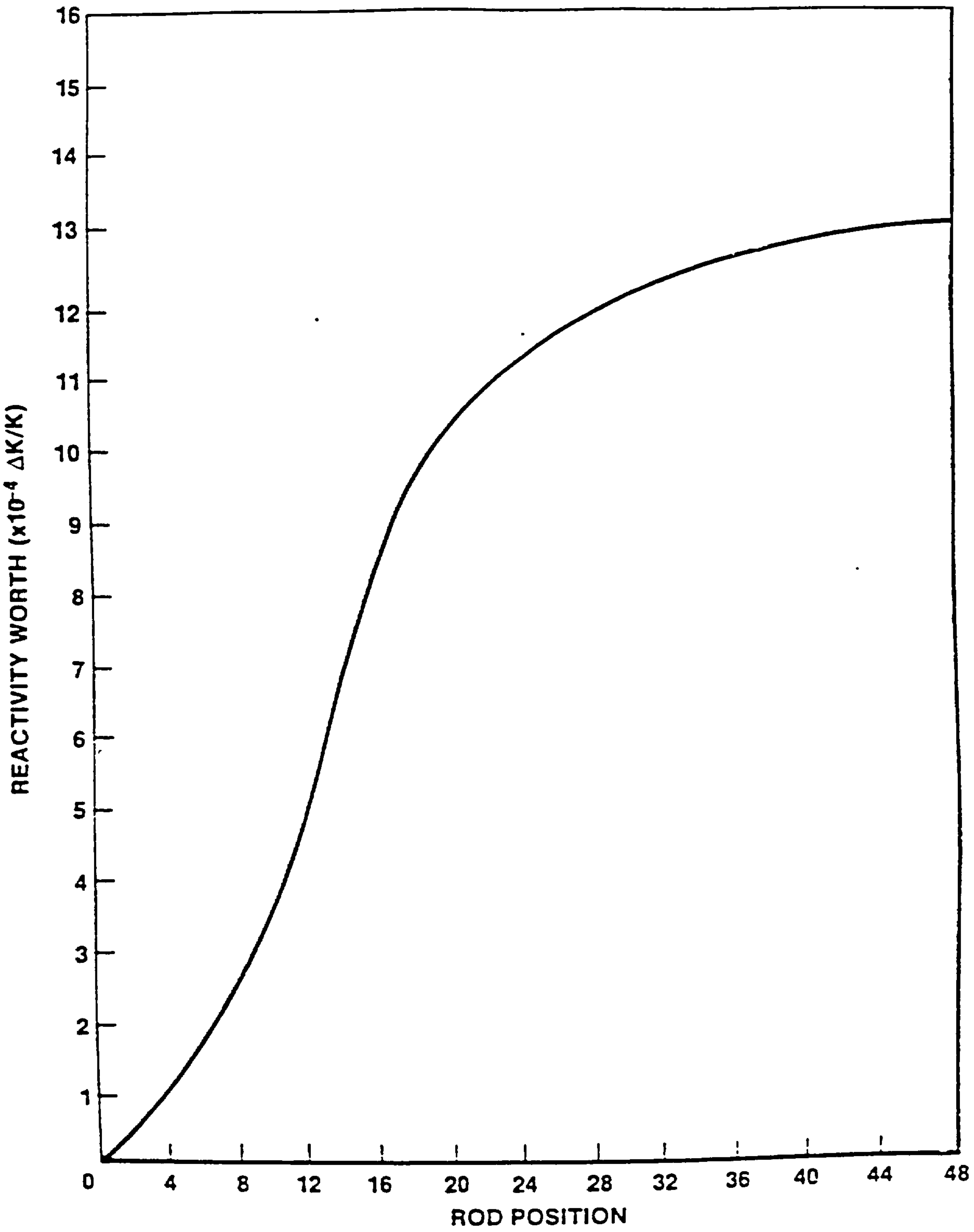


FIGURE 1.7-11 INTEGRAL ROD WORTH

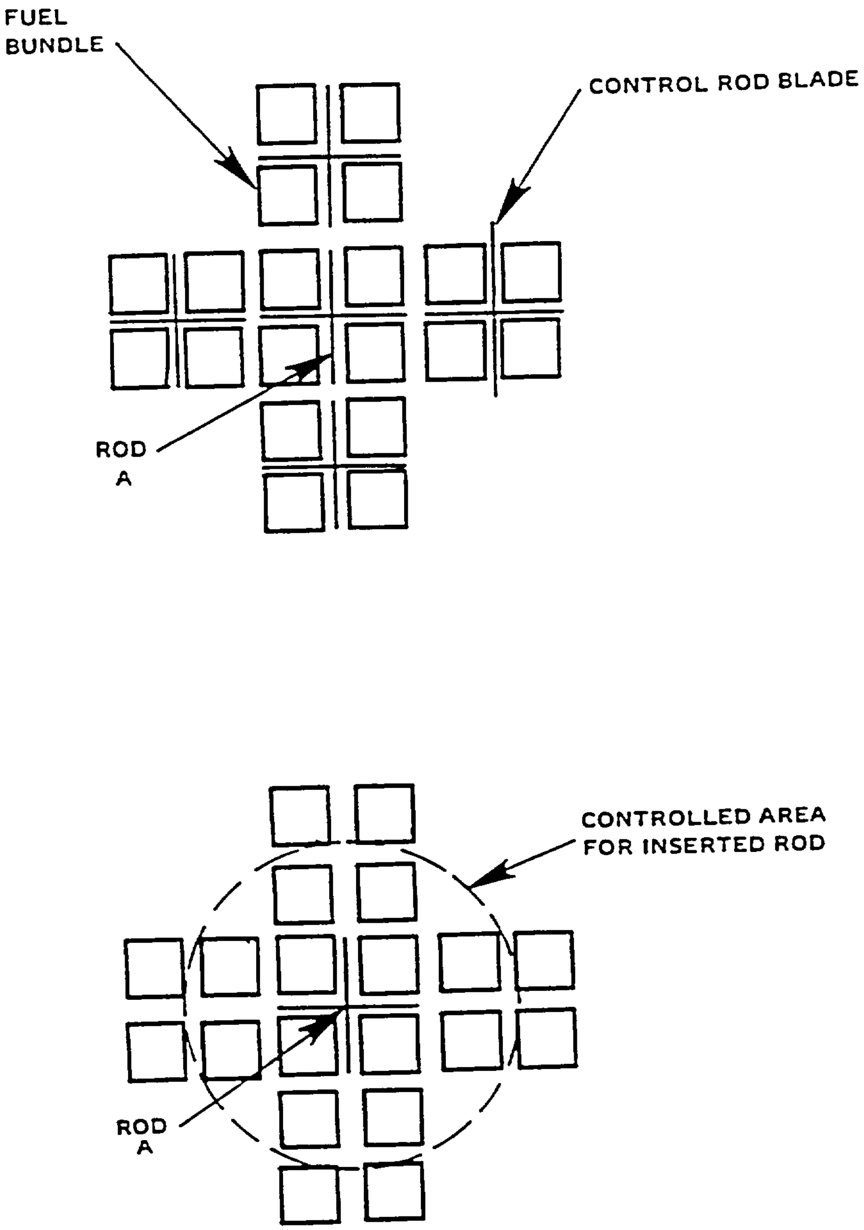


FIGURE 1.7-12 ROD SHADOWING EFFECT

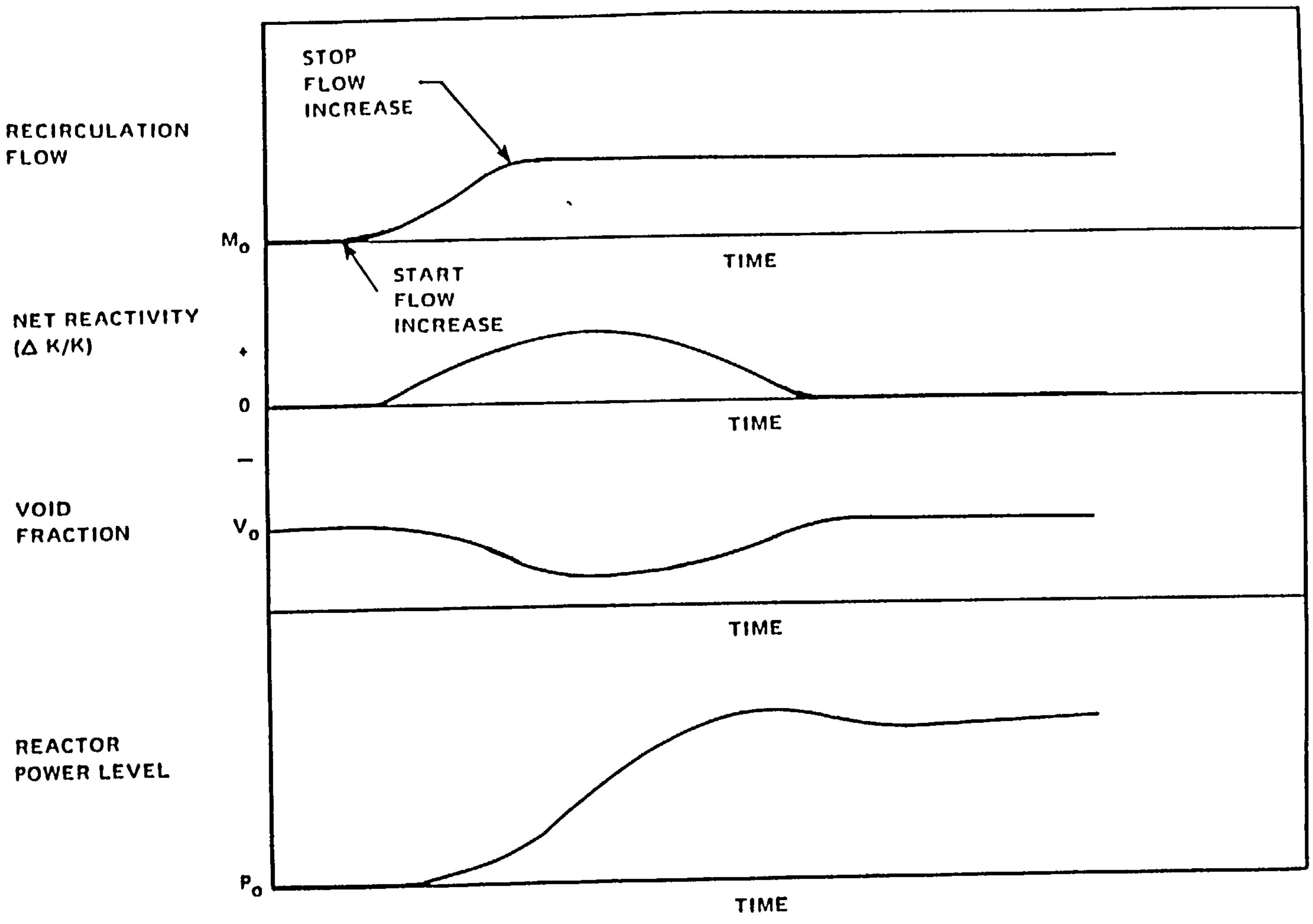


FIGURE 1.7-14 RESPONSE TO INCREASE IN RECIRCULATION FLOW

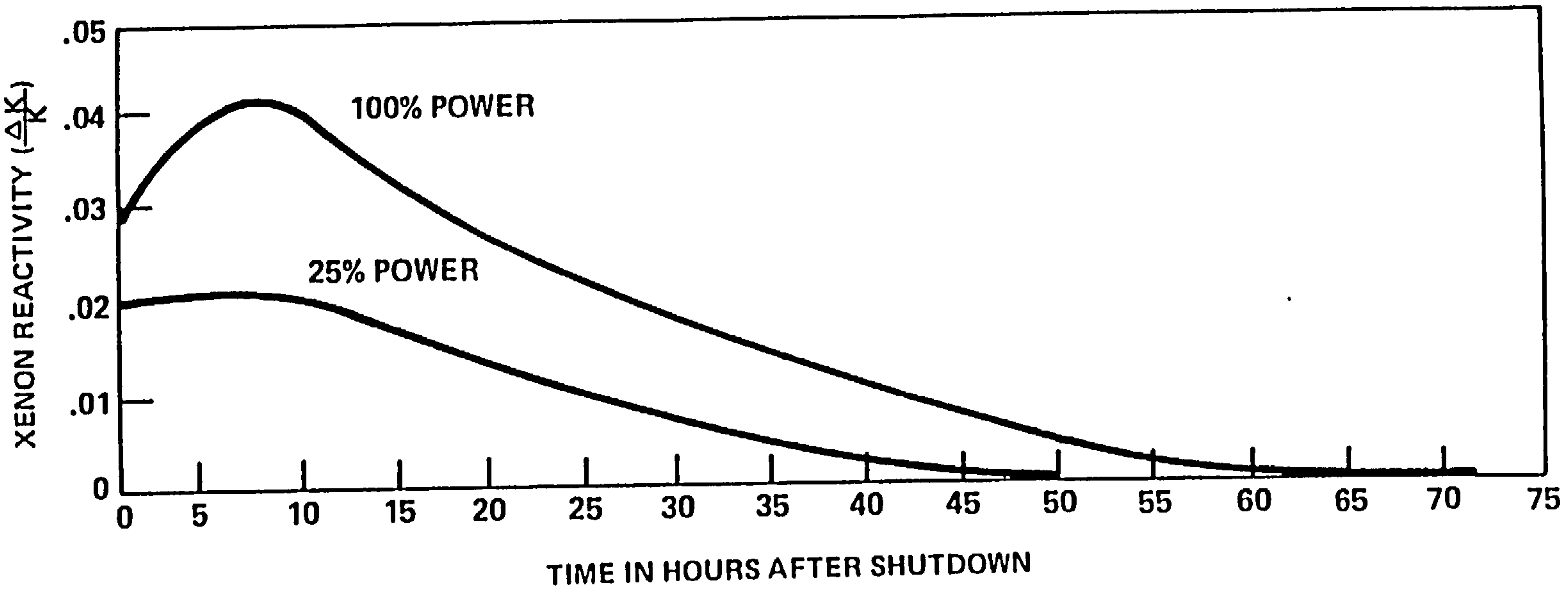


FIGURE 1.7-15 XENON TRANSIENTS FOLLOWING A REACTOR SHUTDOWN

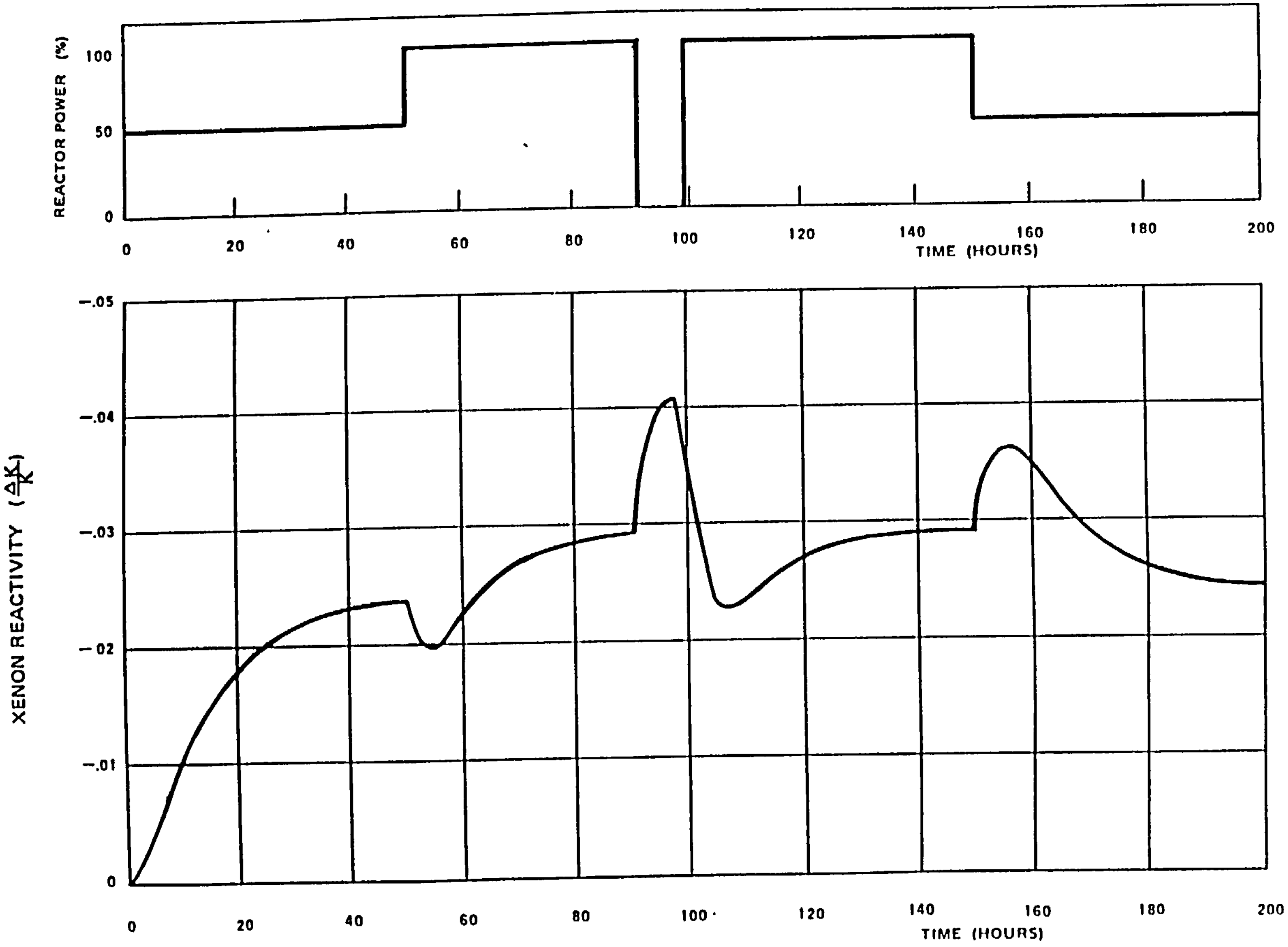


FIGURE 1.7-16 TYPICAL XENON TRANSIENTS

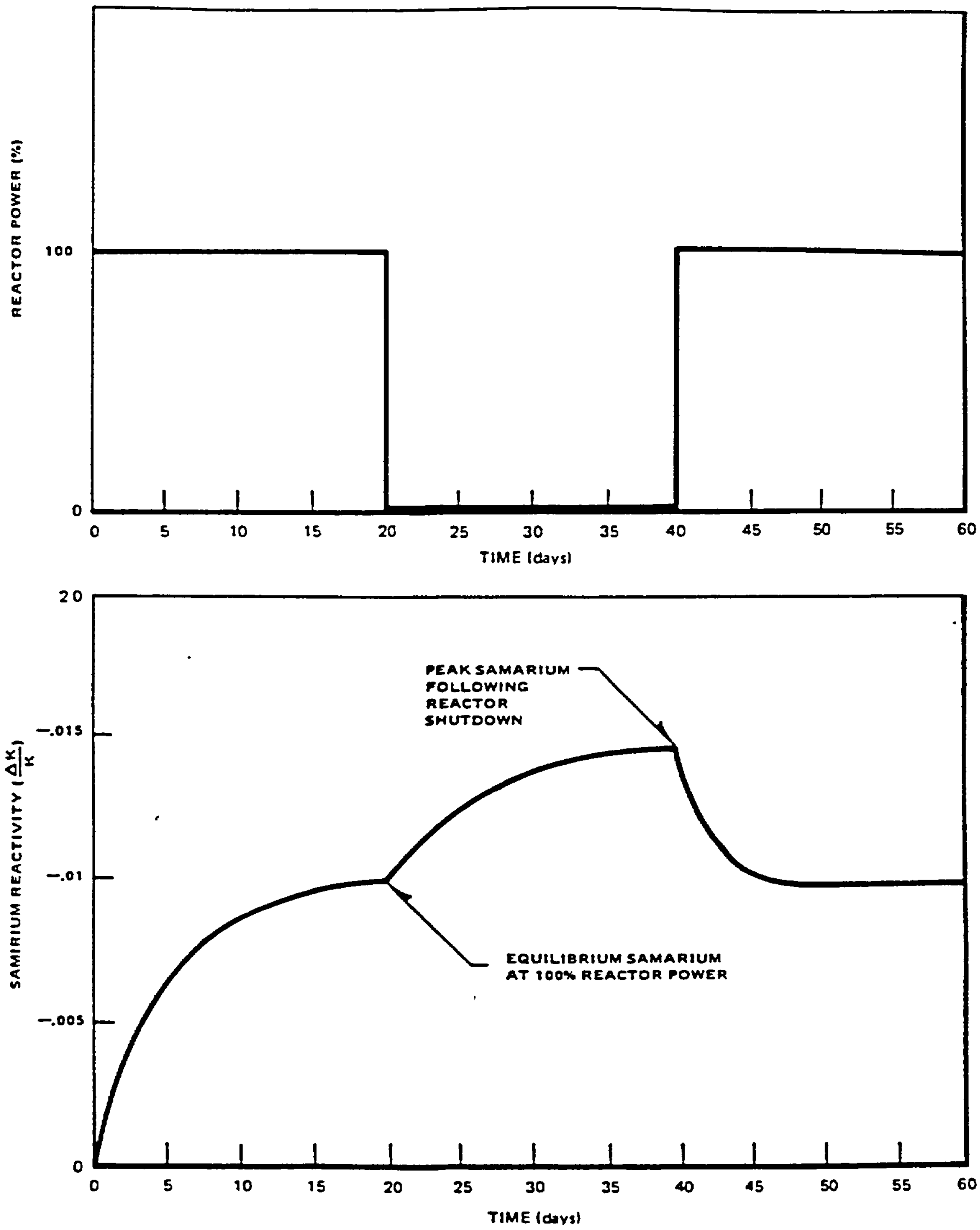


FIGURE 1.7-17 TYPICAL SAMARIUM TRANSIENTS

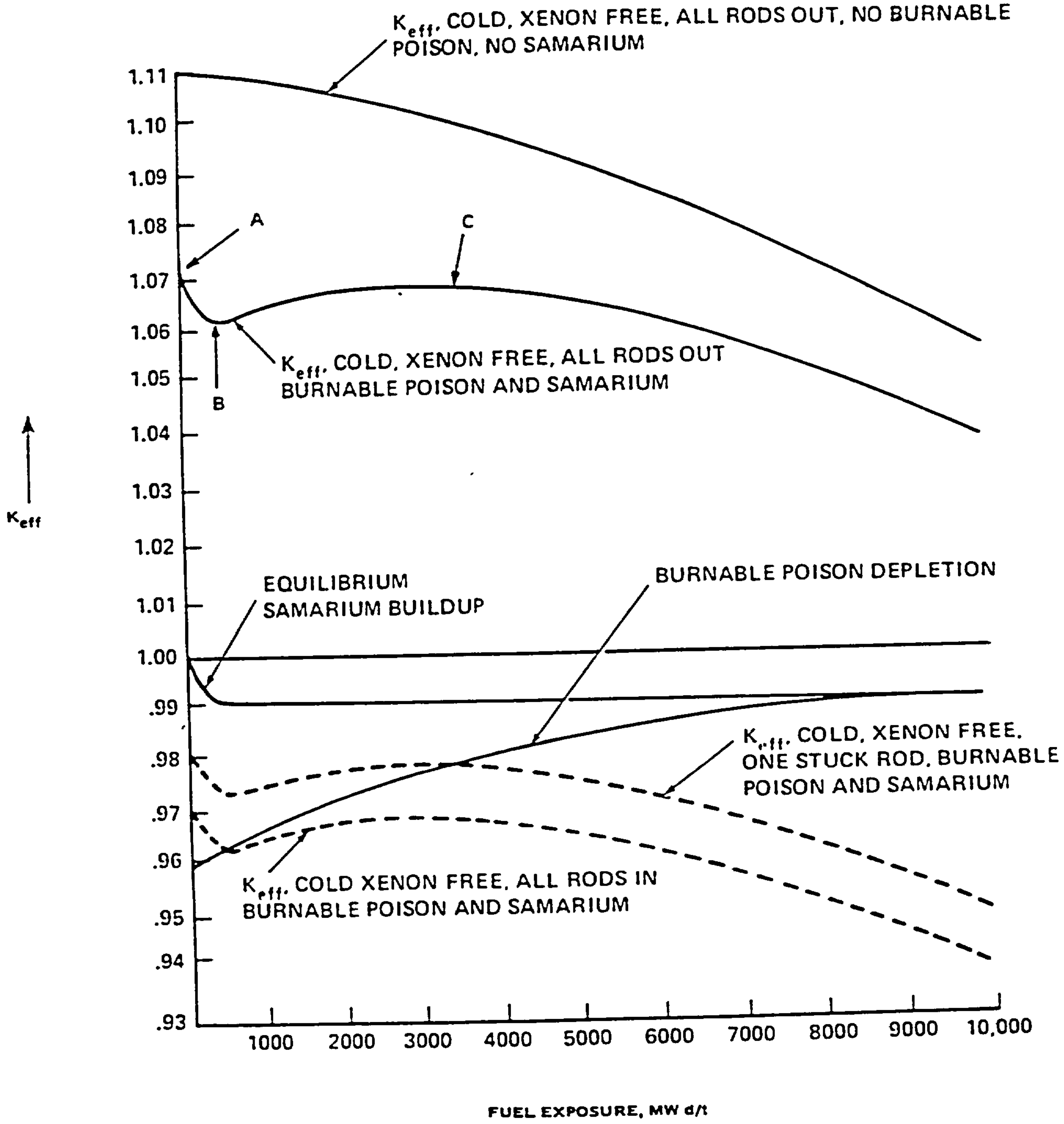


FIGURE 1.7 – 18 SUMMARY OF FUEL DEPLETION AND POISON EFFECTS ON K_{eff}

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1.8 THERMAL LIMITS

Learning Objectives:

1. List the three basic BWR thermal limits and explain the purpose of each.
2. Define fuel damage and explain the mechanisms which could cause fuel damage.
3. Define the following terms used in the study of BWR thermal limits:
    a. Linear Heat Generation Rate (LHGR)
    b. Average Planar Linear Heat Generation Rate (APLHGR)
    c. Critical Power
    d. Critical Power Ratio (CPR)
    e. Safety Limit MCPR
    f. MCPR Operating Limit

1.8.1 Introduction

Thermal limits are provided to minimize the radiological release from the plant during normal operation, abnormal operation, abnormal operational transients, and postulated accidents by restricting plant operation so that the fuel cladding integrity is maintained.

1.8.2 Thermal Limit Description

Thermal limits are provided for normal operation and transient events to maintain the integrity of the fuel cladding. This objective is achieved by limiting fuel rod power density to avoid overstressing the fuel cladding because of fuel pellet-cladding differential expansion and by maintaining nucleate boiling around the fuel rods so that the transition to film boiling is avoided. The thermal limits established for these purposes are the linear heat generation rate (LHGR) limit and the minimum critical power ratio (MCPR) limit.

A thermal limit is provided for postulated accidents to maintain the core geometry by minimizing the gross fuel cladding failure because of the heatup following a loss of coolant accident (LOCA). The thermal limit established for this purpose is the maximum average planar linear heat generation rate (MAPLHGR) limit.

The basic thermal limits are shown in Figure 1.8-1. Table 1.8-1 gives values for BWR/3, BWR/4, BWR/5, and BWR/6 core parameters and is provided for reference only.

1.8.3 Background Information

In order to understand the BWR thermal limits, it is necessary to have an understanding of related background material such as heat transfer and fluid flow characteristics. This subject material is discussed in the paragraphs that follow.

1.8.3.1 Heat Transfer

In light water reactor operation, heat is transferred from the fuel center line to the light water moderator which comes into contact with the outer fuel cladding surface. The heat can be transferred by conduction, convection, or radiation. Conduction and convection are the modes of heat transfer of primary interest in nuclear power plant operations.

Conduction

When heat is applied to a material, the kinetic energy of the atoms or molecules of the material is increased. Because of this increase in kinetic energy, the particles have a greater tendency to collide with each other. When these collisions occur, the particles transmit a portion of their kinetic energy to neighboring atoms. This is conduction.

This is the process by which heat generated in the fuel pellet is transmitted to the outer cladding surface. In relation to the fuel rod, this conduction flow is in a horizontal plane from the fuel center line to the cladding surface.

Convection

Convection is the process of transmitting heat from a heated surface or area to a fluid by circulation or mixing of the fluid. Convection takes place only in fluids.

The application in this case deals with a fluid (moderator) flowing past a metallic fuel cladding surface. Fluids have a tendency to adhere to solid surfaces resulting in the formation of a stagnant film on the surface. This film is normally very thin and heat is transferred across this film by a combination of conduction and convection. After the heat penetrates the film, it is transferred rapidly through the remainder of the fluid by mixing.

The resistance of heat flow is so low that there is virtually no temperature variation through the bulk of the fluid (moderator) at any given elevation along the fuel rod.

Radiation

Radiation heat transfer is the transmission of heat in the form of radiant energy from one object to another across an intervening space. This form of heat transfer is avoided in nuclear power plants because very high temperature differentials are required to transfer a significant amount of heat. If high temperatures were allowed to occur, the materials would degrade.

1.8.3.2 Boiling Heat Transfer

The boiling heat transfer curve is shown in Figure 1.8-2. The amount of heat transferred from the fuel cladding to the coolant is greatly affected by the coolant properties and by the thermal and hydraulic conditions of the coolant. The rate at which heat is transferred from the cladding, the heat flux, is dependent on the specific temperature difference between the cladding and the coolant (ΔT) and the heat transfer coefficient. The heat flux may be plotted against the temperature difference between the cladding and the coolant. This curve can be divided into boiling regions corresponding to the regions shown in Figure 1.8-2. The heat transfer coefficient in each region is controlled by the mode of heat transfer in that region.

The first region is single phase convection heat transfer. The heat flux increases somewhat with increased ΔT.

The second region is associated with subcooled nucleate boiling. Subcooled nucleate boiling is boiling that occurs at the cladding surface while the bulk coolant temperature is not yet at saturation temperature. The steam bubbles may collapse before departing from the cladding surface or they will collapse as they enter the subcooled region after departing from the surface. This mode of convective heat transfer is a complicated mixture of single phase convection and nucleate boiling modes of heat transfer.

The next region is fully developed bulk nucleate boiling. Nucleate boiling is a very efficient mode of heat transfer because of high turbulence created by the boiling process. Annular flow exists at the outlet of high power fuel assemblies during normal operation. The fuel rod surface remains in a well cooled, nucleate boiling type of heat transfer state. Nucleate boiling is maintained in the core in all modes of normal operation and in all transient conditions caused by a single operator error or equipment malfunction.

The heat flux increases as the temperature difference between the cladding and the coolant increases. There is a point, however, at which the heat transfer coefficient no longer increases with an increased ΔT. There is a transition boiling regime where the boiling mode changes from nucleate boiling to film boiling. This region is highly unstable and is characterized by the intermittent physical rewetting of the heated surface by the coolant. The beginning of this region is called onset of transition boiling (OTB) and is labeled so on Figure 1.8-2. The OTB point is avoided in the BWR by remaining within the critical power ratio thermal limit.

The crosshatched region represents temperature oscillations which take place during transition boiling. At a given heat flux the clad surface temperature will oscillate between a point on the right in the crosshatched region and a point on the left in the crosshatched region along a horizontal line. This is caused by intermittent physical rewetting of the clad surface. At the point of onset of transition boiling (OTB), the temperature oscillations reach 25°F in magnitude; this is defined as the critical power point.

1.8.3.3 Fluid Flow

Figure 1.8-3 shows the different flow patterns which can exist in a fuel bundle during normal operation. As coolant (single phase liquid) enters the fuel bundle, it is slightly subcooled, and begins to gain heat from the forced flow convection mechanism. Because of subcooling, there is little or no bubble formation. As energy is gained, the coolant temperature increases until nucleate boiling with its attendant bubble formation begins. Early states of nucleate boiling occur while the bulk coolant in the bundle is below liquid saturation enthalpy, and the bubbles readily collapse as the turbulent flow and their buoyancy sweeps them away from the cladding surface. A point will be reached where the bulk coolant enthalpy is at liquid saturation, (bulk boiling) and the bubbles will no longer collapse in the coolant as they are swept away. The bubbles now begin to exist separately throughout the bulk coolant, causing a significant steam fraction to be present in the coolant. From this point to the bundle outlet, the bubbles continue to form at the fuel rod surface (nucleate boiling) and to be swept into the coolant and begin to coalesce into larger and larger slugs of steam. At the outlet of very highest powered fuel bundles, steam may fill most of the bundle flow area between fuel rods, but a thin annulus of water adheres to the fuel rod surfaces. In this annular flow region, the wetted rod surface is still transferring heat through the nucleate boiling mechanism.

Fuel Channel Parameter Characteristics

Figure 1.8-4 shows a plot of coolant and fuel bundle temperature versus flow path length of an average fuel bundle. Coolant enters the bottom of the core, flows upward around the fuel rods, and absorbs energy from heat transfer originating from the nuclear process. Because of the peculiar characteristics of neutron-caused fission reactions, the average heat flux (Q/A) produced from fission in the core assumes a shape somewhat like that shown in Figure 1.8-4. The highest heat flux is in the core interior, hence some fuel bundles have a higher than average heat flux while some have a lower than average heat flux.

The coolant temperature curve rises as heat is added, until temperature saturation occurs, and coolant bulk boiling begins. From this point the coolant temperature remains constant all the way to core outlet. Because the coolant is changing phase, the coolant temperature profile is not altogether descriptive of coolant energy increase. A better description is obtained by plotting coolant enthalpy change, which continuously increases from core inlet to outlet, with the largest rate of increase at the maximum value of heat flux.

The curve for fuel rod surface temperature rises and then levels at a constant value above coolant temperature. The initial rise is caused by the ΔT across the film required to accommodate the heat flux (Q/A) during single phase forced convection heat transfer. The fuel rod temperature levels off when nucleate boiling starts. Nucleate boiling is an excellent heat transfer mode; therefore, even thought the heat flux increases, the ΔT across the boiling film remains relatively constant.

The curve for fuel rod center line temperature is above that of the fuel surface temperature. The amount that the center line temperature is greater than surface temperature depends directly on the heat flux. The beneficial effects of nucleate boiling on center line temperature can also be seen. As long as nucleate boiling is occurring on the fuel rod surface, the fuel rod surface temperature is only slightly greater than liquid temperature. This, in turn, keeps the fuel center line temperature at a lower value than if single phase convection were the mode of heat transfer from surface to liquid.

Fuel Temperature Profile

Figure 1.8-5 illustrates a typical fuel temperature cross section with nucleate boiling at a high heat flux. The beneficial effect of nucleate boiling can be seen. As long as nucleate boiling is occurring on the fuel rod surface, the fuel rod surface temperature is only slightly greater than liquid temperature. This, in turn, keeps the fuel center line temperature at a lower value than if single phase convection were the mode of heat transfer from surface to liquid.

1.8.4 Fuel Damage

Fuel damage is defined for design purposes as perforation of the cladding, which permits release of fission products. The mechanisms which could cause fuel damage in reactor transients are:

* Rupture of the fuel cladding because of strain caused by relative expansion of the uranium dioxide (UO_2)pellet and the fuel cladding.

* Severe overheating of the fuel cladding caused by inadequate cooling. Fuel damage that results from local overheating of the cladding is conservatively defined as the onset of the transition from nucleate to film boiling.

1.8.5 The LHGR (Thermal-Mechanical) Limit

Linear heat generation rate (LHGR) is the surface heat flux integrated over each square centimeter of cladding material in one linear foot of a fuel rod. Limits on LHGR are set to restrict the strain on the fuel cladding because of relative expansion of the fuel pellets and the cladding. A value of 1% plastic strain of the cladding is conservatively defined as a threshold below which fuel damage due to fuel cladding overstraining is not expected to occur.

Relative expansion arises from several sources: the UO_2 fuel thermal expansion coefficient is approximately twice that of zircaloy; the fuel pellets operate at higher temperatures than the cladding; the fuel pellets undergo irradiation growth as they are exposed; and the fuel pellets crack and redistribute toward the cladding because of thermal stress. Cladding cracking because of differential expansion of pellet and cladding is prevented by limiting fuel pin power so that 1% plastic strain does not occur.

The linear heat generation rate required to cause 1% cladding strain in 8 x 8 fuel is approximately 25 KW/ft for unirradiated fuel, but decreases with burnup to a value of approximately 20 KW/ft at a local exposure of 40,000 MWd/MT.

The design LHGR for 8 x 8 fuel is 13.4 KW/ft, which provides a margin to the 1% plastic strain threshold. The design LHGR for newer 8 x 8 fuel and 9 x 9 fuel is 14.4 KW/ft.

1.8.5.1 Elasticity and Plasticity

Suppose we plot a graph of stress as a function of the corresponding strain. If Hooke's law is obeyed, stress is directly proportional to strain and the graph is a straight line. Real materials show several types of departures from this idealized behavior.

Figure 1.8-6 shows a typical stress-strain graph for a metal such as copper or soft iron. The stress in this case is a simple tensile stress, and the strain is shown as the percent elongation. The first portion of the curve, up to a strain of less than 1%, is a straight line, indicating Hooke's-law behavior with stress directly proportional to strain. This straight-line portion ends at point a; the stress at this point is called the proportional limit.

From a to b, stress and strain are no longer proportional, but if the load is removed at any point between o and b, the curve is retraced and the material returns to its original length. In the entire region ob the material is said to be elastic or to show elastic behavior. Point b, the end of this region, is called the yield point, and the corresponding stress is called the elastic limit. Up to this point the forces exerted by the material are conservative. When the load is removed, the material returns to its original shape, and the energy put into the material in causing the deformation is removed. The deformation is said to be reversible.

If the stress is increased further, the strain increases rapidly, but when the load is removed at some point beyond b, say c, the material does not come back to its original length but traverses the thin line in Figure 1.8-6. The length at zero stress is now greater than the original length, and the material is said to have a permanent set. Further increase of load beyond c produces a large increase in strain (even if the stress decreases) until a point d is reached at which fracture takes place. From b to d, the material is said to undergo plastic flow, or plastic deformation. A plastic deformation is irreversible; when the stress is removed, the material does not return to its original state. If a large amount of plastic deformation takes place between the elastic limit and the fracture point, the metal is said to be ductile; but if fracture occurs soon after the elastic limit is passed, the metal is said to be brittle.

1.8.6 The APLHGR (ECCS/LOCA) Limit

Average planar linear heat generation rate (APLHGR) is the average LHGR of all fuel rods in a given fuel bundle in a given horizontal plane (actually a 6 inch slab or node). This parameter is important in the core heatup analysis for a loss of coolant accident (LOCA).

In the event of a LOCA, the heat stored in the fuel at the time of the accident and the decay heat produced following the accident could significantly damage the fuel. In the case of some LOCAs, nucleate boiling is maintained around the fuel long enough for the majority of the stored energy in the fuel to be conducted to the coolant; thus fuel damage is minimized. In the design basis LOCA, however, the core region is voided of liquid coolant in a relatively short time. If the fuel is operating at a high power level, the stored energy in the fuel could lead to a gross cladding failure and possible severe degradation of core geometry. Once water is removed from the cladding, radiation is the only heat transfer mechanism.

Gross cladding failure is prevented by placing a limit on the power level which would result in a peak cladding temperature (PCT) of 2200°F following a LOCA. The thermal limit specified is the APLHGR which is used because the PCT following a postulated LOCA is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is dependent only secondarily on the rod-to-rod power distribution within an assembly. The peak cladding temperature is calculated assuming an LHGR for the highest powered rod less than or equal to the design LHGR corrected for fuel densification.

The thermal limit in this case is given in terms of the incore maximum APLHGR (the MAPLHGR) and is specified for each individual fuel type as a function of fuel exposure. The units of the MAPLHGR are the same as those of the LHGR (KW/ft) even though the parameters are different. Typical MAPLHGR limiting values are depicted in Figures 1.8-7, 1.8-8 and 1.8-9.

1.8.6.1 ECCS/LOCA and Thermal-Mechanical Limits

The ECCS/LOCA limit and the thermal-mechanical limit can be combined into one number. Current GE BWR MAPLHGR limits (as a function of exposure) are based on the most limiting value of either the ECCS/LOCA limits or the thermal-mechanical design limits. Since the thermal-mechanical design limit is included in the determination of the MAPLHGR limit, it can not be exceeded if the MAPLHGR limit is met. Separate specification of the steady-state thermal-mechanical limit in the Technical Specifications is redundant. Therefore, GE has proposed and the NRC has agreed that the separate, redundant steady-state thermal-mechanical limits be eliminated from Technical Specification. The MAPLHGR limit will continue to provide assurance that the limits in 10CFR50.46 will not be exceeded, and that the fuel design analysis limits defined in NEDE-24011-P-A (GESTAR-II) will be met.

1.8.7 The CPR Safety Limit

Critical power is the fuel bundle power required to cause transition boiling somewhere in the fuel bundle. The critical power ratio (CPR) of a fuel bundle is the ratio of its critical power to the actual fuel bundle operating power. The CPR is a measure of how close to transition boiling a fuel bundle is operating. The minimum value of the CPR for all fuel bundles in the core is the minimum critical power ratio (MCPR) and represents the fuel bundle which is the closest to transition boiling. MCPR limits are imposed to avoid fuel damage due to severe overheating of the cladding.

The MCPR safety limit is set at 1.07 to ensure that in the event of an abnormal operating transient, more than 99.9% of the fuel rods in the core are expected to avoid transition boiling, provided the limit is not exceeded. The margin between an MCPR of 1.0 (onset of transition boiling) and the safety limit is derived from a detailed statistical analysis of uncertainties in monitoring the operating state of the core and in the boiling transition correlation.

1.8.7.1 The GEXL Correlation

The General Electric Critical Quality (X_c) - Boiling Length (L_B) (GEXL) correlation predicts the occurrence of transition boiling. The onset of transition boiling point is a function of many parameters: local steam quality, heat flux, mass flow rate, boiling length (LB), pressure, flow geometry, core inlet subcooling, and local peaking patterns. The General Electric Company (GE) has conducted extensive experimental investigations of the transition boiling event and its relation to these parameters over their design range.

The boiling transition testing was done at the Atlas Test Facility, which was specifically designed to simulate real reactor conditions and to handle transients as well as steady state conditions. The Atlas Test Facility test fuel bundle components (lower tie plate, fuel rod interim spacers, upper tie plate, fuel channel, and fuel bundle rods) were dimensionally the same as those used in the BWR. The fuel rods were electrically heated to simulate nuclear heat. Axial power shapes and local peaking of specific rods were simulated by varying the thickness of the fuel rod conductor. The fuel rods which were expected to achieve the onset of transition boiling (OTB) were instrumented with thermocouples to monitor temperature changes.

The testing procedure involved constructing a bundle with the desired axial power shape and local peaking factor and then establishing a constant system pressure, bundle flow rate, and inlet subcooling. The power in the test bundle was slowly increased until the onset of transition boiling was indicated by a signal from a rod thermocouple. A rod thermocouple was considered to be indicating a boiling transition condition when about 25°F rod surface temperature rise was observed on a strip chart recorder. The bundle power at this point was defined as critical power.

The data for each boiling transition point were recorded and the inlet subcooling was changed for the next test. The inlet subcooling was varied for each test until the desired range had been covered. A similar procedure was used to study the variation in the OTB point relative to changes in reactor pressure, inlet flow, axial power shape, and local peaking.

The test results of the ATLAS Test Facility were analyzed extensively to find a correlation that could be used to predict how close actual operating conditions are to OTB. GE chose the critical quality versus boiling length for correlating the boiling transition data because it is independent of axial flux profile and subcooling, it yields good precision, and it is fairly simple to apply in both design and operation.

Critical quality X_c is defined as the fuel bundle planar quality at the plane where boiling transition occurs, and boiling length L_B is defined as the distance from the plane where bulk saturation conditions are reached to the plane where boiling transition occurs.

The GEXL correlation curve in Figure 1.8-10 shows that, for that specific reactor condition, boiling transition occurs at a distance LB from the plane of saturation if the quality rises to the critical quality X_c. Both the axial elevation where saturation occurs and the quality at any axial elevation can be easily calculated by doing an enthalpy balance on the fuel bundle. This makes the determination of the critical power ratio (CPR) using the GEXL correlation relatively simple to apply to existing operating conditions.

Line (1) of Figure 1.8-10 represents a heat balance plot of average steam quality versus distance along an operating fuel bundle from the start of bulk boiling. For the same initial parameters as the operating bundle (mass flow rate, pressure, etc.), line (3) represents the quality-boiling length points which would result in transition boiling. By successively increasing bundle power from its initial level, a set of curves, located somewhere above curve (1), can be generated until the bundle power is high enough that its curve becomes tangent at some point to the correlation curve (3). The bundle power corresponding to curve (2) is the critical power, or the bundle power that is required to cause transition boiling in the bundle at the reactor conditions of interest.

The GEXL correlation is valid over the range of conditions used in the tests of the data used to develop the correlation. These conditions are listed in Table 1.8-2.

1.8.8 The Thermal Power Safety Limit

The use of the GEXL correlation is not valid for all critical power calculations at pressures below 785 psig or core flows less than 10% of rated flow. Therefore, the fuel cladding integrity safety limit under these plant conditions is established by other means. This is done by establishing a limiting condition on core thermal power with the following basis. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a bundle flow of 28×10^3 lb/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be greater than 28×10^3 lb/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a thermal power of more than 50% of rated thermal power. Thus, a thermal power limit of 25% of rated thermal power for reactor pressure below 785 psig is conservative.

1.8.9 Methods to Prevent Exceeding Thermal Limits

The thermal limits listed in the plant technical specifications include two safety limits and three limiting conditions for operation (LCOs).

The LCO for linear heat generation rate (LHGR) is a maximum value of 13.4 KW/ft (for older style of 8 x 8 fuel). The LHGR limit can be exceeded during abnormal operational transients but is limited by the limiting safety system settings (LSSS) of the Reactor Protection System (RPS). The Average Power Range

Monitor (APRM) System high power scram setpoint value is a maximum of 120% of rated power. Thus, with the core at 100% power and some fuel rod operating at its design LHGR (13.4 KW/ft), the maximum LHGR achieved by that fuel rod is 16.08 KW/ft (13.4 KW/ft x 120%), which is still well below the LHGR required to cause 1% plastic strain.

The LCO for maximum average planar linear heat generation rate (MAPLHGR) is listed in graph form in the technical specifications as a function of fuel type and exposure. Administrative control (procedures and technical specification license requirements) is the only method to prevent exceeding this thermal limit.

The minimum critical power ratio (MCPR) safety limit is a minimum value of 1.07. The limiting safety system settings (LSSS) of the Reactor Protection System (RPS), in conjunction with a higher operating MCPR limiting condition for operation (LCO) discussed in Section 1.8.10.1, prevent exceeding the safety limit even during abnormal operational transient. This safety limit is applicable when reactor pressure >785 psig and core flow >10% of rated flow.

The core thermal power safety limit, applicable when reactor pressure <785 psig or core flow <10% of rated, is 25% of rated core thermal power. The limiting safety system settings (LSSS) of the Reactor Protection System (RPS) (specifically the fixed APRM scram at 15% of rated power with the reactor mode switch not in the run position) prevent exceeding the safety limit.

1.8.10 Thermal Limit or Protection Modification

Some additional considerations must be taken into account in order to ensure that the thermal limits and the protection against exceeding these limits are adequate so that the thermal-limit function is achieved even under unusual circumstances. These considerations are discussed in the paragraphs that follow.

1.8.10.1 MCPR Modifications for Limiting Transients

The required MCPR at steady state operating conditions is derived from the established fuel cladding integrity safety limit MCPR of 1.07, and an analysis of abnormal operational transients. To ensure that the fuel cladding integrity safety limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine those which result in the largest reduction in critical power ratio (ΔCPR). The types of transients evaluated include turbine trip without bypass valves, generator load rejection without bypass valves, feedwater controller failure, pressure regulator failure downscale, loss of feedwater heating, fuel loading error, and rod withdrawal error.

The transients are categorized as rapid pressurization events (turbine trip without bypass, generator load rejection without bypass, feedwater controller failures, pressure regulator downscale failure) and nonpressurization events for the purpose of analysis. The slower, nonpressurization transients are analyzed using either the steady state three dimensional BWR core simulator or the REDY transient model. Rapid pressurization events are analyzed using the "One Dimensional Core Transient Model"

(ODYN) and is performed in accordance with either or both of two methods known as Option A and Option B.

The operating MCPR limit is obtained by addition of the absolute, maximum ΔCPR value for the most limiting transient (including any imposed adjustment factors) to the fuel cladding integrity safety limit MCPR as shown in Figure 1.8-11. The MCPR operating limit at rated power and rated core flow is referred to as MCPR (TRANS-ECCS).

Using Option A, the pressurization transient is analyzed using the ODYN models to obtain the change in the critical power ration (ΔCPR) for the core. The licensing basis OLMCPR is given as:

OLMCPR = 1.044 (Safety Limit CPR +ΔCPR)

Using Option B, a statistical determination of the pressurization transient ΔCPR such that there is a 95% probability with 95% confidence (95/95) that the event will not cause the critical power ratio to fall below the MCPR Safety Limit. Utilities using Option B must demonstrate that their plant's scram speed distribution is consistent with that used in the statistical analysis. This is accomplished through an approved technical specification which requires testing and allows adjustment of the operating limit MCPR if the scram speed is outside the assumed distribution as shown in Figure 1.8-12.

At conditions of less than 100% core flow or less than 100% power, the MCPR operating limit is additionally adjusted. During these operational conditions, transients such as Rod Withdrawal Errors, Feedwater Controller Failures, or recirculation pump runouts become limiting. For this reason, the operating limit MCPR is raised to compensate for such transients. For reactor power conditions between 25% and 30% the MCPR operating limit is provided in Figure 1.8-13. For reactor power conditions greater than 30% power the MCPR operating limit is the greater of either (1) the flow-dependent MCPR limit shown in Figure 1.8-14 or (2) the appropriate kp given by Figure 1.8-13, multiplied by the rated flow and rated power limit obtained from Figure 1.8-12. These reduced flow or power limits are established to protect the core from inadvertent core flow or power increases.

1.8.10.2 APLHGR Modifications

The APLHGR limits obtained from Technical Specification charts are further adjusted for low flow (Figure 1.8-15) or low power (Figure 1.8-16) conditions. These additional adjustments assure that the fuel thermal-mechanical design criteria are preserved during abnormal transients initiated from off rated flow or power conditions.

$MAPFAC_p$ is usually determined from the feedwater controller failure event results. $MAPFAC_f$ is usually determined by the recirculation pump runout event results.

Below P_{Bypass}, there is significant sensitivity to core flow during transients. P_{Bypass} is defined as the power level at which a reactor scram on turbine stop valve position/turbine control valve fast closure is bypassed. For this reason the $MAPFAC_p$ is further defined separately for a high flow (>50% core flow) and a low flow condition (≤50%).

1.8.11 Differences

"ARTS" modifications have been made to many plants. "ARTS" relates to changes in the Average Power Range Monitoring (APRM) System scram setpoints, the Rod Block Monitoring (RBM) System, and the Technical Specifications relating to these systems.Figure 1.8-17 shows thermal limits and modifications for BWR/3, BWR/4, and BWR/5 product line plants not using "ARTS".

1.8.11.1 Modifications Associated with the LHGR Limit for plants not using "ARTS"

Peaking factors are calculated by the Process Computer System and can be used to determine LHGRs. Three separate peaking factors are calculated, then combined as a total peaking factor as follows:

Local peaking factor = maximum fuel rod heat flux in that fuel assembly at a given elevation / average fuel rod heat flux in that fuel assembly at the given elevation.

Radial peaking factor = power produced in a fuel assembly / core average fuel assembly power

Axial peaking factor = maximum heat flux along the length of a given fuel rod / average heat flux of that fuel rod

The product of these three factors, illustrated in Figure 1.8-18, is the total peaking factor (TPF); and the maximum value of the TPFs for all core locations is automatically typed out by the periodic calculation program as the core maximum peaking factor (CMPF). Another definition for TPF or CMPF is:

CMPF = heat flux of the highest power fuel rod segment / average heat flux of a fuel rod segment in the core

In the case of a uniformly loaded core (all the same type 8 x 8 fuel), a design TPF (DTPF) is generated. The value of this DTPF is such that the highest powered fuel rod segment would be operating at its design LHGR when the core is operating at rated power with the DTPF. If the core is operated at rated power with a TPF > DTPF, then a fuel rod segment somewhere in the core would be greater than its design LHGR, thus reducing the margin to its 1% plastic strain limit.

Technical specifications do not directly limit the DTPF, so it is possible to operate the core with TPFs >DTPF. Therefore, the core could be operated at 60% of rated power with the highest powered fuel rod at its design LHGR (i.e., 13.4 KW/ft). Now, a transient could cause power and LHGR's to double before the scram point is reached, which would put the high powered rods in the vicinity of the 1% plastic strain threshold. To preclude this, whenever a TPF >DTPF, the APRM Scram and Rod Block setpoints must be reduced by the factor DTPF/CMPF which results in a maximum LHGR of <120% of design at the scram setpoint.

For cores that have a mixed fuel configuration, CMPF is not as useful a tool, since the DTPFs are different for different type fuels. In these cases, an equivalent expression is used to determine when the APRM setpoints must be reduced:

$$\text{FRP} / \text{CMFLPD}$$

where FRP is the fraction of rated power and CMFLPD is the core maximum fraction of the limiting power density (FLPD) defined by

$$\text{FLPD} = \frac{\text{LHGR (actual)}}{\text{LHGR (design)}}$$

The Process Computer System will automatically print out the value of CMPF or CMFLPD. Whenever the applicable fraction is less than 1.0, the APRM setpoints must be reduced.

1.8.11.2 Modifications Associated with the APLHGR Limit for plants not using "ARTS"

Plants not using the "ARTS" modification do not adjust the APLHGR limits with MAPFRAC_P or MAPFRAC_F.

1.8.11.3 Modifications Associated with the CPR Limit for BWR/6 product lines.

Figure 1.8-21 shows thermal limits and modifications for BWR/6 product line plants not using "ARTS".

These plants use MCPR_f (Figure 1.8-19) or MCPR_p (Figure 1.8-20) as adjustments to MCPR for operation less than rated flow or power conditions. The greater of the two limits obtained from the figures is used as the adjusted MCPR operating limit.

1.8.12 Summary

Thermal limits minimize the radiological release from the plant during normal operation, abnormal operation, abnormal operational transients, and postulated accidents by restricting plant operation so that the fuel cladding integrity is maintained. Limits are imposed on linear heat generation rate (LHGR) and critical power ratio (CPR) to ensure fuel cladding integrity during normal and transient operation. Average planar linear heat generation rate (APLHGR) is limited to meet ECCS criteria.

Two fuel damage mechanisms are considered: fuel cladding cracking caused by high stress and fuel cladding cracking or failure caused by lack of cooling.

LHGR limits are imposed to prevent fuel cladding perforation because of mechanical stress of the fuel pellets. The LHGR limit is set to account for high local power peaking.

MCPR limits are imposed to prevent cladding perforation because of the onset of transition boiling (breakdown of the heat transfer mechanism), and are modified to account for transients and flow conditions less than rated. When outside of the bounds of the correlation normally used to calculate CPR, limits are imposed on core thermal power (CTP).

MAPLHGR limits are imposed to restrict the amount of stored energy in the fuel, thus limiting the rate of cladding heatup on a LOCA. Power and flow dependent corrections are applied to the rated conditions APLHGR to assure that the fuel thermal-mechanical design criteria are preserved during abnormal transients initiated from below rated conditions.

TABLE 1.8-1 COMPARISON OF BWR CORE PARAMETERS

PARAMETER	BWR/3	BWR/4	BWR/5	BWR/6
RATED POWER (MWT)	2527	3293	3323	3579
NUMBER OF FUEL ASSEMBLIES	724	764	764	748
NUMBER OF CONTROL RODS	177	185	185	177
VESSEL SIZE	251"	251"	251"	238"
RATED CORE FLOW (LBS/HR)	98×10^6	102.5×10^6	108.5×10^6	104×10^6
RATED STEAM/FEEDWATER FLOW (LBS/HR)	9.8×10^6	13.4×10^6	14.2×10^6	15.4×10^6
AVERAGE POWER DENSITY (KW/LITER)	36.6	50.7	50.0	54.1
HEAT FLUX (WATTS/CM^2)	41.7	51.8	45.3	49.5
HEAT FLUX (BTU/HR-FT^2)	132,314	164,410	143,740	159,500
PEAK HEAT FLUX (BTU/HR-FT^2)	396,942	428,124	361,000	361,600
AVG. THERMAL POWER/BUNDLE (MWT)	3.49	4.31	4.35	4.78
TOTAL HEAT TRANSFER AREA (FT^2)	62,928	66,214	74,871	73,409
AVERAGE LHGR (KW/FT)	5.72	7.05	5.33	5.90
PEAK LHGR (KW/FT)	17.5	18.5	13.4	13.4
DESIGN OR MAX. TOTAL PEAKING FACTOR (FUEL TYPE DEPENDENT)	3.0	2.6	2.51	2.21
HEAT TRANSFER AREA/FUEL ROD (FT^2)	1.768	1.768	1.581	1.592

TABLE 1.8-2 GEXL BOUNDING PARAMETER VALUES

PARAMETER	VALUE		
Pressure:	800 to 1400 psia		
Mass Flux:	0.1 to 1.25 10^6 lb/hr-ft^2		
Inlet Subcooling:	0 to 100 Btu/lb		
Local Peaking:	1.61 at a corner rod to 1.47 at an interior rod		
Axial Peaking:	Shape	Max./Avg.	
	Uniform		1.0
	Outlet Peaked		1.60
	Inlet Peaked		1.60
	Double Peak		1.46 and 1.38
	Cosine	1.39	
Rod Array:	64 rods in an 8x8 array, 49 rods in a 7x7 array		

TABLE 1.8-3 THERMAL LIMITS CALCULATIONS

MFLCPR = Max Fraction Limiting Critical Power Ratio

$$\text{MFLCPR} = \frac{\text{MCPR Operating Limit}}{\text{MCPR Actual } \dfrac{\text{Critical Power}}{\text{Actual Power}}}$$

MFLPD = Max Fraction of Limiting Power Density

$$\text{MFLPD} = \frac{\text{MRPD}}{\text{RPDLM}} \qquad \frac{\text{Maximum LHGR (Actual) KW/ft}}{\text{LHGR Limit (13.4 KW/ft)}}$$

MAPRAT = Max Fraction of Limiting Average Planar Linear Heat Generation Rate

$$\text{MAPRAT} = \frac{\text{MAPLHGR (Actual Value of Average Planar LHGR) KW/ft}}{\text{MAPLHGR Limits (from T.S.) KW/ft}}$$

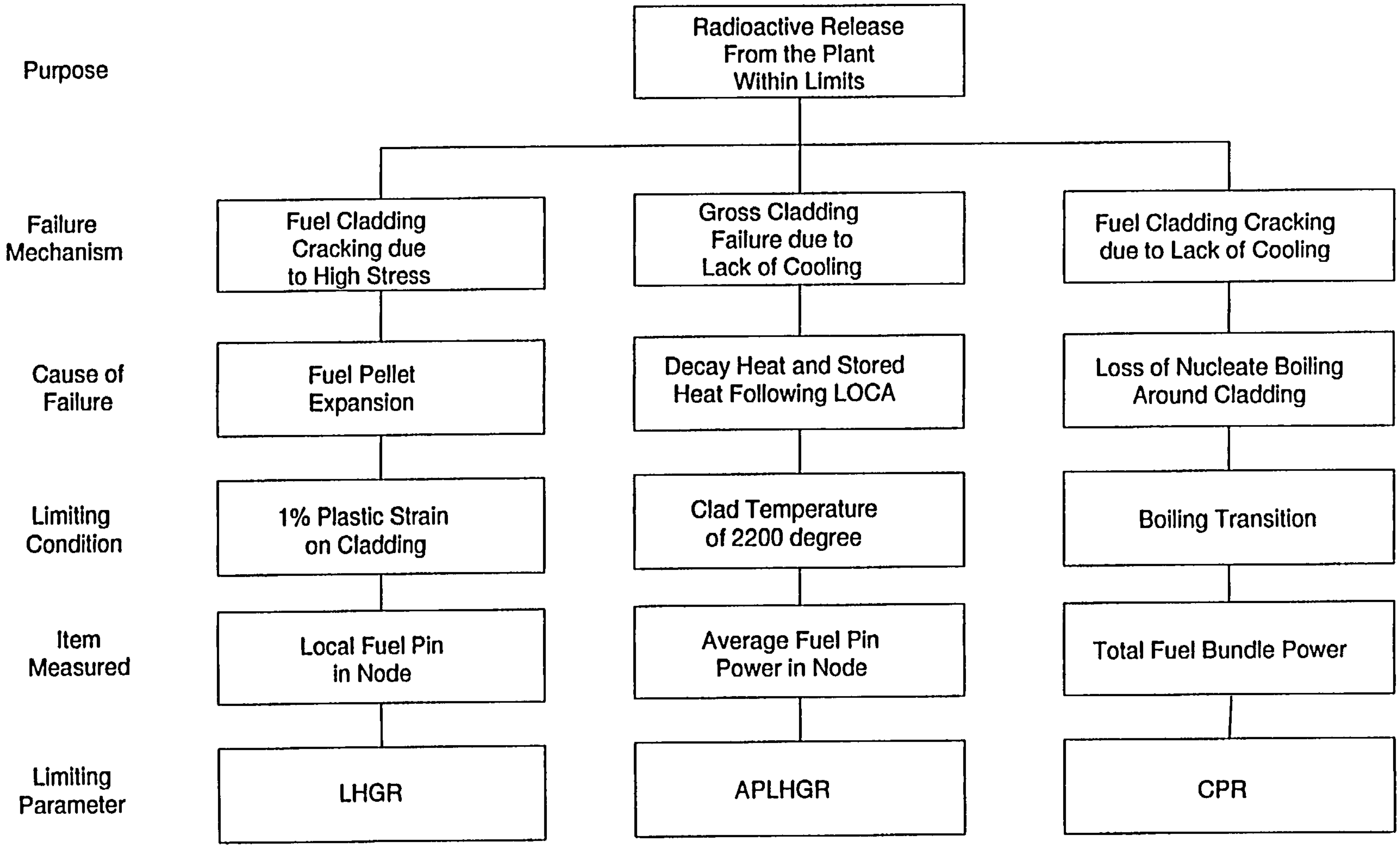


Figure 1.8-1 Thermal Limits

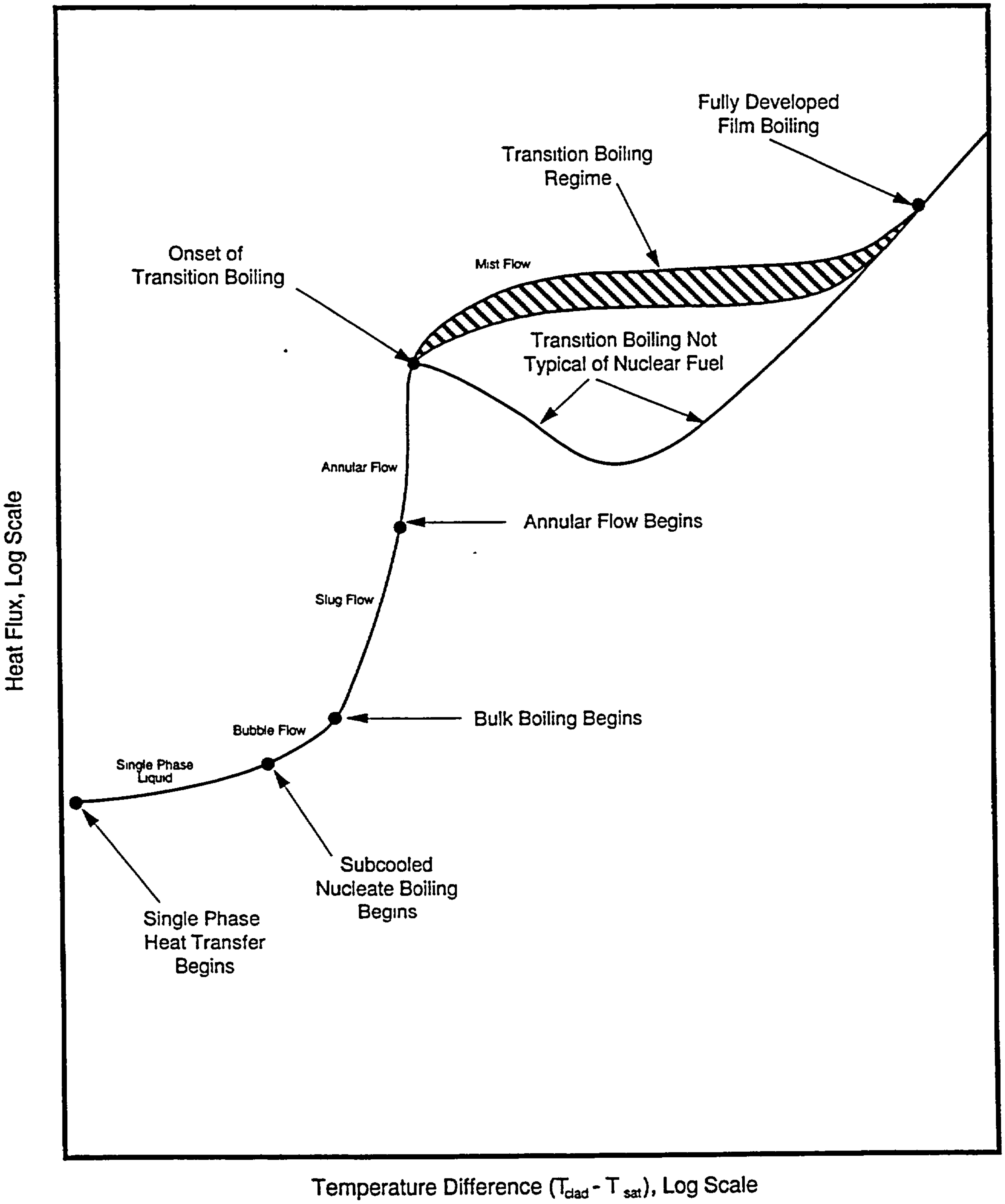


Figure 1.8-2 Forced Convection Boiling Curve

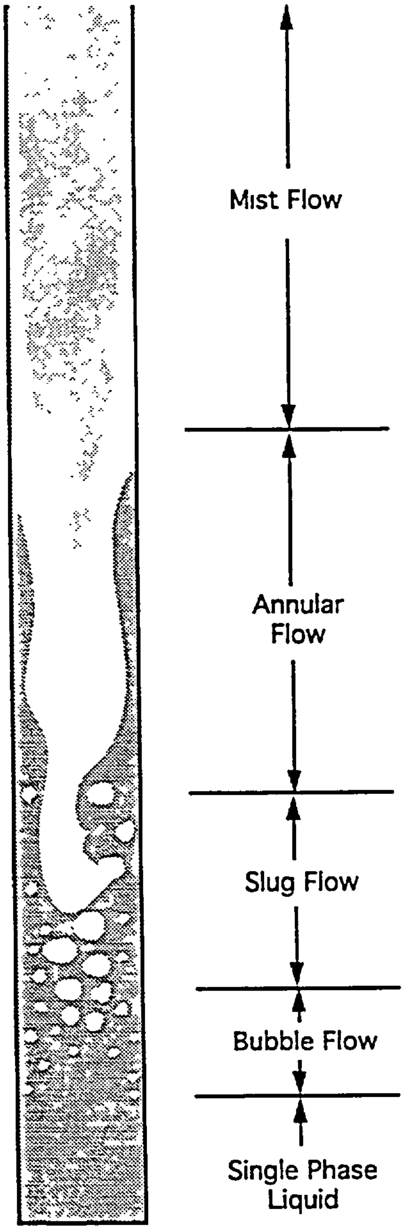


Figure 1.8-3 Regions of Boiling Heat Transfer

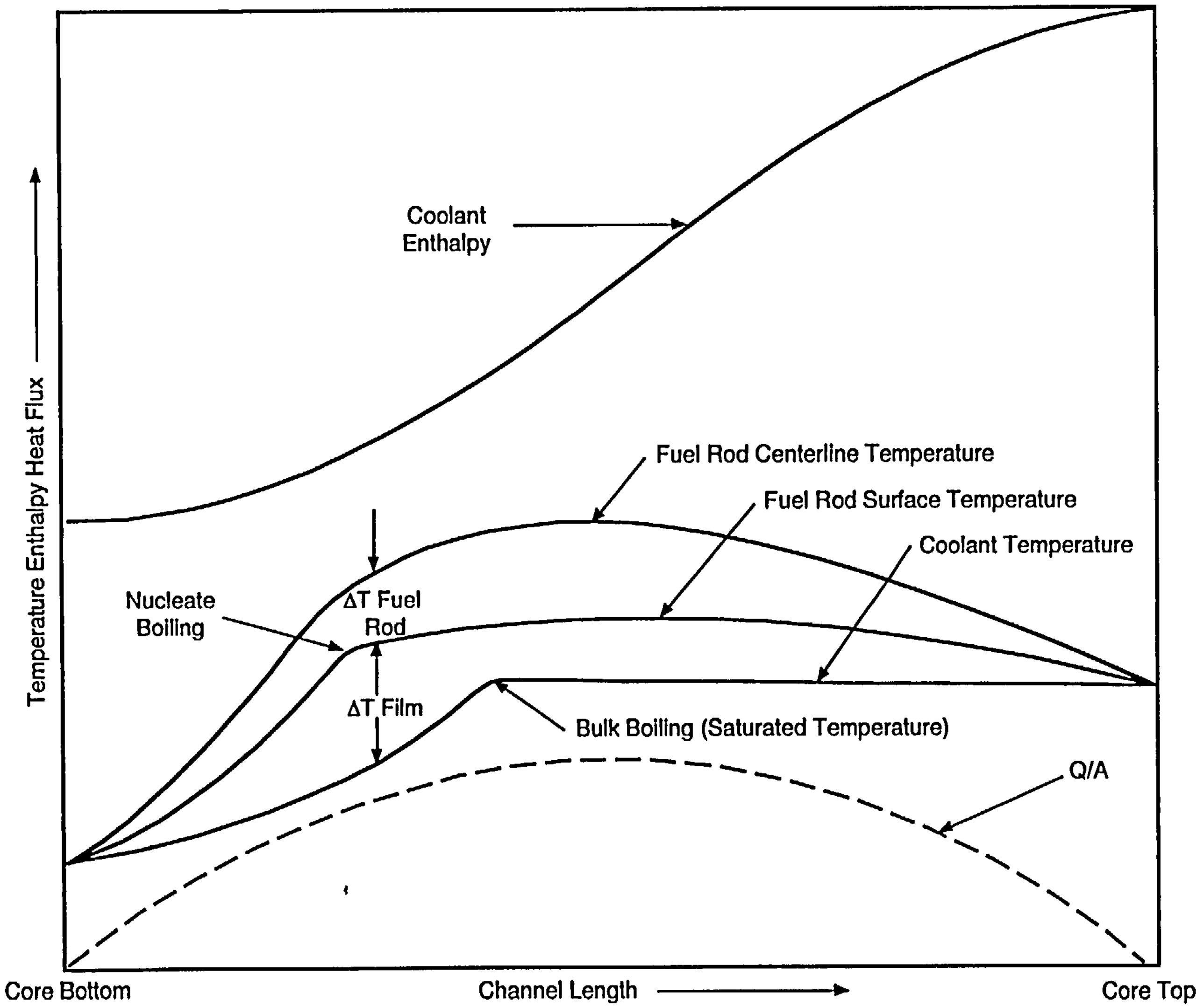


Figure 1.8-4 Plot of Coolant and Fuel Bundle Temperature vs. Flow Path Length

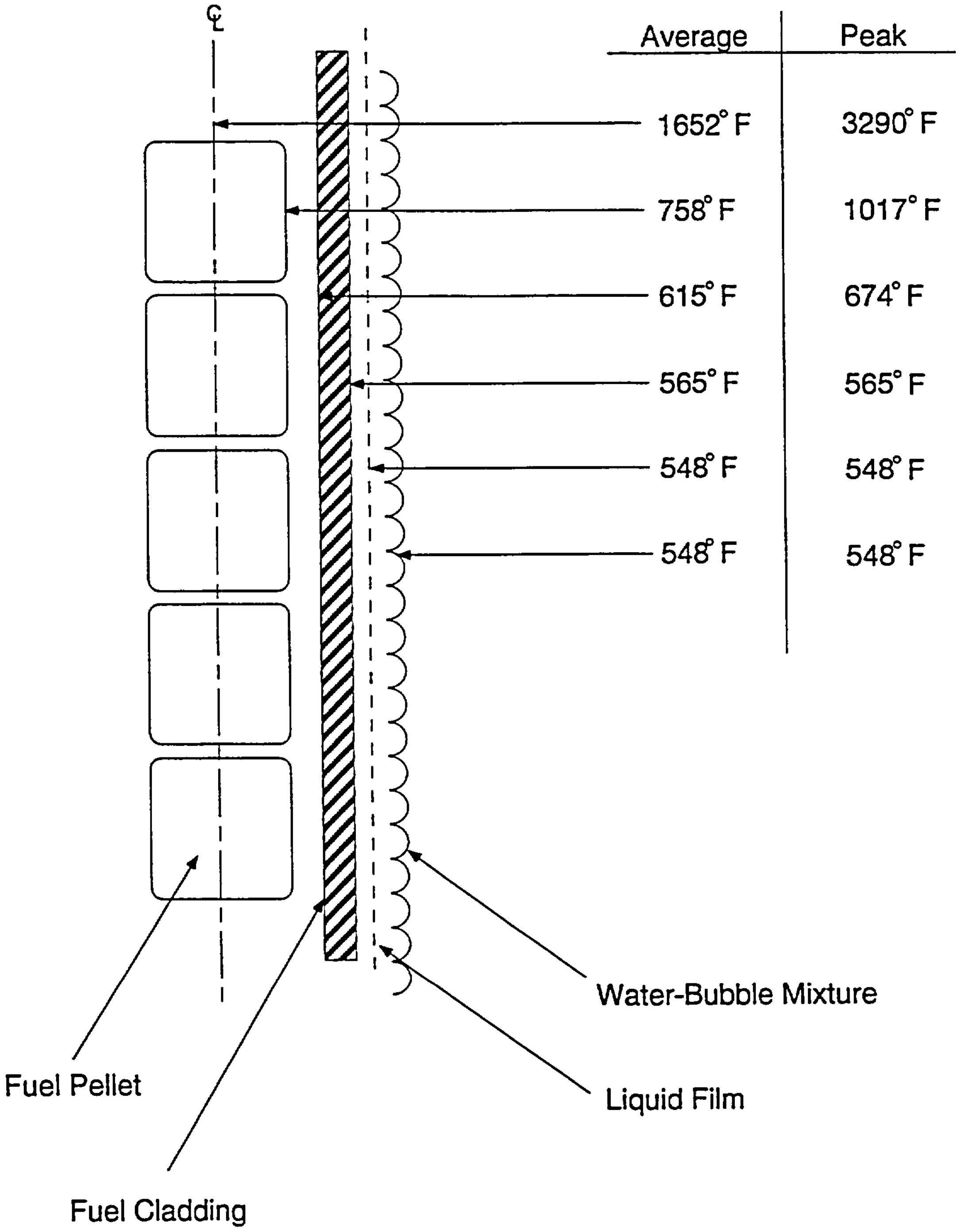


Figure 1.8-5 Fuel Temperature Cross Section

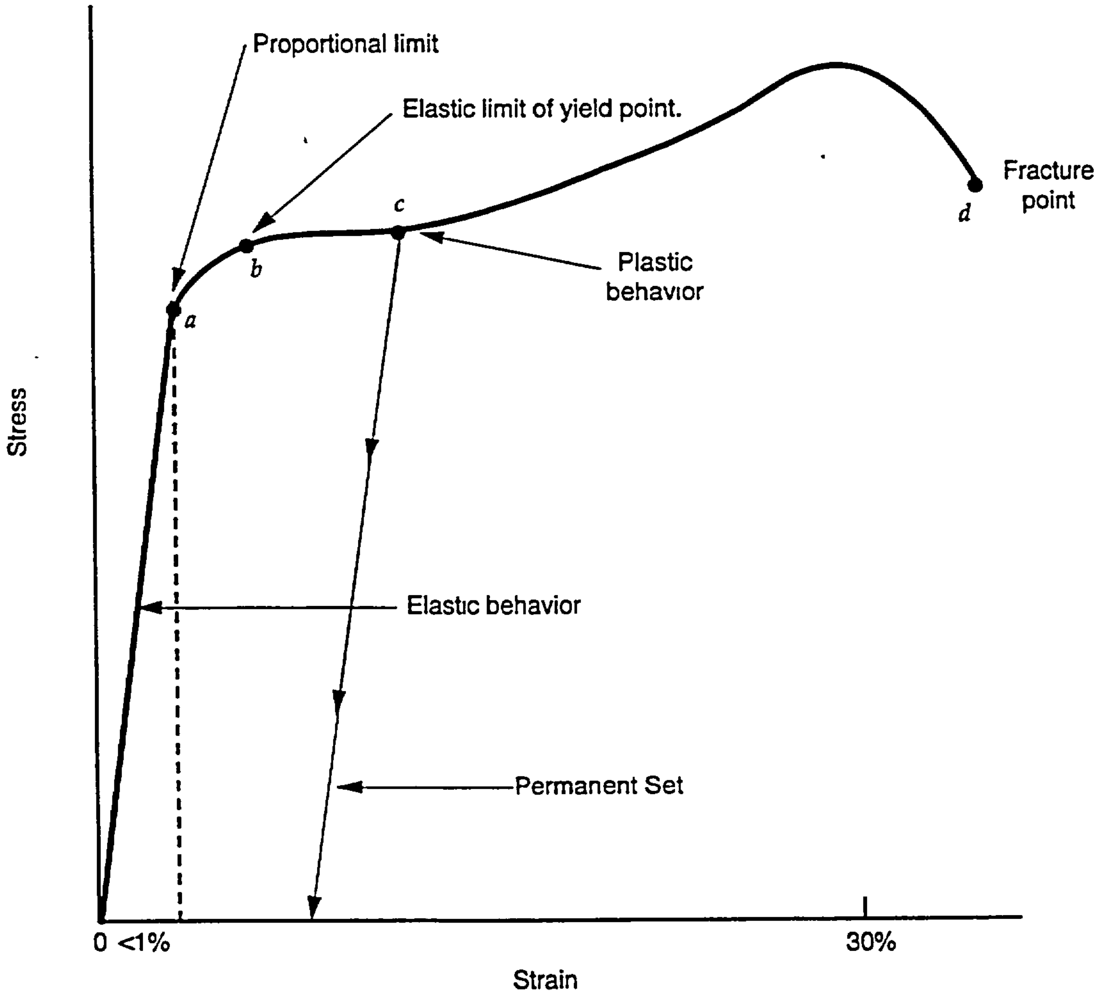
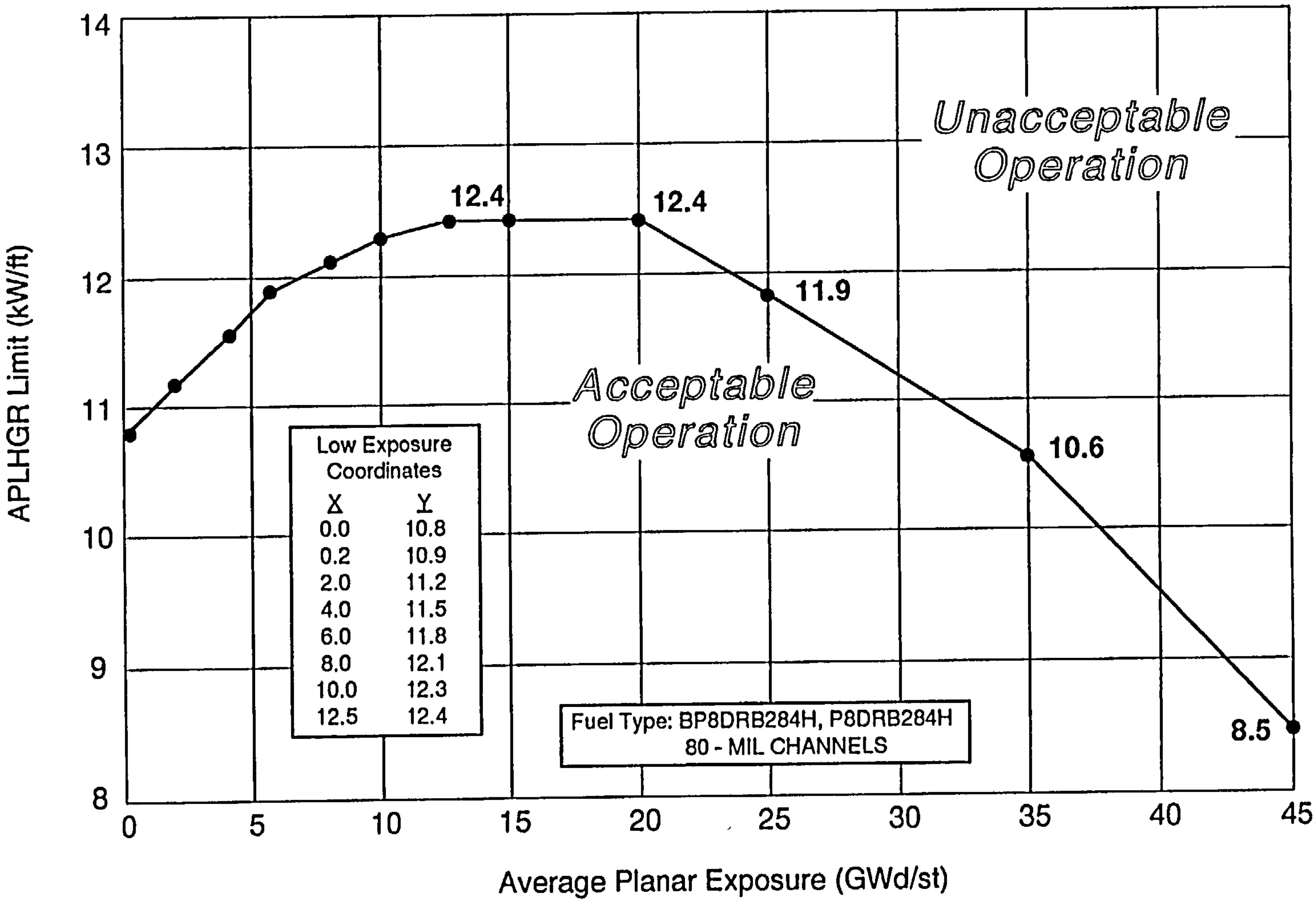


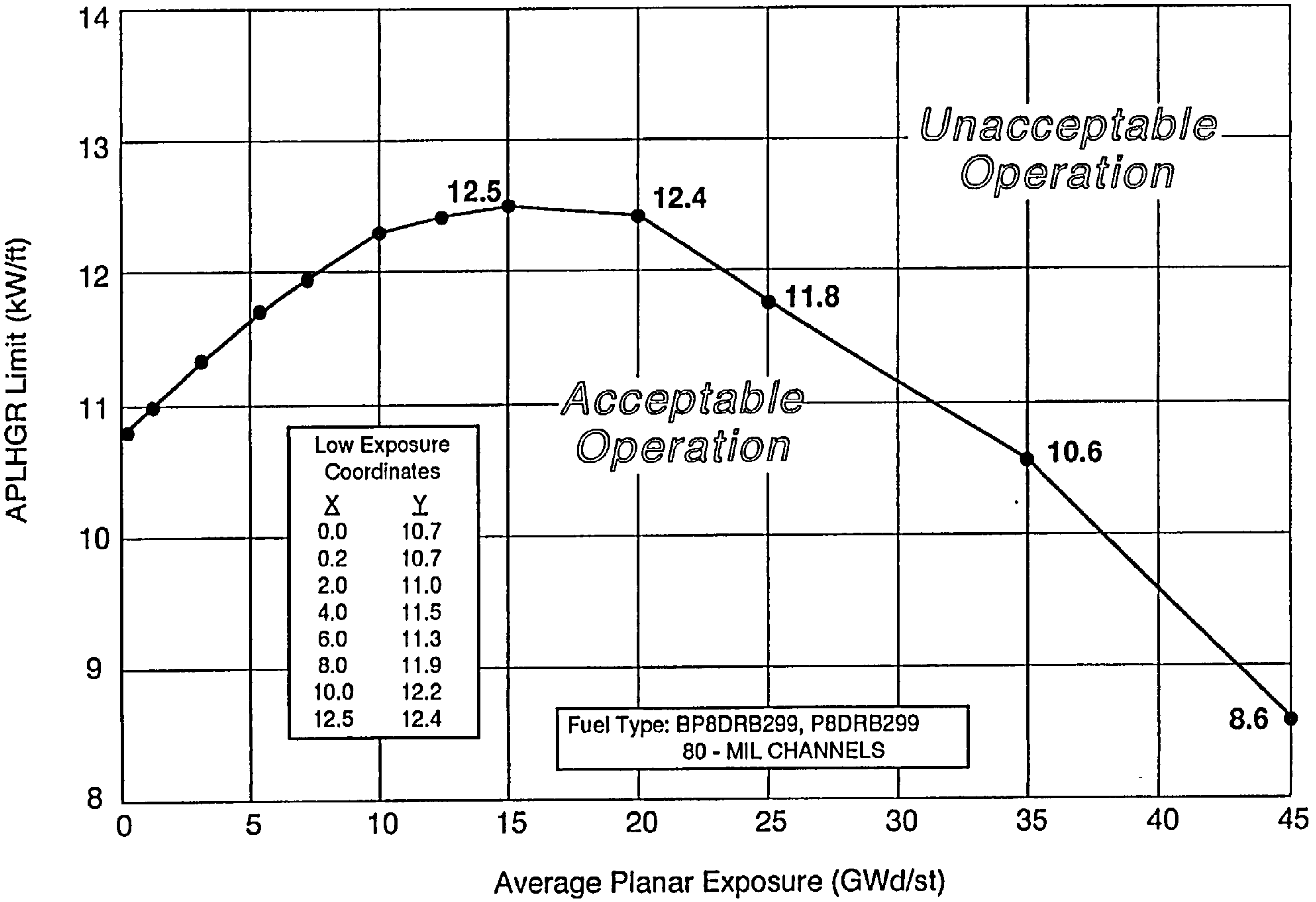
Figure 1.8-6 Typical Stress/Strain Curve for Ductile Metal



Low Exposure Coordinates

X	Y
0.0	10.8
0.2	10.9
2.0	11.2
4.0	11.5
6.0	11.8
8.0	12.1
10.0	12.3
12.5	12.4

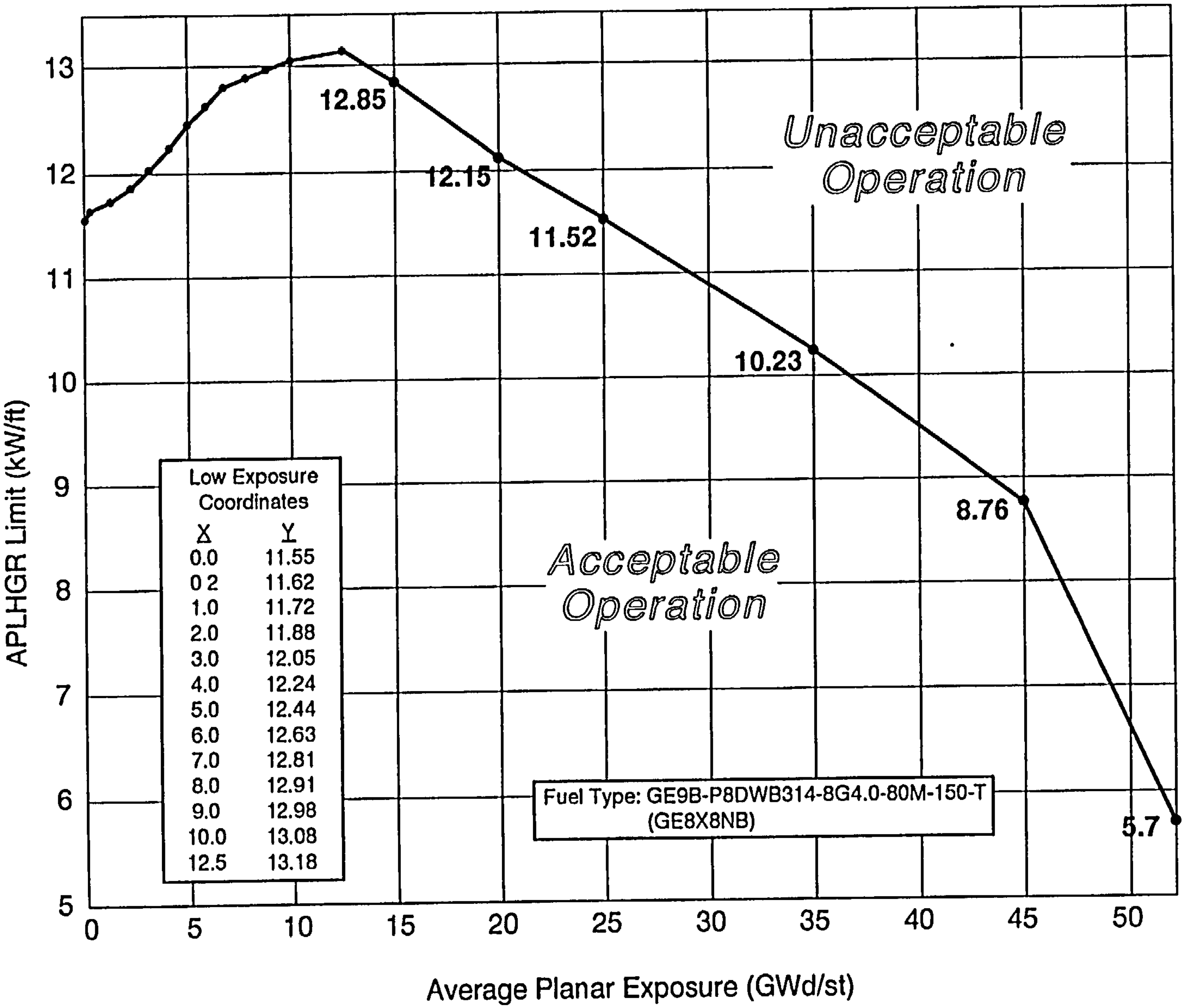
Figure 1.8-7 Average Planar Linear Heat Generation Rate Limit vs. Average Planar Exposure



Low Exposure Coordinates

X	Y
0.0	10.7
0.2	10.7
2.0	11.0
4.0	11.5
6.0	11.3
8.0	11.9
10.0	12.2
12.5	12.4

Figure 1.8-8 Average Planar Linear Heat Generation Rate Limit vs. Average Planar Exposure



Low Exposure Coordinates

X	Y
0.0	11.55
0.2	11.62
1.0	11.72
2.0	11.88
3.0	12.05
4.0	12.24
5.0	12.44
6.0	12.63
7.0	12.81
8.0	12.91
9.0	12.98
10.0	13.08
12.5	13.18

Figure 1.8-9 Average Planar Linear Heat Generation Rate Limit vs. Average Planar Exposure

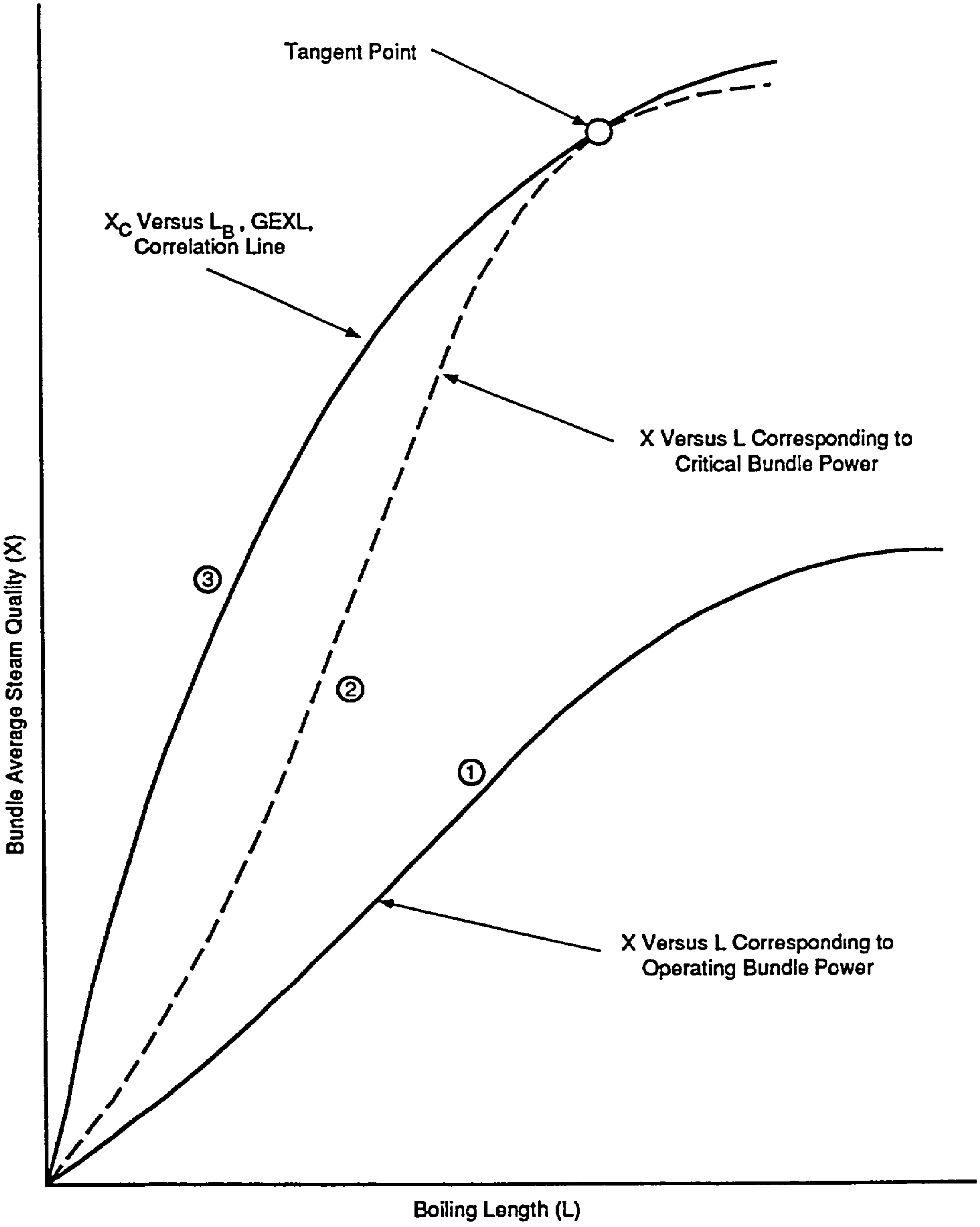


Figure 1.8-10 Critical Power Ratio

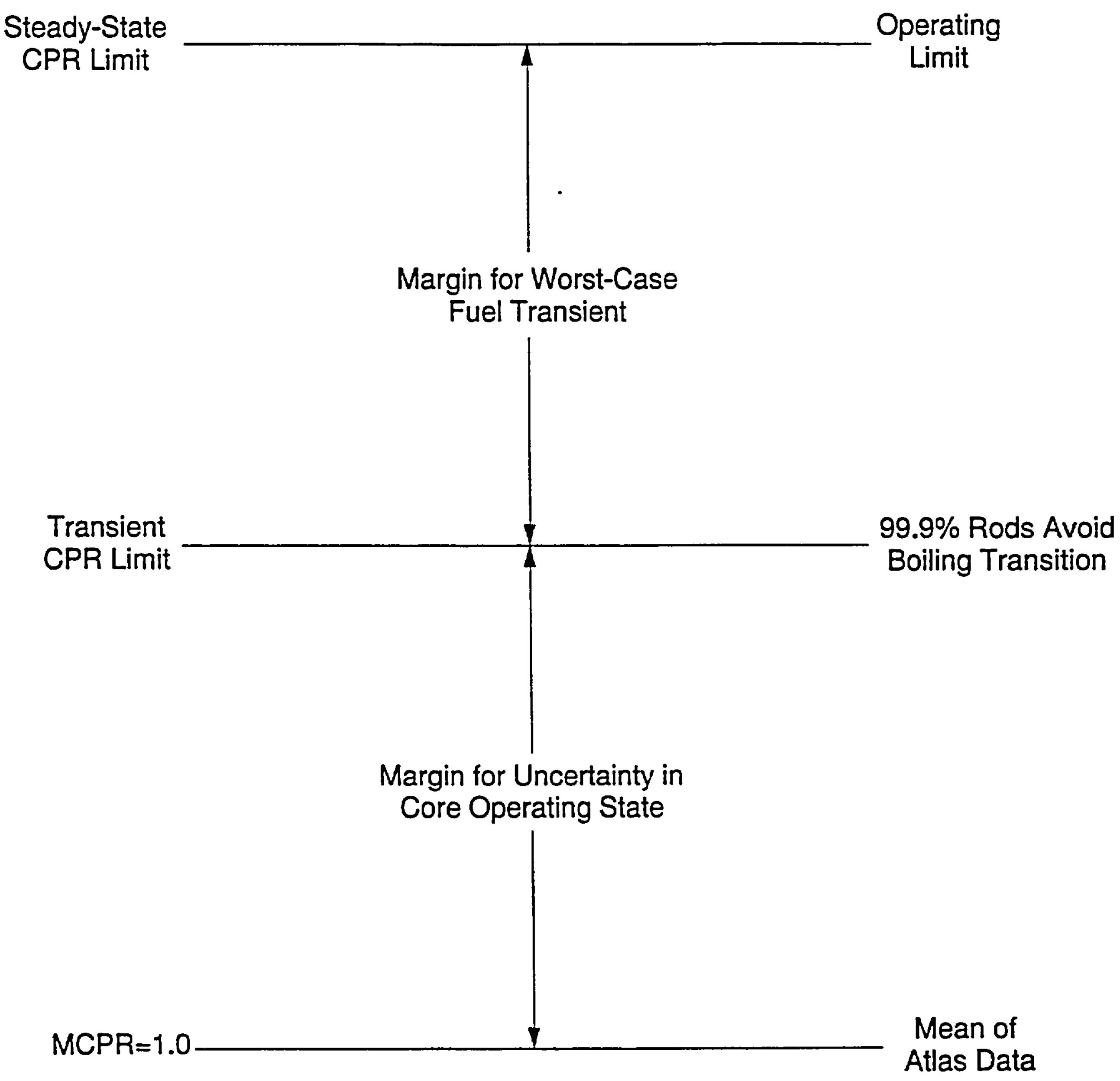


Figure 1.8-11 MCPR Limits

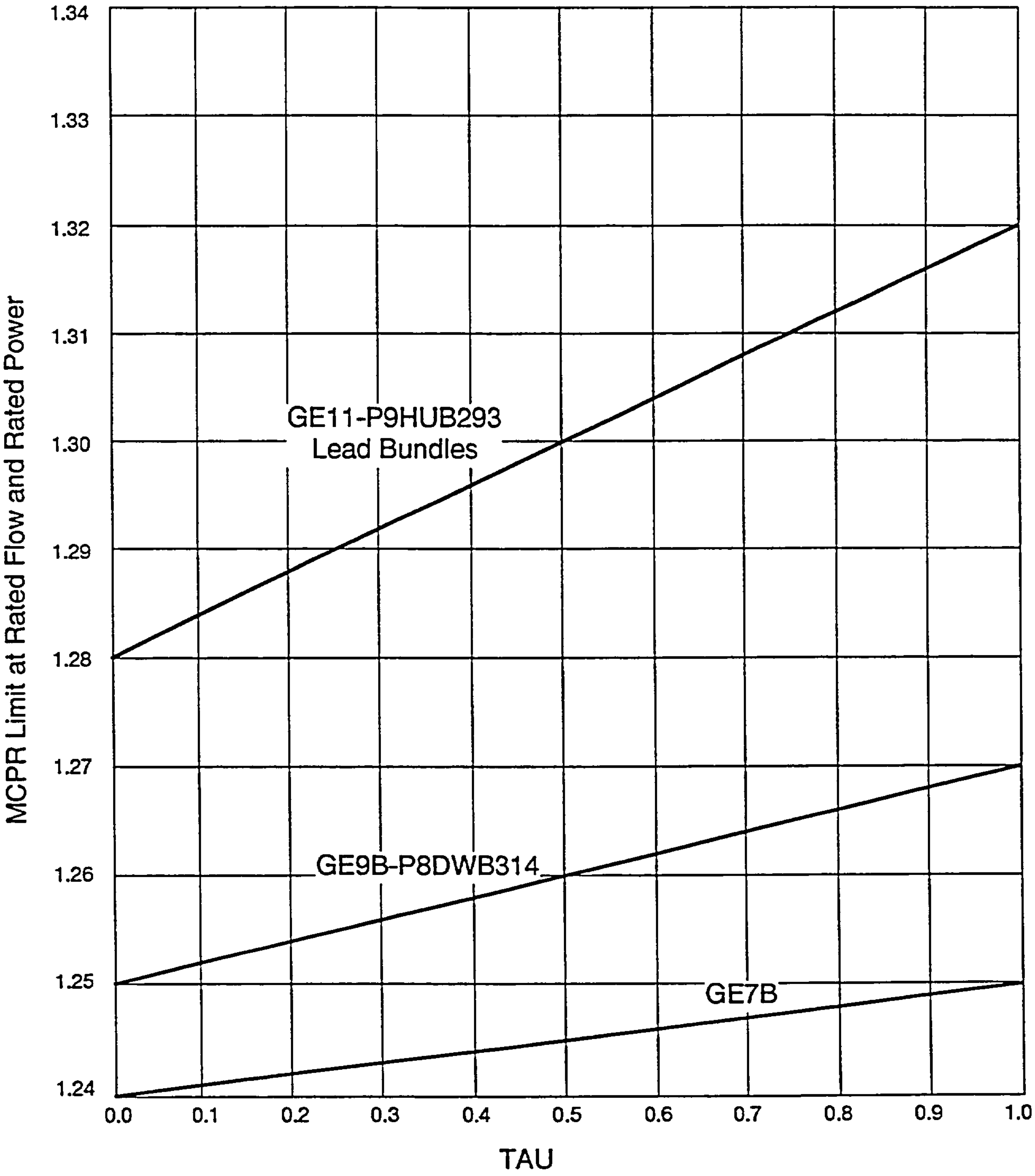


Figure 1.8-12 MCPR Limit as Function of Average Scram Time

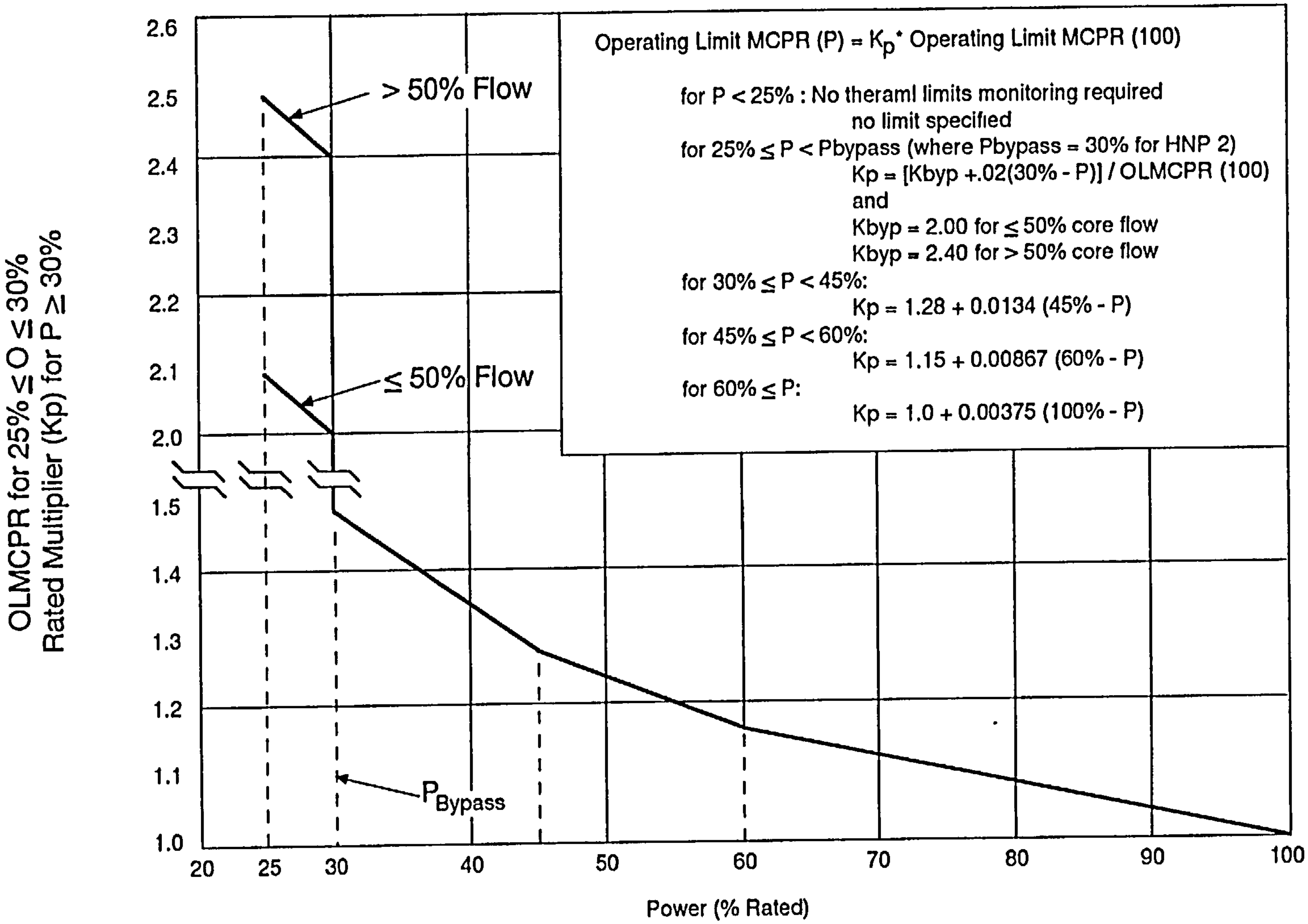
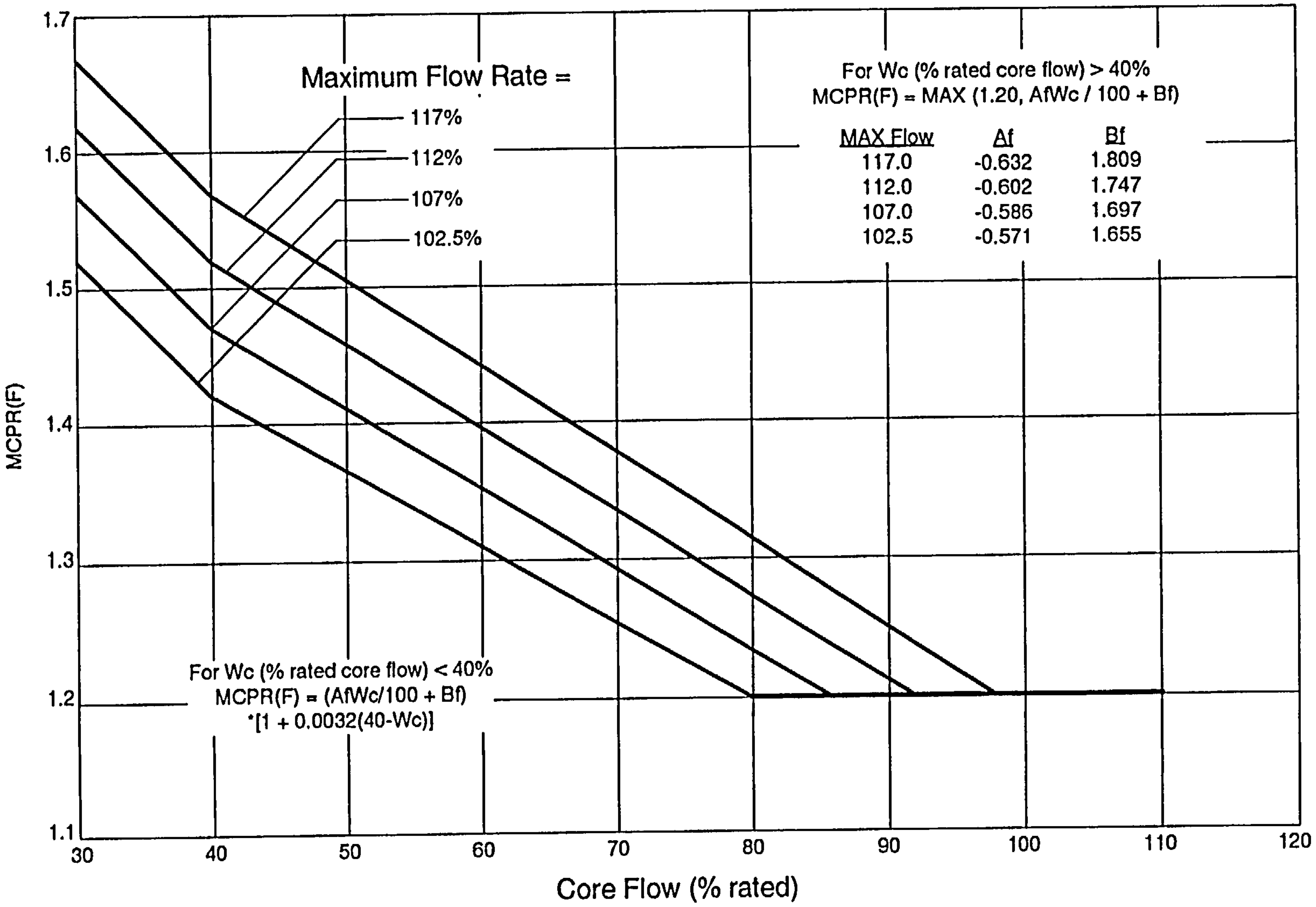


Figure 1.8-13 Power-Dependent MCPR Multiplier (Kp)

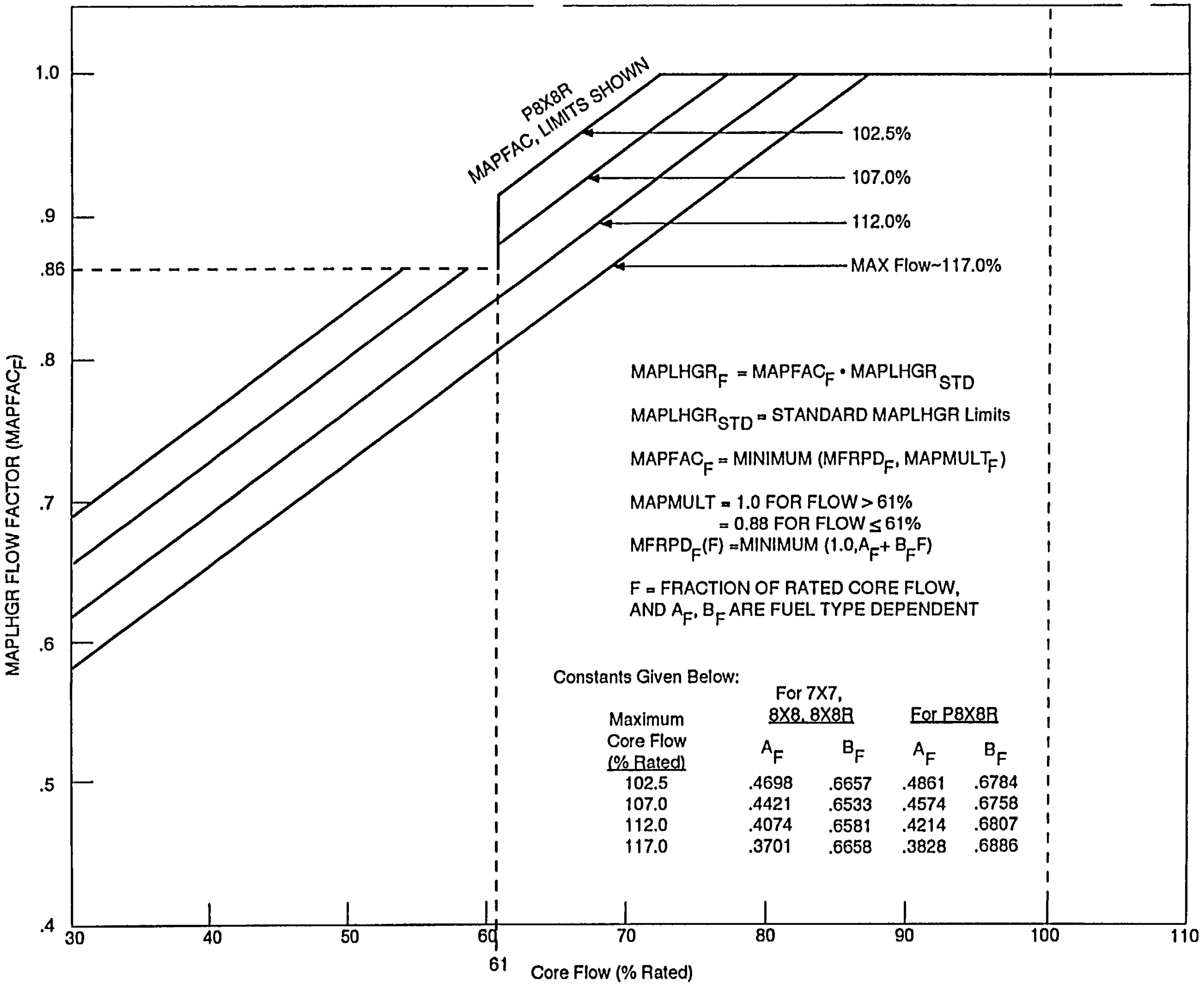


For Wc (% rated core flow) > 40%
MCPR(F) = MAX (1.20, AfWc / 100 + Bf)

MAX Flow	Af	Bf
117.0	-0.632	1.809
112.0	-0.602	1.747
107.0	-0.586	1.697
102.5	-0.571	1.655

For Wc (% rated core flow) < 40%
MCPR(F) = (AfWc/100 + Bf)
*[1 + 0.0032(40-Wc)]

Figure 1.8-14 Flow - Dependent MCPR Limits, MCPR(F)



$MAPLHGR_F = MAPFAC_F \cdot MAPLHGR_{STD}$

$MAPLHGR_{STD}$ = STANDARD MAPLHGR Limits

$MAPFAC_F = \text{MINIMUM} (MFRPD_F, MAPMULT_F)$

MAPMULT = 1.0 FOR FLOW > 61%
= 0.88 FOR FLOW ≤ 61%

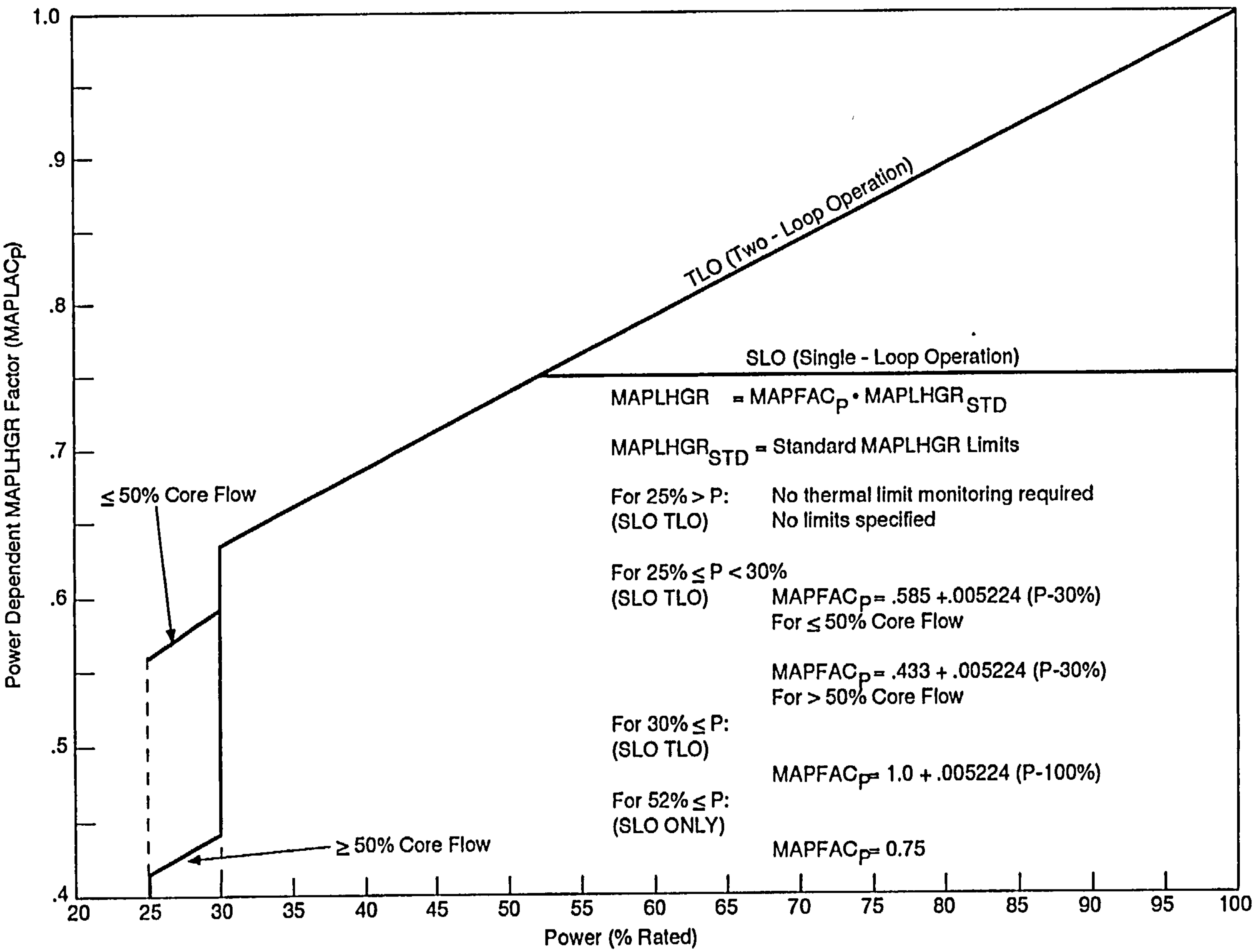
$MFRPD_F(F) = \text{MINIMUM} (1.0, A_F + B_F F)$

F = FRACTION OF RATED CORE FLOW,
AND A_F, B_F ARE FUEL TYPE DEPENDENT

Constants Given Below:

Maximum Core Flow (% Rated)	For 7X7, 8X8, 8X8R		For P8X8R	
	A_F	B_F	A_F	B_F
102.5	.4698	.6657	.4861	.6784
107.0	.4421	.6533	.4574	.6758
112.0	.4074	.6581	.4214	.6807
117.0	.3701	.6658	.3828	.6886

Figure 1.8-15 $MAPFAC_F$



$MAPLHGR = MAPFAC_P \cdot MAPLHGR_{STD}$

$MAPLHGR_{STD}$ = Standard MAPLHGR Limits

For 25% > P: (SLO TLO) No thermal limit monitoring required. No limits specified

For 25% ≤ P < 30% (SLO TLO)

$MAPFAC_P = .585 + .005224\ (P-30\%)$ For ≤ 50% Core Flow

$MAPFAC_P = .433 + .005224\ (P-30\%)$ For > 50% Core Flow

For 30% ≤ P: (SLO TLO)

$MAPFAC_P = 1.0 + .005224\ (P-100\%)$

For 52% ≤ P: (SLO ONLY)

$MAPFAC_P = 0.75$

Figure 1.8-16 $MAPFAC_P$

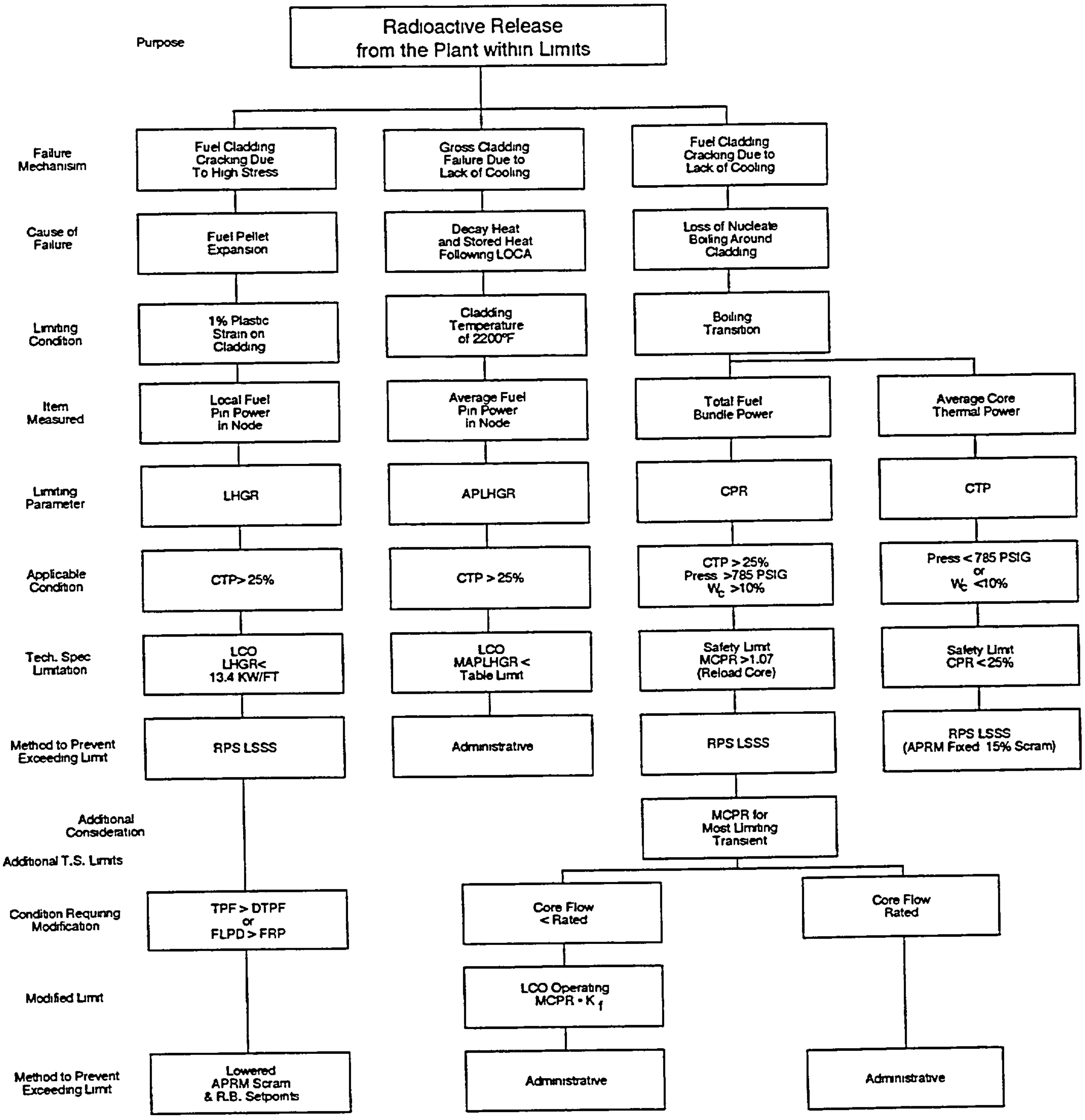
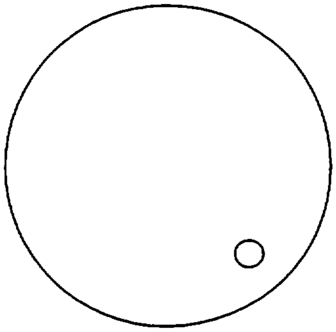
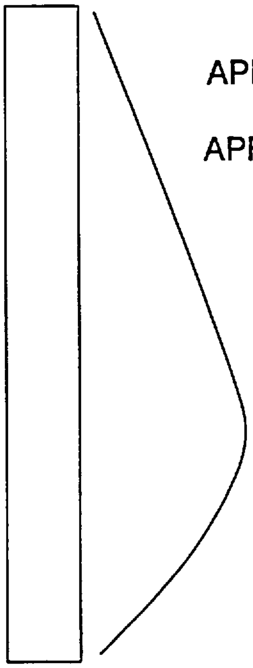


Figure 1.8-17 Thermal Limits & Modifications (BWR/3/4/5, without "ARTS")

RPF - Radial Peaking Factor

$$\text{RPF} = \frac{\text{Bundle Power}}{\text{Average Bundle Power}}$$



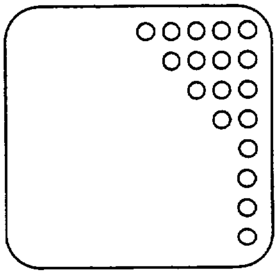


APF - Axial Peaking Factor

$$\text{APF} = \frac{\text{Nodal Power}}{\text{Average Nodal Power}}$$

LPF - Local Peaking Factor

$$\text{LPF} = \frac{\text{Pin Power}}{\text{Average Pin Power}}$$



TPF - Total Peaking Factor

TPF = RPF x APF x LPF

Maximum TPF = Highest Powered Pin in Core

Figure 1.8-18 Peaking Factors

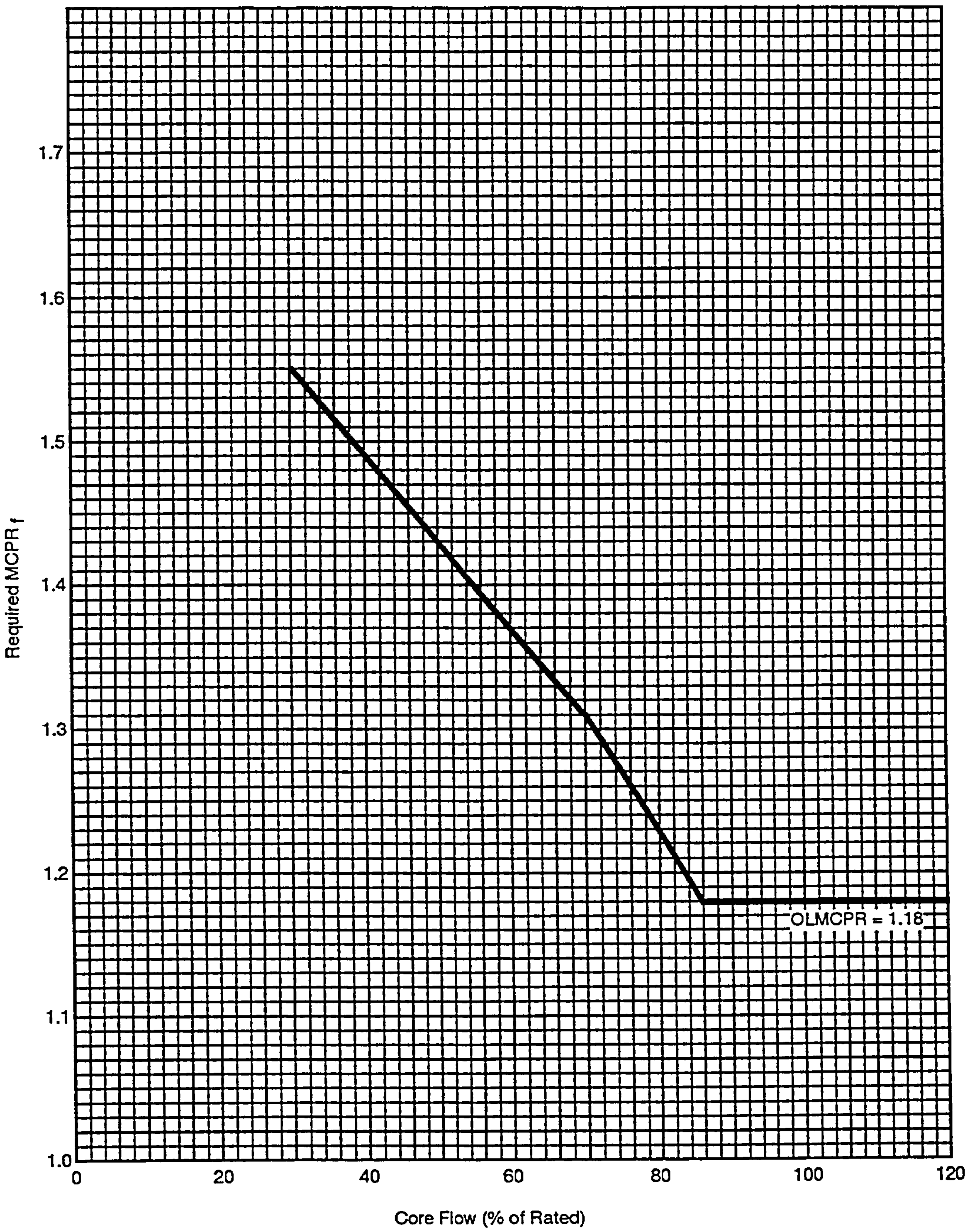


Figure 1.8-19 MCPR$_f$

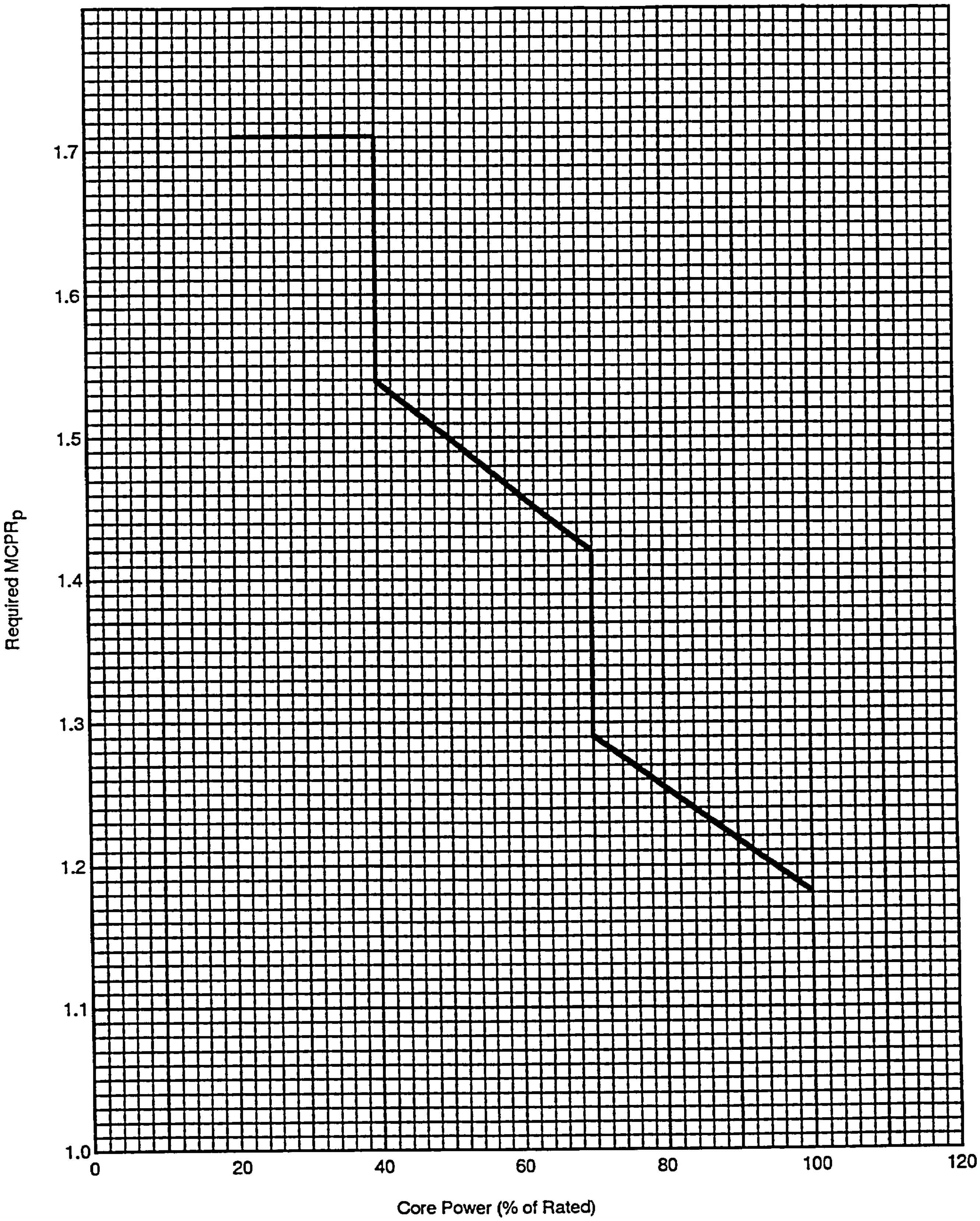


Figure 1.8-20 $MCPR_p$

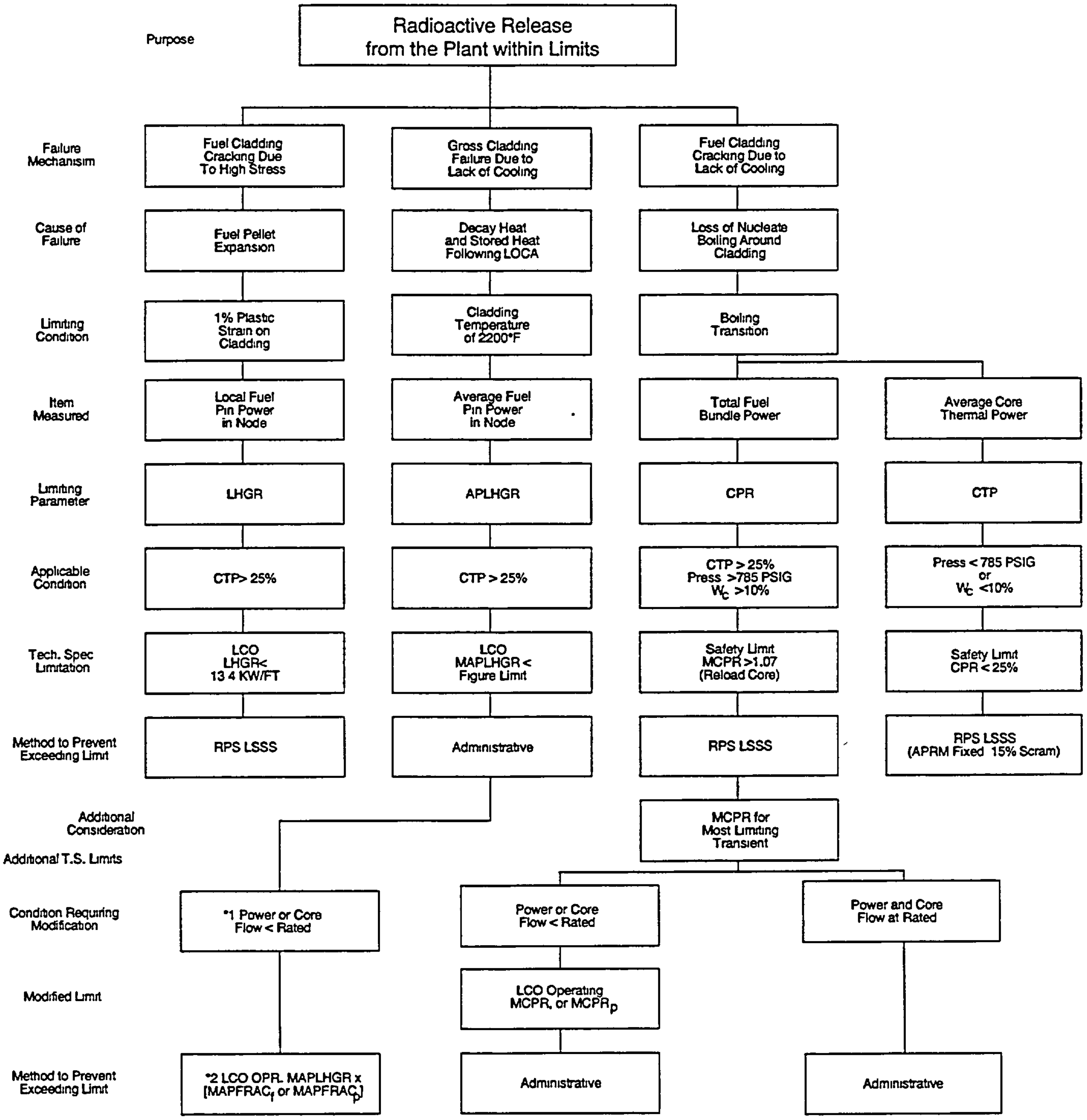


Figure 1.8-21 Thermal Limits & Modifications (BWR/6)

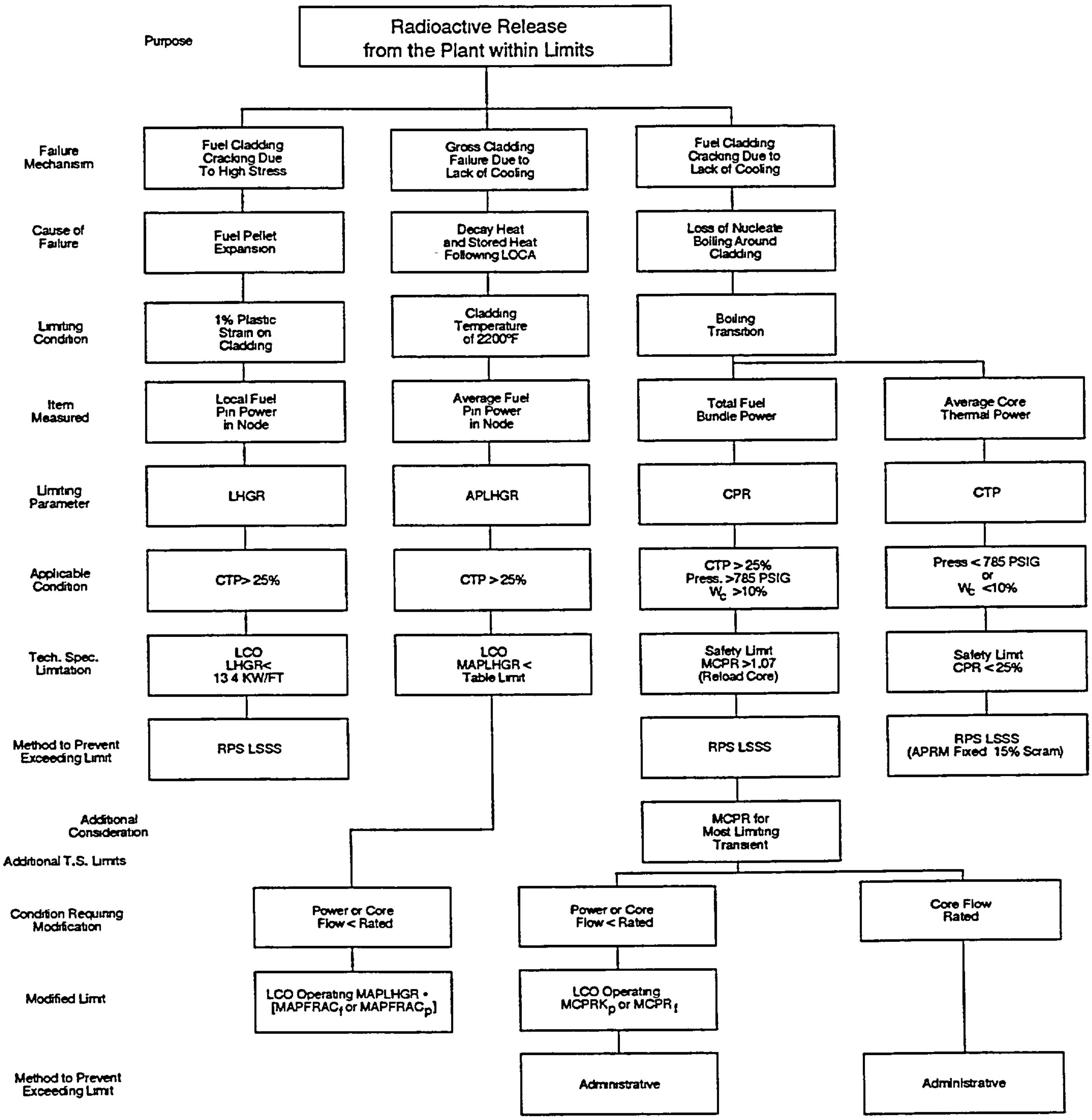


Figure 1.8-22 Thermal Limits & Modifications

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1.9 BWR CHEMISTRY

Process chemistry in a nuclear power plant involves water chemistry for control of the water quality similar to that used in conventional plants and radiochemistry that is unique to a nuclear plant. The presence of the radioactive materials does not affect the water chemistry but it does make the processing of wastes and controlling of discharges more complex. The radionuclides in the process streams simplify many analytical measurements and can be used as tracers to diagnose and evaluate plant performance.

The first part of this section deals with water quality within the plant and the means that are used to maintain the water purity. Water purification is an important part of plant operations and is done by conventional ion exchange and filtration processes to maintain the required purity of the coolant and to control radioactivity in the plant. Conventional chemical parameters, such as conductivity, chlorides, and silica are discussed as well as the water quality limits for various process streams. Basic radiochemistry considerations are also covered in this section.

1.9.1 The BWR Cycle

A boiling water reactor (BWR) is a closed cycle, water processing plant that uses very high purity water in all parts of the system. Water quality in one part affects the quality of the water in other parts.

The BWR closed cycle water system is shown in Figure 1.9-1. Condensed steam is collected in the hotwell which is located below the cooling tubes of the condenser. Possible tube leaks in the condenser can result in circulating water entering the hotwell and introducing impurities. When the circulating water is seawater, the consequences of very small tube leaks can be quite significant. Therefore, special precautions are required in seawater cooled plants. The hotwell water is deaerated to reduce the content of oxygen dissolved in the water. It then passes through full flow condensate demineralizers in order to obtain the purest possible feedwater. The condensate demineralizers remove insoluble corrosion products by filtration, and dissolved ionic materials (i.e., metals and salts) by ion exchange. They afford limited protection from condenser tube leaks until corrective action is initiated. This corrective action can range from injecting sawdust into the circulating water, to isolating the appropriate water box, to scramming the reactor to shut down the plant.

The feedwater passes through the feedwater heaters and enters the reactor at about 420°F. Since there is no blowdown of the reactor water as there is in a fossil boiler (except for a very small amount of primary system leakage) and there is very low carryover (<0.1% with the steam), the reactor water concentrates all of the soluble and insoluble materials that enter via the feedwater. The Reactor Water Cleanup System (Section 2.8) removes soluble corrosion and fission products from the reactor water, but it has only a limited ability to remove insoluble corrosion products. Thus, extremely pure feedwater is a necessity.

No additives are used in a BWR to control pH, oxygen, or reactivity. Pure, low conductivity water with very low chloride levels is the most important chemical criterion for the reactor water.

1.9.1.1 Control of Water Quality

The primary reasons for controlling water quality are to control the corrosion of materials used in the plant and to minimize the levels of radioactive nuclides in the coolant. In a nuclear plant, two types of corrosion are of concern: the general loss of metal and stress corrosion cracking.

The primary system is built of corrosion resistant materials, mainly stainless steels. These stainless steels have very low corrosion rates, but they are susceptible to stress corrosion cracking (SCC). Therefore, special limits on chloride must be strictly adhered to in order to avoid conditions that might lead to stress cracking.

Other parts of the plant use less corrosion resistant materials such as low alloy steels in the steam lines, turbine, and condenser. The corrosion products released from the large surface areas of these materials must be removed from the feedwater before it enters the reactor vessel in order to minimize the amount of material carried to the reactor which could deposit on heat transfer surfaces of the fuel, and foul critical flow areas. Reducing the amount of material that enters the reactor vessel also helps to minimize the amount of activated corrosion product (crud) buildup in primary system locations out of the core area. Thus the principal emphasis in a BWR system is placed on the removal of impurities from the water cycle rather than on treatment of impurities already in the operating system.

1.9.1.2 Control Parameters

The BWR cycle water environment is based on neutral high purity water chemistry. Some pressurized water reactors (PWRs) and fossil fueled plants employ a basic pH environment using various chemical additives to control conditions. The use of high purity neutral pH water environment in BWRs greatly simplifies and facilitates the monitoring of system chemistry conditions. The primary monitoring parameter is electrical conductivity (specific conductance). Conductivity instrumentation is relatively simple and reliable. Conductivity is defined as the reciprocal of the resistivity. The units of conductivity are mho/cm. Since the conductivity is very low in a BWR, conductivity units of 10^{-6} of a mho/cm or µmho/cm are used. Conductivity of theoretically pure water at 25°C is 0.054 µmho/cm. Several process streams in the BWR cycle approach the theoretical minimum value. Conductivity measures dissolved ionic species in the water and, since the water in a BWR is of very high purity, the conductivity value allows estimation of the maximum values of other chemical parameters. Thus conductivity provides a simple cross-check on chemical test results and instrumentation.

A second important monitoring parameter is pH, mathematically defined as the reciprocal of the logarithm of the hydrogen ion concentration or

$$pH = -\log[H+].$$

where [H+] is the concentration of H+ ions in equivalents per liter. Neutral water has a pH of 7 derived from the physical fact that, at a given temperature, a finite number of water molecules will dissociate into H+ and OH- ions. For water at 25°C, the dissociation constant is $K = 10^{-14}$.

$$K = [H+][OH-]$$

Figure 1.9-2 shows the relationship between conductivity and pH at 25°C.

The normal operating limit for the reactor water is 1 µmho/cm, at which point the pH must be within the range of 5.6 to 8.6. If the conductivity is above the normal operating range of 0.2 to 0.3 µmho/cm, something is wrong: either the input is high or the cleanup system is not removing soluble species as it should.

If conductivity is known, pH can be inferred. As shown in Figure 1.9-2, as the conductivity becomes lower, the possible pH range becomes small and approaches a value of 7 at 0.054 µmho/cm. Since the feedwater has a conductivity approaching the theoretical minimum, there is no need to measure its pH. In the reactor water, with a conductivity of >1 µmho/cm, the possible pH range gets broader and should be checked.
There is no possible combination of acids, bases or salts that can give a point below the curve, thus the indicated impossible region in Figure 1.9-2.

Chloride concentration is a third important parameter in BWR chemistry because of the relationship of chloride to stress corrosion cracking in stainless steels. There is no continuous monitor for chloride, so these analyses must be made in the lab on grab samples. However, conductivity provides some assistance. Figure 1.9-3 is a plot of conductivity versus chloride concentration. For any given low conductivity, the maximum sodium chloride concentration is indicated because the curve is based on sodium and chloride ions being the only conductive species present.

Conductivity, pH, and chloride are the three most important chemical parameters in the chemistry of the BWR. Some other parameters which are routinely monitored are:

* dissolved oxygen in the reactor water

* insoluble iron, sometimes referred to as insolubles or turbidity (generally measured by filtration and laboratory analysis of the filters)

* oxygen monitored in the feedwater by means of a galvanic cell

* silica monitored because it is an early warning of anion demineralizer breakthrough
* boron checked to ensure that the Standby Liquid Control System (Section 7.4) is not leaking

1.9.1.3 System Chemistry Levels

This section covers the limits placed on the various parameters which are monitored and controlled in a BWR system and the interplay of various system components.

A general set of chemical levels is shown in Table 1.9-1. Refer also to Figure 1.9-1 during the following discussion.

The concentrations are readily achievable with properly operated plant equipment. Through filtration and ion exchange in the condensate demineralizer system, a marked improvement is seen through the feedwater system.

The reactor vessel is the single most important part of the primary system, for it is here that the fuel and other critical nuclear components are located. The feedwater enters the reactor vessel through a feedwater sparger and is heated to form steam as it is pumped through the core. The amount of reactor water inventory in the primary system seems large but actually represents only 3 to 4 minutes' worth of steam production. Thus, the reactor water is boiled 15 to 20 times an hour. Impurities present in the feedwater remain behind in the reactor water when it turns to steam.

The only way to remove the impurities is carryover, which is negligible, or to employ the cleanup system. If the cleanup system flow is 1% of the feedwater flow and cleanup system is 100% effective, then for a truly soluble material or one that will stay suspended or dispersed, the reactor water concentration will be 100 times the concentration of this same material in the feedwater. This holds true for solubles such as sodium chloride or sodium hydroxide.

Corrosion products (iron and copper), however, do not follow this same rule because they do not stay dissolved or suspended in the reactor water. Instead, the corrosion products in the feedwater quickly disappear from the reactor water. They are insoluble and about 90% deposit on surfaces such as pipe walls, fuel, other core components, and crud traps. Only about 10% of the input remains suspended long enough to be removed by the cleanup system. The importance of this is that the levels of ionic materials in the reactor water can be measured by conductivity (Figure 1.9-3) and are related to the input rates. The crud levels in the reactor water are no indication of how many corrosion products are being put into the system or of the problems this input will cause. The input of metals must be measured in the feedwater.

Oxygen limits in the reactor water do not show the 100 concentration factor for several reasons. Being a gas, oxygen is continuously leaving the reactor vessel with the steam and is removed from the condenser through the steam jet air ejectors. Note the high levels of oxygen in the steam. The increase of oxygen in the reactor water over that in the feedwater is due to the radiolytic decomposition of water in the high flux of the core. The high oxygen limit of reactor water at depressurized conditions represents saturated conditions for water in contact with air (i.e., reactor vessel head removed for refueling and maintenance).

The reasoning behind the several sets of limits for reactor water comes from the fact that chlorides can induce stress corrosion cracking in stainless steels. The relationship of chloride stress failure as a function of both chloride and oxygen is shown in Figure 1.9-4. During operation, the oxygen level of the reactor water is about 0.2 to 0.3 ppm which indicates 1 ppm chloride is an acceptable limit for a short period of time. During startups or when steaming rates are less than 100,000 lb/hr, the oxygen levels in the reactor water can rise to several ppm, and the chloride limit under these conditions is 200 ppb. Additional information on SCC has dictated that the minimum short term chloride should be reduced to 0.5 ppm during operation and 100 ppm during startup.

Reactor water limits imposed by fuel warranty and technical specifications can be maintained only by imposing strict limitations on the level of impurities in the reactor feedwater. The reactor vessel acts as a big concentrating pot to concentrate feedwater impurities by a factor of 100. Therefore, to meet the normal operating limit of 200 ppb for chloride in the reactor, the feedwater must contain 2 ppb or less. Since this is well below the normal analytical level of detection, it is necessary to rely on the reactor water measurements to show that the feedwater is within specification requirements. This is the only indirect measure of feedwater quality.

To achieve the recommended parameter levels in the vessel shown in Table 1.9-1, the normal limits imposed on feedwater quality are as follows:

Conductivity	<0.10 µmho/cm
Metals	<15 ppb, no more than 2 ppb Cu
Dissolved O_2	>20 ppb <200 ppb
Chloride	Level shall be such that reactor water limits are met.

Because metallic corrosion products are insoluble in the reactor, a strong effort must be made to minimize their inputs. For an 1100 MWe plant operating at the limit of 15 ppb of corrosion products in the feedwater, the input can reach 1800 lb/yr. Once deposited in the reactor vessel, the only removal methods are: departure when fuel is replaced every 12 to 18 months or deposition on primary system surfaces such as piping and recirculation pumps where it can contribute to radiation zones.

Installed plant chemical systems provide the means of minimizing impurity inputs to the reactor vessel.

1.9.1.4 Plant Chemical Systems

The plant components responsible for producing high quality feedwater are the condensate demineralizers (sometimes called polishers). The primary functions of the condensate demineralizers are to act as a barrier for the reactor against condenser tube leakage, and to minimize the input of metallic material from the feedwater system.

There are presently two types of condensate demineralizing systems available: deep bed systems which use bead type ion exchange resins and powdered resin systems which use pulverized resins precoated onto nylon fiber or stainless steel filter septa.

Ion exchange is the basic chemical operation for removing soluble chemical species in both types of units. The resins are composed of reactive sites fixed to a chemical chain structure.

For cation resins, the reactive sites are negatively charged radicals which normally hold H+ ions. In the ion exchange process, the H+ ions are given up for a positive ion which binds itself tighter to the resin than the H+.

For anion resins, the reactive sites are positively charged radicals which will give up the OH- ion to collect other negative ions. Thus if an ionic solution is allowed to flow through a mixed resin bed, both positive and negative ions in that solution are collected by the resin.

Two points can be made about these reactions. The result of the cation exchange is free hydrogen ion (H+) and the result of the anion exchange is free hydroxyl ion (OH-) which combine to give water (H_2O). The ion exchange process is also reversible, allowing the chemical regeneration of the resins.

Another important principle of ion exchange is that resins exhibit a selective affinity of ions in solution. The relative selectivity for various cations and anions is shown in Table 1.9-2. The strongly held ions are at the top of each column; the weakly held ions are at the bottom. In general, an ion will replace any ion which is below itself in relative affinity. Since the hydrogen ion and the hydroxyl ion are near the bottom of each respective column, most cations replace the positively charged hydrogen ions and most anions replace the negatively charged hydroxyl ions. The resins, therefore, are recharged with hydroxyl and hydrogen ions. The fact that water is formed from the ions pairs is also important.

Fluoride is low on the selectivity listing and leaks through the condensate demineralizer resins of a boiling water system more rapidly than other anions. Chloride will replace hydroxyl, but it is not held as strongly as some of the other anions; thus, if strict chloride control is required, care must be used that the resin is not depleted to the point where other ions displace chloride. Fortunately, iodide such as fission product I-131 is easily removed by resins.

1.9.1.5 Deep Bed Demineralizers

Deep bed condensate demineralizer systems are used on all saltwater cooled plants in which the added ion exchange capacity is necessary for protection against condenser cooling water leakage. Resins are loaded as regenerated mixed resins into spherical or cylindrical tanks (150 to 200 ft^3 resin per tank). A typical spherical tank is shown in Figure 1.9-5.

Two factors limit the length of the operating cycle of a bed: exhaustion of ion exchange capacity and reduced flow because of pressure drop due to crud buildup in the resin bed.

In the absence of condenser leakage, pressure drop becomes the limiting factor. Resin bed performance is optimized by a combination of ultrasonic cleaning and chemical regeneration. Ultrasonic cleaning is a mechanical cleaning process that removes the suspended particles or crud which have been collected in the resin bed. Since no chemicals are used in this cleaning process, the radwaste generated in this process is relatively easy to process. Ultrasonic cleaning is used when the pressure drop across the bed is excessive but the ion exchange capacity is adequate. In a normally operating BWR, several ultrasonic cleanings are performed on resin before it is necessary to chemically regenerate the bed.

When the ion exchange capacity of the resin bed is depleted, a more involved chemical regeneration process is required. In this process, the cation resin beads are subjected to a strong acid solution and the anion beads to a strong basic solution. The H+ and OH- ions replace the impurity ions at the reactive sites and the ion exchange capacity of the resin is restored. The strong acid (H_2SO_4) or base (NaOH) solutions are, however, difficult to process in the radwaste systems.

The advantage of a deep bed system is that its relatively large ionic capacity provides the margin required to allow an orderly plant shutdown following a significant condenser tube leak. The disadvantage of the system results from the large volumes of concentrated spent regenerant solutions plus the volumes of rinse and transfer water associated with each resin regeneration cycle. These large volumes must be treated in the plant radwaste system before disposal or return to the cycle.

1.9.1.6 Powdered Resin Filter Demineralizers

Another type of condensate demineralizing system which is used in fresh water cooled plants uses powdered resin filter demineralizers. Powdered resin systems basically consist of ground up mixed bed resins which are precoated onto a filter system, much the same as for a conventional precoat filter.

The quantity of resin used in such a system is small compared with that used in the deep bed system. Because of the lower ionic capacity of powdered resin systems, they are not used for feedwater purification where saltwater condenser cooling is employed. Powdered resin beds do not afford the reactor protection needed following the start of a saltwater condenser tube leak. The powdered resin system does, however, provide suitable protection for small leaks in a fresh water plant.

One of the advantages of the powdered resin system is that liquid radwaste requirements are lower because the resin is not regenerated or cleaned but is discarded and placed in barrels after use. Another advantage is that the finely ground resin constitutes a much better filter than deep bed resin. Powdered resin also affords a great deal of flexibility in that it can be mixed with a chemically inert filter medium to improve filtering characteristics or it can be mixed with variable cation-to-anion ratios in order to selectively remove specific ions.

A typical filter element (Figure 1.9-6) consists of nylon wound over a stainless steel center element. Six 1 foot long elements are normally used to make each individual septum. Each filter unit contains about 300 of these 6 foot long septa. Other designs employ fine screen or textured wire wrapped stainless steel septa.

The resin precoat has a maximum thickness of approximately 3/8 inch with much thinner layers in some areas. In contrast to a deep bed demineralizer system, which operates at 50 gpm/ft^2, a powdered resin unit is designed to operate at about 4 gpm/ft^2.

A definite advantage of the powdered resin system is that insoluble removal efficiencies are better than corresponding deep bed efficiencies. Powdered resin systems in many cases remove more than 90% of the insoluble material. However, there are always thin resin layer areas that have very rapid ionic breakthrough. As a result, effluent conductivities are not as good as those for deep bed plants.

Representative breakthrough curves for the deep bed and powdered resin systems are shown in Figure 1.9-7. For the deep bed system, little leakage is expected until a good deal of the available capacity has been exhausted; whereas, for powdered resin, the breakthrough is approximately linear with percent resin exhaustion. This linear breakthrough characteristic occurs because there are areas of very thin resin layers which become exhausted almost immediately. This initial starting point for powdered resin systems depends upon the evenness of the individual precoats. The curves shown are only approximate and should not be construed as actual data.

Operation of a powdered resin system is somewhat simpler than a deep bed system because no resin cleaning or regeneration is required. Resin precoats are used for one operating cycle and are disposed of as solid wastes.

Powdered resin filter demineralizer systems are equipped with fine mesh strainers downstream of each polisher unit. These strainers will intercept gross quantities of powdered resins in case of a septum failure or other loss of the precoat. The strainers will do little to stop a small number of resin particles moving individually in the effluent stream. The present design for all Reactor Water Cleanup (RWCU) Systems include powdered resin demineralizers operating at reactor pressure. Performance of the powdered resin units has been satisfactory. Cleanup units are physically smaller than the powdered resin condensate polishers because they treat a smaller quantity of water per unit of time.

1.9.2 BWR Radiochemistry

The preceding discussion of BWR plant chemistry has established the reasons for and application of the selected chemical environment. The radiochemistry of the BWR results from the nuclear aspects of the cycle, is compatible with the selected chemistry, and has little or no effect on the selected environment. The radionuclides in the cycle do provide an additional analytical tool which in many cases facilitates chemical control. However, the plant radiochemistry is most important from the standpoint of understanding radionuclide sources and how these nuclides can be controlled to meet plant access needs and regulatory requirements.

1.9.2.1 Coolant Radiochemistry

There are three classifications of radionuclides associated with the coolant. These are activation products of the coolant, corrosion products from materials of construction, and fission products from the fuel materials. Each of these classifications is discussed separately.

Activation products are formed when a stable nuclide absorbs a neutron or experiences another nuclear reaction that results in the atom being formed in an unstable state. Energy in the form of alpha rays, gamma rays, x-rays, electrons, or positrons emitted from the unstable nuclide is termed radioactivity.

1.9.2.2 Activation Products of Water

In the high flux of the reactor core, the coolant water molecules are broken down, and a fraction of the component atoms become activated. Since the disassociation products have been discussed earlier, only the activation products will now be covered.

The important activation products of water are nuclides normally considered as gases. In the reactor, these products are distributed between the gaseous (steam) and liquid phases (e.g., nitrogen-16 (N-16) and nitrogen-13 (N-13) are two principal coolant activation products).

The most common activation products of water are listed as follows:

Nuclide	Half-Life	Formation Mechanism
N-16	7.1 seconds	$O^{16}(n, p)N^{16}$
O-19	29 seconds	$O^{18}(n, \gamma)O^{19}$
N-13	10 minutes	$O^{16}(p, a)N^{13}$
F-18	110 minutes	$O^{18}(p, n)F^{18}$
H-3 (tritium)	12.33 years	$H^{2}(n,\gamma)H^{3}$ and tertiary fission

This table also indicates the formation mechanism of the nuclides in the standard notation which allows us to trace these displacements on a chart of the nuclides.

The more volatile constituents are carried with the steam phase and are removed with the condenser offgas. Since the half-lives of the water activation products are relatively short, most of the activity formed decays away within the reactor or offgas before being released from the plant. The production rate of the activation products of water is nearly proportional to the power level of the plant. The quantity of energy associated with a particular mode of disintegration is expressed in MeV or KeV (millions or thousands of electron volts). For example, the 7.1-second N-16 nuclide has a 7 MeV gamma radiation. This high-energy gamma is responsible for most of the radiation field from the turbine which is sometimes referred to as "turbine shine." It is also the primary source of radiation associated with steam piping.

1.9.2.3 Activated Corrosion Products

These radionuclides are typically soluble and insoluble materials which have been corroded from the turbine condenser, feedwater piping, feedwater heaters, and reactor system components. These materials are transported into the flux zone and become activated while circulating through the core or while plated out on surfaces in the core area.

Some of these materials remain on the heat transfer surfaces, some are removed by the Reactor Water Cleanup System, and a small fraction become part of the out of core deposits. Reactor water leakages and controlled rejections can introduce these products to the radwaste system where they contribute the bulk of long lived radioactive waste processed in the nuclear plant. The fraction of these products which deposit out of core become a major portion of the radiation field during primary system maintenance.

Several of the most commonly activated corrosion products and the production mechanisms are listed in Table 1.9-3. Along with fission products, the longer lived corrosion products are the common nuclides which the chemist must keep in inventory, since these nuclides will make up a large contribution to the waste that will be eventually discharged from the plant.

1.9.2.4 Fission Products

The third important group that can contribute to the total plant radioactivity consists of the fission products. The distribution of specific fission products released from fuel imperfections is studied to evaluate the condition of the fuel. An extremely important concept to remember is that the inventory of fission products in the fuel is large but, even with severe fuel defects, only a very minute fraction of this inventory is released to the coolant.

The relationship between the fission yield and the mass number for the fission products of uranium-235 (U-235) is shown in Figure 1.9-8. The two peaks represent the most probable products formed upon fissioning or splitting of the U-235 nucleus. The products that have the higher fission yields are normally the nuclides that are significant to the power plant radiochemist. Note that several radioactive nuclides of each element exist and the sum of all independent yields totals 200% since there are two nuclides produced for each fission. The maximum yield in each of the peaks is approximately 6% and the halogens and noble gases are prominent in both peaks.

The fission products can be subdivided into groups by chemical and physical behavior. These are the iodines, the noble fission gases, the particulates from the fission gases, and the soluble and insoluble fission products. These materials are released through defects in the fuel rod cladding. Additionally, small amounts of fission products are released from the tramp or impurity uranium in the cladding.

Iodine

There are five iodine isotopes with half-lives greater than 85 seconds that are detectable. These are listed as follows:

Nuclide	Half-Life	Chain Fission Yield (%)
I-134	52.3 minutes	7.176
I-132	2.28 hours	4.127
I-135	6.7 hours	6.386
I-133	20.8 hours	6.762
I-131	8.06 days	2.774

The production rate in µCi/s and the relative distribution of the iodine nuclides help the trained observer to evaluate the integrity of the fuel cladding. Because of the volatility of the iodine, a fraction of that produced will be transferred out of the reactor vessel with the steam. Radioiodines that escape the cladding and enter the coolant are removed by three means:

* decay in the reactor water

* removal in the Reactor Water Cleanup System

* conveyance with the steam to the hotwell and then removal in the condensate demineralizers

Noble Gases

There are 22 noble gases (xenons and kryptons) resulting from the fission of U-235 which must be summed for release reporting.

These gases are prominent in the peak regions of Figure 1.9-8 as relatively high yield fission products. The half-lives of the 22 noble gas nuclides range from less than 1 second to more than 10 years. Because of certain characteristics of the 22 isotopes (i.e., half-life, fission yield, and energy of gamma rays emitted), only six major noble gas nuclides are shown in Table 1.9-4. These gases are reported as the sum of the six nuclides.

Iodines are measured in the reactor water but the life cycle of the fission gases is different because of

* release from the fuel via a cladding defect to the primary coolant

* transport with the primary steam through the turbine to the condenser

* removal from the main condenser by the steam jet air ejectors along with gases from radiolytic decomposition of water and air inleakage

* processing before release to the outside environment to reduce activity content

Such processing may include extended holdup via compression, adsorption on charcoal, or cryogenic processing. Since atom quantities are being considered, these fission gases contribute essentially nothing to the volume of the radiolytic gas and air inleakage. Fission gases also have a tendency to spike following large pressure increases or decreases. The fission gases are used as the basis for the evaluation of fuel integrity and also must be routinely quantitatively measured in order to ensure that site release limits are being met. An important concept is that offgas release rates are normally determined by the sum of six nuclides as measured at the air ejector, whereas stack releases refer to all 22 of the noble gases. Relationships have been developed which combine the sum of the six at the air ejector and the holdup time to arrive at the sum of 22 noble gases. The amount of these gases at the air ejector that are released to the stack depends on the amount of decay before discharge. Where recombiner charcoal systems are employed, of the six nuclides, only xenon-133 (Xe-133) gives a significant contribution.

There is one long lived fission gas that will continue to build up in the environment for many years and that is krypton-85 (Kr-85) which has a 10.76 year half-life. However, since most of the Kr-85, like the other fission gases, remains in the fuel, it will not be released until the fuel is reprocessed.

Sources

The release rate of each individual noble gas isotope can be expressed by the following empirical relationship:

$A/y\lambda = K\lambda^{-b}$ for the nuclides discussed (six noble gases at the air ejector)

A = release rate in µCi/s of individual nuclide

Y = fission yield

λ = decay constant

K = dimensional constant related to the level of release

b = unitless constant establishing the relative amount of each nuclide

Of these factors, A is routinely determined by the plant chemistry staff. The factors Y and 1 are established radiochemistry constants for each nuclide. These three parameters can be used to plot the log $[A/(Y\ \lambda)]$ versus log λ to obtain an approximate straight line relationship for the several nuclides. The slope of this plot can be correlated with the type of the release. The three classic modes of defining fission gas release are known as recoil, equilibrium, and diffusion. Each of these modes is characterized in the sections that follow.

Recoil

The recoil pattern of release is typical of tramp uranium which has plated out on core internals or fuel surfaces. This type of release is characterized by the release of the fission products to the coolant upon formation (i.e., there is no delay between the fission event and release of the fission product). Thus, the spectrum of nuclides observed is that determined by the fission product yield distribution.

The plot of log $[A/(Y\ \lambda)]$ versus log 1 for the noble gases results in a straight line of zero slope. This is the characteristic of a recoil release plot. Figure 1.9-9 plots representative data. In recoil releases, iodine nuclide data are also used in the evaluation, and the release rate is plotted on the same scale as the noble gases ($A/Y\ \lambda$ versus λ 1). Since there is no delay or holdup of the fission products to skew the release toward longer half-life isotopes, the iodine plot should coincide in magnitude and shape with the noble gases.

Release rates associated with recoil release can vary from <100 µCi/s because of core impurities during startup and early operation to several hundred thousand µCi/s with loss of UO_2 to the coolant and core surfaces resulting from severe cladding failures. Since recoil release results only from UO_2 fuel material in direct contact with the coolant, the release is linearly proportional to core power.

Equilibrium

The fission product release from very small, or pinhole, perforations in the fuel cladding is termed equilibrium release. In this type of release, there is no direct contact between the coolant and the UO_2 fuel material. The fission product gases are held up or delayed within the fuel material and cladding body until the gases diffuse to the pinhole defect point. Because of this delay after the fission event, the release products show a dominance of longer lived nuclides over the shorter lived ones which undergo some decay within the fuel. The plot of log $[A/(Y\lambda)]$ versus log l reveals major differences from the recoil pattern (Figure 1.9-10).

- * The slope of the line is approximately 1 showing a swing toward the longer lived nuclides.
- * The iodine nuclide plot is considerably below that of the gases (10 to 500 times), but with a similar slope.

This lowering of the iodine line as compared to the recoil plot is primarily the result of differences in the chemical reactivity characteristics of the iodines as compared to the gases. Some of the iodines will leave a pinhole leak as a gas or vapor; a major portion, however, will remain in the fuel material.

Some fuel defects yield an equilibrium release that increase exponentially with increased power; others yield releases that are nearly linear with power. The total equilibrium offgas release from the core can be quite high (100,000 to 200,000 µCi/s); the individual release from each defect, however, may be quite small.

Diffusion

The pattern between recoil and equilibrium is termed the diffusion type and is typical of large cladding defects in which the UO_2 column in the rods is exposed to the coolant.

The slope of the $A/Y\ \lambda$ versus λ for a diffusion mixture is 0.5 (Figure 1.9-11), and the iodines usually match the gases.

The release rate per defect of a diffusion type is greater than that with an equilibrium distribution, and the release rate is always exponential with power.

During startup and early stages of core life, a well defined recoil pattern is usually experienced. After initial plant operation, a sharp increase in offgas release is usually found to be a clearly defined equilibrium mixture. This is to be expected, since recoil is steady or decreasing with time and any minor cladding defect releases an aged mixture from rod inventory. At elevated release rates, a mixture of equilibrium and diffusion patterns is generally experienced. The separation and evaluation of the respective contributions becomes a very complex task.

Fission Gas Daughters

The particulate fission products resulting from the decay of the fission gases are commonly referred to as particulate daughter products of the fission gases. Some of the more important particulates and their respective precursor gases are given in Table 1.9-5.

Since the particulates are formed with the gases in transit, these nuclides are distributed throughout the plant system steam piping, hotwell, offgas pipe, etc. A filter located in the Offgas System (after the holdup pipe) is designed to remove these particulates before their release to the stack or inlet to the charcoal system.

The efficiency of these filters is routinely measured with the aid of particulate nuclides.

The nuclide of primary concern in this group is the strontium-90 (Sr-90) isotope with its long half-life and daughter product, yttrium-90 (Y-90), which emits a very high energy ß particle.

Soluble and Insoluble Fission Products

This group of fission product nuclides remains in the primary system (i.e., they are not carried in the steam vapor phase). A representative number of the more important fission product nuclides in this classification is given in Table 1.9-6. Where two nuclides are shown together, a parent/daughter relationship is indicated. At equilibrium, both nuclides are always present.

The soluble species are removed in the cleanup filter demineralizers by ion exchange, and a small fraction of the insoluble species is removed by the filtration. However, a fraction of the insoluble species is deposited on primary system surfaces which contributes to out of core dose rates. This classification of fission products becomes particularly important in a plant with a significant recoil fission product source.

1.9.3 General Electric Fuel Warranty Operating Limits (FWOL)

The FWOL is an integral part of the General Electric Company's (GE's) fuel supply contract with a utility. The document defines and specifies operating limits and requirements such as heat flux,water quality, and fuel handling to ensure the fuel is able to meet its warranty exposure.

Historically, there were initially few fuel warranties. There were some limits on local power and suggested reasonable water chemistry limits. One of the early reactors, Humboldt Bay 3, started developing large amounts of crud on the fuel. This prompted GE to start specifying water quality limits in order to keep crud levels down.

Reactor water and feedwater characteristics and impurity limits are imposed so that a suitable chemical environment is provided. These characteristics and limits, as well as sampling frequency, are compiled in Table 1.9-7. The limits are presented for all levels of reactor operation as nominal limits. Also provided are maximum limits that are acceptable for brief periods of time. When an operating value is outside the nominal limits, steps must be taken to correct it as soon as practicable, within the permissible cumulative time interval allowed above normal limits.

Quality limits specified in Table 1.9-7 cover both reactor water and feedwater. High concentrations of insoluble materials in the reactor water are essentially due to feedwater impurities. Overall emphasis, therefore, is concentrated upon feedwater purity. Water conductivity, measured in both water systems, is the controlling property.

1.9.4 Summary

BWR reactor water is maintained in a high state of purity, without any chemical additives to:

* Limit corrosion of reactor materials

* Limit fouling of heat transfer surfaces

* Minimize activation of corrosion products

* Minimize coolant radionuclide levels

Water purity is maintained by limiting corrosion product input to the reactor and continuous cleanup of the reactor water. Conductivity and chloride ion content are monitored to detect water chemistry that is off standard.

TABLE 1.9-1 BWR WATER CHEMISTRY (NORMAL OPERATIONS)

	Iron	Copper	Chloride	Oxygen	(μmho/cm at 25^0 C)F	pH at 25^0 C
Condensate	15 to 30	3 to 5	<20	<20	0.1	7
Condensate treatment effluent	5 to 15	<1	0.2	20 to 50	<0.1	7
Feedwater	5 to 15	<1	0.2	20 to 50	<0.1	7
Reactor water - Normal Operation	10 to 50	<20	<20(200)*	100 to 300	0.2 to 0.5 (1)*	7(5.6-8.6)*
Reactor water - Shutdown			<20(100)*		<1 (1)*	7
Reactor water - Hot standby			<20(100)*		<1(1)*	7
Reactor water - Depress-urized			<20(500)*	8,000	<2(10)*	6 to 6.5 (5.3-8.6)*
Steam	0	0	0	10,000 to 30,000	0.1	
Control rod drive cooling water**	50 to 500		<20	<8,000	1	6

Concentrations, ppb

* Typical Technical Specifications Limits

** This water may be close to air-saturated demineralized water in which the conductivity and pH are primarily due to absorbed carbon dioxide gas from the air.

Table 1.9-2 RESIN SELECTIVITY

Cation Resin	Anion Resin
Fe+++	I-
Ba++	NO3-
Sr++	Br-
Ca++	HSO4-
Cu++	Cl-
Zn++	HCO3-
Ni++	IO3-
Co++	SiO2-
Fe++	OH-
Mg++	F-
Ag+	
Tl+	
Cu+	
Cs+	
Rb+	
NH4+	
K+	
Na+	
H+	
Li+	

TABLE 1.9-3 ACTIVATED CORROSION PRODUCTS

Nuclide	Half - Life	Formation Mechanism
Cr-51	27.8 days	$Cr^{50}(n,\gamma)Cr^{51}$
Mn-54	312 days	$Fe^{54}(n, p)Mn^{54}$
Mn-56	2.58 hours	$Fe^{56}(n, p)Mn^{56}$ and $Mn^{55}(n, \gamma)Mn^{56}$
Fe-59	45 days	Fe58(n, γ)Fe59 and Co59(n,p)Fe^{59}
Co-58	71 days	$Ni^{58}(n, p)Co^{58}$
Co-60	5.24 years	$Co^{59}(n,\gamma)Co^{60}$ and $Ni^{60}(n, p)Co^{60}$
Cu-64	12. hours	$Cu^{63}(n, \gamma)Cu^{64}$
Zn-65	243 days	$Zn^{64}(n, \gamma)Zn^{65}$
W-187	24.0 hours	$W^{186}(n, \gamma)W^{187}$

TABLE 1.9-4 FISSION GASES

Nuclide	Half-Life	Fission Yield (%)
Sum of six nuclides at steam jet air ejector		
Xe-138	14.2 minutes	6.235
Kr-87	76 minutes	2.367
Kr-88	2.79 hours	3.642
Kr-85m	4.4 hours	1.332
Xe-135	9.16 hours	6.723
Xe-133	5.27 days	6.776
Other important nuclides		
Xe-135m	15.7 minutes	0.05
Kr-85	10.76 years	0.27

TABLE 1.9-5 FISSION GAS DAUGHTER PARTICLES

Nuclide	Half-Life	Gas Precursor
Rb-88	17.7 minutes	2.79 hours Kr-88
Cs-138	32.2 minutes	14.2 minutes Xe-138
Sr-89	50.8 days	3.18 minutes Kr-89
Sr-90	30 years	32.3 seconds Kr-90
Sr-91	9.67 hours	8.6 seconds Kr-91
Sr-92	2.69 hours	1.84 seconds Kr-92
Ba-139	83.2 minutes	40 seconds Xe-139
Ba-140	12.8 days	13.6 seconds Xe-140
Ce-141	32.5 days	1.7 seconds Xe-141
Ce-144	284 days	9 seconds Xe-144

TABLE 1.9-6 SOLUBLE AND INSOLUBLE FISSION PRODUCTS

Nuclide	Half-Life	Fission Yield (%)
Mo-99, Tc99m	66.6, 6.0 hours	6.136, 5.399
Zr-95, Nb-95	65.5, 35.1 days	6.503, 6.505
Ba-140, La-140	12.8 days, 40.2 hours	6.300, 6.322
Cs-137	30.2 years	6.228
Sr-89	50.8 days	4.814
Sr-90	28.9 years	5.935
Ce-141	32.5 days	5.867

Sample	Normal Operational Limit	Warranty Limits		Sampling Frequency	Sampling Method
		Maximum Operational Limit	Internal Allowed Above Normal Limits		
REACTOR WATER					
At Steaming Rates >1% Rated Steam Flow					
1. Conductivity (μmho/cm at 25°C)	≤1.0	≤10.0	14 days/ 12-mo-period	Continuously[1,6]	Conductivity probe-recirc or C.U. loop
2. Chloride(ppm)	≤0.2	≤0.5	14 days/ 12-mo period	Daily[2,3]	Grab sample-recirc or C.U. loop
3. Hydrogen ion concentration(pH)	5.6 to 8.6 at 25°C	≥4≤10	24 hours (continuous)	Every 4 hours when conductivity ≥1.0 μmho/cm	Flow through cell-recirc or C.U. loop
At Steaming Rates <1% Rated Steam Flow					
1. Conductivity (μmho/cm at 25°C)	≤2.0	≤2.0	--	Continuously[1,]	Same as at rated pressure
2. Chloride(ppm)	≤0.1	≤0.1	--	Daily[2,3]	
Nonpressurized (Reactor at or below 212°F)					
1. Conductivity (μmho/cmat 25°C)	≤10.0	≤10.0	--	Continuously[1,6]	Same as at rated pressure
2. Chloride(ppm)	≤0.5	≤0.5	--	Daily[2]	
3. Hydrogen ion concentration(pH)	5.3 to 8.6 at 25°C	5.3 to 8.6 at 25°C	--	Every 4 hours when conductivity ≥2.0 μmho/cm	
FEEDWATER					
Operation >50% Rated Power					
1. Metallic analyses - filtrate (ppb)					
a. Fe	≤15.0[4]	≤2.0[5]		Weekly	Downstream continuous feedwater
b. Cu	≤2.0[5]	≤2.0[5]	14 days/ 12-mo period	Weekly	
c. Fe,Cu,Ni,Cr	≤15.0[4]	≤50.0[4]	14 days/ 12-mo period	Monthly	Millipore sampling loop - ion exchange
2. Metallic analyses - filterable solids(ppb)					
a. Fe	≤15.0[4]			Weekly	Millipore filter-continuous. Filters changed each 1-7 days
b. Cu	≤2.0[5]	≤2.0[5]		Weekly	
c. Total filterable metallics	≤15.0[4]	≤50.0[4]	14 days/ 12-mo period	Weekly	

TABLE 1.9-7 MEASUREMENTS TO DEMONSTRATE OPERATION WITHIN FUEL WARRANTY LIMITS

Sample	Normal Operational Limit	Maximum Operational Limit	Internal Allowed Above Normal Limits	Sampling Frequency	Sampling Method
d. Chemical balance of one week composite sample				Quarterly	
3. Oxygen(ppb)	20 to 200	≤200	14 days/ 12-mo period	Continuously[6]	Oxygen analyzer probe before the outboard isolation valve (after last heater)
Operation ≤50% Rated Power					
1. Metallic impurities filtrate and solids (ppb)	≤15.0	≤100.0	14 days/ 12-mo period	Same as at >50% rated power	
Condensate Demineralizer Effluent					
1. Oxygen(ppb)	≥20.0	≤200.0	14 days/ 12-mo period	Continuously[4]	Oxygen analyzer probe after the condensate demineralizer
2. Conductivity (μmho/cm at 25#C)	≤0.1	≤0.2	4 hours/ occurrence	Continuously[1,6]	Conductivity probe

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NOTES

1. Calibrate weekly with laboratory cell placed in stream or compare cleanup input reading with recirculation reading. If comparison method is used calibrate one of the monitors against laboratory cell monthly. Report date of calibration.
2. If chemistry personnel are limited to 5 days/week coverage, weekend and holiday readings may be omitted except in cases of operation within the maximum operational limits.
3. If readings are abnormal, samples are to be drawn at 4-hour intervals until the value drops to within normal limits.
4. Warranty limit for total metallic impurities (filtrate and solids) is equal to or less than 15 ppb.
5. Warranty limit for total copper (filtrate and solids) is less than 2 ppb.
6. When continuous monitoring equipment is inoperative, measuring frequency is to be consistent with Technical Specifications requirements.

TABLE 1.9-7 MEASUREMENTS TO DEMONSTRATE OPERATION WITHIN FUEL WARRANTY LIMITS (CONTINUED)

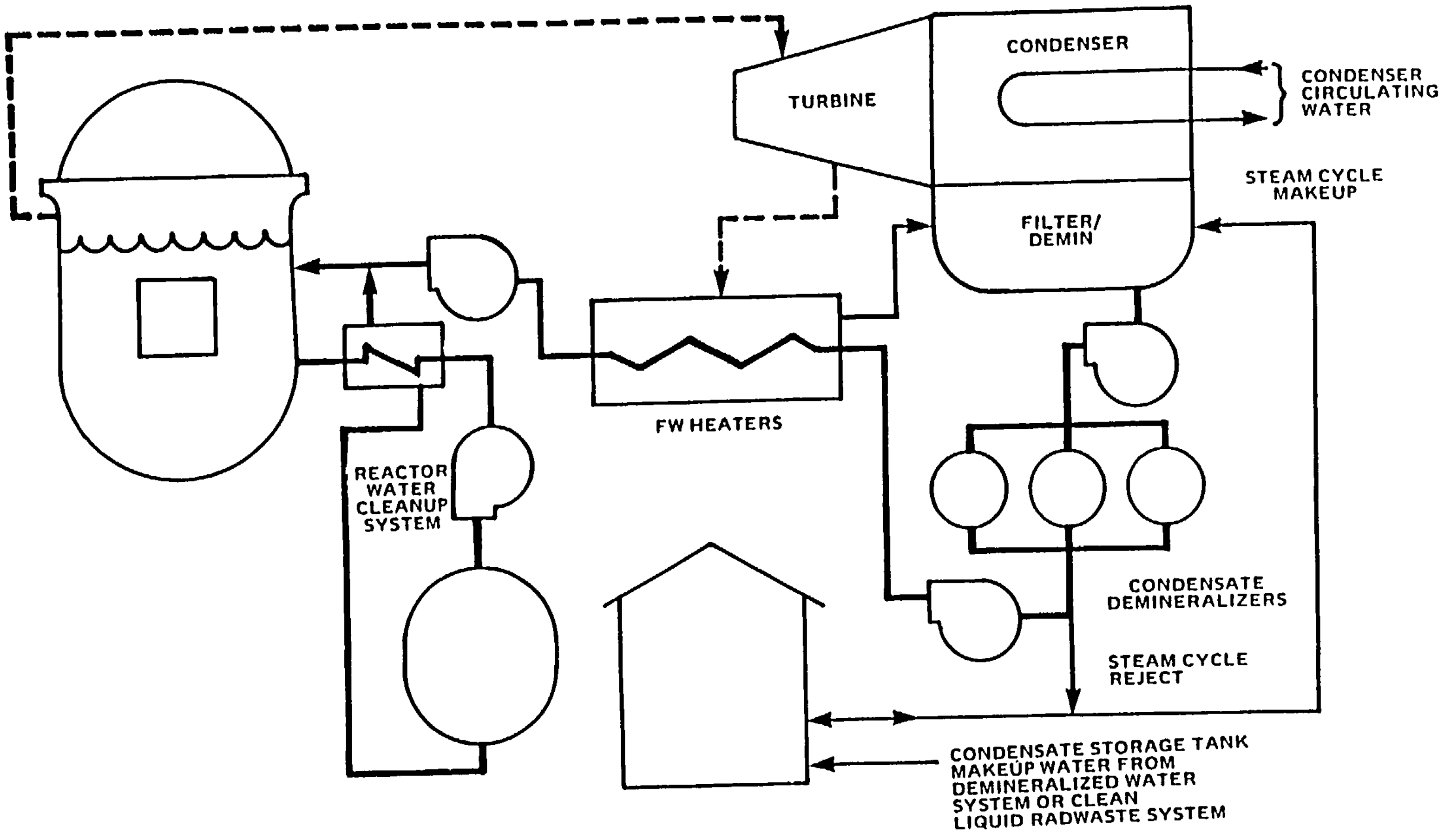

FIGURE 1.9–1 BWR WATER CYCLE

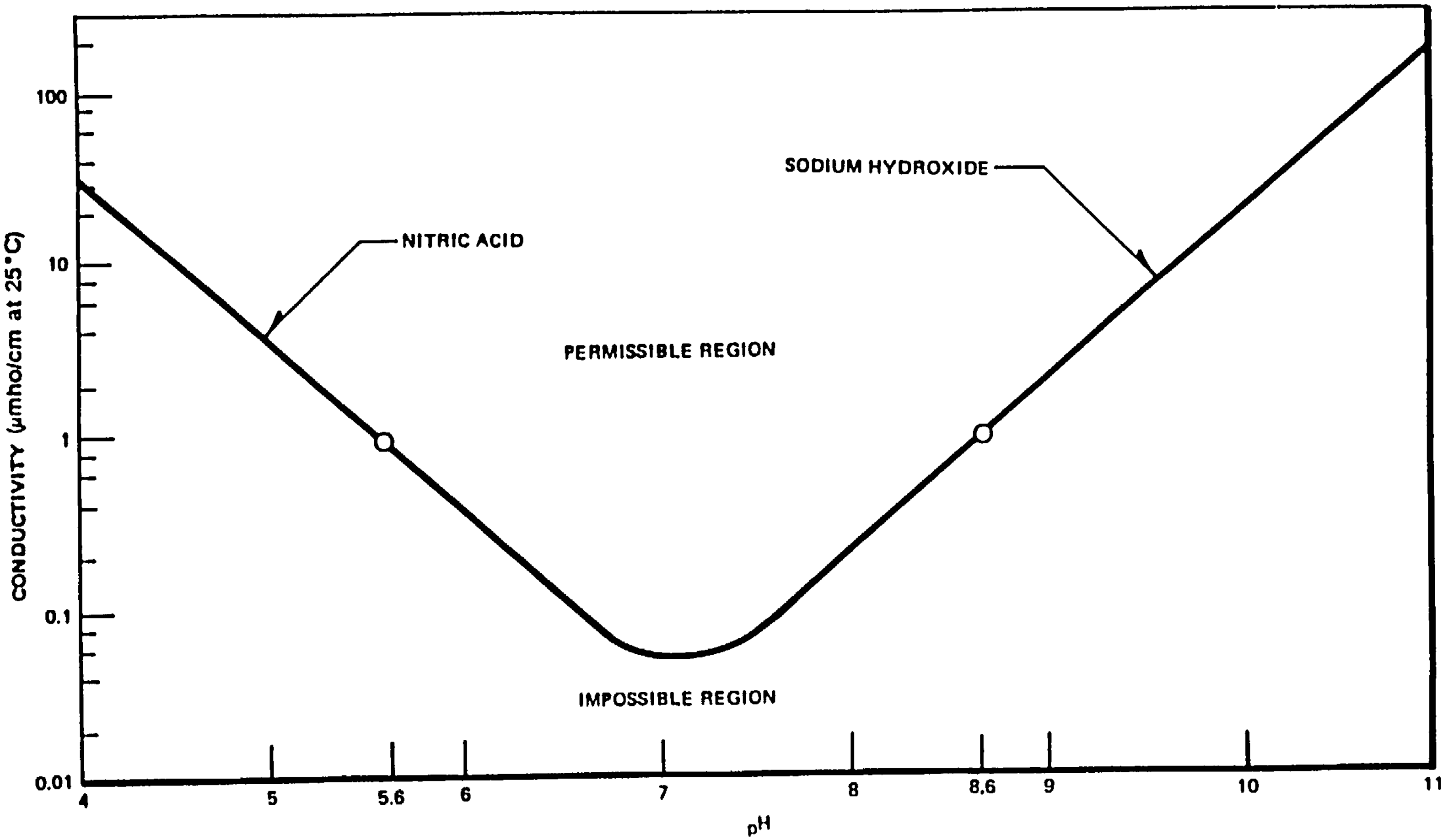

FIGURE 1.9 – 2 CONDUCTIVITY AND pH OF AQUEOUS SOLUTION AT 25° C

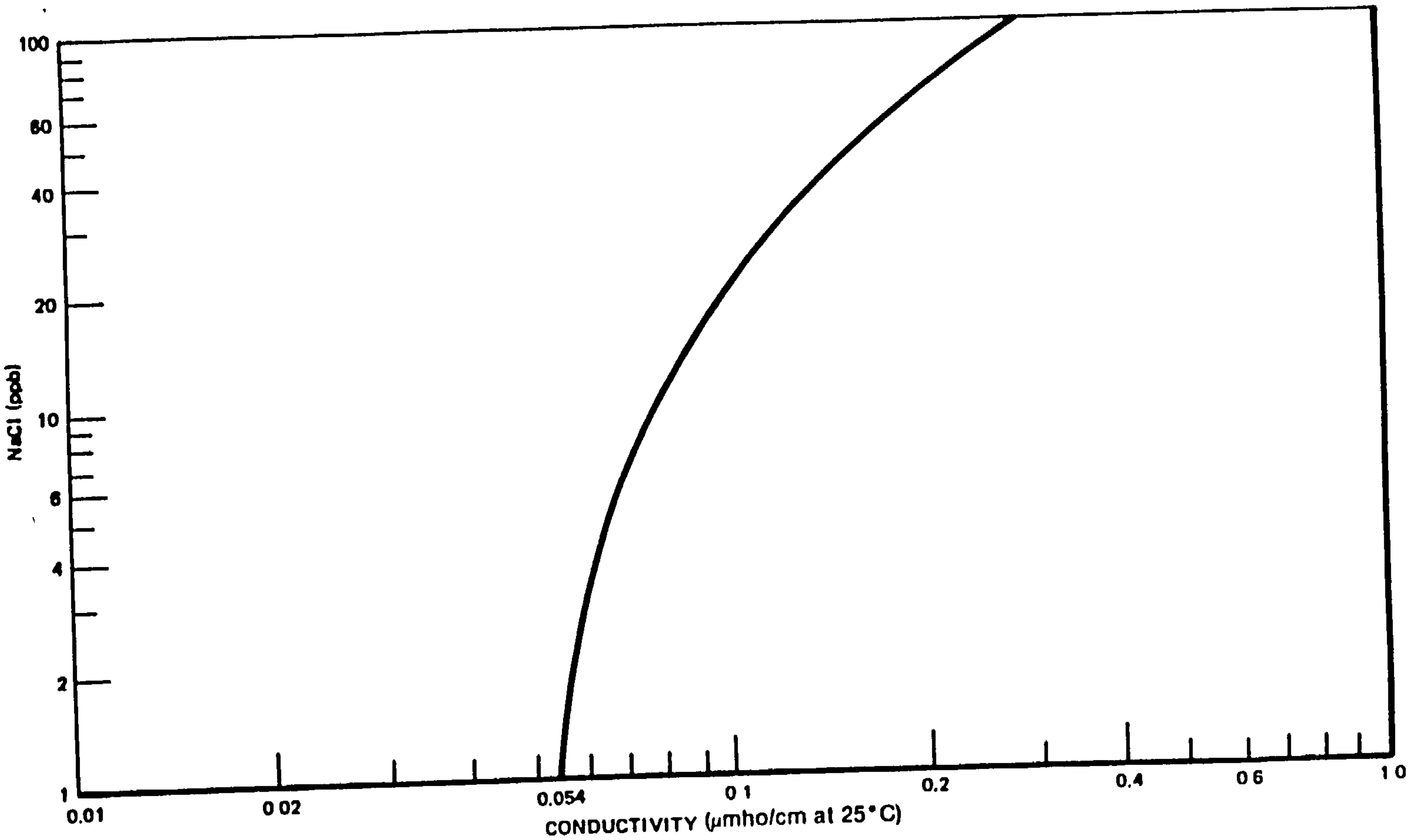

FIGURE 1.9 – 3 CONDUCTIVITY VERSUS CHLORIDE CONTENT

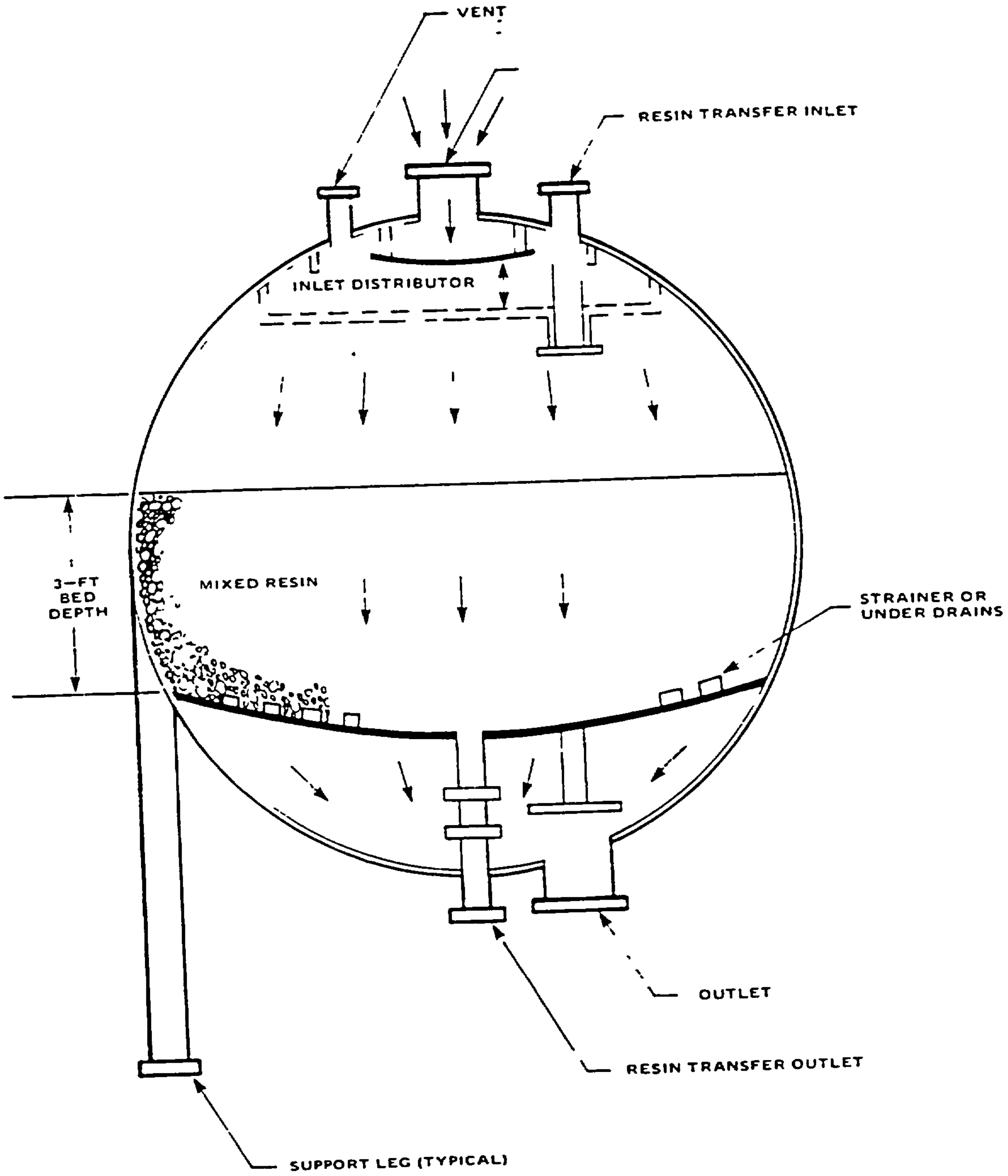

FIGURE 1.9-5 DEEP BED DEMINERALIZER

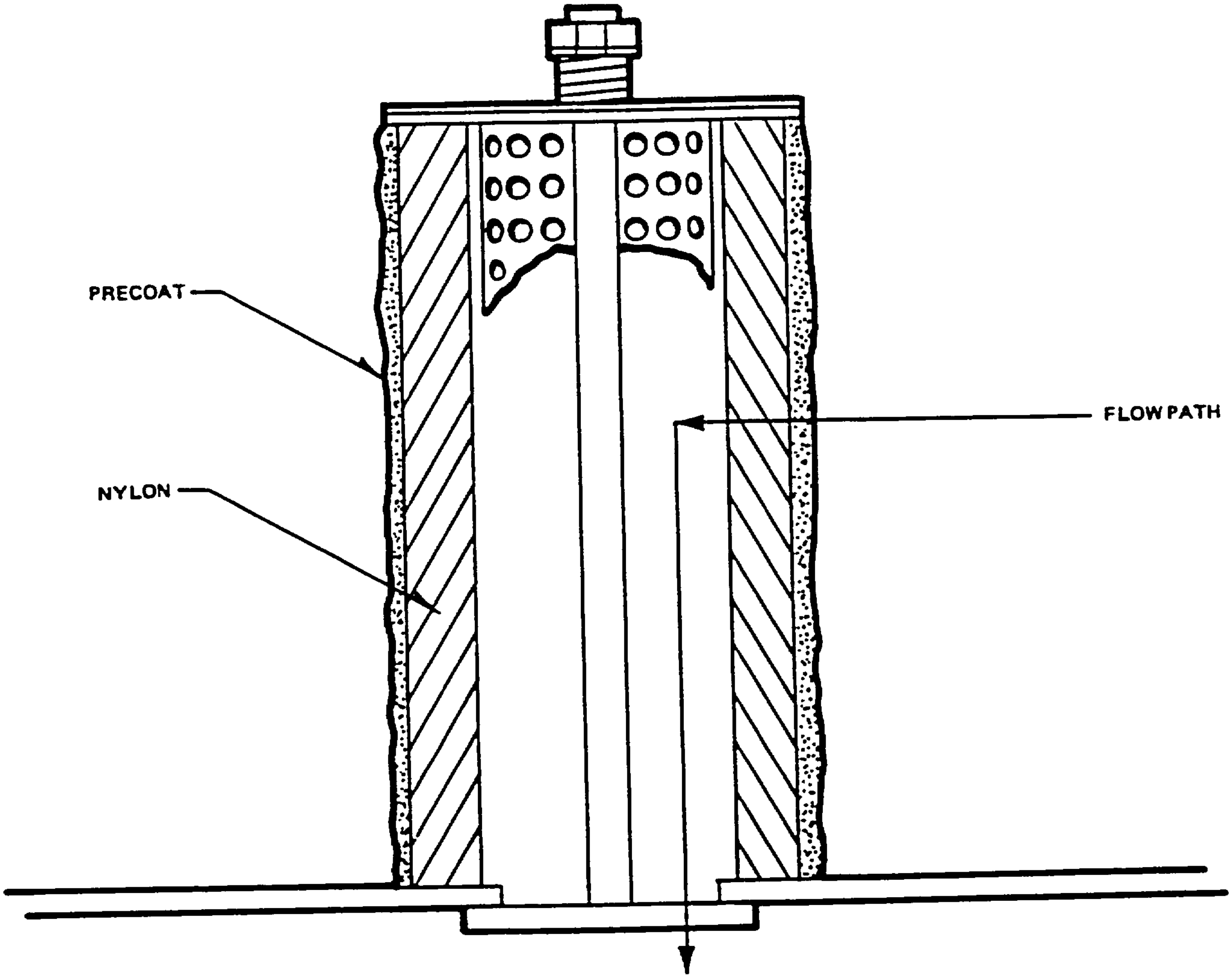

FIGURE 1.9 – 6 FILTER DEMINERALIZER PRECOATED ELEMENT

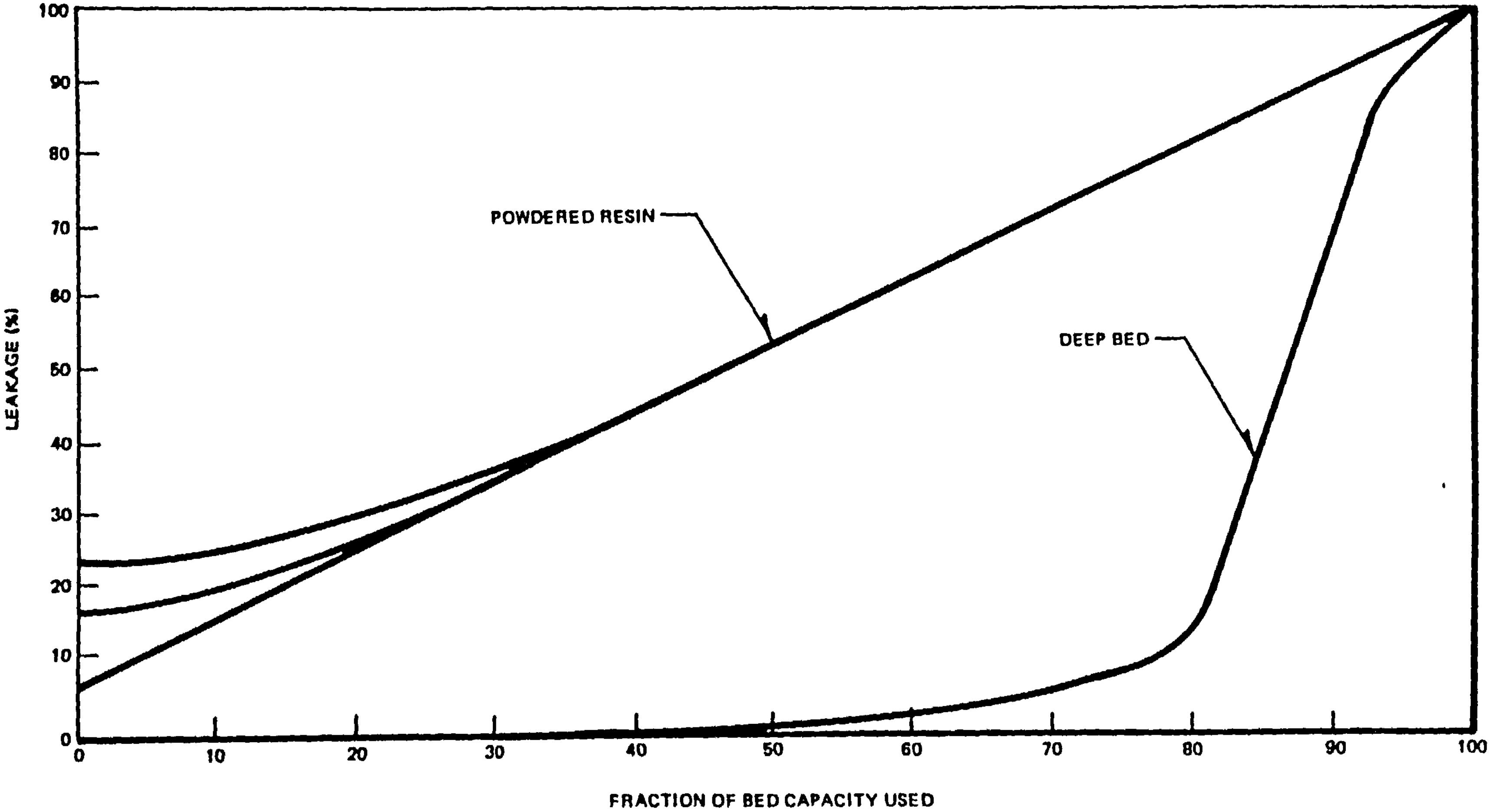

FIGURE 1.9 -- 7 PERCENT LEAKAGE VERSUS FRACTION OF BED CAPACITY USED

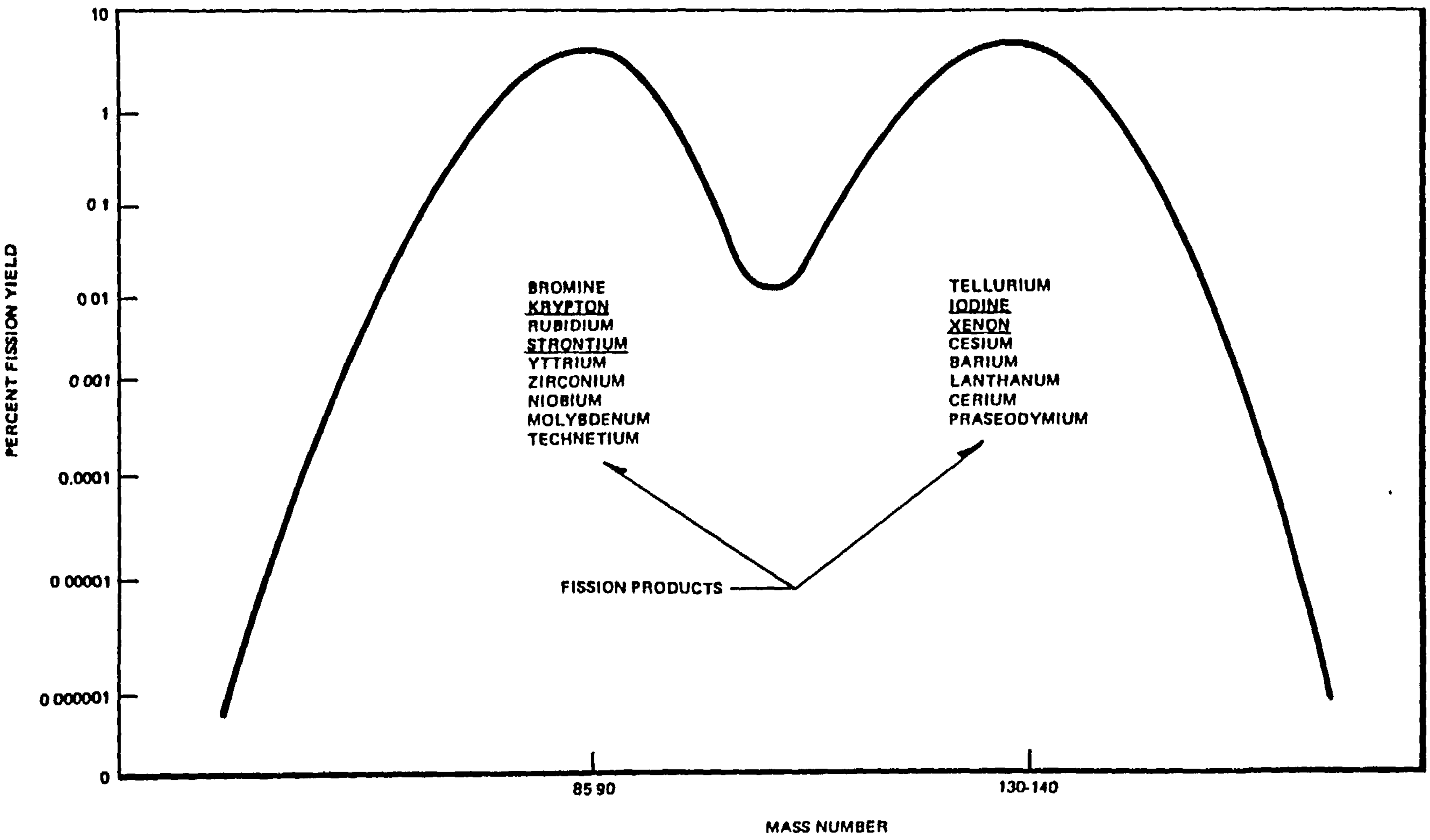

FIGURE 1.9 – 8 FISSION YIELD VERSUS FISSION PRODUCT MASS NUMBER FOR URANIUM–235

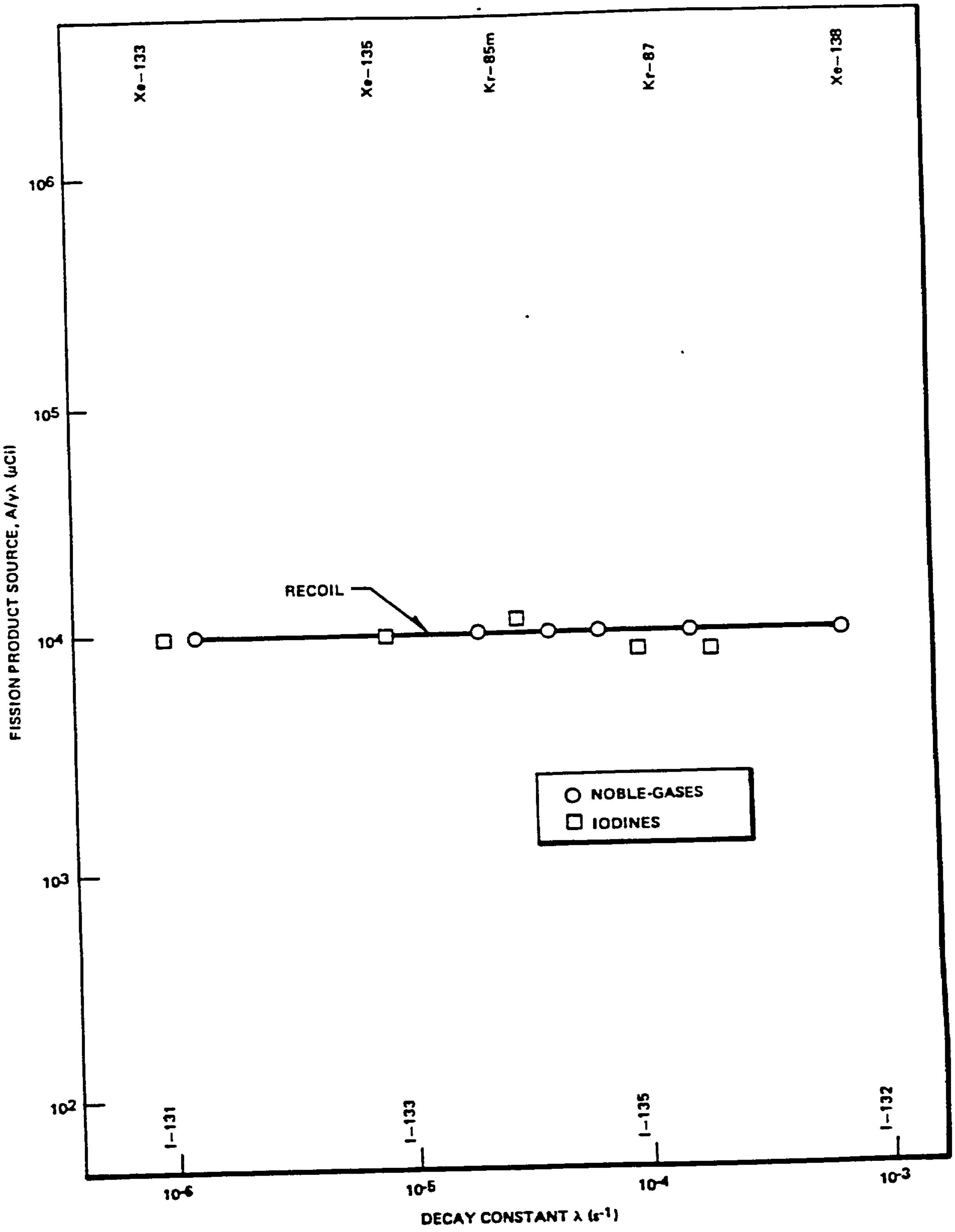

FIGURE 1.9 – 9 RECOIL–TYPE RELEASE

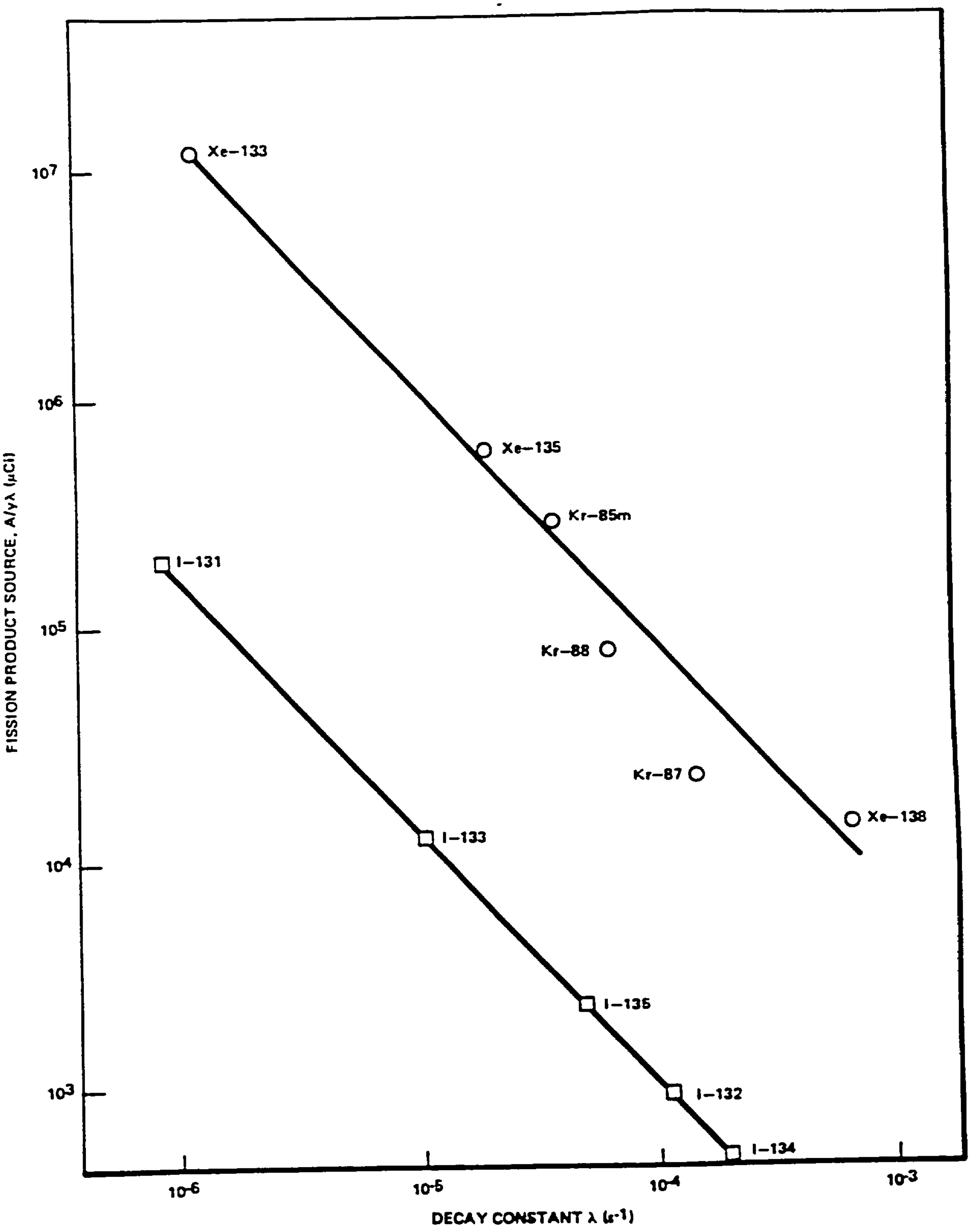

FIGURE 1.9 – 10 EQUILIBRUM–TYPE RELEASE FROM PINHOLE DEFECT

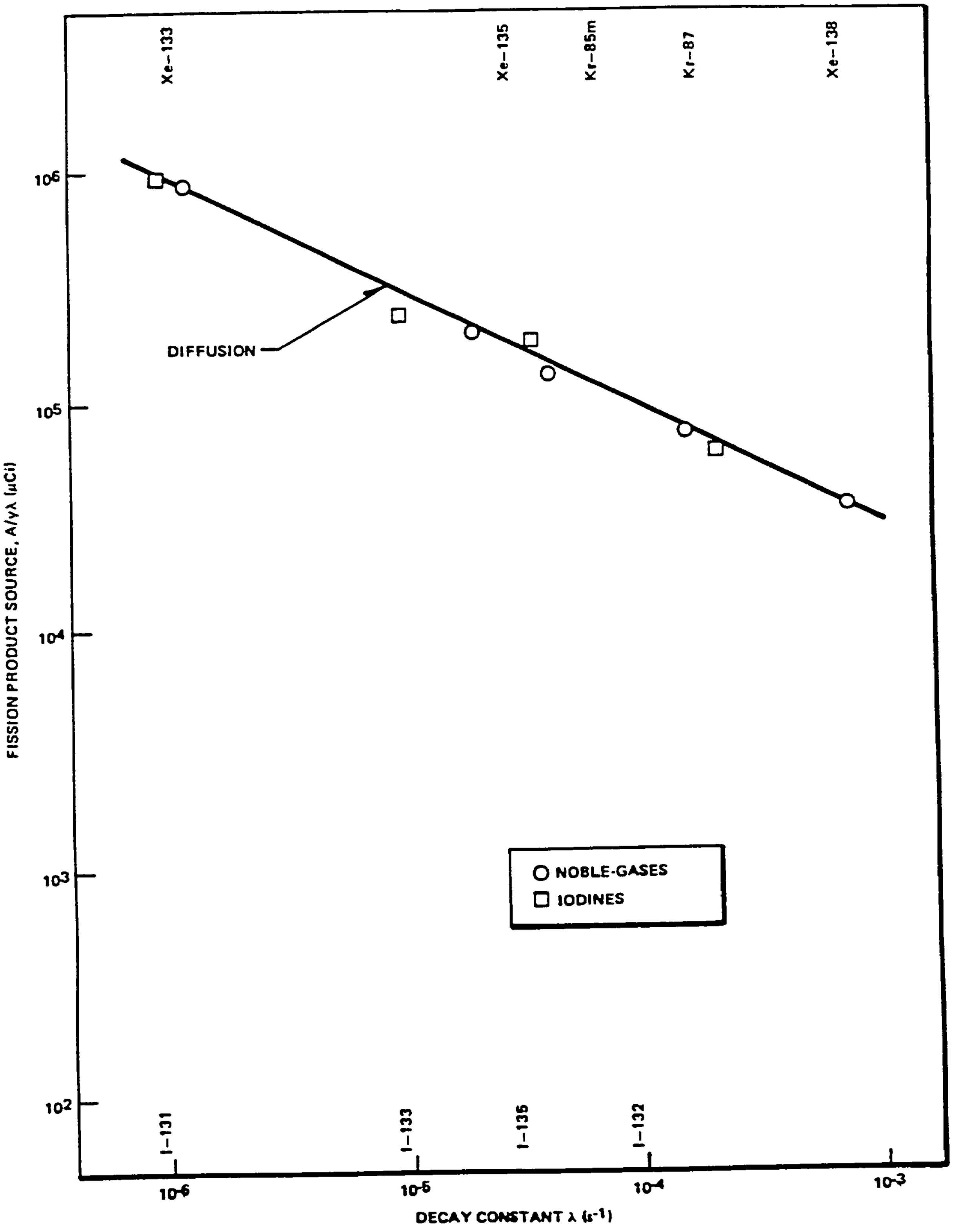

FIGURE 1.9 – 11 DIFFUSION–TYPE RELEASE